SAFETY ASSESSMENT FOR DECOMMISSIONING

Annex I, Part A

Safety Assessment for Decommissioning of a Nuclear Power Plant

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CONTENTS

FOREWORD

The purpose of Annex I is to provide a demonstration of the application of the DeSa safety assessment methodology described in the main report. For that purpose three examples of facilities to be decommissioned were selected by the DeSa project participants for evaluation. The chosen test cases are broadly representative of ongoing or completed decommissioning projects.

The test cases selected for evaluation were:

- A nuclear power plant (NPP);
- A research reactor; and
- A nuclear laboratory.

The facilities were selected because they represented a range of differing types of facility and because the operating organizations had committed to provide all necessary technical information to allow safety assessments to be conducted.

Once the safety assessments for the decommissioning of NPP, research reactor and the nuclear laboratory had been developed, each test case report was reviewed by the Regulatory Review Working Group and the Graded Approach Working Group to provide a simulation of a regulatory review and to demonstrate that the regulatory review procedure developed for DeSa (see Annex III) and the recommendations on the graded approach (see Annex II) are robust.

Part A of Annex I deals with the NPP test case. The subject of this test case is a Boiling Water Reactor (BWR) that was used for electricity generation, but which had reached the end of its operating life. The reactor had been defueled and fuel removed from site where two of these units were located. A plan for the transition and preparatory phase for decommissioning was available, but the planned decommissioning work was at an early stage and had not been subject to detailed safety assessment. The purpose of the safety assessment for this unit was to support the decommissioning plan for immediate dismantling.

The safety assessment for a large NPP is extensive and comprises a large suite of documentation. Due to the time constraints on the DeSa project, it was not considered practicable to address the decommissioning of a whole NPP, nor was it considered necessary as the safety assessment approach and methods would be similar for many of the phases of the overall project. It was decided therefore that by selecting two radiologically significant decommissioning tasks (dismantling of two systems) a satisfactory demonstration of the DeSa safety assessment methodology would be achieved that would be broadly representative of most decommissioning projects for light water reactors. The NPP Test Case therefore:

- Deals broadly with the whole decommissioning project and the supporting safety assessment; and
- Specifically addresses two significant decommissioning tasks for the purpose of demonstrating the DeSa methodology.

1. INTRODUCTION

1.1. BACKGROUND

Evaluation and demonstration of safety is an essential component of the successful planning, performance and completion of decommissioning activities. This has been highlighted by the international safety standards on decommissioning of nuclear power plants (NPPs), research reactors medical and research facilities [1], [2] etc. Recognizing the need for exchange of information and experience and consolidation of the best experience and lessons learned in these areas, the International Atomic Energy Agency (IAEA) launched the international project on Evaluation and Demonstration of Safety of Decommissioning of Facilities Using Radioactive Material (DeSa) in 2004 [3]. This project aimed the development of a harmonized methodology for evaluation and demonstration of safety during decommissioning and the development of safety assessments for selected facilities by applying this methodology. The methodology developed, as part of the project and documented in the main report was tested in several test cases – an NPP, a research reactor and a nuclear laboratory. All three test cases are presented in this Annex I.

The three test cases provide practical illustration of the application of the methodology and also illustrate the need of and the application of a graded approach to development of safety assessment, due to the complexity and hazards of the facilities. These may also be of assistance to operators and regulators in the Member States.

The test cases present safety assessments for facilities with different complexities and hazards, following the individual steps of the methodology (see the main report). By developing these cases, practical issues related to the use of the methodology were identified, such as the criteria for selection and justification of scenarios and models, definition of types of uncertainties and approaches for their treatment. Decisions on the importance of input data required, the use of generic vs. site specific data, as well as the depth for safety assessment necessary for demonstration of safety for decommissioning of various facilities with different hazard potential are also be addressed in the project. The formulation of the test cases very much relied on the information provided by volunteer facilities and the knowledge and experience of the participants in the DeSa project.

It should be noted that, although the test cases are based on real information about the facilities, they do not represent a specific safety assessment carried out for these facilities.

The safety assessment for a nuclear power plant (NPP) was developed in Phase 2 of the project and reviewed by the Regulatory Review and the Graded Approaches Working Groups [3]. The recommendations made by these project working groups are also reflected in this Part |A of Annex I. Part B presents the safety assessment of for the decommissioning of the research reactor and Part C the safety assessment for the nuclear laboratory.

1.2. SCOPE

Part A of Annex I documents the safety assessment for the decommissioning of one of two light water nuclear reactors at a NPP site. It concentrates on the activity associated with the removal of a contaminated part of the reactor primary circuit after spent fuel removal, its size reduction until the waste is packaged and handled from the reactor building to on-site processing and storage or further off-site disposal. Waste processing, disposal and clearance of material are not within the scope of the assessment and this report. Non-radiological hazards are not considered, however the report highlights the importance of consideration of these hazards in the development of a decommissioning plan and authorization of decommissioning activities.

The information on the facility, the site and the surroundings presented in the NPP Test Case exceeds the needs of the particular safety assessment for the two selected systems 321 and 322. However, in the safety assessment for the whole reactor, the level of detail and information required would be also based on national legislation and on the specific requirements of the Regulatory Body in a Member State and on other stakeholders.

The host country has a mature regulatory and legal environment, with substantial powers delegated to individual regulators and to regulatory bodies. The host country has a developed national system for disposal of radioactive waste. The waste acceptance criteria for disposal are established. Waste treatment processes have been established and the types of containers for waste disposal have been decided upon. This means that the aims and objectives for waste management at the decommissioning NPP are known. These in turn determine the nature of the tasks to be carried out for decommissioning.

For the purpose of the NPP Test Case, the safety assessment was based to large extent on information from a volunteered facility and also on series of assumptions. The assessment was developed in support of a high level decommissioning plan for the NPP, shown in Fig. 1.

Planning scenario for decommissioning - DeSa project "Test Case A"

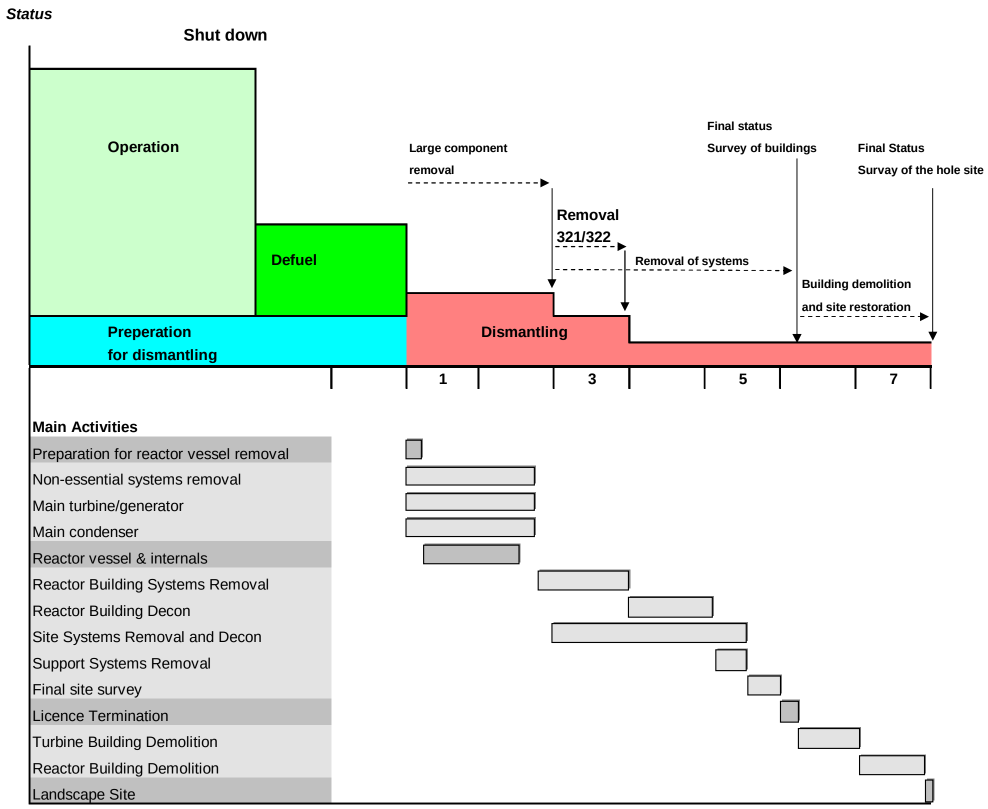

FIG. 1. Planned NPP decommissioning activities.

However, the range of decommissioning tasks and supporting activities necessary is very large, with the decommissioning stage of the NPP life cycle lasting 7 years. The complexity of the NPP is indicated by the systems line diagram shown in Fig. 2.

The following more specific decommissioning activities at unit 1 are envisaged to be carried out during the decommissioning period (see Fig. 1.):

(a) Design and fabrication of temporary and permanent shielding to support removal and transportation activities, construction of contamination control envelopes, and the procurement of special tools;

(b) Procurement of shipping canisters, cask liners, and industrial packages;

(c) Removal of piping and components no longer essential to support decommissioning;

(d) Transfer of the steam separator and dryer assemblies to the dry separator pool for segmentation;

(e) Removal, segmentation and packaging of the feed water sparge system;

(f) Disassembly, segmentation, and packaging of the core shroud and in-core guide tubes;

(g) Removal, segmentation, and packaging of the remaining internals, including the fuel support castings and core plate assembly;

(h) Draining and decontamination of the reactor well. Installation of shielded platform for segmentation of reactor vessel;

(i) Disconnection of the control rod drives and instrumentation tubes from reactor vessel lower head. The lower reactor head and vessel supporting structure are then segmented;

(j) Removal of the reactor recirculation pumps. Exterior surfaces are decontaminated and openings covered;

(k) Demolition of the biological shield activated concrete by controlled demolition;

(l) Removal of remaining components, equipment, and plant services in support of the area release survey(s) – including systems 321 and 322;

(m) Conduct of final radiation survey of land and remaining structures and buildings are in compliance with relevant criteria (see Section 2);

(n) Building demolition and site restoration; and

(o) Preparation of the final decommissioning report.

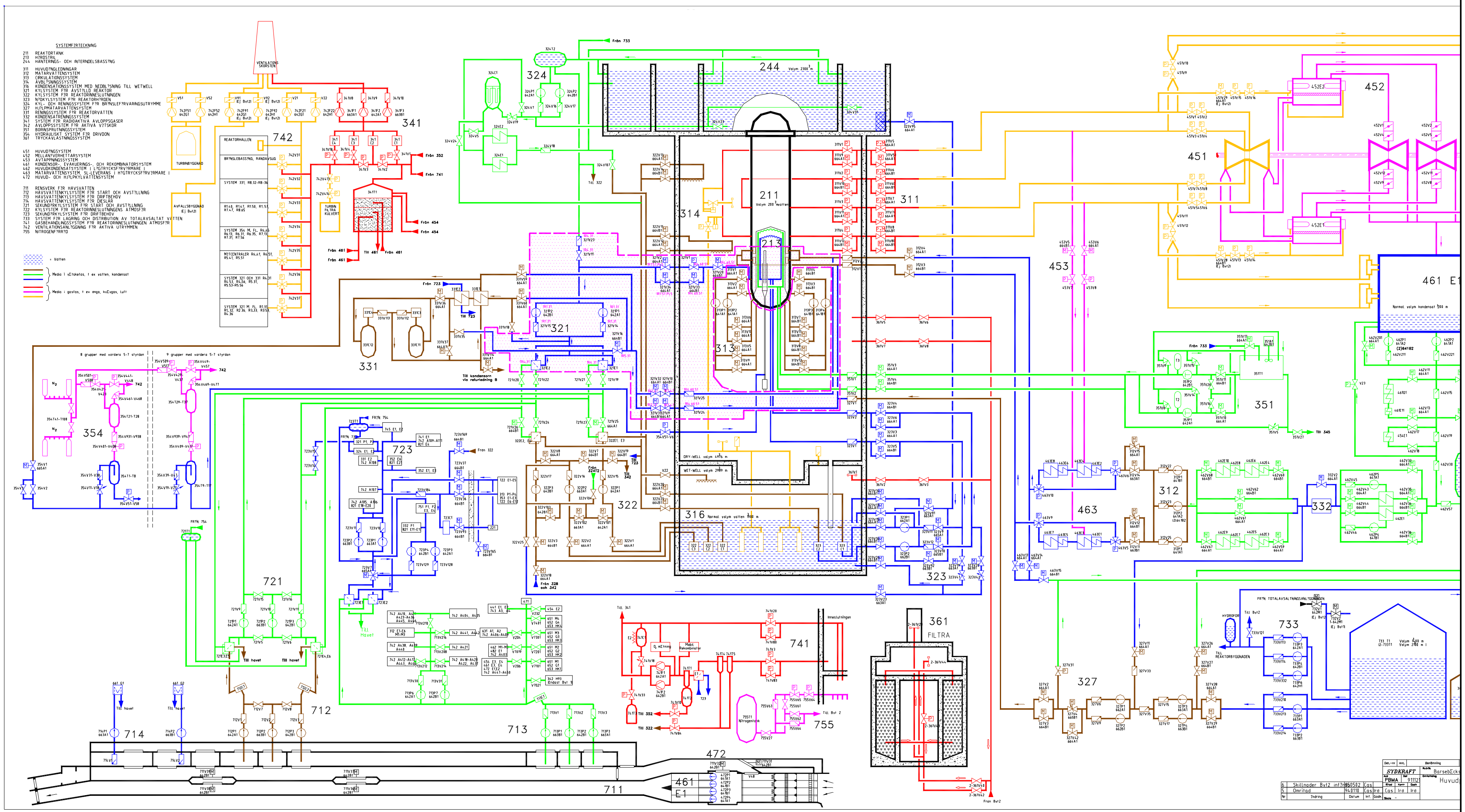

FIG. 2. Unit 1 systems diagram.

Given such a large decommissioning project and the constraints of the three year DeSa project, it was impracticable in this Test Case report to present a comprehensive safety assessment for all decommissioning activities at the unit 1. Safety assessments for large projects are always divided to support individual phases of the overall project. For the purpose of demonstrating the safety assessment for a NPP, two specific decommissioning tasks have been selected for presentation of the safety assessments methodology. These are decommissioning of:

- The cooling system for the shut down reactor system (System No. 321); and
- The containment spray cooling system (System No. 322).

These two systems were chosen for the NPP Test Case because they are representative of the type of decommissioning work to be undertaken; they are amongst the more significant tasks in terms of radiological exposure to workers resulting from planned work and the radiological and conventional industrial risks under accident conditions. In addition, the differences in the hazards presented by the two systems allow demonstration of the application of the graded approach.

1.3. OBJECTIVES

The aim of this report is to illustrate the application of the safety assessment methodology (see the main report) to an NPP, representing a large and complex nuclear facility.

