Chapter 3
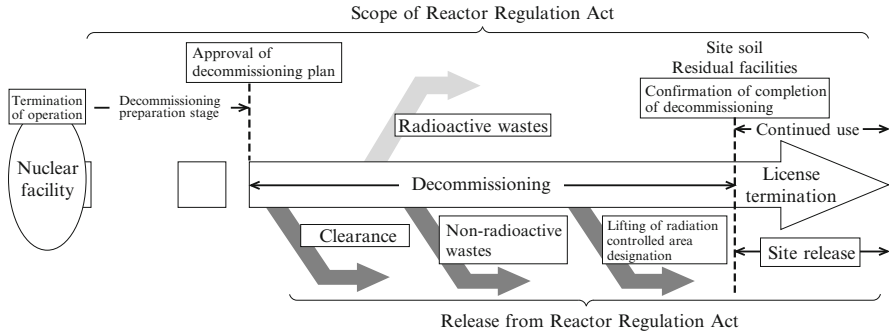# Decommissioning of Nuclear Facilities

Taro Shimada

3.1 What Is Decommissioning?

Decommissioning is a series of measures taken after the main activities associated with a licensed nuclear activity have been terminated and before the regulations set forth in the Act on the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors [1] (hereinafter referred to as "Reactor Regulation Act") are fulfilled. Decommissioning includes the transfer of nuclear fuel material, removal of contamination caused by nuclear fuel material, and disposal of nuclear fuel material or other materials contaminated with nuclear fuel material. Therefore, the dismantling of facilities, which is undertaken after the main activities have been terminated, is also included in decommissioning. Decommissioning is thus a process to reduce the residual radioactivity of such facilities to the levels necessary for fulfilling the regulations set forth in the Reactor Regulation Act. Because these measures produce various types of radioactive wastes in large amounts in a short period of time, the concept of radioactive waste management needs to be actively incorporated into the planning and implementation of decommissioning. If a decommissioning plan is not adequately formulated, there is a possibility that material that does not need to be handled as radioactive wastes may be improperly classified and handled as such. Furthermore, depending on the dismantling method selected, the amount of secondary wastes generated

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The nuclear regulatory system in Japan has been changed significantly after the Fukushima Daiichi Nuclear Power Station accident in March 2011. Descriptions in this chapter have been translated from the book originally published in Japanese before the accident, with minimal update where appropriate. The Nuclear Regulation Authority newly established after the accident has not completed its review for the guidelines and regulations established by the former Nuclear Safety Commission. In this chapter, guidelines set by the NSC have been adopted.

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Fig. 3.1 Nuclear facility decommissioning process flow

may increase or decrease and the disposal method for the wastes may also vary. It is therefore important to develop a decommissioning plan based on analytical evaluation, operating history surveys, measurement evaluation and other advance surveys as well as the latest dismantling technology studies. As explained above, there is a close relationship between decommissioning and radioactive waste management.

Instead of leaving radiologically-contaminated facilities or sites without any treatment and possibly affecting the health of the general public in their vicinity, removing residual radioactivity at the earliest stage possible is desirable also from the perspective of reducing the risk of radiation exposure to the public in the surrounding areas. Because removing the radioactivity allows buildings and sites to be used for other activities and thereby benefits society, safe and reasonable decommissioning activities are crucial.

As described in Fig. 3.1, decommissioning involves dismantling nuclear facilities that are regulated by the Reactor Regulation Act, as well as managing radioactive equipment and structures generated in the course of such facility dismantling as radioactive wastes in accordance with the provisions of the Reactor Regulation Act. Items with no history of contamination as well as those decontaminated by physically removing contaminated parts are handled as industrial wastes under the classification of "non-radioactive wastes". Contaminated items with radioactivity concentrations that meet or fall below the clearance level, which is set as a reference for a sufficiently low concentration level, are released from the regulations of the Reactor Regulation Act by applying the clearance system (see Chap. 4, Clearance) designed to facilitate the effective use of resources and the reduction of radioactive wastes. After equipment is removed from nuclear facilities, the remaining building structures are reused or dismantled as appropriate by applying the clearance, site release or other appropriate systems along with the lifting of the radiation controlled area designation according to operational safety program. The site will also be released from the regulations of the Reactor Regulation Act when the criteria for site release, which will be developed in the future, are met and after the confirmation of the completion of decommissioning measures is obtained from the regulatory authority.
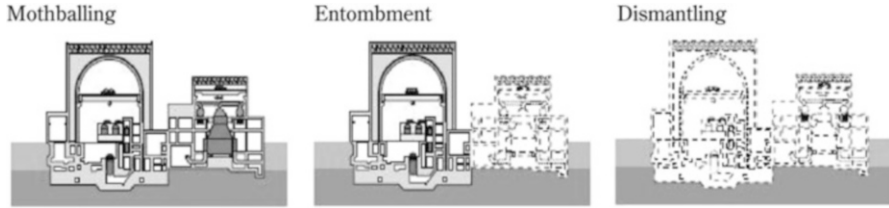
Fig. 3.2 Reactor facility decommissioning methods
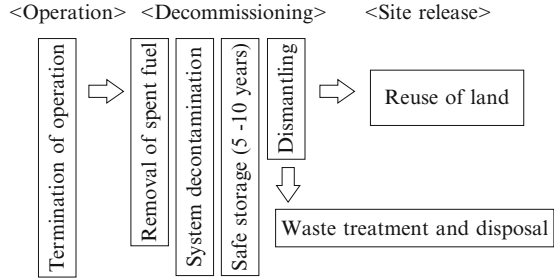
3.1.1 Decommissioning Method

While the methods used for decommissioning reactor facilities vary depending on the siting conditions of the facilities, regulations and standards used in each country, policies adopted by the licensee and other factors, there are three main types of decommissioning methods in general as shown in Fig. 3.2: mothballing, entombment and dismantling.

1. Mothballing: This is a method in which the entire facility is shut down and placed under appropriate control after spent fuel removal and system decontamination are completed. This method is also called safe storage.
2. Entombment: This is a method in which highly radioactive parts of the inside of the reactor building are entombed (openings are sealed by filling with concrete or other materials) and placed under appropriate control after spent fuel removal and system decontamination are completed, and the rest of the facility is dismantled.
3. Dismantling: This is a method in which the entire facility is dismantled after spent fuel removal and system decontamination are completed.

International Atomic Energy Agency (IAEA) classifies immediate dismantling, deferred dismantling and entombment. The U.S. Nuclear Regulatory Commission (NRC) also classifies deferred dismantling (SAFSTOR), entombment as in-situ disposal (ENTOMB), and immediate dismantling (DECON) [2]. In-situ disposal is a method in which a facility is mothballed over a long period of time and subsequently disposed of by burying it on site. This method, however, is no longer a possible option for decommissioning since it was recently decided in the U.S. that decommissioning must be completed within 60 years of the termination of operation. It should also be noted that the concept of "in-situ disposal (ENTOMB)" referred to above differs from that used in the field of radioactive waste disposal.

While all these classifications exist, decommissioning projects in practice are in most cases a combination of mothballing, which is carried out for a certain length of time to allow the radioactive decay, and dismantling, which takes place after the mothballing process.

Fig. 3.3 Standard process
of decommissioning

<Operation> <Decommissioning>   <Site release>

| Termination of operation | ⇨ | Removal of spent fuel | System decontamination | Safe storage (5 -10 years) | Dismantling | ⇨ | Reuse of land |

Waste treatment and disposal

The U.K. and some other European countries recently changed their policy on decommissioning and now intend to reduce risks by dismantling as early as possible facilities that have been mothballed for deferred dismantling. This change was made based on the understanding that while mothballing allows the decrease of residual radioactivity due to radioactive decay, it does little to reduce risks because radioactive materials remain on site during the process.

Meanwhile, Japan's standard approach to the decommissioning of a reactor facility is to dismantle it as soon as possible in principle following the termination of the reactor operation, while maintaining cooperation with the local community and continuing to effectively utilize the land as a site for nuclear power generation, based on the premise of ensuring safety.

The Advisory Committee for Energy of the Agency for Natural Resources and Energy of Japan proposed in 1985 that the standard process for the decommissioning of 1,100-MWe-scale nuclear power plants should involve 5–10 years of safe storage and then subsequent dismantling of facilities over a span of 3–4 years (Fig. 3.3). Based on the proposed standard process, the decommissioning cost was estimated to be about 30 billion yen per reactor. The Reserve Fund for Dismantling Nuclear Power Facilities was set up in 1989, and Japanese electric utilities have since been building up the reserve. The cost of the treatment and disposal of radioactive wastes produced by dismantling commercial nuclear plants was also added to the reserve in 2000. Since its amendment in 2005, the Reactor Regulation Act provides that decommissioning may be started once spent fuel has been removed from the reactor core.

3.1.2 Amount of Wastes Generated from Decommissioning

Large amounts of wastes are generated when a nuclear facility is dismantled. Table 3.1 shows the estimated amounts generated at the nuclear power plants currently undergoing decommissioning. The data indicate that most of the wastes generated from decommissioning nuclear plants are cleared items or non-radioactive waste, which are not required to be handled as radioactive

Table 3.1 Amounts of wastes generated from decommissioning nuclear power plants (unit: ton)

Radioactivity level classification		Tokai Power Station (GCR)	Advanced thermal reactor Fugen	Hamaoka Nuclear Power Station (BWR)	
				Unit 1	Unit 2
Low-level radioactive wastes	Waste with relatively high radioactivity concentration	1,530	260	100	100
	Waste with relatively low radioactivity concentration	8,870	1,380	1,000	1,200
	Waste with very low radioactivity concentration	13,080	45,460	6,300	7,900
Wastes not required to be handled as radioactive materials (cleared items)		40,160	510	11,200	13,400
Non-radioactive wastes		128,700	141,000	192,700	249,500

Source: Websites of the Japan Atomic Power Company, Japan Atomic Energy Agency, and Chubu Electric Power Co., Inc.

materials; therefore, radioactive wastes account for only a fraction of the total amount of wastes.

Upon disposing of radioactive wastes, the landfill disposal method is used for wastes with a very low concentration of radioactive materials, while the concrete vault disposal method is used for wastes with a relatively low concentration of radioactive materials. For wastes that are removed from inside the reactors and contain a relatively high concentration of radioactive materials, the subsurface disposal method is envisaged (Chap. 6, Radioactive Waste Disposal). The clearance system, which provides that materials that are generated from nuclear facilities and have extremely low levels of radioactivity concentration are not required to be handled as radioactive materials, was introduced in December 2005 and has been applied to some nuclear power plants and research facilities.

3.2   Formulation of Decommissioning Plan

The licensee of a nuclear facility who intends to decommission the facility is required to draw up a decommissioning plan pursuant to the Reactor Regulation Act and apply for the approval of the regulatory authority. It is required that the decommissioning plan include: the dismantling methods; removal of contamination caused by nuclear fuel materials; disposal of nuclear fuel material or other materials contaminated with the nuclear fuel materials; and the decommissioning process. In addition to the above information to be included in the main text of the document, the decommissioning plan must also be accompanied by supplementary documents

explaining the management of radiation exposure; consequences of credible accidents; distribution of contamination caused by nuclear fuel materials and its evaluation method; and facilities and performance that need to retain their functions during decommissioning and the duration required for those functions.

The criteria for the approval of a decommissioning plan are: (1) spent fuel has been removed from the reactor; (2) plans for the management and transfer of nuclear fuel materials are appropriate; (3) plans for the management, treatment and disposal of contaminated materials are appropriate; and (4) plans for the prevention of disasters resulting from the reactor or materials contaminated with nuclear fuel materials are appropriate.

With respect to criteria (3) and (4), the subsequent sections provide detailed explanations on the evaluation of the distribution of contamination caused by radioactive materials, development of a dismantling work plan, management of radiation exposure, and the evaluation of the consequences of credible accidents (safety evaluation). These are considered particularly important among all information to be included in the main text of a decommissioning plan and its accompanying documents.

3.2.1   Evaluation of Residual Radioactive Materials

The evaluation of radioactive materials remaining inside a facility to be decommissioned involves the evaluation of activated radioactivity, which is the source of contamination, and the evaluation of secondary contamination caused by various factors such as the migration of the generated radioactivity or fission products.

3.2.1.1   Evaluation of Activated Radioactivity

Activation is a process in which neutrons generated by nuclear fission or other processes undergo nuclear reactions with coolant, air, or the materials that constitute nearby equipment and structures, and produce radionuclides in them. Radioactivity generated through activation is called activated radioactivity or induced radioactivity. Typical nuclear reactions that cause activation are (n, r), (n, p), $(n, 2n)$, (n, np), (n, d), and (n, α) reactions. During the operation of a reactor, radionuclides are generated constantly through activation and transform to different nuclides due to radioactive decay. Figure 3.4 illustrates an example of the activation reactions and decay series of a representative element in a metallic structure. To evaluate activated radioactivity, the generation and decay of radionuclides produced through activation are evaluated based on the distribution of the neutron fluence rates, which are obtained by calculation and measurement, inside the operating reactor and its surroundings. The results are then used to determine the concentration and quantitative distribution of radionuclides at a specific point in
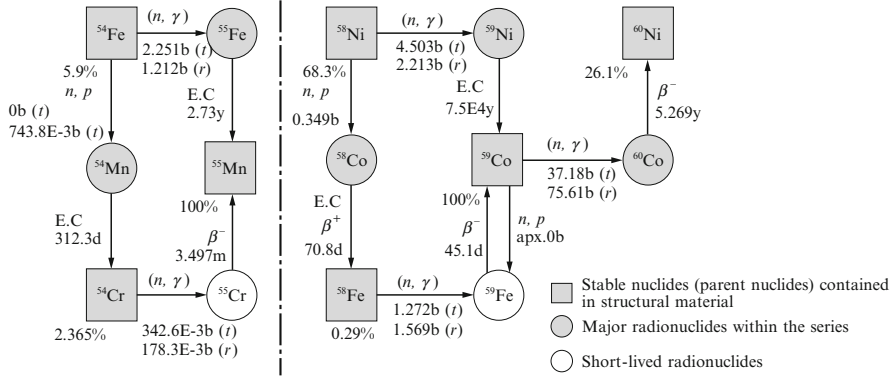
Source: Chart of the Nuclides (JAEA) and Radiation Data Book (Chijinshokan Co., Ltd., Tokyo)

Fig. 3.4 Example of reaction series in metallic structure induced by neutron irradiation – b barn, (*t*) thermal neutron capture, (*r*) resonance integral

time. An example of a flow used for calculating activated radioactivity is presented in Fig. 3.5. The three main procedures are described next.

1. Evaluation of neutron fluence rate distribution

The calculation of neutron fluence rates is made in principle by numerically solving the Boltzmann transport equation using the Sn method (discrete ordinate method). The radius of activation induced by neutron irradiation during the reactor operation is limited in part to equipment and structures surrounding the reactor. The general calculation procedure when using transport calculation codes of the Sn method, such as the two-dimensional DORT or three-dimensional TORT, is as follows. First, create the geometric model of the structure subject to evaluation (from the reactor to the biological shield) while creating a neutron macroscopic effective cross section library according to the material composition of each region using nuclear data libraries such as JENDL-4.0 [3] and ENDF/B-VII [4]. Secondly, input source strength data including the neutron energy spectrum in the reactor core as well as the number and distribution of neutrons generated in the entire reactor core area, and set the Legendre expansion order for scattering as well as other calculation parameters for the Sn method. The neutron fluence rate distribution in each mesh region of the geometrical model should then be calculated using transport calculation codes of the Sn method. For areas where neutron streaming occurs, such as ducts and experimental holes, do calculations separately using the Monte Carlo method and make appropriate corrections where necessary.

2. Evaluation of activated radioactivity

Activated radioactivity is calculated using burnup calculation codes, most typically ORIGEN. The amount of radionuclide *i* generated in a specific area (dXi/dt) is calculated based on neutron fluence rate φ using the following equation.
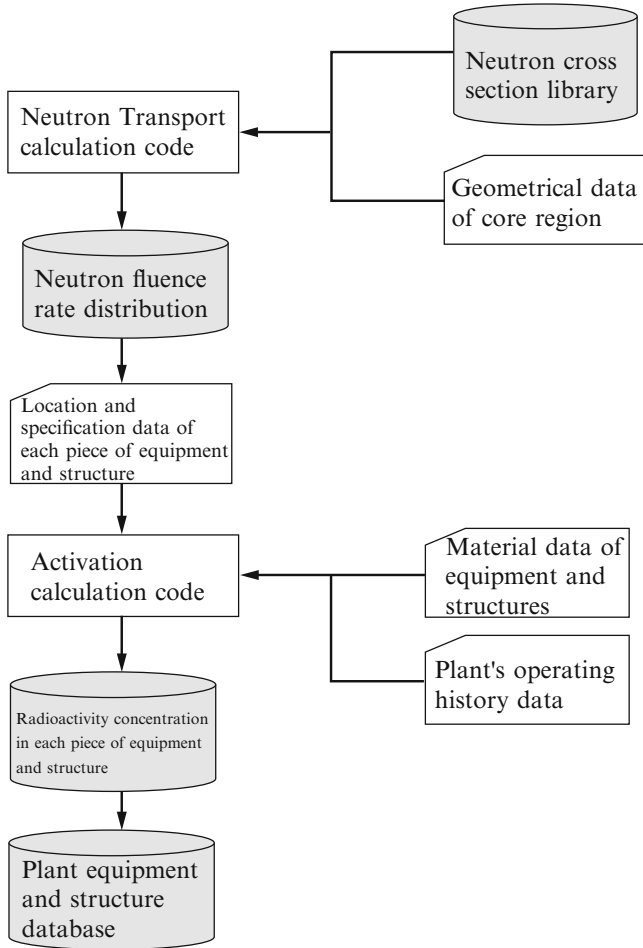
Fig. 3.5 Example of activation calculation flow

$$\frac{dX_i}{dt} = \phi \sum_{k \to i} f_{ik} \sigma_k X_k + \sum_{j \to i} l_{ij} \lambda_j X_j - (\lambda_i + \phi \sigma_i) X_i + y_i F \tag{3.1}$$