This report also aims to illustrate the evaluation of the radiological hazards expected to result from the conduct of decommissioning in the time and with the resources available following a decommissioning phased approach. Given the limited resources available in the DeSa project, it was not possible to analyse the decommissioning of the entire unit. Therefore, for the purpose of this report, it was decided to limit the scope to a limited number of decommissioning tasks. The scenarios and work packages have been chosen to illustrate a number of assessment areas, including:

- Assessment of hazards to workers arising from decommissioning operations; and
- Assessment of radiological releases and effective doses to the public due to anticipated or accidental releases to the environment via either gaseous or liquid pathways (involving doses attributable to inhalation and ingestion of released material via agricultural pathways, etc.).

The safety assessment for this NPP has provided feedback to the Regulatory Review and Graded Approach Working Groups of the DeSa project.

1.4. STRUCTURE

Part A of Annex I is structured as follows:

- Section 1 provides a background, scope and objectives of the report;
- Section 2 provides the safety assessment framework;
- Section 3 describes the NPP, and sets the context of the scope of this test case report in the overall larger decommissioning work for the whole NPP;
- Section 4 presents the identification and screening of hazards;
- Section 5 the evaluation of the normal and accidental scenarios during decommissioning, and the associated modelling;
- Section 6 describes the engineering analysis;

- Section 7 presents the analysis of results and identification of safety measures;
- Section 8 discuses the approach to and decisions made for application of the graded approach in the development of the safety assessment of the NPP;
- Section 9 contains the confidence building measures applied to the safety assessment development;
- Section 10 provides a summary of the lessons learnt and conclusions from the preparation of this test case report;
- Appendix I contains the detailed analysis that supports the assessment of consequences for workers from normal decommissioning activities, and details of the proprietary computer code that was used to do this;
- Appendix II contains the detailed analysis that supports the assessment of consequences for the public (critical group) from normal decommissioning activities, and details of the proprietary computer code that was used to do this; and
- Appendix III contains the detailed analysis that supports the assessment of consequences for workers and the public from accident conditions that could arise during decommissioning activities.

2. SAFETY ASSESSMENT FRAMEWORK

This Section provides an outline of the context in which the safety assessment for the decommissioning of two systems of the NPP is considered and developed. It also presents the objectives, endpoints, timeframes, approach and boundaries for conduct of this assessment.

2.1. CONTEXT OF SAFETY ASSESSMENT

The safety assessment is performed in preparation for final decommissioning of one of two units at a NPP site for which the immediate dismantling strategy has been selected. The whole NPP was shut down in November 1999; defueled two years later and in a state of surveillance since then (see Fig. 1 in Section 1). A detailed decommissioning plan is under preparation as part of the supporting information for the application for authorization of decommissioning of the facility and release of site from regulatory control with remaining buildings on the site.

The safety assessment is developed in support to the final decommissioning plan for the unit that is necessary for obtaining of authorization for decommissioning.

2.2. SCOPE OF THE ASSESSMENT

The assessment covers two specific systems of unit 1 (one of two reactor units) with common support systems located at an NPP site. The assessment concentrates on the activity associated with the removal (i.e. cutting of systems, size reduction and segregation) of contaminated parts of the reactor primary circuit (Systems 321 and 322, see Section 4), their emplacement in containers for transportation for processing, storage on the NPP site. Evaluation of safety during waste treatment, conditioning, storage and disposal is not within the scope, but waste handling within the NPP is included up to the boundary of this scope where waste arrives at the on-site waste treatment plant. Waste handling in the context of this test case is limited to handling waste from the decommissioning of the two systems as they are removed from their site of installation, placing this waste into temporary containment and temporary transport containers and moving these containers through the reactor buildings to the point of the boundary with the waste handling plant.

It should be noted that the safety assessment does not cover the erection and operation of a supporting facility that needs to be built to carry out decommissioning of the reactor.

Due to the time constraints of the DeSa project the analysis is reduced to two systems within the unit 1 (systems 321 and 322). The two systems have different levels of contamination which will show the application of the graded approach.

2.3. OBJECTIVES

The objectives of the NPP safety assessment are:

(a) To demonstrate safety of workers and public during the planned decommissioning activities and to show compliance with regulatory requirements and criteria;
(b) To confirm the existing or suggest new safety related systems and controls;
(c) To give confidence that the use of selected NPP systems demonstrates that the methodology is applicable to the whole NPP; and
(d) To be independently reviewed within the DeSa project framework (i.e. Regulatory Review Working Group and the Graded Approach Working Group).

2.4. TIMEFRAMES

The timeframes specified at site level and relevant for the safety assessment are:

— The whole NPP decommissioning intent is to take the NPP out of regulatory control 7 years from the start of decommissioning; and
— Unrestricted use of the buildings is achieved after 5 years from the start of decommissioning.

The timeframes specified at the work package level are 9 months for the dismantling of each system 321 and 322 that result in 18 months work in total.

2.5. END POINTS AND END STATE OF THE DECOMMISSIONING

The end point for the NPP Test Case is the removal of the two systems from the unit 1, the removal of any residual contamination in the rooms and the removal of any associated decommissioning equipment.

The end state for the whole unit 1 decommissioning is that all reactor nuclear and auxiliary systems are removed and the building structures are below the release values and can be released from regulatory control (see Section 2.6). The intention is that the site will be removed from regulatory control after completion of decommissioning. The owner is then free to develop the site for any purpose that may consider suitable.

The radioactive waste arising will be in a buffer store that will remain at a part of the NPP site under nuclear regulatory control until the waste is removed for disposal and this part of the site is also designated as suitable for unrestricted use.

2.6. REGULATORY REQUIREMENTS AND CRITERIA

The safety assessment is based on the following regulatory requirements and criteria that will apply to the planned decommissioning activities at unit 1 and for which compliance will need to be demonstrated:

2.6.1. Workforce criteria from normal activities

The radiation exposure predicted during the planned decommissioning needs to comply with the criteria specified in International Basic Safety Standards [4] and IAEA Safety Requirements for Decommissioning [2], i.e. the limit for potential effective doses to workers will not exceed:

(a) An effective dose equivalent of 20 mSv per year averaged over five consecutive years;
(b) An effective dose equivalent of 50 mSv per year in any one year;
(c) An effective dose equivalent to the lens of the eye of 150 mSv in a year; and
(d) An effective dose equivalent to the extremities (hands and feet) or the skin of 500 mSv in a year.

2.6.2. Public criteria from normal activities

For the relevant members of critical groups (the public):

(a) The estimated average effective dose equivalent (from all sources) shall not exceed 1 mSv in a year;
(b) The dose constraint for the NPP site is 0.3 mSv/y [5], which is divided equally between the waste handling facility and the two units under decommissioning. The dose constraint for the decommissioning of the NPP unit 1 is thus 0.15 mSv/y;
(c) An equivalent dose equivalent o the lens of the eye of 15 mSv in a year; and
(d) An equivalent dose equivalent to the skin of 50 mSv in a year.

Notwithstanding the dose limits and constraints, work needs to be planned for optimization of safety so that predicted exposure can be demonstrated as low as reasonably achievable (ALARA).

2.6.3. Accident criteria

Consequence-based, defence-in-depth criteria were used in this Test Case to determine the acceptability of the safety controls (as presented in Table 1.). As the defence-in-depth criteria were met in the Test Case, no specific risk criteria were used directly, although Appendix III (the radiological accident analysis) shows that the risk is adequately low.

These criteria are appropriate for short-term decommissioning tasks, and can even be applied to one-off operations, for which risk-based criteria may be less suited. Using these more deterministic criteria usually avoids the need for extensive frequency calculations, resulting in a less complex radiological accident analysis that is easier to understand. The relatively modest consequences of the radiological accident analysis support this approach.

TABLE 1. DEFENCE-IN-DEPTH CRITERIA

Unmitigated Consequences[1]	**Number of Independent Complete Safety Measures Required** [2, 3]
Higher > 20 mSv to a worker [4] > 1 mSv to the public critical group [4]	Two
Significant 2 to 20 mSv to a worker 0.01 to 1 mSv to the public critical group	One
Insignificant < 2 mSv to a worker[4] < 0.01 mSv to the public critical group[5, 6] [6]	None

Notes to Table 1:

1 ‘Unmitigated’ assumes failure of safety controls.

2 An independent complete safety measure must be:

- Independent of the initiating event and of any other complete safety measures;
- Capable of detecting a failure (the initiating event), deciding what must be done, and terminating or suitably mitigating the accident scenario. This may for example require an equipment safety control (such as an alarm) to detect a failure, an operator deciding what needs to be done in response to the alarm, and another equipment safety control used by the operator to terminate or mitigate the scenario. ‘Suitably mitigating’ means that the consequences are reduced to below the low end of the relevant consequence range, meaning that more than one mitigating safety control may be required just for the mitigation part of a single independent complete safety measure.

3 This number can vary depending on the initiating event frequency, but this complexity was not needed in the NPP Test Case.

4 One tenth of the ‘higher’ consequences value of 20 mSv.

5 Based on IAEA RS-G-1.7, [6] which treats this as a trivial annual dose.

6 All doses are effective dose equivalent.

The requirement for an appropriate number of independent complete safety measures is fundamentally based on the ‘single failure and redundancy/independence’ (para. 3.73 to 3.80 in Ref. [7]).

2.6.4. Clearance values

Three sets of clearance values are used in the NPP Test Case:

a. Activity concentration

The values for activity concentration for clearance of bulk material from regulatory control are the ones presented in Ref. [6], and in Table 2 below.

TABLE 2. CLEARANCE VALUES FOR ACTIVITY CONCENTRATIONS

Radionuclide	**Activity Concentration Value (Bq/g)**
Mn-54	< 1
Fe-55	< 1000
Co-60	< 1
Ni-59	< 100
Ni-63	< 100
Tc-99	< 1000
Pu-239	< 1
Pu-240	< 1
Pu-241	< 100
Am-241	< 1
Cm-244	< 1

As well as compliance with limits for individual isotopes given in the Table 2 above, the composite activity must also meet the criterion that, summed across all isotopes *i* then,

$$\sum_i \frac{\overline{C}_{i,a}}{C_i} \leq 1 . \qquad (1)$$

b. Surface contamination

For the site end state, the intention is to remove the remaining building structures from regulatory control; the criterion for surface contamination will be that there is no removable surface contamination. Ingrained radioactive material that cannot be removed by treatment of the surface will be addressed through the criterion (a) above.

The surface areas of the room are to be decontaminated to levels [8] that allow their release from regulatory control. Clearance is treated on the basis of the concept of 10 μSv/y individual dose. The set of isotopes is based on analysis of the primary circuit and the confirmation that these are the isotopes of relevance to the protection of the workers, the public and the environment during decommissioning.

It is has been considered also appropriate to use a set the criteria shown in Table 3, which are based on the occupational dose limits for the workers subject to dose control to the occupational limit of 20 mSv/a and the associated health surveillance.

TABLE 3. CLEARANCE VALUES FOR SURFACE ACTIVITY

Radionuclide	**Activity Concentration Value (Bq/cm^2)**
All β,γ emitters	< 4
All α emitters	< 0.4

c. Site release

The applicable dose constraint for the public after the release of a site needs to be expected to be no higher than that applied for the operational phase of the practice. According to Ref. [5] this dose constraint should take into account multiple pathways of exposure and should not exceed 300 µSv in a year above background.

The dose limit of 1 mSv in a year for members of the public represents an upper bound on the sum of effective doses from all possible combinations of exposures arising from practices.

These criteria are relevant for the decommissioning of the unit 1 and release the land after decommissioning completion. However, these criteria are not directly related to the decommissioning of the two systems 321 and 322 but their removal makes a positive contribution to the final end state.

Monitoring and surveillance is required during and after decommissioning until the site is released from regulatory control [2, 5].

For the scope of this test case, metallic scrap material will be assessed by a combination of monitoring of surface contamination and dose rates by conventional radiation protection instruments and by more detailed analysis using more sensitive spectrometric instruments such as high purity gamma spectrometry. It is then the intent to release as much as possible of the metallic scrap into the commercial scrap market so as to realise its economic value.

For the civil structures, the intent is to monitor it *in-situ* so that it can be passed out of regulatory control. The likely route for the civil structures is demolition, with the material arising being disposed of by normal landfill.

2.6.5. Waste management

The waste classification used in the NPP Test Case is based on the revised IAEA waste classification as presented in Ref. [9]:

(a) *Exempt waste (EM)* - Exempt waste contains such small concentrations of radioactive material that it does not require radiation protection provisions, irrespective of whether it is disposed of in conventional landfills or recycled.

(b) *Very short-lived waste (VSLW)* - Very short lived waste contains only radionuclides of very short half-life with concentrations above the clearance levels. Such waste can be stored until the activity has fallen beneath the levels for clearance, allowing for their clearance waste and management as conventional waste. However, in general the management option of storage for decay will only be applied for radionuclides with a half-life in the order of 100 days or less.

(c) *Very low level waste (VLLW)* - it is expected that for these waste with a moderate level of engineering and controls, a landfill facility can safely accommodate waste containing artificial radionuclides with activity concentrations of one or two orders of magnitude above the levels for exempt waste. In the case of naturally occurring radionuclides the acceptable activity concentrations will be in general more limiting in view of the long half-life radionuclides involved. An adequate level of safety for such waste may be achieved by their disposal in engineered landfill type facilities.

(d) *Low level waste (LLW)* - Low level waste in the classification scheme set out in this publication is waste that is suitable for near surface disposal. This is a disposal option suitable for waste that contains such an amount of radioactive material that it requires containment and isolation for

limited periods of time up to a few hundred years (i.e. up to around 300 years). A limit of long lived alpha emitting radionuclides of 4000 Bq/g in individual waste packages and to an overall disposal facility average of 400 Bq/g has been adopted by an increasing number of countries for near surface disposal facilities.

(e) *Intermediate level waste (ILW)* - Intermediate level waste in this classification scheme contains long lived radionuclides in quantities that need a higher degree of containment and isolation from the biosphere than provided by near surface disposal. Disposal in a facility at a depth between a few tens and a few hundreds of meters is indicated.

(f) *High level waste (HLW)* - The high level waste class contains large concentrations of both short and long lived radionuclides, so that, as compared to ILW, a higher degree of containment and isolation from the biosphere, usually provided by the integrity and stability of deep geological disposal, with engineered barriers, is needed to ensure disposal safety. HLW generates significant quantities of heat from radioactive decay, and normally continues to generate heat for several centuries. The NPP site has no stocks of HLW now that spent fuel has been removed.

The waste material generated during the decommissioning of systems 321 and 322 is planned to be size-reduced within the working area tent such that it can fit into the transport box specific to the facility. This box is transported to the site waste management facility where further processing takes place to consign the waste as ILW, LLW or release from regulatory control in accordance with regulatory requirements. The transport box is required to have a maximum dose rate at 10 cm from the surface [10]. The maximum will be 2 mSv/h for LLW and 60 mSv/h for ILW generated during decommissioning. These values are actually those derived for transport in the public domain, but for the purpose of the NPP Test Case it has decided to apply these to internal transport and the movement of waste arising.

2.6.6. Chemical and other industrial safety considerations

The applicable national occupational health and safety regulations will also apply to the control of effects to workers from non-radiological hazards. However, these aspects are not addressed in the DeSa project and in the NPP Test Case, but they are discussed for illustrative purposes. The hazard analysis considers the affect of decommissioning processes and unique chemical applications used in decommissioning as initiators to release events, or as they could affect or hinder the ability of workers to respond to an event (e.g. beryllium).