X_i: atom density of radionuclide i generated through activation [n/m^3]

X_k: atom density of target nuclide k [n/m^3]

σ_k (E): neutron absorption cross section of target nuclide k [m^2]

f_{ik}: generation rate of nuclide i when absorbed by nuclide k

λ_i: decay constant of radionuclide i [s^{-1}]

l_{ij}: generation rate of nuclide i when radionuclide j decays

φ: neutron fluence rate averaged in required space integrated over the entire energy range [n/(cm · s)]

σ_i: one-group reaction cross section averaged over spectrum load [m^2]

y_i: fission yield of nuclide i

F: fission rate

The activated radioactivity concentration in each mesh region is calculated by constructing a histogram (hour-thermal output histogram) for the activation calculation based on the operating history of the reactor to be evaluated. For a short-cut calculation, the effective full power year (EFPY) value may be used instead of a histogram. The value is multiplied by the volume of the region to determine the total amount of activated radioactivity contained in the region.

In activation calculations, some trace elements that are negligible when creating the neutron cross section may affect the accuracy of calculations. The amount of impurities contained in a metallic material or concrete may have a significant effect on the calculation results. It is therefore necessary that these values are set properly according to mill certificates or actual measurements.

3. Comparison with measurement results

While the concentration of activated radioactivity can be evaluated by calculation in principle using the above procedure, it is difficult to make an accurate evaluation on the entire system of a large-scale reactor. For such systems, evaluation needs to be made after measuring the radioactivity of samples taken from representative locations, comparing the results with calculated values and making necessary corrections. For example, the results of measurements taken by a neutron detector or gold samples irradiated during operation should be compared with calculation results when evaluating neutron fluence rates. For the concentration of activated radioactivity, comparisons should be made between calculation results and the measurement results of the samples taken from the biological shield by boring machine.

3.2.1.2 Evaluation of Secondary Contamination

Secondary contamination occurs due to various phenomena such as the leakage of nuclear fuel materials or fission products from the fuel, the release of fission products from uranium adhered to the surface during fuel manufacturing, the dissolving of metals in structural materials, the activation of structural materials, or the deposition or peeling of structural material surfaces where in contact with liquid. The contamination spreads over the entire system including equipment and piping. If a leakage of liquid or gas containing radioactive material occurs, the material may adhere and deposit on the surfaces of building structures and cause surface contamination. The level of contamination caused by radioactive material on the inner surface of the system in contact with liquid varies depending on the operating history of the facility, fuel conditions and other factors. For this reason,

the evaluation is usually made using the values actually measured on site although attempts are being made to evaluate the level of contamination using calculation codes based on the radioactivity balance. For example, because the compositions of radionuclides found inside pipes carrying coolant with common characteristics, such as those in a reactor cooling system, are mostly the same, the level of radioactivity contamination remaining in each system is evaluated by combining sample collection at representative points and the measurement of dose rates taken outside the pipes.

3.2.2 Development of a Decommissioning Plan

In order to carry out decommissioning smoothly, important steps include formulating a dismantling plan, estimating the number of workers required, and evaluating necessary costs before work is commenced. Upon preparing the dismantling plan, it is recommended that evaluation be made in advance on the characteristics (weight, shape, material, amount of radioactivity, location, etc.) of the equipment and structures present in the facility and a database be created to allow necessary information to be retrieved efficiently. Dismantling work for a nuclear facility consists of a combination of various steps including dismantling using conventional methods, remote dismantling, decontamination and measurement of radioactivity in buildings, and the dismantling of buildings. Each of these steps also involves preparation, cutting, storage and cleanup work. In the planning, these work steps are broken down to formulate a work breakdown structure (WBS). A decommissioning plan is developed by first identifying all necessary work steps as well as determining work conditions and procedures, followed by the restructuring of the work steps. This will allow relevant management data to be calculated including the number of required man-hours, exposure doses of workers, costs, and the amounts of wastes generated, thereby facilitating the development of a rational plan.

One of the tools developed to assist the formulation of a practical decommissioning plan is a Code System for Management of Reactor Decommissioning (COSMARD) [5], a computer program that uses a work package in which work configuration and conditions necessary for the dismantling work are assigned as the input conditions. And it calculates management data for each basic work step (work unit) or each set of such work steps based on information prepared in advance on the physical quantities and residual radioactive materials of the equipment and structures used in the facility (physical inventory database), dose rates of the work areas (work environment database) and calculation models for the management data (work unit database). The work unit database is an aggregation of calculation models written in a simplified language including work unit factors, which indicate the number of man-hours required for dismantling work for each unit weight. Management data associated with the dismantling of equipment and structures are calculated using calculation formulas and numeric values.
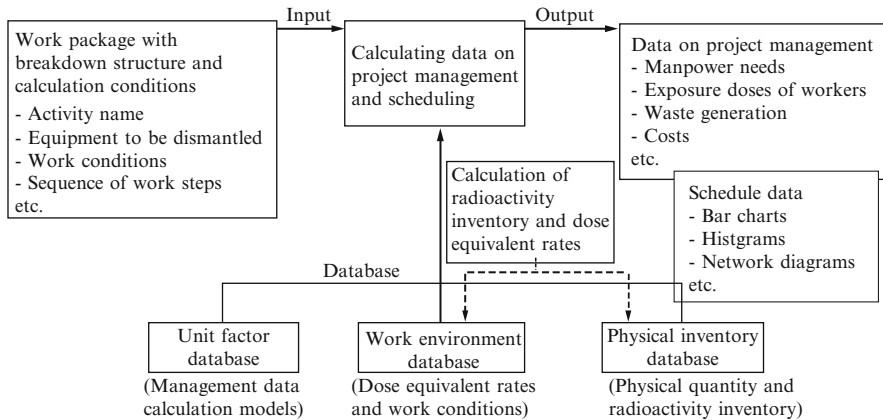
Fig. 3.6 Concept of COSMARD [5]

The concept of the management data calculation using COSMARD is described in Fig. 3.6. COSMARD consists of a management data calculation program as well as an auxiliary program to calculate residual radioactive materials and dose rates, which are necessary for calculating management data. Regarding the work configuration required as part of the input conditions, a work unit or a set of work units can be written using the work breakdown structure. In addition, work conditions can be assigned to each work unit individually where applicable.

3.2.3 Evaluation of Safety

During dismantling work, radioactive gases or particulates may be released into the work environment due to operations such as cutting of objects contaminated with activated radioactivity or radioactivity from secondary contamination. Some of these gases or particulates may be carried through the filters of the exhaust system of the facility building and released into the atmosphere from the ventilation stack. When liquid is used to cut or treat objects, radioactive materials may be transferred to the liquid phase, and some may be discharged into the ocean from the outlet of the facility as liquid waste through the drainage system.

These substances can cause radiation exposure of the general public in the vicinity of the facility. When developing a decommissioning plan, safety must be confirmed in advance by evaluating the expected exposure dose for each exposure pathway through the calculation of the amount of radioactivity released each fiscal year based on the residual radioactivity inventory as well as the types and processes of dismantling work.

Radioactive materials may also be released into the atmosphere in a short period of time if an accident such as a fire or explosion occurs during cutting or other operations that use a flame. Safety must be confirmed in advance by identifying

accidents that may occur during decommissioning and calculating the amount of radioactivity released into the atmosphere in the event of such accidents to evaluate the exposure doses of the general public in the vicinity of the facility. It is also necessary to calculate the expected collective exposure doses based on the work plan to ensure that the decommissioning plan will limit the radiation exposure of workers engaged in the dismantling work to the level as low as reasonably achievable.