Under the transition from operation to decommissioning of a reactor and later to the site of unrestricted use the dominant risk-focus is rapidly changing from reactor safety and radiological risks to more conventional industrial safety risks (see Fig. 3). This leads to application of a graded approach in the evaluation of safety during decommissioning phases, as it is required by Ref. [2].

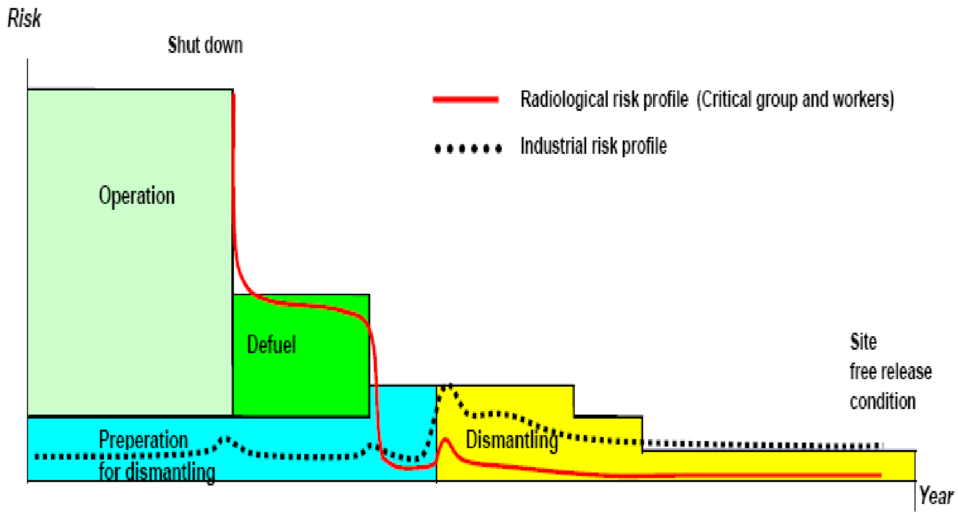

FIG. 3. An example of an overall radiological and industrial risk profile during the defueling and dismantling periods.

2.7. ASSESSMENT OUTPUTS

The assessment outputs for the NPP Test Case presented in this report comprise of the following:

— An estimate of the effective dose to workers and member of the public from both normal (planned) and accident conditions during decommissioning of the selected systems 321 and 322;
— The operational limits, controls and conditions for the specific decommissioning tasks considered for systems 321 and 322; and
— Recommendations for improvements to engineering (Safety Related Systems, Structures and Components - SSCs) and administrative measures (procedures) for decommissioning of systems 321 and 322, so that expected doses to workers and the public are ALARA.

2.8. SAFETY ASSESSMENT APPROACH

The safety assessment framework depicted in Fig. 4. (as discussed in detail in the main report) was followed in a deterministic manner [11].

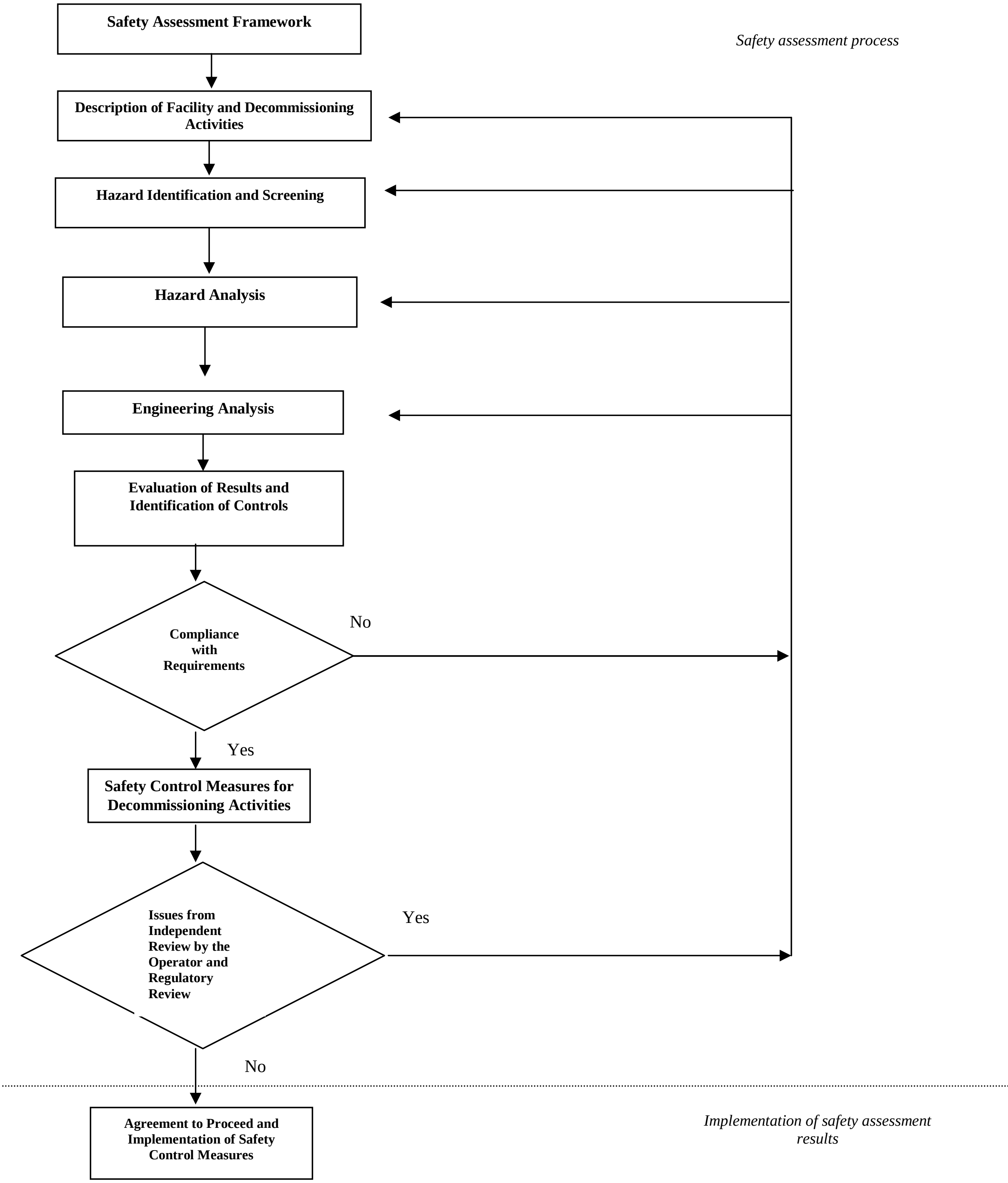

FIG. 4. The safety assessment methodology applied to the NPP Test Case.

Hazards were identified using the HAZOP[1] methodology. The HAZOP analysis technique uses a systematic process to:

- Identify possible deviations from planned activities; and

- Ensure that appropriate controls are in place to prevent or mitigate potential accidents. The HAZOP technique uses systematically considers all credible deviations from normal conditions.

The characteristics of HAZOP include (see also stands for Hazard and Operability Analysis Technique Volume I):

- A systematic, structured assessment conducted by a multidisciplinary team using a brainstorming to generate a comprehensive list of hazards, conditions and potential control mechanism(s) to prevent or mitigate events;
- It is used mostly as a system or process level risk assessment technique;
- Generates primarily qualitative results, although some basic quantification is possible.
- It identifies the hazardous material, initiating events and processing parameters associated with the planned decommissioning activities to identify potential events that could result in the release of material or expose receptors to unnecessary potential for harm. Standard hazard analysis techniques were applied which use a series of screening tools and techniques to focus on a set of events that are representative and bounding, so that there is assurance that the resulting controls offer an appropriate level of protection. The evaluation is iterative and limits and conditions are reviewed and consolidated to provide the minimum set of manageable controls to address the relevant hazards during decommissioning.
- The approach to hazard analysis (further described in Section 4) adopted accident screening criteria which were used to eliminate any low consequence/frequency accident sequences that do not make a significant contribution to overall risk. Results of the screening performed for the NPP Test Case are provided in Section 4.

2.9. EXISTING SAFETY ASSESSMENT

No preliminary decommissioning assessments are available. The NPP has prepared safety assessments for operational and care and maintenance phases of the plant. Data from the volunteered NPP safety cases was used for the NPP Test Case. It contains useful material, such as plant descriptions, services descriptions and relationship between releases to the environment and doses to the critical group that has been used in the analyses included in this test case. All NPP systems have been characterized for their radioactive inventory and the data used in this test case are based on these documents.

2.10. SAFETY MANAGEMENT MEASURES

The NPP has prepared a management system for decommissioning that includes: organizational structure for decommissioning with clear responsibilities and authorities; change control procedures; work control procedures; maintenance and testing procedures; personal protective equipment; training and testing programmes; trained personnel; radiation protection programmes and procedures; occupational safety programmes and emergency preparedness programmes; quality assurance programme, as well as procedures for documentation and record keeping. The NPP has a good and

[1] HAZOP stands for Hazard and Operability Analysis

known operational record and maintains a good safety culture that also applies to the NPP decommissioning.

It is required that these safety measures will remain in place until decommissioning is completed.

3. DESCRIPTION OF THE FACILITY AND DECOMMISSIONING ACTIVITIES

The NPP has been in a care and maintenance regime since December 2001, after all spent fuel had been transported from the site. Care and maintenance is comprised of the following: closed building, controlled ventilation with filtration still present, irradiated components covered by water in the ponds, instrumentation systems in operation for fire monitoring/fire fighting and for radioactivity measurements. There are still contaminated and activated components in the reactor building.

3.1. SITE DESCRIPTION AND LOCAL INFRASTRUCTURE

3.1.1. Site description

The NPP area is located in the southern part of a peninsula enclosed by a bay in the north and south. The bay is 1.8 km long with approximately 2.4 km wide opening. The water depth is about 6 m at the deepest spot. In the southern part of the bay, there is a shallow zone about 1 km wide, where the deepest spot is about 6 m. The transition between the shallow zones close to the shore and the outer, deeper zones, where the water depth is varying between 10 and 20 m, is rather fast in the whole area described (see Fig. 5).

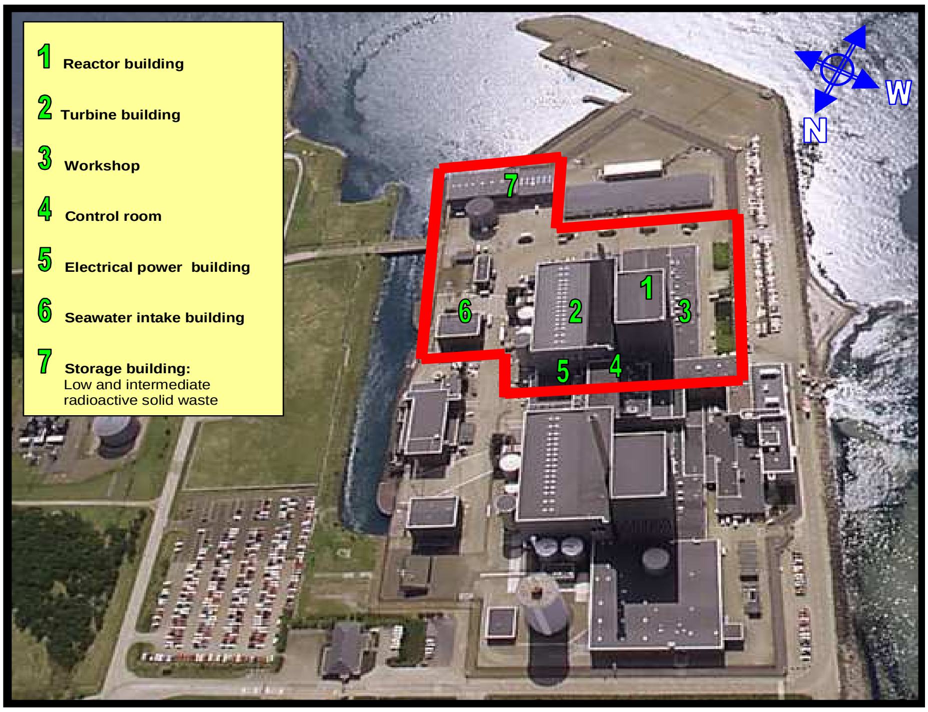

FIG. 5. Overall view of the NPP site (unit 1 marked in red).

The site is situated southwest approximately 1.5 km from the nearest village and 16 km north of the nearest city. The closest big city is situated about 20 km from the NPP.

The cooling water channel that previously supplied cooling water to the units, runs east of the facility through an open canal from the south side of the power plant level surface, until it reaches the coolant intake building for unit 2 (see Fig. 5).

The cooling water was returned to the sea via culverts which discharged at the shore on the west side of the facility. The return channel is dimensioned for a water flow of approximate 60 m^3/s, corresponding to a depth of 4.0 m, bottom width of 36 m and inclination of 1:2. On the south side of the power plant level surface, the opening of the canal is protected by two breakwaters, and in front of the inlet to the cooling water canal are dredged sedimentation pools reaching to a level of -7.5 m.

Leading to the cooling water canal is a dredged sailing channel, with bottom level at -6.0m. Inside the breakwaters is a small harbour that is used to ship out radioactive waste. The water depth in the harbour is the same as in the sailing channel, i.e. -6.0 m.

At the NPP area, two power units are in place (see Fig. 5). The outside dimensions for the reactor buildings above ground are 34 x 36 m. The building sides are directed towards the north-east and south-west respectively, that is, with approximate 45 degrees deviation from the main cardinal points. The height of the reactor buildings is 60 m. Connecting chimneys reach another 50 m, resulting in a total height of 110 m above sea level.

Every reactor building is connected to a turbine building, consisting of two parallel, windowless annexes of different lengths, directed in the same way as the reactor buildings. The dimensions of the turbine building are: length 86 m and width 35 m. The turbine building has a total height of 33.5 m above sea level.

The electricity buildings are placed between the turbine buildings for each block. In the space between the electricity building and the reactor building, there is an auxiliary cooling water building, a workshop and a storage facility for unit 2. The workshop and storage facilities are placed south of the reactor building.

On the south side of the NPP area, at the inlet of the cooling water canal, there is a building for storage of with low level waste (LLW) and intermediate level waste (ILW).

The waste treatment building, the coolant intake buildings and the facility for desalination are located east of the turbine buildings and placed along the inlet of the cooling water canals.

Next to the waste treatment facility for both units, there is a hydrogen making facility that during normal operation produced hydrogen for dosing to both units. During the care and maintenance period no hydrogen is produced, and there is no residual hydrogen in the factory.

North of the turbine building of unit 2 is a cylindrical building (FILTRA), built with diameter 20 m and height 40 m. At the top, a 10 m high chimney is connected. The pipes connected to the pressure relief valves in the secondary containment at each block are connected to FILTRA via an outer culvert in the ground. This building provides emergency cooling to the secondary containments on both NPPs in the event of a loss of containment accident. No such accident ever occurred at this NPP.