3.2.3.1   Evaluations of the Exposure Doses of the General Public in the Vicinity of the Facility at Normal Situation

These evaluations can be classified into two steps: released radioactivity evaluation, which covers the period to the release of radioactive gases or airborne particulates into the environment due to operations such as the cutting of equipment or structures; and exposure dose evaluation, which assesses the exposure doses expected after the release of radioactivity.

Released radioactivity evaluation is done for the assumed dismantling work including the cutting and decontamination of specific objects and the storage of the object into waste containers. It also includes the assumed radioactive particulates generated during the dismantling work that are transferred to the gas or liquid phase, and that pass through filters or leak from a contamination control enclosure or building and are released into the atmosphere or ocean. Figure 3.7 illustrates the atmospheric discharge pathways of radionuclides. Released radioactivity is evaluated separately for each of these pathways. Transfer to the gas phase is evaluated taking into consideration the dispersion rate of each nuclide and the collection efficiency of filters. As the inside of the building is usually maintained under negative pressure at normal situation, the leakage from the building is considered negligible and the leakage rate is therefore set as zero in the calculations. In the case of dispersion to the liquid phase, the amount released into the ocean is evaluated taking into consideration the decontamination factor of each nuclide by the liquid treatment system.

The evaluation of exposure doses expected following the release of radioactivity estimates the amount of surface deposition of radionuclides released into the atmosphere using meteorological data such as wind direction and speed for each bearing and taking into consideration radioactivity concentration in the atmosphere and surface deposition velocity. The external and internal exposure doses of the general public in the vicinity of the facility are then calculated for each exposure pathway.

Furthermore, exposure dose from direct and skyshine radiation caused by temporarily stored waste containers with dismantled objects is also evaluated by calculating the amount of the containers generated and taking into consideration the layout of the containers inside the building. The series of evaluations is coordinated with work processes and conducted each fiscal year.
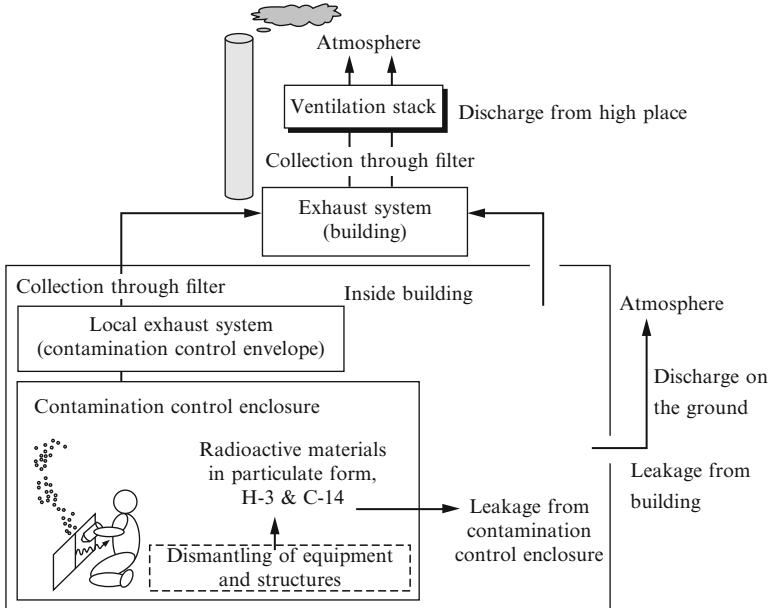
Fig. 3.7 Atmospheric discharge pathways of radionuclides at normal situation

3.2.3.2 Evaluation of the Exposure Doses of the General Public in the Vicinity of the Facility at an Accidental Situation

Table 3.2 lists postulated events at the time of an accident during decommissioning as well as the locations of major mobile inventories. A mobile inventory is the mass of radioactive materials that may possibly be dispersed in the work environment due to combustion or other phenomena when an event occurs. Because combustion temperature is low in a normal fire, the dispersion of radioactive gases or airborne particulates caused by the melting or evaporation of activated base metal is not factored into the evaluation.

This evaluation is performed in the order of: (1) calculation of each mobile inventory accumulated according to the types of dismantling work; (2) calculation of the amount of radioactivity released into the atmosphere based on the mobile inventory expected when a postulated event occurs and the dispersion rate of the mobile inventory into the air; and (3) evaluation of the exposure dose of the general public in the vicinity of the facility for each exposure pathway.

In (1), some of the results of the evaluations conducted on released radioactivity at normal situation are used to calculate the amount of accumulated or deposited mobile inventory that may disperse into the air when a postulated event occurs. The calculation is made separately for each object to be dismantled, taking into consideration the collection efficiency of the filters and the adherence and sedimentation of radioactive particulates. More specifically, evaluation is made by referring to the

Table 3.2 Postulated events and locations of major mobile inventories at the time of occurrence of each event

Classification	Postulated events	Locations of major mobile inventories
Fire	Local filter fire	Inside of filter (immediately before filter replacement)
	Building filter fire	Inside of filter (immediately before filter replacement)
	Contamination control enclosure fire	Adhered to inner walls of a contamination control enclosure or objects to be cut
	Fire from cardboard boxes for temporary storage	Temporary waste storage
Explosion	Unplanned explosion of explosives intended for controlled blasting	Wastes, etc. associated with temporary storage near explosive storage area
	Accidental explosion during controlled blasting operation	Biological shielding concrete
	Explosion of flammable gas	Adhered to inner walls of a contamination control envelope
Filter damage	Local filter damage	Inside of filter (clogging)
	Building filter damage	Inside of filter (clogging)
Loss of electric power	Loss of external power supply	Airborne radioactive material in work space
Drop	Waste container drop	Dross from cutting operation, waste ion exchange resin, etc.
	Cut piece drop	Pieces from cutting activated or contaminated equipment, etc.
	Fuel assembly drop	Spent fuel in transit

general specifications of HEPA filters and setting the maximum amount of mobile inventory that can be held by filters in a usable condition, as well as the amount held by filters when mobile inventory is accumulated to the point that filters are damaged by rising differential pressure. The amount of accumulated radioactive particulates at the point that the above set amount is reached is then calculated separately for each nuclide.

In (2), the total mobile inventory where an accident has progressed to the final stage is calculated. The amount of radioactivity released into the atmosphere in a short period of time during such an accident is also calculated for each nuclide and mobile inventory.

Calculations in (3) are made independently from the evaluation method that uses statistically processed meteorological data for normal situation. This step calculates external exposure to radiation caused by radioactive gases, airborne particulates or other substances in the atmosphere, as well as internal exposure doses from inhalation of such radioactive materials, after radioactivity is released into the environment as a result of a certain event. Out of 8,760 relative concentration and dose values calculated based on meteorological observation data recorded for each hour according to the meteorological guide [6] for the time of an accident, values that represent 97 % annual cumulative frequency of occurrence are applied to these calculations.

3.2.3.3 Exposure Doses of Workers

A collective external exposure dose (D_{ex} [Sv]) of workers engaged in dismantling work is calculated by multiplying the average dose rate (R_{ijk} [Sv/h]) at the position of a worker of each occupation by work hours (T_{ijk} [h]), and is presented as follows.

$$D_{\mathrm{ex}} = \sum_{i=1}\sum_{j=1}\sum_{k=1} R_{ijk} \cdot T_{ijk} \tag{3.2}$$

i: occupation (worker, supervisor, technical supervisor, radiation control manager, etc.)

j: dismantling work classification (preparation, dismantling or cleanup)

k: daily work classification (entry to or exit from area, preparation, dismantling (main or assistance), cleanup, etc.)

The average dose rate at the position of a worker of each occupation changes with the progress of the dismantling work, and such changes vary depending on the layout of equipment and the order of removing the equipment. In normal cases, equipment having a high dose rate is removed first to reduce worker exposure. However, in cases where such equipment is not easily accessible, other pieces of equipment located around it are dismantled first. In many cases the work is performed in crew units consisting of multiple occupations such as a supervisor, workers and a radiation control manager. While workers undertake the cutting operation near equipment, the supervisor is responsible for providing directions and assistance from a location where the entire operation can be observed. A different dose rate therefore needs to be set for each different occupation.

The position of a worker in a work area and the amount of time spent therein vary depending on the worker's occupation and the classification of work performed. Work classifications within a work process are preparation, dismantling and cleanup, while daily work classifications consist of work units such as entry to or exit from the area, preparation, dismantling and cleanup. Work hours for each work unit and effective work hours for each occupation taking into account the allocation of time in a day can be evaluated using the WBS formulated in Sect. 3.2.2, "Development of a decommissioning plan."