East of the road leading up to the NPP, i.e., outside guarded area, are the outer facilities built. Within this area there is a gas turbine facility, with oil tanks and the electrical switchgear for 400 kV and 130 kV cables. At the end of the road, south of the gas turbine buildings, there is an information building.

North of the gas turbine facility is a village camp used as accommodation by temporary workers who were hired for the overhaul outages.

Within the reactor building are the primary and secondary systems of the NPP. The NPP Test Case will assess only the safety of decommissioning of two of the unit 1 systems, 321 and 322, to allow a simple test case report. As mentioned in Section 2 and as described in more detail in Section 4, these systems are chosen to represent the extremes of high and low hazards to allow application of the graded approach.

3.1.2. Local infrastructure

A road leads away from the NPP, and connects to the nearest highway connects, located 3.5 km from the NPP. A separate analysis has showed that the risks for transports of dangerous goods to affect the NPP are negligible.

The nearest railway station is located 16.5 km from the NPP. No part of the railway stretch is in the vicinity of the NPP facilities.

The nearest airport is located approximately 21 km from the NPP, and there is frequently air traffic in the vicinity of the NPP, as well as some local air traffic. A separate investigation, dated 1999, shows that an airplane crash in the vicinity of the plant has very low probability. The probability of an airplane crash was estimated to be lower than $1.8.10^{-6}$ per year and thus the potential event does not require analysis for this decommissioning safety assessment.

3.1.3. Population distribution and critical group

The closest village is situated 1.5 km northwest of the site with a population of 200 people. The population within a 15 km radius of the site is approximately 20 000 people. In the immediate surroundings there are also a fishing port, agriculture and summerhouses etc. The closest major city is about 20 km from the plant. In an area stretching from the north-western to the south parts there are also some deciduous forests.

Although the closest village is situated at 1.5 km from the site, it is assumed for conservative evaluation, that the members of critical group live at the boundary of the site where the distance from the stack is 500 m. They are assumed to spend 24 hours every day a year in their house, and they eat vegetables, meat and milk which were cultivated at the boundary. The amounts of food ingestion for individual adult of members are assumed to be 310 g/d for leaf, 400 g/d for root, 250 g/d for meat, and 1000 g/d for milk and milk products according to their dietary habit.

Because no liquid effluent is released into the sea, pathways of public dose such as swimming, fishing at the sea and ingestion of seafood are excluded in the evaluation of system 321 and 322 dismantling. The analysis models used would be able to calculate doses from this pathway if needed.

3.1.4. Current and future land use

The site has two boiling light water reactors that were used for electricity production. These units are at present at a care and maintenance stage prior to decommissioning. At the NPP area, there is a port which first and foremost is used to ship waste to a final disposal site for radioactive waste.

After completion of decommissioning, the site is intended for release from regulatory control after decommissioning, that is to say, release from regulatory control. At the end state for the site, buildings are envisaged to be left for other industrial purposes.

3.1.5. Meteorology

The meteorological data from the national weather stations close to NPP provided meteorological data about the temperature, precipitation and wind. The given data are mean- and extreme values of air temperature, precipitation and wind. These data are taken from the operational safety case of the NPP. They form a good basis for this test case, as the decommissioning exercise assessed by the test case follows quickly on from the operational period.

a. Air temperature

The highest temperature measured since 1936 is 34˚C and the lowest is -28˚C. These temperatures are measured in a thermometer cage 1.5 – 2 m above ground. At the cage level, above grass, the temperature can rise to an estimated 35-40˚C and fall to -35˚C and -40˚C under extreme circumstances.

b. Precipitation

Mean precipitation per month (mean value for the period 1961-1990) is 603 mm, of which approximately 10% is constituted by snow.

The highest measured precipitation value is 65 mm/d. The highest value reported in the region is 300 mm per day. The upper limit in the country is estimated to be between 300 and 400 m per day (24 h).

In extreme cases, an amount of 35 to 50 mm of precipitation can fall during a 10 min. period. This is in cases of rain or rain with hail. There are no representative values calculated for frozen rain and snow, but the frequency is lower than for rain. The frequency for 20-25 mm precipitation falling within a 10 min. period is estimated to 10^{-2}/a.

It is recorded that flooding from extreme precipitation has never happened and so this can be dismissed from the list of external hazards shown in Section 4, and in any case flooding would not be a relevant hazard for the decommissioning of systems 321 and 322.

c. Winds

The wind speed rarely exceeds 15 m/s.

Table 4 shows the calculated wind frequencies (% of time) divided into direction and speed, measured at 10 m above ground at the NPP during the period 1987-1992.

TABLE 4. WIND FREQUENCIES AT THE NPP

Speed [m/s]	N	NE	E	SE	S	SW	W	NW	Sum
0.5-2.5	2.0	2.1	3.6	1.5	1.5	1.0	1.5	1.9	15.1
2.5-4.5	2.4	2.3	4.5	2.6	2.8	2.6	3.4	3.2	23.8
4.5-6.5	1.7	1.4	3.4	2.5	3.0	3.5	4.4	3.2	23.1
6.5-8.5	0.87	0.60	1.9	1.8	2.4	3.3	4.0	2.6	17.5
8.5-10.5	0.36	0.20	0.88	0.99	1.5	2.3	2.8	1.7	10.7
10.5-12.5	0.13	0.05	0.34	0.43	0.78	1.3	1.6	0.93	5.5
12.5-14.5	0.04	0.01	0.11	0.14	0.33	0.54	0.71	0.44	2.3
14.5-16.5	0.01		0.03	0.04	0.11	0.18	0.26	0.18	0.81
16.5-18.5			0.01	0.01	0.03	0.05	0.08	0.06	0.24
18.5-20.5					0.01	0.01	0.02	0.02	0.06
Sum	**7.5**	**6.6**	**14.7**	**10.0**	**12.5**	**14.8**	**18.8**	**14.2**	**99.1**
Still									**0.09**

The extreme wind speeds at NPP have been calculated for two different heights above ground (10 m and 50 m). The calculated extreme winds (see Table 5) are the highest values, that with 99% probability not will be exceeded during one year, i.e., wind speeds occurring once every 100 years. As a comparison, Table 6 shows the values that are exceeded once per year. Mean wind is a mean value during 10 minutes whereas the wind gusts have a mean value for 3 seconds.

TABLE 5. CALCULATED EXTREME WIND SPEEDS OCCURRING ONCE EVERY 100 YEARS AT TWO DIFFERENT HEIGHTS ABOVE GROUND AT NPP

Height [m above ground)]	Mean Wind [m/s]	Wind Gust [m/s]
10	30.6	43.0
50	39.8	50.5

TABLE 6. CALCULATED EXTREME WIND SPEEDS OCCURRING ONCE EVERY YEAR ON TWO DIFFERENT HEIGHTS ABOVE GROUND AT NPP

Height [m above ground]	Mean Wind [m/s]	Wind Gust [m/s]
10	22.1	31.1
50	28.7	36.5

The wind speed in a tornado can during short periods of time reach 70-100 m/s. The frequency of such events at the NPP is estimated to 10^{-5}/y. For normal winds, the given maximum value during a period of 50 years is 45 m/s.

d. Changes in air pressure

A change in air pressure of around 10 kPa/h has been observed in the region. Such pressure falls are estimated to be occurring approximately once every 100 years. Above the ocean, a change in the air pressure of 53 kPa/h has been observed in connection with a very intense low pressure. A change of this magnitude is utterly extreme and can be an approximate upper limit for fast changes in air pressure over the country.

e. Lightning

A normal lightning flash is defined in such a way that 2% result in major impacts. Usually, a normal lightning flash is defined in such a way that 10% of all flashes result in higher risks than a normal flash. Table 7 shows data for both of the normal flashes.

TABLE 7. DATA FOR NORMAL LIGHTNING FLASHES

Share of Flashes with Unfavourable Values	**10%**	**2%**
Total time t_o (s)	0.4	0.9
Rise time t_1 (microsecond)	0.9	0.7
Half-time value t_2 (microsecond)	45	100
Number of strokes per flash (N)	7	12
Charge Q (As)	90	160
Peak current I (kA) (stroke No. 1)	60	110
Steepness dI/dt (kA/μs) (stroke No. 2)	25	80

The highest values of N, I and dI/dt do not occur in the same lightning flash, and therefore the real safety level is higher than 90% and 98% respectively.

3.1.6. Geology

The area of the NPP is located on a layer of approximately 30 m, consisting of highly compressed moraine clay and muddy moraine, on the edge of the so called a depression in the underlying limestone rock. Only the top layer, down to a few meters depth, consists of less compressed gravel and sand. The depression is a valley created by erosion of the limestone in connection with a glacial period more than 150 000 – 200 000 years ago. The depression and the surrounding area has since then been filled with sediment, consisting of moraine clay, gravel and sand in several layers, compressed during the following glacial periods.

The facilities within the NPP area have mainly been erected on the highly compressed moraine clay. Only the top layer, down to a few meters depth, consists of less compressed gravel and sand.

The terrain is almost completely flat, which is distinct from the otherwise forested and sea-rich geography of the country. There are hardly any mountains or even hills or lakes or forests. Denser forests are only found in the north-eastern parts that border to the heavily forest dominated province.

3.1.7. Hydrogeology

Some of the NPP structures penetrate below the groundwater table, and as a result all drainage and ground water are collected in a sump. This is instrumented and fitted with pumps to ensure and control the water level around and under the buildings.

(a) Sea water level

The water level is mainly determined by the winds and the air pressure, which primarily affect the water level in the region.

These conditions, generally causing extreme water levels in the area, arise during the period September/October until April.

Fast changes can cause so called Seicher-waves; long waves, reflected back and forth between the neighbouring coasts with a period of approximately an hour. Obviously, these can also increase an extreme water level with a few decimetres, for a short period of time.

Measurements of the water level have been collected for a consecutive period of 1937 to 1968, although this was only with a few measurements per day.

The statistic material below contains information about the mean water levels, maximum and minimum levels. Data for 1937-1993 is summarized in Table 8 below. The land rise coefficient is 0.03 mm/y.

It is recorded that flooding from extreme sea levels has never happened and so this can be dismissed from the list of external hazards shown in Section 4, and in any case flooding would not be a relevant hazard for the decommissioning of systems 321 and 322.

TABLE 8. WATER LEVEL NEAR THE NPP, MEASURED IN 1937-1993

	Level [cm]	**Day of Registration**
Highest water level on record	101.0	1962-02-17
Mean value of annual HHW[1]	70.0	
Mean value of annual MW[2]	0.0	
Mean value of annual LHW[3]	-54.0	
Lowest water level on record	-97.0	1941-11-12

Note: [1]HHW = Highest water level
[2]MW = Mean water, with the land rising considered
[3]LLW = Lowest water level

An extreme value was reached during a storm from northwest in December 1902. This has been verified through studies of literature from the period. With a careful localization of such data and from measurements, it has been established that the water level at this point was as high as 180 cm above the MW-value above. The data from reference needs also to have an estimated margin of error of plus minus 40 cm, for the value of 180 cm. from this Table 9 is derived to show the extreme values of high and low sea level and their return period.

TABLE 9. PROBABILITIES FOR EXTREME SEA WATER LEVELS IN RELATION TO THE MEAN WATER LEVEL, MW.

Recurrence Time [y]	**Extreme High-water Level [m]**	**Extreme Low-water Level [m]**
100	1.31	-1.32
1 000	1.57	-1.71

From this it can be shown that variations in sea level are extremely unlikely to present a hazard to the site during the decommissioning period.

b. *Ice formation*

Frazil icing can occur due to strong winds in combination with temperatures below zero and open water. The water temperature on these occasions is approximately 0°C. Ice crystals are created on the water surface and drawn down to several meters depth by the turbulence. They then form clusters that can stick to constructions below the water surface. Gale winds can make them reach down to 10 m. depth.

The ice is often drifting back and forth due to winds and currents. The ice sheets are small, almost as crushed ice (partly due to the ship traffic in the strait). Snow and ice slush are also occurring. If the onshore wind is strong, the ice is pressed towards the coast. A packed zone of ice slush, a so called ice slush bank, is produced, and has occasionally been measured to reach 6-7 m below the water surface.

When the wind is less strong and when the water level is decreasing, the ice slush bank will run ashore and remain to several meters depth on the ferrule. If the weather is cold enough, the ice will freeze and create a thick, immobile ice cover.

3.1.8. Seismology

The NPP is located in a region with low earthquake activity. There are, however, geographical variations in the activity – mainly two regions with noticeable earthquake activity. The platform tectonics, with dynamic propagation from the nearest ridge, and the land rise, continuing since the last glacial period, have been presented as two main hypotheses of why earthquakes occur in the region. Whichever the reason is, fissures or zones of weakness in the bedrock are required to release accumulated energy in case of relative movement between blocks. Such a sudden displacement, when the friction in the rock no longer can hold back the potential movement, constitutes the earthquake itself. To find out the exact occurrence of earthquakes in an area, good knowledge about the distribution and character of the fissures is required. For the NPP area, such detailed knowledge is missing, and as the seismic statistics is limited, a reliable correlation analysis of the known faults and the probability for earthquakes can not be conducted.

Earthquakes are a phenomenon reaching over millions of years, and the national statistics comprise, with varying reliability, only the last 900 years. Registrations based on instruments are only available since the beginning of the 20th century, whereas earlier statistics are based on documented descriptions of the consequences of the earthquakes. For earthquakes with a Richter-magnitude up to 4.5, there are enough statistics in the region to find geological/geographical relations, whereas this is not established for higher Richter-magnitudes. In more active regions in the world, with known active faults, there is a linear relationship between small and large earthquakes. For this region, known active faults are missing. On the other hand, earthquakes with magnitudes over 4 occur occasionally even in areas that do not have heightened seismicity. It has therefore not been possible to determine a maximum value for earthquakes in the region, or a regional distribution of large earthquakes. It is conservatively assumed that the distribution is even over the region.

3.1.9. Natural resources

The land area surrounding the NPP consists of highly fertile agricultural land. For instance, 90% of the country's sugar beets are grown in the region. Figure 6 shows the pattern of agriculture in the region surrounding the NPP. The soil in southwestern of the region is among the most fertile in the world. The ellipse marks the location of the NPP.

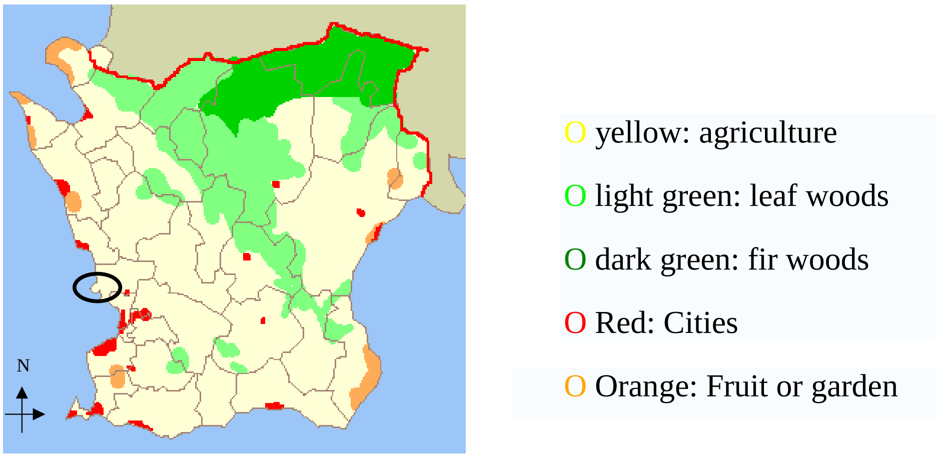

FIG. 6. Overview of the regional agriculture.