Meanwhile, a collective internal exposure dose (D_{in} [Sv]) is calculated by multiplying the radioactivity concentration in the air inhaled by a worker ($C_{ijk}(t)$ [Bq/m^3]) by work hours (T_{ijk} [h]), breathing rate (R_{ik}) and effective internal exposure dose conversion factor [7] for each nuclide (A_m [Sv/Bq]), and is presented as follows.

$$D_{\mathrm{in}} = \sum_i\sum_j\sum_k C_{ijk}(t) \cdot T_{ijk} \cdot R_{ijk} \cdot \frac{1}{W_p} \cdot A_m \tag{3.3}$$

W_p used in the above equation indicates a protection factor, which is a number quantifying the effectiveness of protective equipment. For example, the protection

factor is 10 for a half-face respirator, 50 for a full-face respirator and 2,000 for a ventilation suit.

Radioactivity concentration in air at the work space changes significantly in terms of space and time in the course of each operation. The evaluation of radioactivity concentration in air must be conducted applying area and hours that are accurate in light of the actual conditions of each operation. For example, when evaluating the exposure of workers engaged in the cutting operation inside a contamination control enclosure based on the radioactivity concentration in air, calculation should use the average radioactivity concentration of the small area around the cut section. Evaluation for workers not engaged in the cutting operation should use the average radioactivity concentration in air in the entire contamination control enclosure. Similarly, the amount per unit of time of radioactive gases and particulates, which are generated and dispersed in the air due to the cutting of equipment or structures, is dependent on the kerf width and cutting speed of the method used; therefore, data that correspond to the actual cutting conditions must be prepared in advance of evaluation.

3.3 Decommissioning Techniques

3.3.1 Decontamination Techniques

Decontamination conducted upon decommissioning is an effective technique for reducing worker exposure during dismantling work as well as for reducing radioactive wastes. Decontamination can be divided into system decontamination and decontamination after dismantling depending on the timing of the work applied.

Objects subject to decommissioning may be of various materials and shapes, and can have various forms of contamination. There are also many decontamination techniques based on different principles. Optimum techniques should be selected when carrying out decontamination, taking into consideration the purpose of the decontamination work, characteristics of objects to be decontaminated and the cost-effectiveness of using each technique. Representative decontamination techniques are listed in Table 3.3.

An overview of decontamination techniques applicable to decommissioning is given below for system decontamination and decontamination after dismantling depending on the timing of work applied.

3.3.1.1 System Decontamination

While system decontamination is a method also used during service, it is used in decommissioning prior to dismantling work to remove contaminants attached to the inner surface of the system by supplying a decontamination solution into piping that

Table 3.3 Representative decontamination techniques

Object	Classification	Principle	Decontamination techniques
Metallic equipment or structures	Physical	Mechanical method	Brush polishing, grinding, wiping
		Jet method	Steam, high – pressure, ultrahigh – pressure
		Blast method	Dry ice, CFC, water, ice, metallic particle (steel, zirconia, alumina), sponge, etc.
		Vibration method	Ultrasound, vibrating machine
		Laser method	YAG, excimer, CO_2 lasers
		Other	Microwave, melting, strippable coating, thermal shock
	Chemical	Immersion method	Acid (HCl, H_2SO_4, HNO_3, HF, etc.), alkali, oxidant, reductant
		Foam method	Chelator, surfactant, corrosion inhibitor
		Gel method	Combination of Gallant and acid (HCl, H_2SO_4, HNO_3, HF, etc.)
		Paste method	Decontamination agent + filler + carrier
		Other	Formic acid decontamination method (formic acid for reductive dissolution)
			NP – LOMI method (potassium permanganate for redox dissolution, formic acid, vanadium, picolinic acid)
			CORD method (permanganic acid for redox dissolution, oxalic acid, hydrogen peroxide solution UV)
			DfD method (fluoroboric acid for redox dissolution, potassium permanganate)
			Cerium (IV) decontamination method (cerium sulfate for oxidative dissolution)
	Electrochemical	Electrolytic method	Electrolytic polishing, dilute sulfuric acid, phosphoric acid, etc.
	Combined	Electrochemical & physical	ECB method (Electrolytic polishing and mechanical polishing)
		Chemical & physical	Chemical solution and vibration (ultrasound and machine)
		Chemical & electrolytic regeneration	Redox ($Ce^{3+} \rightarrow Ce^{4+}$), hydroxyl radical ($Ag^+ \rightarrow Ag^{2+} \rightarrow \cdot OH$)
Concrete surface	Cutting	Mechanical	Milling cutter (planer), grinder
		Cutting	Water jet, blast (several types of abrasives)
	Crushing	Impact crushing	Scabbling, breaker
		Splitting	Spaller
	Thermal		Laser, microwave
	Chemical		Chemical dissolution, electrochemical, biochemical, and coating methods

forms a single loop, such as of a reactor cooling system. This will significantly reduce the dose rate around the piping, which can help reduce the level of worker exposure. The representative techniques include the LOMI method, which uses reductive dissolution, and the CORD method, which uses oxidation-reduction (redox) dissolution. When a facility is in operation, the use of powerful decontamination solutions that may have serious effects on the base metal is avoided in order to maintain the equipment performance. On the other hand, because such effects do not need to be considered during decommissioning, the cerium (IV) decontamination method and the DfD method, which can dissolve base metal, are effective. However, there may be some variations in the effectiveness of system decontamination as a result of the effects of the oxide film on the surface layer or the flow conditions inside the system. Another disadvantage is a large volume of secondary waste generated from filling decontamination solutions in the piping and other parts of the system, necessitating the treatment of decontamination solutions and disposal of secondary wastes.

3.3.1.2 Decontamination after Dismantling

Contrary to system decontamination, decontamination after dismantling is a method carried out after equipment is removed from its original position. Techniques include physical, chemical, electrochemical and thermal decontamination. Decontamination after dismantling does not have a substantial effect in reducing worker exposure; it is rather effective in lowering the classification of radioactive wastes by one or more levels. For example, a radioactivity concentration of waste subject to concrete vault disposal without decontamination may possibly be reduced to the level only requiring landfill disposal after decontamination, thereby reducing the cost of disposal. It is important to fully understand the surface contamination density and characteristics of contamination in advance and determine to what level the surface contamination density can be reduced by decontamination after dismantling.

While the immersion method is used for chemical decontamination in general, this method requires flow of the decontamination liquid. It is also difficult to remove both the oxide film and base metal together, and additional treatment may be needed to dissolve the base metal. The blast decontamination and polishing methods can remove the base metal through physical grinding and are therefore effective for decontamination after dismantling; however, contamination is transferred to the blasting material or the abrasive, for which treatment or disposal needs to be arranged. Laser decontamination is capable of removing the oxide film and base metal at one time by turning up the output level, and is very effective in treating localized contamination. Nevertheless, it is inefficient in decontaminating a wide area in light of its treatment area and rate, which leaves room for improvement in its applicability.

3.3.2 Remote Techniques

In nuclear facility dismantling, the facility may have some areas that cannot be accessed easily due to equipment or structures highly activated or contaminated with radioactivity. Work in these areas are carried out by using remote devices or remote operations such as underwater cutting of equipment, which is performed while equipment is placed in water used as shielding material.

Remote devices employed in dismantling work in the past have included a robot manipulator type, mast (rotating pillar) type, and mast-arm type, which is a combination of the first two. A manipulator with a mast allowing rotation, upward and downward movements and multiple degrees of freedom is attached with a cutting device to its tip to perform dismantling operations. There are several methods of controlling a robot manipulator, such as by sending real-time commands for rotation angle of 6–7 axes from so-called master-slave type equipment, or by specifying the locus of the coordinates of the tip in advance on the computer for adjustment of automatic standoff (distance between the tip of the torch and the surface of the object to be cut) and attitude control to ensure optimum cutting performance. The most desirable method is selected after considering cost-effectiveness and other factors concerning each method.

In the underwater cutting of reactor core internals, dross generated from the operation increases the turbidity of water, leading to reduced visibility and increased radioactivity concentration in the water. Particulate matter floating in the water needs to be collected by filters or other means to prevent turbidity, and measures must be prepared to protect workers from exposure while engaging in this process.

There are also other technologies under development, including a mouse-like robot that has a built-in radiation detector and is able to move freely inside piping by remote control to measure the distribution of contamination inside piping.

3.3.3 Dismantling Techniques

Representative dismantling techniques are listed in Table 3.4. Cutting methods used in dismantling work are classified broadly into thermal cutting methods (plasma arc, laser, oxidation reaction heat methods, etc.), which are intended for metallic equipment and structures, and mechanical cutting methods (abrasive water jet method, nibbler, shearing machine, wire saw, disk cutter, etc.), intended for both metals and concrete. Blasting techniques (shaped charge pipe cutting, controlled blasting of concrete, etc.) as well as electric discharge machining techniques for cutting metals are also used in dismantling operations.

While thermal cutting provides relatively high cutting rates leading to high levels of work efficiency, processes involve the melting and evaporating of base metals, which produce large amounts of airborne particulates in the work environment. In cases where an object to be dismantled has a high radioactivity concentration or surface contamination density, measures to prevent the spread of

Table 3.4 Representative dismantling techniques

Object	Classification	Principle		Dismantling techniques
Metallic equipment or structures	Thermal cutting	Laser		CO, CO_2, YAG, and iodine lasers
		Electric	Plasma	Plasma arc
			Arc	Arc saw, TIG, MIG, consumable electrode water jet
		Combination of electric and oxidation reaction heat		Oxygen gas, powder/gas, oxygasoline torch, thermite reaction lance
		Oxidation reaction heat		Oxy-arc, Combination of gouging and gas (G&G method)
		Electric discharge		Electric discharge machining (EDM)
		Metal disintegration		Metal disintegration machining (MDM)
	Mechanical cutting	Grinding		Abrasive water jet, abrasive
		Physical	Reciprocating motion	Nibbler, shearing machine, hacksaw, reciprocating saw, guillotine saw, milling cutter
			Rotational motion	Band saw, chain saw, diamond wire saw, disk cutter, carbide tipped circular saw, roller cutter, pipe cutter, hinge cutter, clamshell lathe
		Blasting		Shaped charge
Concrete structures	Cutting	Mechanical		Diamond wire saw, core boring, disc cutter
		Cutting		Abrasive water jet, shaped charge, liquefied gas
	Crushing	Impact		Breaker, steel ball
		Shock		Controlled blasting
		Pressure		Crusher, jack, lock jack, burster
		Dilatational deformation		Static crushing agent, shape memory alloy
	Heat	Heat		Flame cutting, thermite, laser, electromagnetic induction, and direct electrification methods

contamination as well as protection measures for workers, such as ventilation suits, are necessary.

3.4 Dismantling of Nuclear Facilities

Methods used for dismantling a nuclear facility vary depending on the type and size of the facility. The following sections introduce actual cases of dismantling, with separate discussions for reactor facilities and nuclear fuel cycle facilities.

3.4.1 Dismantling of Reactor Facilities: Examples

A reactor facility, while in operation, continuously generates neutrons in its core through fission reactions, and there are typically large amounts of contamination from activation around the reactor at the time of decommissioning. Of all the radionuclides produced by activation, the radioactivity of short-half-life nuclides decays in a short period of time following the termination of reactor operation. Therefore, by ensuring an adequate cooling period length with safe storage, the exposure dose of workers engaged in dismantling work as well as the amount of radioactive wastes can be reduced.

Two representative examples of the dismantling of reactor facilities are described below.

3.4.1.1 Japan Power Demonstration Reactor [8]

The Japan Power Demonstration Reactor (JPDR) owned by the former Japan Atomic Energy Research Institute's Tokai Research Institute was Japan's first BWR for power generation (initial thermal output 45 MW; post-reconstruction output 90 MW; electric output 12.5 MWe); it reached its initial criticality on August 22, 1963, generated its first electrical power on October 26 of the same year, and ceased operation in March 1976.

The dismantling of the facility started in December 1986. After first removing the dump condenser equipment (equipment used to condensate steam that had doubled in volume after reconstruction) to secure temporary storage space for wastes, the dismantling of equipment inside the reactor containment vessel was commenced. In order to reduce the external exposure of workers and minimize the amount of radioactive gases and particulates released into the environment, remote handling devices and underwater cutting methods were effectively used to dismantle highly radioactive equipment with existing ventilation system.

The dismantling of equipment in the reactor containment vessel was undertaken in the following order. (1) Equipment located around the containment vessel was removed to secure space for installing remote devices such as master-slave type manipulators and mast type devices. (2) After remote handling devices were installed, reactor core internals were dismantled by underwater plasma arc cutting. The reactor pressure vessel was dismantled using the underwater arc saw cutting technique. (3) The dismantling of biological shielding concrete was carried out using techniques such as the core boring method, diamond saw method, water jet cutting device and controlled blasting method. (4) Remaining equipment was removed, and the floors, walls and ceiling planes were decontaminated using scabblers and other devices. (5) Overall measurements were conducted to confirm the completion of decontamination, and the designation of radiation controlled area was lifted. (6) The reactor containment vessel was demolished using conventional techniques.

The amount of radioactive wastes generated during the JPDR dismantling was 3,770 tons. Metals and other radioactive wastes having low radioactivity concentrations were stored in 200-liter drums or 1-m^3 or 3-m^3 steel containers. Highly activated metallic equipment was stored in designated shielding containers specially fabricated for the purpose. On-site shallow landfill disposal was selected for concrete fragments amounting to 2,000 tons, which had a very low radioactivity concentration, as a buried disposal verification test (see Fig. 6.3 in Chap. 6). The number of man-days expended to complete the JPDR dismantling work was 145,000. The collective exposure dose of the workers was 0.3 man-Sv, which was 1/3 of the planned value.

3.4.1.2   Shippingport Atomic Power Station [9]

The Shippingport Atomic Power Station in the U.S. was a PWR built for power generation with an electric output of 100 MW. Its operation began in 1957 and was terminated in 1982. The plant also went through reconstruction during the course of its operation. Decommissioning started in 1985 and was completed in 1989, during which the reactor pressure vessel was removed as an assembly without being cut or separated into segments. This approach was employed after it was decided that removing the reactor pressure vessel assembly as a whole, instead of cutting it in parts and storing them in waste containers, would provide more benefits such as reduced cost, shorter term of work and lower worker exposure. As shown in Fig. 3.8, the pressure vessel was placed in the outer neutron shield tank, and reactor core internals, filters and other elements were stored in the pressure vessel. This assembly was then processed into a waste form, and was shipped long-distance (Fig. 3.9) through land transportation by truck trailer and water transportation by a carrying vessel to the low-level radioactive waste burial site located in Hanford, Washington, where it was buried for disposal. The decommissioning cost was about 90 million dollars, and the population exposure dose of the workers was 1.55 man-Sv.

3.4.2   Dismantling of Nuclear Fuel Cycle Facility: Examples

Unlike in reactor facilities, neutron irradiation of equipment and structures does not need to be considered in nuclear fuel cycle facilities and there is no contamination from activation. Radionuclides used in nuclear fuel cycle facilities have a long life, and it is not necessary to allocate a specific time interval between the termination of facility operation and the start of dismantling work because radioactive decay cannot be expected sufficiently. A nuclear fuel cycle facility consists of many small caliber pipes, small equipment, tanks, vessels and other components, and its systems have complex structures. Large amounts of nitric acid-resistant stainless steel as well as Ti and Zr alloys are often used as component materials. The
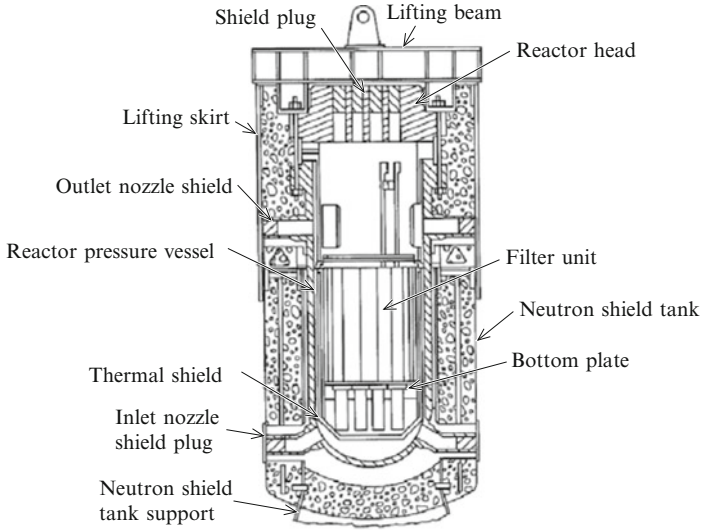
Fig. 3.8 Configuration of the waste form of the Shippingport reactor pressure vessel removed as an assembly [10]
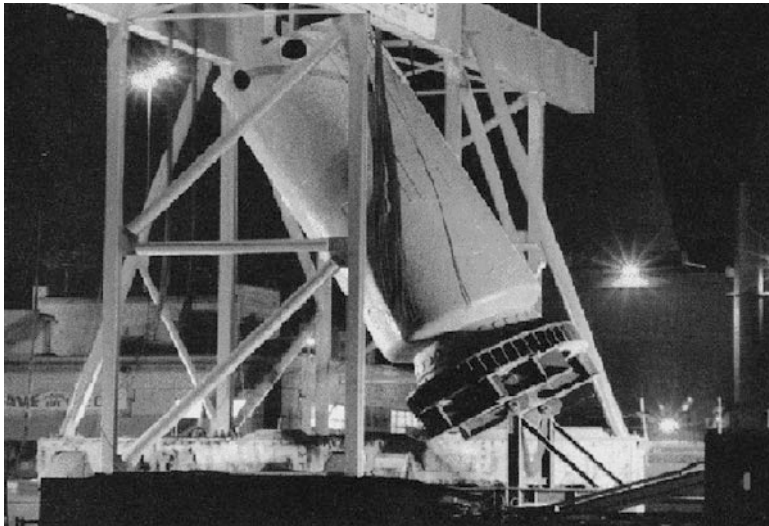
Fig. 3.9 Waste form of the Shippingport reactor pressure vessel being carried out of the facility [11]

dismantling work environment includes many narrow areas, where radionuclides of uranium and transuranium such as plutonium are used in various forms including gases, liquids and powders. The work must be conducted by selecting appropriate methods according to the types of radionuclides.