3.2. FACILITY DESCRIPTION

This Section contains a description of the NPP's construction, function and performance during the care and maintenance operation period; including barriers, protection functions, requirement systems and handling of radioactive waste (see Fig. 7).

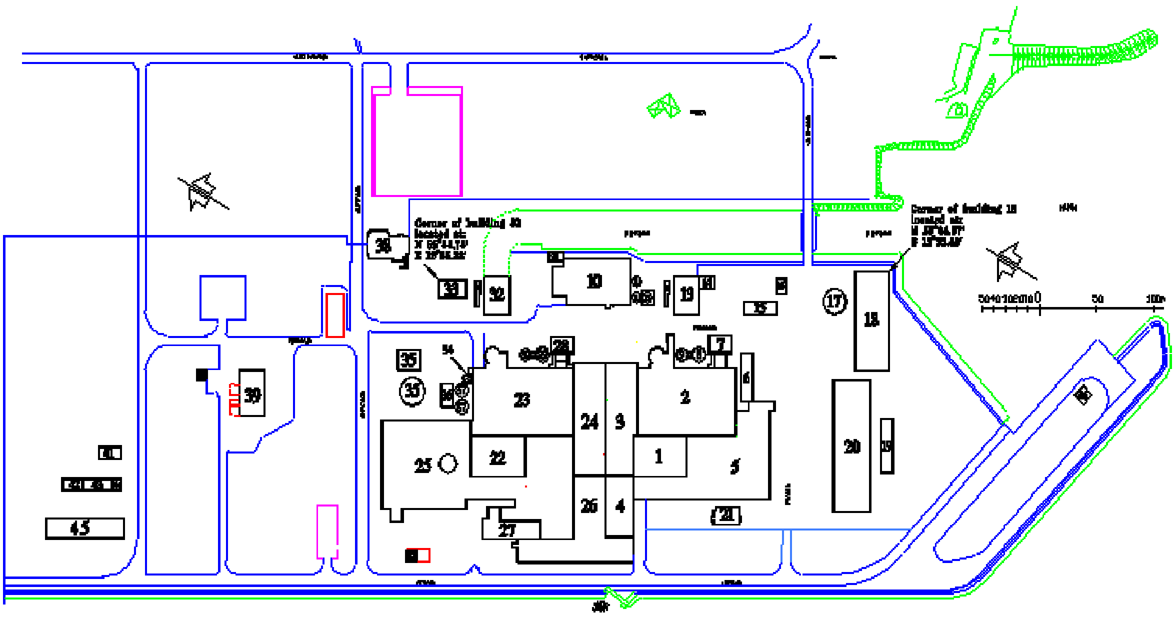

FIG. 7. NPP site layout.

3.2.1. Description of the NPP

The NPP consists of two boiling light water reactors for electricity production with thermal reactor output 1800 MW and electrical generator output 615 MW from each unit (see Fig. 8). Both units are closed down earlier than planned, respectively NPP closed down in 1999 and unit 2 in 2005.

3.2.1.1 Systems, large components and the buildings

The main buildings, components and systems are presented in Fig. 8 to Fig. 11 and are also summarized in Table 10 below.

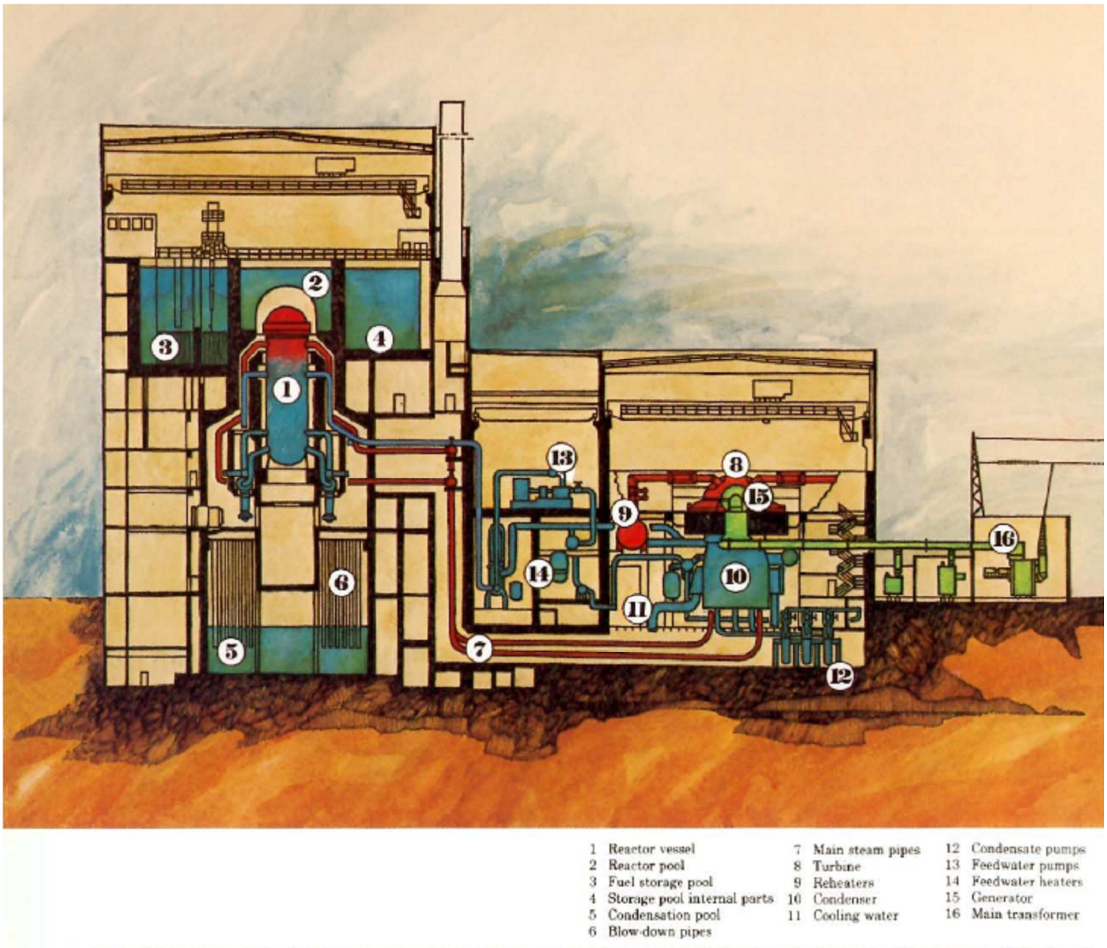

FIG. 8. Vertical cross-section of unit 1

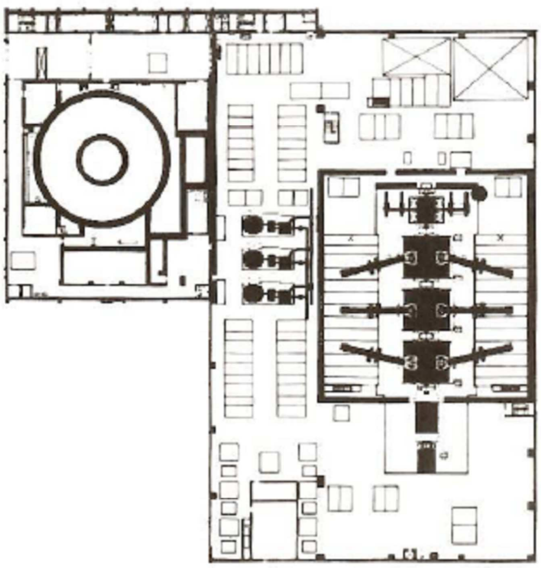

FIG. 9. Horizontal cross-section of unit 1.

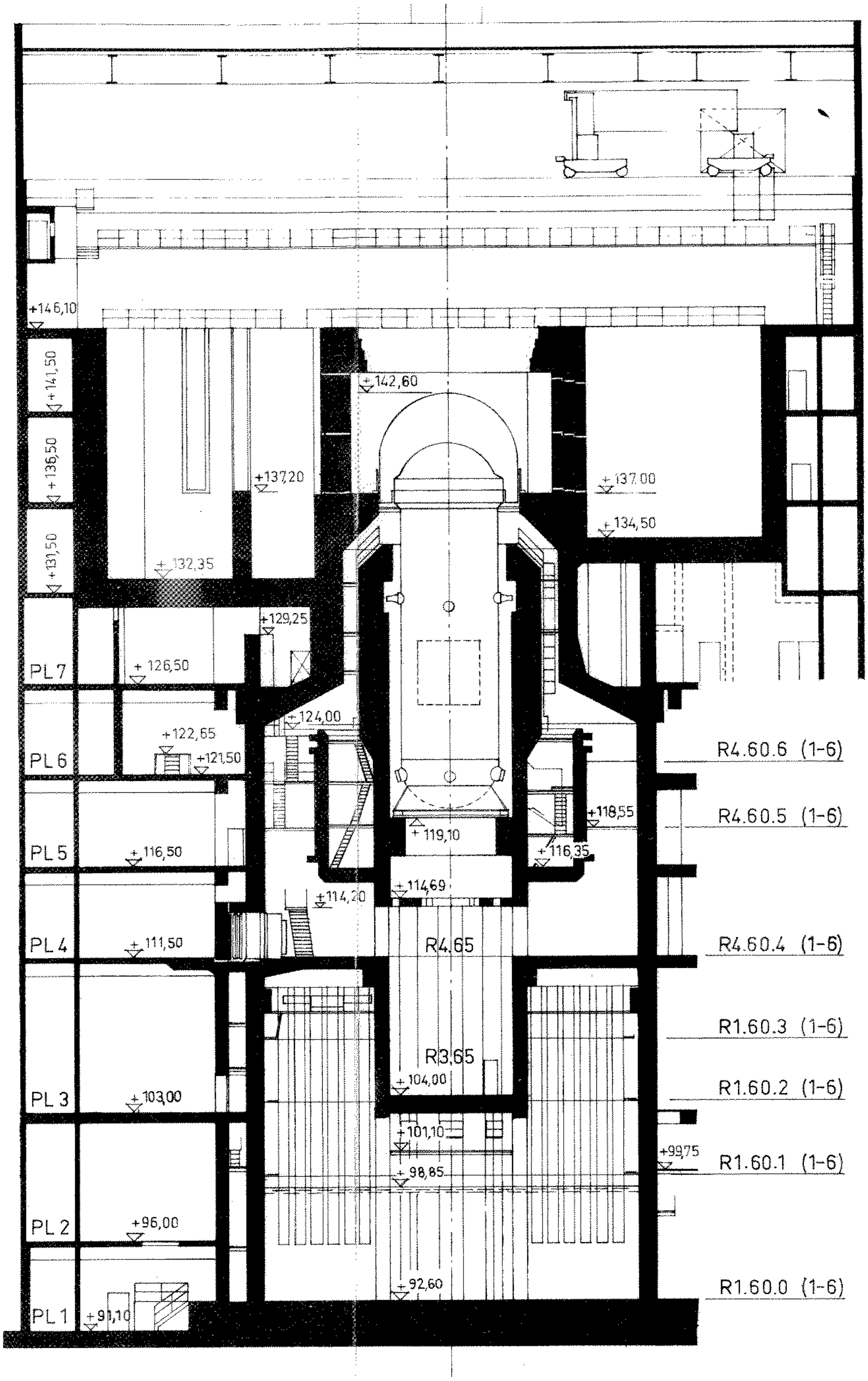

FIG. 10. Cross section of unit 1.

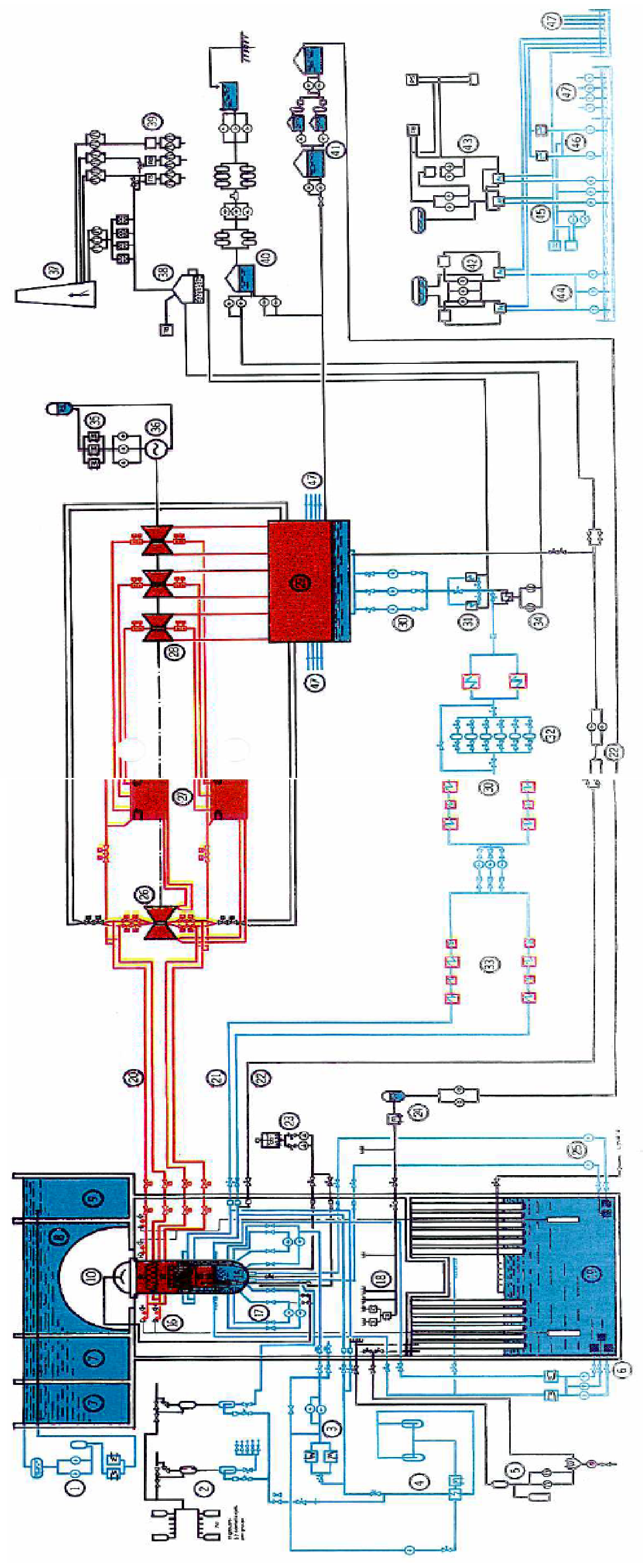

FIG. 11. Main process systems of unit 1.

TABLE 10. LIST ON MAIN SYSTEMS OF UNIT 1

1. Cooling and cleaning system for fuel storage pool	20. main steam pipes	39. Ventilation plant for other active spaces
2. Hydraulic system for control rod drive mechanisms	21. Feed water system	40. System for total demineralization, storage and distribution of demineralized water
3. Cooling system for shutdown reactor (system 321)	22. Auxiliary feed water system	41. System for processing liquid radioactive waste
4. Cleaning system for reactor water	23 Boron injection system	42. Secondary cooling system for starting and shutdown
5. Gas processing system for reactor containment atmosphere	24. Drainage system for reactor core	43. Secondary cooling system for service requirements
6. Cooling spray system (system 322)	25. Sprinkler system for reactor core	44. Seawater cooling system for starting and shutdown
7. Storage pool for irradiated fuel assemblies and control rods	26. High-pressure turbine	45. Seawater cooling system for service requirements
8. Water radiation protection pool	27. Reheaters and moisture separator	46. Seawater cooling system for diesel engines
9. Storage space for internal components	28. Low-pressure turbine	47. Condenser and auxiliary water system
10 Sprinkler system for reactor vessel lid	29. Condenser	
11. Moisture separator	30. Main condensate system	
12. Steam separator	31. Condenser, evacuation and recombiner system	
13. Core sprinklers	32. Cleaning system for condensate, powder	
14. Reactor core	33. Feed water system	
15. Control rod	34. Leakage and auxiliary steam system	
16. Blow off system	35. Generator cooling system (external circuits)	
17. Circulation system	36. Generator	
18. Leakage monitoring system	37. Stack	
19. Condensation system	38. System for radioactive waste gases	

(The NPP Test Case systems 321 and 322 are highlighted in blue)

(a) Safety related systems, structures and components

Strategic components that are needed during the decommissioning period have been analysed from their lifetime service in light of the new timescales associated with the care and maintenance period, with future new investments in replacement systems considered. This set of most relevant safety related structures, systems and components (SSCs) for the whole NPP is presented in Table 11. It needs to be remembered that the NPP Test Case will only consider the removal of the two plant systems already specified.

TABLE 11. SUMMARY OF SAFETY RELATED SYSTEMS, STRUCTURES AND COMPONENTS

System of NPP	Requirement/ Safety Function	Decommissioning Service
BUILDINGS		
Outer facilities		
Bridge (N)		X
Harbour (O)		X
Culverts (K)		X
Main building parts		
Reactor building (R)	X	X
Turbine building (T)	X	X
Electricity building		X
Waste building	X	X
AB- and C-storage	X	X
Water plant building		X
Workshop-, storage-, and auxiliary cooling water building (V)	X	X
Other buildings		
Building for transport containers, ATB-garage (Q)	X	
REACTOR WITH SERVICE EQUIPMENT		
Equipment for disassembly of internal parts		X
Bolt tensioners for the reactor vessel lid		X
Equipment in the storage pools		
Equipment in the storage pool for irradiated fuel cartridges and control rods	X	
Equipment in the storage areas for internal parts	X	X
Sealing between the vessel and the reactor containment		X
Sealing plate		X
WASTE SYSTEMS		
Equipment for treatment of liquid radioactive waste	X	
Equipment for treatment of solid radioactive waste	X	X
Cleaning equipment (decontamination equipment)		X
Floor drains in controlled areas	X	X
Oil spill handling (only BVT2)		X
Safety- and drainage systems		
Drainage system for the reactor part		X
MONITORING EQUIPMENT		

System of NPP	Requirement/ Safety Function	Decommissioning Service
Equipment shared by the programmable control systems		
Programmable electronic systems, shared by several systems	X	X
Control- and information network, shared by several systems	X	X
Man-machine interface, shared by several systems	X	X
Facility network, shared by several systems		X
Communication equipment reaching outside the facility		X
Shared control equipment		
Control boards		X
Desks		X
Apparatus- and relay lockers, apparatus lockers and voltage distribution		X
Switchboard, switch boxes and distribution frames		X
Cables, cable insulators and cable support systems		X
Process monitoring		
Process measuring		X
Valve maneuvering		X
Miscellaneous process maneuvering		X
Equipment for monitoring of radioactivity		
Radioactivity measurements in the main chimney	X	X
Radioactivity measurements in other rooms and areas	X	X
Monitoring of radioactivity in the surrounding area		X
External measuring equipment		
Meteorological measuring equipment	X	X
ELECTRICAL EQUIPMENT		
6 kV switchyard		X
400 V switchyard		X
Prioritized AC		
230 V AC inverter-secured grid	X	X
DC		
Rectifier		X
Batteries		X
DC distribution		X
110 V grid	X	X
Equipment shared by the electrical back-up power systems		
Protection equipment		X
Internal earth wire grid		X
SERVICE SYSTEMS		
Cooling systems		

System of NPP	Requirement/ Safety Function	Decommissioning Service
Ventilation cooling system		X
Water treatment- and distribution systems		
Storage and distribution of completely desalted water	X	
Ventilation facility		
Ventilation for other active areas	X	
Ventilation for uncontrolled areas		X
Water supply systems inside buildings		
Heating system for 742 (mixture of water and glycol)		X
Station heating		X
Hot water		X
Fire extinguishing water	X	X
Sanitary waste water		X
Runoff water, incl. from roof tops		X
Circulation system for utilitarian hot water		X
Auxiliary steam boiler		X
OTHER EQUIPMENT		
Lifting- and transport devices		
Traversers		X
Lifts		X
Harbour crane		X
Inventories		
Decontamination equipment		X
Workshop equipment		X
Transport radiation protection		X
Special tools		X
Gates		X
Radiological indication- and protection equipment		X
Lighting and electric sockets		
Power, indoor lighting		X
Outdoor lighting		X
Communication- and alarm systems		
Local telephone net, direct telephone lines		X
National telephone lines		X
Alarm facility		X
Fire alarm	X	X
Surveillance systems		
Passage control	X	X
Area surveillance	X	X
Fire protection system		
Fire protection equipment	X	X

(b) Common systems needed for decommissioning

The two units have mostly independent systems that allow their independent decommissioning. The exceptions are:

- The radioactive waste management treatment and storage facility;
- Site electricity supply;
- Common drainage systems;
- Common emergency arrangements for e.g. fire fighting; and
- Common management team and management systems.

3.2.2. Description of systems 321 and 322

Shut down cooling and clean up system 321 (number 3 in Table 11 above for main process system) consists of two parallel pumps, with one pump that was used in normal operation (see Fig. 12 and Fig. 14 below).

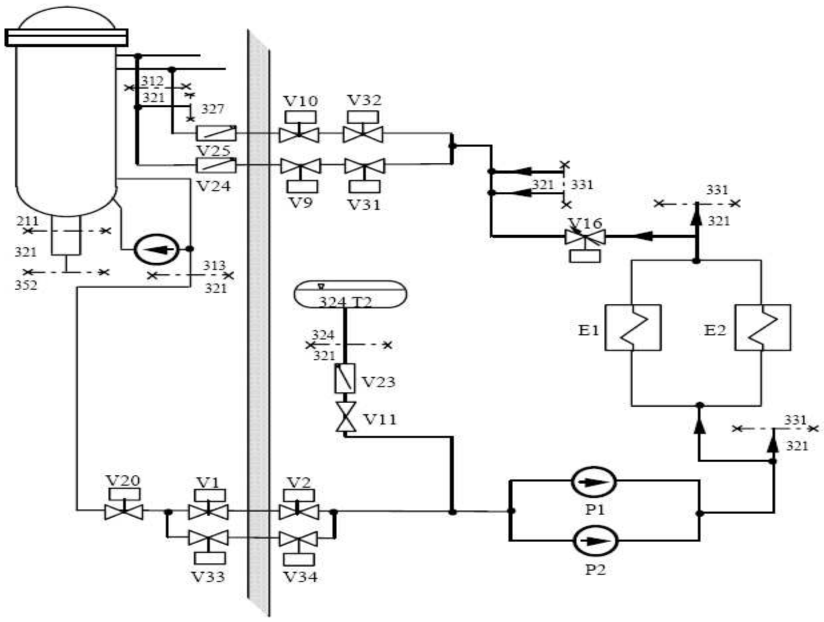

FIG. 12. Outline diagram of system 321.

Both heat exchangers are connected and cool the decay-heat from the reactor vessel until the vessel head is dismantled. System 321 cools the reactor from 10 bar (180°C) and steam dumps to the turbine-condenser or when there is no access to the condenser dumps it to the containment condensation pool (see Fig. 13).

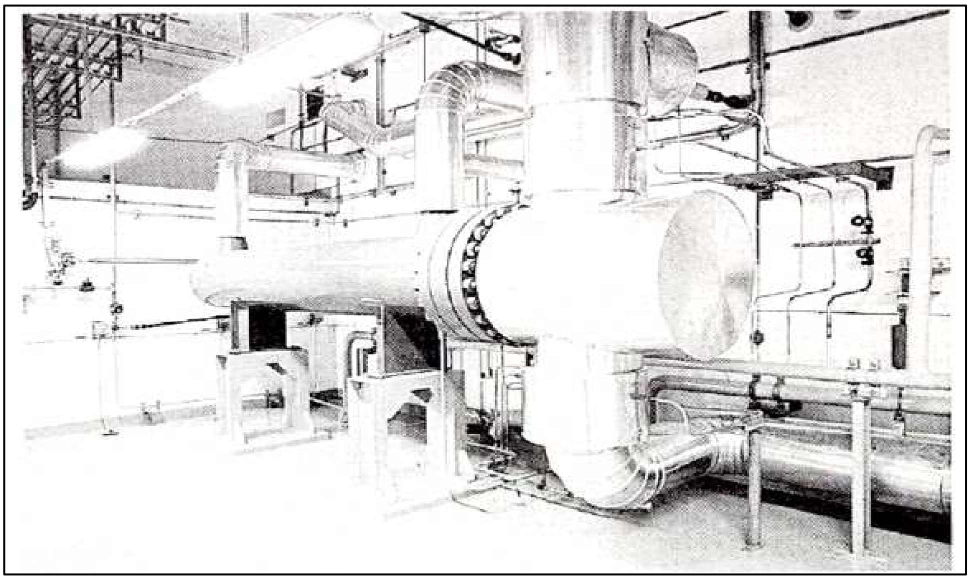

FIG. 13. Cooler from system 321.

The pipes of system 321 are of austenitic stainless steel, the pumps of stainless cast steel and the heat exchangers of tubes and end walls are manufactured in austenitic stainless steel, the header in compound plate and mantle in steel for pressure vessels.

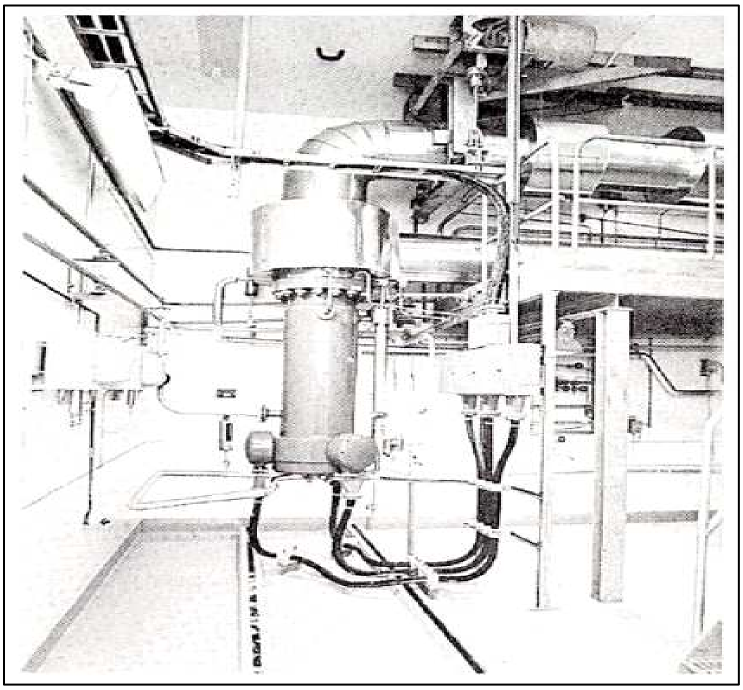

FIG. 14. Main pump of system 321.

The containment spray system 322 contains two separate circuits, each with one pump, P1 and P3 and also an extra pump P2, which provides with water from the condensation pool. Each circuit has 100% capacity and one pump during normal operation of NPPs (see Fig. 15 below).

The function of the containment spray system 322 during NPP operation was:

— In the event of a pipe break in the containment sprinkler the containment to decrease pressure and temperature, flush condensable fission products and achieve a circulation of the containment atmosphere; and

— In the event of a pipe break within the containment structure, to flush the strainers in the system 322 and 323 (an associated system for cooling of the reactor core after a loss of coolant with decreasing water level under a certain level in the reactor vessel).

The spray system 322 is held ready to cool the containment condensation pool. When the temperature is over 25 °C, the system automatically starts the pumps and mix the water with help from the so-called “cooling chain”, consisting of system 322-721-712 (secondary 721 and primary circuit of seawater 712).

The pipes of the system 322 are made of austenitic stainless steel and the pipes that connect to 323 are made of carbonized steel. The pumps are made of stainless cast steel and the impeller in of austenitic stainless steel (see Fig. 16). The heat exchangers E1, E2 have tubes manufactured in aluminium-brass and a shell made from steel. The heat exchangers E3 and E4 plate heat exchanger (see Fig. 17), with the plates made from austenitic stainless steel. The strain and sprinkler nozzles are made of austenitic stainless steel (see Fig. 18).

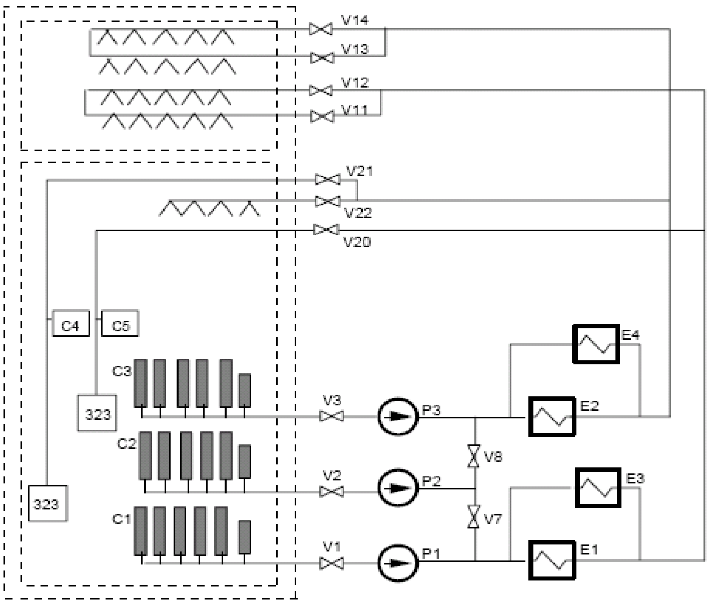

FIG. 15. Schematic of containment spray system 322.