Three representative examples of the dismantling of nuclear fuel cycle facilities are described below.

3.4.2.1 Dismantling of Hanau Fuel Fabrication Plant [12]

Siemens AG, based in Germany, decommissioned its uranium and MOX fuel fabrication plants and hot cells that had been constructed in Hanau. The Hanau MOX fuel fabrication plant was in operation from 1965 to 1991, where 26,000 MOX fuel rods were fabricated from 8.5 tons of plutonium. After the operation of the plant was terminated, the treatment of residual nuclear fuel materials was carried out until early 2001, and the dismantling of the fabrication equipment followed. Two hundred forty glove boxes installed in the plant were dismantled and disposed of during the process. Radioactive wastes generated from dismantling the plant were solidified through cementation and have been stored in containers that met the burial conditions of the Konrad repository. The dismantling work was completed in September 2006, and the site was released.

3.4.2.2 Reprocessing Test Facility [13]

The Japan Atomic Energy Research Institute (predecessor of the current Japan Atomic Energy Agency) Reprocessing Test Facility (JRTF) was constructed in 1966 as Japan's first engineering-scale reprocessing research facility that studied fuel reprocessing based on the PUREX method. It conducted aqueous reprocessing tests between 1968 and 1969.

Upon dismantling installations and equipment, the surface contamination density and dose rates of the work area were measured first, and wipe decontamination, contamination fixation and radiation shielding were applied where needed. Based on the degrees of the contamination of the installations and equipment, contamination control enclosures with up to four compartments and local exhaust systems were installed to prevent radioactive particulates and other material from leaking outside the work area.

For dismantling glove boxes and large tanks, mechanical cutting tools such as band saws, pipe cutters and nibblers were mainly used as appropriate taking into consideration the material, size, structure and other characteristics of each component. Some of the large tanks were sealed without cutting and carried out of the facility by creating an opening from the building. The tanks were then transferred to a designated dismantling facility where they were shredded and stored in containers. Once the dismantling of installations and equipment is completed, contaminated concrete will be removed from the building and the designation of the controlled area will be lifted after it has been confirmed that there is no residual contamination in the building. The building will be demolished subsequently, and the site will be cleared and prepared for future use.

3.4.2.3   Reprocessing Facilities in Other Countries [14–16]

As with the dismantling of reactor facilities, the reconstruction and dismantling of reprocessing facilities are essential tasks for nuclear nations. The development of dismantling plans for reprocessing facilities or actual dismantling has already begun in the U.S. and European countries.

Reprocessing facilities currently in the dismantling process are West Valley in the U.S., the Eurochemic Reprocessing Plant in Belgium, B204 Primary Separation Plant in the U.K., AT-1 and B211 in France, and WAK in Germany. These facilities are being dismantled by applying existing techniques or techniques developed as the work proceeds, considering their individual characteristics.

For the reason that equipment is small in scale as well as from the perspective of preventing secondary contamination, conventional mechanical cutting techniques are often used in these facilities when equipment is dismantled. When dismantling buildings, one or more techniques including diamond saw, core boring, abrasive water jet and wire saw methods are chosen depending on the conditions of the buildings. For the decontamination of installations, many facilities have had system decontamination carried out following the termination of their operation, or contamination fixation undertaken to confine alpha contamination. Current technical development is primarily focused on remote dismantling techniques. At AT-1 and WAK, equipment inside cells was dismantled using hydraulic manipulators.

3.5   Site Release

Upon completing decommissioning, an application must be filed pursuant to the Reactor Regulation Act to request that the regulatory authority confirm the completion of decommissioning with regard to whether the results of the decommissioning work conform to the established standards. When the confirmation is obtained, the license for activity or installation the nuclear reactor ceases to be effective.

It is required that the application requesting the confirmation of the completion of decommissioning should contain the implementation status of the measures put forward in the decommissioning plan, and also include information on the distribution status of contamination caused by nuclear fuel materials. The criteria for confirmation are: the transfer of nuclear fuel materials has been completed; the site soil and residual facilities are in a state that they do not require measures for preventing radiation hazards; the disposal of contaminated objects has been completed; and radiation control records have been submitted to the national government.

There are several possible uses of a site (land and buildings) on which decommissioning has been completed. In the case of nuclear power plants in Japan, many of them have multiple units built on one site. Even when one of the

units is decommissioned, the site may possibly continue to be used as a nuclear plant while designating the decommissioned unit and its peripheral area as a monitoring area. In the case of small-scale experimental facilities and research reactors, it is largely viewed that after a facility is dismantled, the building and land or the land can be excluded from nuclear regulatory control and put into general use without restriction.

3.5.1   Criteria for Site Release

Site release includes two concepts: unconditional release and conditional release. Unconditional release allows a site to be used freely without restriction once site release is approved. Conditional release, on the other hand, requires a facility with a certain level of residual radioactive materials to be placed under institutional control to prevent the exposure dose of individuals reusing the facility from exceeding the reference dose. Institutional control may include the designation of restricted access areas and a restriction on the duration of time spent in specified areas.

3.5.1.1   Reference Dose

The safety guide [17] issued by IAEA explains reference doses based on the following logic.

Materials once released from regulations through the clearance system, which is one example of release from regulatory control, may possibly enter into trade with a broad range of potential uses internationally and the exposure doses from such cleared items would add to the dose constraint; therefore, the reference dose is set below the order of 10 μSv/y, which is regarded as negligible in terms of the risk level. On the other hand, land (soil) and buildings continue to remain in place after their release from regulatory control; therefore, safety can be ensured as long as the exposure dose does not exceed the dose constraint of 300 μSv/y. In other words, the 300 μSv/y individual dose to a member of the general public based on the dose constraint is the starting point of the argument in this guide. It then suggests that each member country should set its own reference dose within the range of several tens to 300 μSv/y. The basis for this argument is that the dose constraint set for the boundaries of a site while the facility is in operation can be applied to the site itself upon the release of the site from regulatory control, which takes place after the operation is terminated and the facility is dismantled.

In the U.S. where many cases of site release have already occurred, the reference dose has been set at 250 μSv/y to allow some margin from the 300 μSv/y dose constraint [18]. For conditional release, the country's standard requires that the individual dose should not exceed 1 mSv/y even in the event that institutional control fails to function effectively.

In Germany, site release is positioned as part of the country's clearance system, and the reference dose is set at 10 μSv/y. The concentration of each radionuclide has been calculated based on this reference dose as explained below, and the calculated concentration values are set out in the national ordinance governing the clearance system.

3.5.1.2   Reference Radioactivity Concentration

While reference doses such as above are stipulated, confirming extremely low doses below 300 μSv/y through direct measurement is difficult. Therefore, in practice, the surface densities of the contamination of buildings and radioactivity concentrations in the soil, which correspond to the relevant reference dose, are calculated for each possible form of contamination and exposure pathway. The calculated values are then compared with values obtained through actual measurement. This process is used to judge whether site release is possible, and is the same method used to determine the clearance of wastes generated during decommissioning (see Chap. 4.2 and 8).

In the U.S., any licensee who wishes to release its site from regulatory control is required to calculate exposure doses using parameters appropriate for the characteristics of the site. It must also calculate radioactivity concentrations in the soil or surface contamination densities that correspond to the reference dose (derived concentration guideline level: DCGL) for each nuclide, and obtain approval for the calculated values through a license termination plan (LTP). The RESRAD code, which enables easy calculation of concentrations, has been developed in the U.S. It has been made widely available through the Internet and is used in different countries around the world. Germany has also used RESRAD, after making some improvements to part of the code, to calculate radionuclide concentrations based on the 10 μSv/y reference dose. Germany has set uniform reference values for radioactivity concentration and surface contamination, which have been calculated without factoring in site-specific scenarios and parameters.