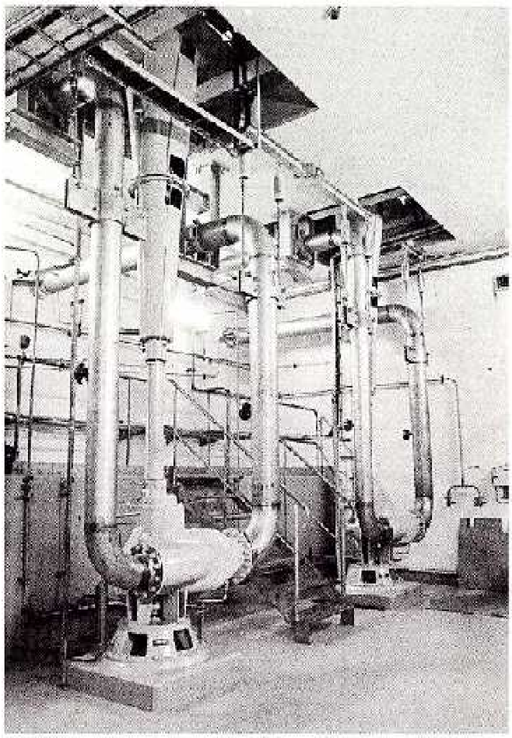

FIG. 16. System322 main pumps system 322 (P1 and P2). The electric motors are placed on the upper level.

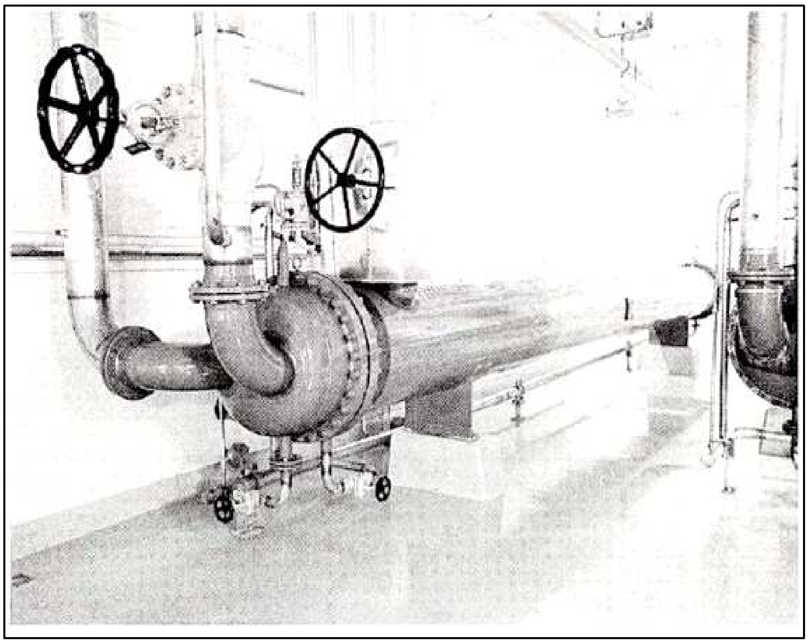

FIG. 17. System 322 heat-exchanger.

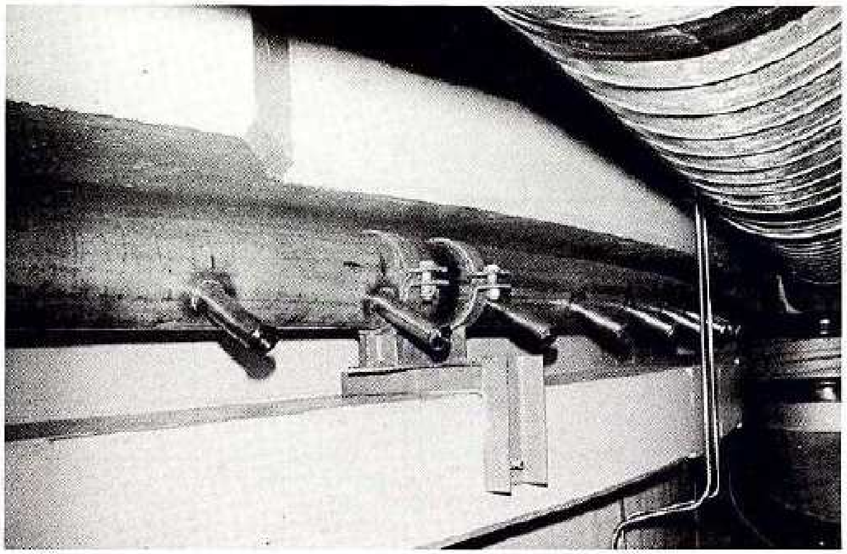

FIG. 18. System 322 upper containment spray nozzles.

3.3. RADIOACTIVE INVENTORY

3.3.1. Nuclear power plant

All nuclear fuel from both units was removed from the NPP site in 2001. The amount of induced activity in structures and activity contamination at the plant, not including fuel or operational waste, has been calculated by combining measurements and calculations 48 000 TBq (Co-60: ca 4700 TBq). The total radioactive contaminated waste has been estimated to approximately 5500 ton of that is 900 ton concrete waste.

The radioactive inventory estimation for NPP was performed on the basis of a combination of activity measurements during the NPP operation and after its shutdown. In order to quantify the degree of contamination of NPP after shutdown an extensive dose rate measurement and surface contamination sampling was performed. Different activity vectors categorized the material. These vectors include all other radionuclides, not only those that are easily measured by gamma measurements. The following techniques were used:

(a) A special sampling campaign was made after the NPP shutdown. These samples were evaluated by radionuclide specific gamma measurements. Samples taken from by drilling included:

- 2 samples from the biological shield outside reactor vessel, 150 cm deep in concrete; reinforcement steel; and insulation - caposile (a type of asbestos);
- 4 samples from concrete constructions, 15 cm deep; and
- 50 samples from floors, 4 cm deep; and (iv) 10 turbine samples, oxide layer.

(b) Direct radionuclide specific measurement with, normally about 30 measuring points;

(c) Dose rate measurements were made on same 250 points every time under consistent conditions. For example, if the measurements are done with water filled pipes it must always be done at the same conditions, the answers are in mSv/h. This measurement gives trends on increasing or decreasing of contamination and dose rates since the start of the NPP;

(d) Groundwater measurements around the NPP. Grab samples have been taken quarterly from drainage and groundwater sumps below the buildings. The samples are checked for example appearance of tritium. Normally no tritium is detected (limit of detection H-3 < 60 Bq/kg). During plant operations, there were a few reports of tritium being detected at above the limit of detection. Since the closure of the NPP, no such findings have been made. Tritium contamination could have an effect on the clearance of the site, because of the possibility that parts of the foundations of the reactor building are contaminated.

(e) Modelling of neutron induced activity for biological shield of the NPP. Contamination profiles for different materials and surfaces were also specified. Induced activity in core internals was calculated by using neutron activation code. All components were specified by elementary composition, weight, exposure to neutron flux and history of activation.

(f) Soil samples taken from outside the waste storage areas, used for waste containers, spent turbine parts, etc. have been checked for gamma emitters by soil sampling. In a few well-defined areas Cs-137 has been detected. The contamination is low (< 60 Bq/kg) and no further measure has been taken. Soil and gravel outside exits from controlled area is checked for beta/gamma emitters monthly. The values are normally below detection limit (Minimum detectable activity - MDA for Co-60 ~20 kBq/m²). If activity is indicated samples will be taken and gamma spectroscopy performed (MDA < 10Bq/kg). Over the years traces of Co-60 has been detected only rarely.

A database was made covering the system identity, room identity, dose rates, floor surfaces, tank surfaces, contaminated system surfaces and mass of components. The database was used to summarize the total amount of contaminated material, weight and activity.

3.3.2. Systems 321 and 322

The radionuclide specific values for contamination of the inner surfaces of systems 321 and 322 are presented in Table 11. This data is obtained as a result of radiological survey of system 321 before decontamination. The survey was conducted by taking smears and direct measurements of residues from cut pipes (metal). The calculated radioactive inventory based on measurements taken in July 2005 for system 321 is summarized in Table 12.

Cs-137 is absent from Table 12 as a result of the integrity of the fuel used during the operational lifetime of the NPP. There are some weak beta emitting radionuclides that are not present on the inventory, such as Nb-93. Given their low radiotoxicity and their lack of gamma radiation emission, this omission is not significant for the assessment of the safety of decommissioning activities. These isotopes will, however be significant in the assessment of the long term performance of any disposal facilities where this material is placed.

The sampling and direct measurements were also complemented by calculation of the concentration of the radionuclides using a proprietary computer code (BWRCrud) that simulate the chemistry and radiochemistry in BWRs. This programme simulates the activity concentration in reactor water and activation products on surfaces by define reactor specific: corrosion rates of different materials; deposition- dissolution rates of metals on core surfaces; neutron activation flux on core surfaces and internals and activity deposition rates on pipes and other surfaces.

TABLE 12. SURFACE CONTAMINATION OF INNER SURFACES OF SYSTEMS 321 AND 322 AS OF 2005

Isotope	System 321	System 322
	Activity [Bq/m^2]	
Mn-54	2.6×10^8	2.6×10^6
Fe-55	3.8×10^9	3.8×10^6
Co-60	3.4×10^9	3.4×10^6
Ni-59	1.5×10^7	1.5×10^4
Ni-63	2.1×10^9	2.1×10^6
Tc-99	9.8×10^2	9.8×10^1
Sb-125	1.8×10^8	1.8×10^5
Pu-238	1.5×10^3	1.5
Pu-239	1.7×10^2	1.7×10^1
Pu-240	2.7×10^2	2.7×10^1
Pu-241	6.3×10^4	6.3×10^1
Am-241	1.2×10^2	1.2×10^1
Cm-244	1.9×10^3	1.9

The calculated result from BWRCrud code is validated by:

- Radionuclide specific measurements of activity surface density [Bq/m^2] on surfaces as the system 321 and other system surfaces, totally about 8-12 every other year during outages;
- Measured dose rates on 100-150 different dose points during outage; and
- On-line radionuclide specific surface contamination measurements at one point in the system 321.

Other data that is used in the programme is crude sample analyses, feed water and reactor water analyses on metals and activity and fuel failure history and transuranics (TRU) analyses of reactor water. The output from BWRCrud is a diagram in time of activity surface density [Bq/m^2] on system surfaces for each radionuclide.

The calculated results were found to correspond very well with actual measured data. The program has been used to calculate consequences of power upgrade, change in feed water iron input and accelerated corrosion in fuel spacers, as some examples. During recent years the programme has been useful in safety assessment calculations. This is simplified and made as steady state calculations by an excel calculation sheet.

Hard to measure radionuclides, such as TRU, Ni-59, Ni-63 and Fe-55 are calculated the same way as other radionuclides, by using the calculated reactor specific constants for activity uptake, corrosion rate, etc. Specific measurements of these radionuclides have also been used to verify the result of the calculations.

System 322 took water from the condensation pool, system 316, a water reservoir of 2000 m^3. The Co-60 activity concentration of this water during operation was about 1000 Bq/kg. Other radionuclides are mostly long lived radionuclides, as the system 316 only was contaminated during outage, when water from fuel pool could contaminate this system. In general the 322 system was tested once a week by circulating water from the condensation pool through pipes and heat exchangers. The temperature of the system was always kept low, and the activity in pipes and heat exchangers is mainly due to particle contamination.

The contamination level of the system 322 was actually measured radionuclide by radionuclide specific surface measurements at one of the heat exchangers until 1999, when it was stopped. Also, dose rates were measured until 1998 when the yearly measurements unfortunately were stopped. Activity concentration data from the water of the system 322 has been collected to the end of operation. When looking at dose rate data one can see that there was an increasing trend during the 1990's. The contamination level expressed as contact dose rates of the most contaminated components is about 10% the level of the system 321. In general the system shows a low contamination level. However, this does not mean that the system 322 can be considered "non radioactive".

This means that the system 322 is mostly a low contaminated system where dose rates are low, but in some parts of the system, especially heat exchangers, the dose rates can make a contribution to collective doses to workers during decommissioning, especially if highly contaminated systems as system 321 have been chemically decontaminated, and system 322 is left not decontaminated.

Measured dose rates for the system 321 at the distance 0.5 m from the equipment, which is considered as the working distance during dismantling, are in the range of 0.02 to 0.48 mSv/h. The dose rates for the system 322, except the heat exchangers, are in the range of 0.005 to 0.030 mSv/h. The reason for dose rate for items of system 322 is not the contamination of these items, but the presence of items of other contaminated systems, like system 321, in the same rooms. The only contaminated items of the system 322 are the heat exchangers, where the dose rates for four inventory items is in the range of 0.015 to 0.030 mSv/h.

3.4. OPERATIONAL HISTORY

3.4.1. Nuclear power plant

In the entire operating life of the NPP, all operating parameters were within normal limits and the operational history is known to a high degree of accuracy. This will allow, for example, the flux history of irradiated components to be calculated. The normal operational mode, the duration of that operation, interruptions etc., is presented in Fig. 19.

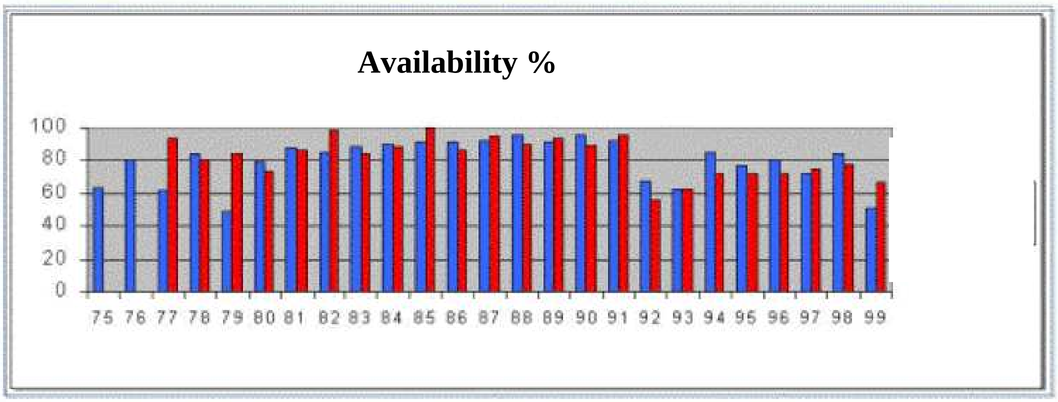

FIG. 19. Availability of the two NPP units during the operational period 1975-1999.

(a) Significant modifications

The NPP has been subject to some modification since it was constructed and commissioned. All modifications were carried out in a controlled and managed manner, and plant drawings are available that accurately describe the plant as it is now. When the NPP was shut down in 1999 a project was realized to increase the electricity supply to auxiliary feed water pumps by using NPP diesel generators. In addition, at the NPP site the following modifications were implemented:

— New building for storage and handling of LLW and ILW solid waste;
— New service building, mechanical and electrical workshops in the radioactive area;
— New hydrogen plant;
— Installation of hydrogen dosage into the feed water; and
— Programme on mitigation of intergranular stress corrosion cracking.