When carrying out exposure dose evaluation, scenarios for the post-decommissioning use of the site need to be developed to evaluate the following: (1) the external exposure of a resident or worker or an individual temporarily entering the site to the residual radioactive material contained in the surface soil of the site; (2) their internal exposure caused by inhaling particulates; (3) exposure due to agricultural practices; and (4) internal exposure caused by intake of agricultural or livestock products cultivated or raised directly on the decommissioned site. In the case of site release, shielding material such as cover soil, which is factored in the evaluation of radioactive waste disposal (e.g., Fig. 6.3 in Chap. 6), does not exist on the site. Consequently, external exposure by direct radiation from radioactive materials in the soil contributes significantly to exposure doses. The evaluation should be made on the assumption that the agricultural and livestock products are grown directly on the decommissioned site, and radioactive materials contained in the soil transfer to the products through plant roots.

3.5.2 Verification for Site Release

In order for a site to be released from nuclear regulations, verification must be made based on the history of use of the site, measured values and other relevant data to confirm that the radioactivity concentrations or surface contamination densities of the site are kept below the values calculated through evaluation. Several countries have established verification procedures for site release, which are to be carried out in combination with necessary remedial actions on the premise that the site is contaminated with radioactive material.

The U.S. has established its radioactivity measurement and evaluation methods concerning site release by introducing statistical methods, and published a verification manual MARSSIM [19] explaining the methods. Because sites will be released from the control of nuclear regulations and put into general use, this manual was compiled with mutual consensus of multiple U.S. government agencies including the Department of Energy, Environmental Protection Agency, Department of Defense, and NRC.

As explained in Fig. 3.10, the MARSSIM approach classifies different areas of the land and buildings into those that may be affected by radioactive materials
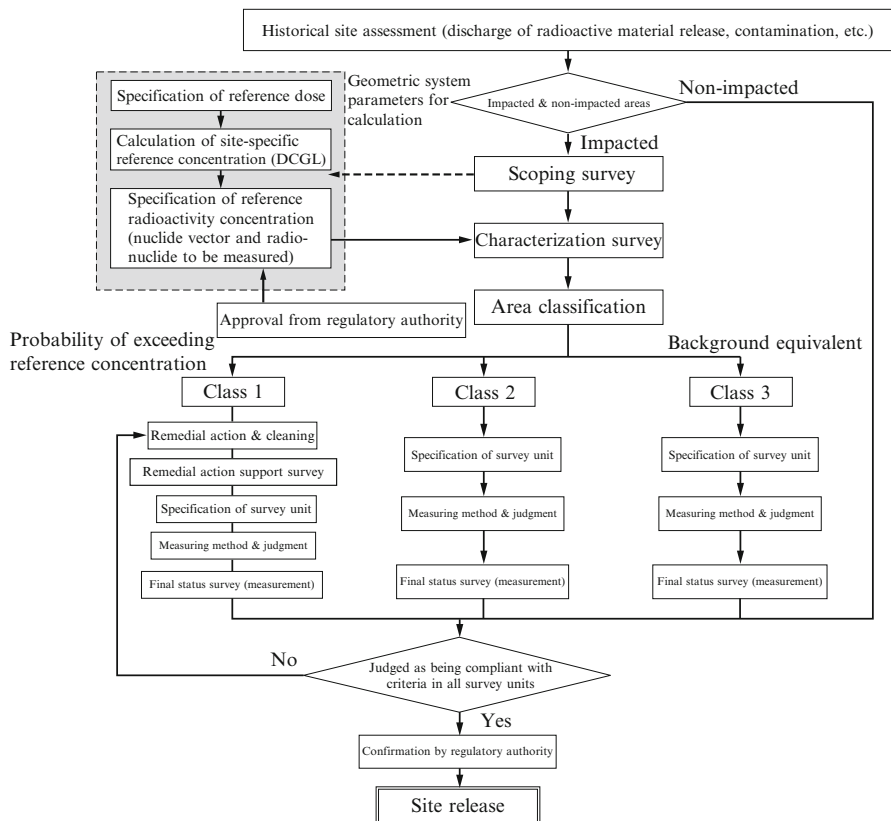

Fig. 3.10 Site release procedure in MARSSIM

(impacted areas) and those not affected by radioactive materials (non-impacted areas) based on the operating history of the site. Non-impacted areas are considered to have already met the criteria, and no measurements will be taken. For impacted areas, on the other hand, scoping and site characterization surveys are conducted followed by area classification. The percentage of the area to be covered by scans is then specified according to the area classification, and a scanning survey is conducted using a scanning instrument that has the minimum detection limit of no more than 50 or 10 % of DCGL to confirm that there are no unacceptable hot spots. Next, using a statistical method, an appropriate number of samples are taken from the site soil as well as the concrete from the building surfaces to measure their radioactivity. Upon evaluating the measurement results, the Sign test is applied when the radionuclide to be evaluated is not present in background, while the Wilcoxon Rank Sum test is applied when it is present, to determine the compliance with the criterion. The Sign test is used to test the hypothesis that the difference median is zero between the continuous distributions of two random variables. The Wilcoxon Rank Sum test is used to determine whether the distribution of the observed values of two groups is the same. For example, when ^{137}Cs is the nuclide to be evaluated, the Wilcoxon Rank Sum test is applied because ^{137}Cs is in the soil due to global fallout. In MARSSIM, Type I error or false positive, in which a null hypothesis is rejected when it is true, and Type II error or false negative, in which a null hypothesis is accepted when it is false, are each specified at 5 % in most cases.

Exercises

1. Calculate the radioactivity concentration [Bq/g] of ^{60}Co remaining in the pressure vessel material at 10 years following the termination of the operation of a reactor that had been operated for 30 effective full power years (EFPYs) at a thermal neutron flux of 1×10^{10} n(cm^2·s). The pressure vessel is made of carbon steel (density 7.9 g/cm^3), and the initial elemental ratio of ^{59}Co therein is 100 ppm. Refer to Fig. 3.4 for the thermal neutron activation cross section of ^{59}Co and the value of the half-life of ^{60}Co. Because the decrease in the number of ^{59}Co atoms due to activation is significantly smaller than the initial level, it can be assumed that the number of ^{59}Co atoms is constant.

2. Calculate the exposure dose of the general public in the vicinity via the inhalation & ingestion and surface deposition pathways when the in-air cutting of 150-m, JIS 300A × sch40 piping with a ^{60}Co surface contamination density of 1×10^4 Bq/cm^2 is performed inside a contamination control enclosure using the plasma arc technique. The calculations are to be made on the following assumptions. In order to improve the storage efficiency, the pipe is halved in the vertical direction when storing it in a 200-liter drum (diameter: 0.566 m, height: 0.8 m). Taking account of the 0.8-m drum height, the pipe is cut at intervals of 0.75 m (including the kerf width). The kerf width is 0.005 m, the contamination radioactivity dispersion rate is 30 %, no leakage from the building or contamination control enclosure is present, and the duration of exposure resulting from the dismantling work and the released radioactivity is 1 year. The collection efficiencies of HEPA or other filters of the local and building exhaust systems

are 99.0 %, respectively. The relative concentration at the site boundary is 2.2×10^{-4} (Bq/m^3)/(Bq/h), the adult breathing rate is 22.2 m^3/day, and the internal exposure dose conversion factor for ^{60}Co is 3.1×10^{-2} μSv/Bq. In addition, the surface deposition velocity is 0.01 m/s, the external exposure dose conversion factor from gamma rays associated with surface-deposited ^{60}Co is 2.2×10^{-2} (μSv/y)/(Bq/m^2), and the general public in the vicinity will be exposed, for the duration of 1 year, to surface-deposited radioactivity that has been accumulated for 1 year. It is recommended that the former Nuclear Safety Commission of Japan's Special Committee report "Evaluation of Exposure Dose of the General Public during Safety Evaluation of Light Water Nuclear Power Reactor Facilities" be used as reference when making the above calculations.

3. Calculate the minimum detectable concentration for scanning (ScanMDC) when carrying out a scanning survey of ^{60}Co on the floor surface of a turbine building. A ScanMDC can be calculated using the following formula:

$$\text{ScanMDC} = \frac{\text{MDCR}}{\sqrt{p}\,\varepsilon_i\,\varepsilon_s \dfrac{\text{Area of detector window}}{100 \text{ cm}^2}}$$

The ScanMDC is to be determined for a background level of 1,350 cpm and a 2-second interval. The performance requirements are a true positive fraction of 95 % and Type I error of 25 %. A surveyor efficiency of $p = 0.5$, instrument efficiency of $\varepsilon_i = 0.24$, and surface efficiency of $\varepsilon_s = 0.25$ are assumed. Section 6.7.2 of MARSSIM may be referred to when making calculations.

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