(b) Incidents or accidents

In the history of the operational period with NPP there were no significant incidents or accidents related to the safety of the decommissioning. There were very few small incidents with fuel leakage (about 15 instances in all) but there was no identification of any alpha- activity in the reactor systems or the buildings. The damage was caused by abrasion from foreign objects and pellet-clad interaction. The fuel damage was slight and did not lead to contamination of the systems with fission products. These events are judged to not affect free-release and dismantling of the facility. Equally, alpha emitting isotopes are not problematic in the assessments made for this test case, and neutron flux from transuranics has been dismissed as a hazard.

A complete record has been compiled of events that may have generated radioactive spillage into the plant have been listed. Personal interviews and meetings with employees took place to confirm that these events were connected to normal NPP operation.

As mentioned in Section 3.3, a special sampling campaign of concrete was performed after the NPP was shut down to confirm areas of highest contamination. Most relevant places of NPP that where affected by these incidents were identified:

— Sump for receipt of any spillage from condensation pool and underneath the reactor condensation pools;
— Spills of material from ion exchangers in different filter rooms, giving rise to ingrained contamination in the concrete floors in these rooms; and
— Sump for liquid waste under the reactor tank (In this example dominated Co-60 over Cs-137. The value for Co-60 was 1.63 kBq/cm^2, while Cs-137 was only 0.079 kBq/cm^2).

The consequences of these spills above indicate that radioactive contamination deeper in the concrete of the reactor building of NPP is to be expected.

No previous decommissioning activities have been implemented at the NPP.

3.4.2. Systems 321 and 322

There have been no serious incidents or accidents that have an impact on the safety of the decommissioning of the systems 321 and 322. The radioactive inventory of these systems has been established with a high degree of accuracy.

3.5. DECOMMISSIONING ACTIVITIES AND TECHNIQUES

The decommissioning activities planned for NPP are presented and discussed in Section 1 and are shown in Fig 1.

3.5.1. Decommissioning activities at systems 321 and 322

The decommissioning schedule for systems 321 and 322 includes the dismantling of systems, decontamination of building surfaces and final radiological survey of building surfaces in rooms where the systems are located. From the point of view of safety of planned decommissioning activities, the dismantling of systems is the critical activity, being performed at dose rate fields 0.5 m from the dismantled equipment and in the average dose rate fields of the rooms where the equipment are located. Other listed activities are evaluated as reference activities.

The overview schedules for decommissioning of systems 321 and 322 are presented in Table 13. Start date for decommissioning activities for the system 321 is the 3rd quarter of 2005. The decommissioning of system 322 starts after dismantling of system 321. The detailed schedules according to individual rooms, as generated in MS Project software, are presented in Appendix I.

One team of workers per each type of activity performs the activities according to the extent of inventory items within the rooms and according to the extent of rooms involved. The average number of members of the working group for dismantling is 7 workers, for decontamination of building surfaces 5 workers and for radiological survey of building surfaces 3 workers.

TABLE 13. OVERVIEW SCHEDULE FOR DECOMMISSIONING OF SYSTEMS 321 AND 322

Activity of systems 321, 322	3rd Q. 2005			4th Q. 2005			1st Q. 2006			2nd Q. 2006			3rd Q. 2006			4th Q. 2006			1st Q. 2007			2nd Q. 2007			3rd Q. 2007			4th Q. 2007		
	7	8	9	10	11	12	1	2	3	4	5	6	7	8	9	10	11	12	1	2	3	4	5	6	7	8	9	10	11	12
Dismantling of system 321	■	■	■	■	■	■	■	■																						
Decontamination of building surfaces of system 321		■	■	■	■	■	■	■	■	■	■																			
Radiation survey of building surfaces of system 321											■	■	■	■																
Dismantling of system 322									■	■	■	■	■	■	■	■	■	■	■	■	■	■	■							
Decontamination of building surfaces of system 322												■	■	■	■	■	■	■	■	■	■	■	■	■						
Radiation survey of building surfaces of system 322															■	■	■	■	■	■	■	■	■	■	■					

Decommissioning of the system 321 is planned to be conducted without prior decontamination. The approach for evaluating the decommissioning activities is that for each decommissioning activity, the general and safety related decommissioning parameters are calculated individually. The extent of specific decommissioning activities need to be defined based on the inventory database. For this purpose, the detailed inventory database was developed for individual items of the systems 321 and 322 and for relevant rooms, for the purpose of calculating the decommissioning parameters. The approach for implementing of the planned decommissioning activities is the room-by-room sequence (see Appendix I).

The room oriented approach includes implementing the set of preparatory activities for each room to prepare the conditions for proper and safe dismantling, the dismantling of items according to the inventory content for each room and finally the set of finishing activities in order to properly finish the dismantling and to leave the rooms in conditions ready for carrying out other decommissioning activities, in this model case the decontamination of building surfaces.

In order to balance the extent preparatory and finishing activities with the dismantling activities, the preparatory and finishing activities were applied only for limited number of rooms due to fact that the dismantling is only partial. The inventory items other than that of systems 321 and 322 are in principal present in the rooms evaluated. The approach selected was, that the preparatory and finishing activities were implemented for rooms where number of inventory items is larger than 10.

The preparatory and finishing activities, as implemented for the system 321 according to the above discussed approach, are presented in the Table 14 for the system 321 and in the Table 15 for the system 322. The data in the tables are defined for rooms according to the room database for both systems. System 321 and system 322 are planned to be decommissioned independently by the same groups of workers.

TABLE 14. PREPARATORY AND FINISHING ACTIVITIES FOR SYSTEM 321

Floor	Room	Preparatory Decommissioning Activities												Finishing Decommissioning Activities : Removal of …							
		Radiation survey	Protective foils	Local ventilation	Scaffolding	Electricity & media	Delineation of cuts	Transport of tools	Isolation of equipment	Preparation of tools	Protective tenting	Preparation of the WG	Preparation of containers	Protective foils	Local ventilation	Scaffolding	Electricity & media	Protective tenting	Tools	Containers with waste	Final cleaning of the room
+91.10 m	1R1.31	x	x	x		X	x	X		x	x	x	x	X	x		x	x	X	x	x
	1R1.32	x	x	x		X	x						x	X	x	x	x		X	x	x
+91.10/ +92.60 m	1R1.30	x	x				x						x	X			x		X	x	x
+92.60 m	1R1.57	x	x	x		X	x	X		x	x	x	x	X	x		x	x	X	x	x
+97.00 m	1R2.36																				
+103.00 m	1R1.57PL3																				
	1R3.31																				
+111.50 m	1R1.57PL4																				
	1R4.31	x	x	x		X	x	X		x	x	x	x	X	x		x	x		x	x
	1R4.60	x	x	x	x		x				x		x	X	x	x		x	X	x	x
+116.35 m	1R4.60.51	x	x	x	x	X	x			x	x	x	x	X	x	x	x	x	X	x	x
+116.5 m	1R1.57PL5																				
	1R5.31	x	x	x	x	X	x				x		x	X	x	x	x	x		x	x
+118.80 m	1R5.66																				
+121.00 m	1R4.60.61																				
	1R4.60.62																				
	1R4.60.63																				
	1R4.60.64																				
	1R4.60.65																				
	1R4.60.66																				
	1R4.60.67																				
+121.50 m	1R1.57PL6																				
+126.5 m	1R7.47	x	x	x	x	X		X		x	x	x	x	X	x	x	x	x	X	x	x

The preparatory activities for decommissioning within individual rooms are the following:

- Survey of radiological situation in the room for confirmation of the data used in the planning of dismantling activities;
- Covering of the floor with protective foils to inhibit the floor contamination;
- Installation of local ventilation to suppress the aerosols from dismantling;
- Installation of scaffolding for dismantling activities;
- Installation of temporary connections for electricity and other media needed;
- Delineation of cuts on equipment;
- Transport of dismantling tools to the dismantling sector;
- Isolation/check of equipment from electrical connection or operating media;
- Preparation of dismantling tools for the work;
- Installation of protective tenting for suppress the spreading of aerosols;
- Preparation of the working group (WG) for the decommissioning work; and
- Preparation of containers for waste from dismantling.

After preparation of rooms, the dismantling of individual items follows, using the appropriate techniques, depending on the physical properties of inventory items (material composition, size, etc.) and also radiological properties. Totally, 373 inventory items were defined for the system 321.

The finishing activities for dismantling within individual rooms are the following (see Table 15):

- Removal of protective foils on floors;
- Removal of local ventilation;
- Removal of scaffolding;
- Removal of temporary electrical connections and media for dismantling;
- Removal of protective tenting;
- Removal of dismantling tools;
- Handling of waste containers; and
- Final cleaning of the room after dismantling.

Decommissioning of system 322 will commence after completion of decommissioning of system 321 (estimated to last 8 months). The same approach is planned to be applied to system 322 as presented in Table 15. Totally 1031 inventory items were defined for this system.

TABLE 15. PREPARATORY AND FINISHING ACTIVITIES FOR SYSTEM 322

Floor	Room	Preparatory Dismantling Activities												Finishing Decommissioning Activities : Removal of …							
		Radiation survey	Protective foils	Local ventilation	Scaffolding	Electricity & media	Delineation of cuts	Transport of tools	Isolation of equipment	Preparation of tools	Protective tenting	Preparation of the WG	Preparation of containers	Protective foils	Local ventilation	Scaffolding	Electricity & media	Protective tenting	Tools	Containers with waste	Final cleaning of the room
+91.10/ +92.60 m	Virtual room	x	x		x		x					x	x	x		x				x	x
	1R1.08	x	x	x	x	x	x	x		x	x	x	x	x	x	x	x	x	x	x	x
	1R1.09	x	x	x	x	x	x	x		x	x	x	x	x	x	x	x	x	x	x	x
	1R1.28																				
	1R1.40	x	x	x	x	x	x	x		x	x	x	x	x	x	x	x	x	x	x	x
	1R1.40PL1																				
	1R1.40PL2	x	x	x		x						x	x	x	x		x			x	x
	1R1.40PL7																				
	1R1.47	x	x	x	x	x	x	x		x	x	x	x	x	x	x	x	x	x	x	x
	1R1.47PL1																				
	1R1.47PL2																				
	1R1.47PL5																				
	1R1.50	x	x	x	x	x	x	x		x	x	x	x	x	x	x	x	x	x	x	x
	1R1.50PL2																				
	1R1.50PL7																				
	1R1.57	x	x	x	x	x	x	x		x	x	x	x	x	x	x	x	x	x	x	x
	1R1.57PL4																				
	1R1.57PL5																				
	1R1.60	x	x	x	x	x	x	x		x	x	x	x	x	x	x	x	x	x	x	x
	1.R1.60.04																				
	1.R1.60.05																				

TABLE 15. PREPARATORY AND FINISHING ACTIVITIES FOR SYSTEM 322 (CONT.)

Floor	Room	Preparatory Dismantling Activities												Finishing Decommissioning Activities : Removal of ...							
		Radiation survey	Protective foils	Local ventilation	Scaffolding	Electricity & media	Delineation of cuts	Transport of tools	Isolation of equipment	Preparation of tools	Protective tenting	Preparation of the WG	Preparation of containers	Protective foils	Local ventilation	Scaffolding	Electricity & media	Protective tenting	Tools	Containers with waste	Final cleaning of the room
+97.00 m	1R2.08	x	x	x	x	x	x	x		x	x	x	x	x	x	x	x	x	x	x	x
	1R2.09	x	x	x		x	x	x		x		x	x	x	x		x		x	x	x
	1R2.10	x	x				x					x	x	x						x	x
	1R2.28	x	x	x		x	x	x		x		x	x	x	x		x		x	x	x
	1R2.30	x	x				x					x	x	x						x	x
	1R2.31	x	x	x	x	x	x	x		x	x	x	x	x	x	x	x	x	x	x	x
	1R2.35	x	x	x	x	x	x				x	x	x	x	x	x	x	x		x	x
	1R2.36																				
+103.00 m	1R3.08																				
	1R3.10	x	x				x					x	x	x						x	x
+111.50 m	1R4.51																				
	1R4.60	x	x	x	x	x	x	x		x		x	x	x	x	x	x		x	x	x
	1R4.60.42																				
	1R4.60.43																				
	1R4.60.44																				
	1R4.60.45																				
	1R4.60.62																				
	1R4.60.63																				
	1R4.60.64																				
	1R4.60.65																				
+116.5 m	1R5.41																				
+126.5 m	1R7.47																				

Similar sequence of activities was applied for decontamination of building surfaces for the system 322 after dismantling of equipment in the rooms as for the system 321. The planned preparatory activities are the following:

— Survey of contamination of building surfaces in the room;
— Covering of the floor to inhibit the contamination of the floor;
— Installation of local tent (for mechanical decontamination);
— Installation of local ventilation to suppress the aerosols from decontamination;
— Installation of scaffolding;
— Installation of temporary connections for electricity and other media needed;
— Delineation of areas for decontamination by various techniques;
— Transport of decontamination tools and equipment to the room;
— Preparation of decontamination tools and equipment for the work;
— Preparation of the working group for the work; and
— Preparation of containers for waste.

For each room involved, the items for chemical and for mechanical decontamination were defined with estimated areas for decontamination. In real decommissioning projects, these data need to be defined based on radiological sampling of rooms surfaces. The finishing activities involved after the decontamination are the following:

- Removal of scaffolding;
- Removal of temporary electrical connections and media;
- Removal of decontamination tools and equipment from the room;
- Removal of local tent (if mechanical decontamination was used);
- Removal of protective foils;
- Removal of local ventilation; and
- Removal of containers with waste;
- Cleaning of the room; and
- Construction of temporary closure of the room for inhibiting the re-deposition of contamination from other rooms.

As the last decommissioning activity planned in the NPP Test Case is the final radiological monitoring of building surfaces. The same sequence of activities was applied – preparatory activities, monitoring and finishing activities. The extent of rooms and involvement of preparatory and finishing activities according to individual rooms is the same like in dismantling and decontamination of building surfaces. Preparatory activities for final radiological monitoring of building surfaces are the following:

- Installation of scaffolding;
- Delineation of areas for radiological monitoring;
- Transport and preparation of measuring instruments; and
- Preparation of the working group for the work.

For radiological monitoring of building surfaces, the hand held instruments or instruments placed on manual or remote positioned small vehicles will be used, depending on shape and dimensions of individual rooms. The finishing activities for radiological monitoring are the following:

- Removal of scaffolding;
- Removal of instruments and tools for monitoring;
- Cleaning of the room; and
- Elaborating the documentation.

The approach described above enables the safety evaluation of each individual decommissioning activity. Using this approach and taking into account the composition of the working groups according to individual professions, the safety related decommissioning parameters can be evaluated on the level of individual workers (individual effective dose) and not only as collective data as is the case of more approximate approaches. The resulting safety related data reflects the procedure expected in real decommissioning projects.

3.5.2. Decommissioning techniques

The inventory items developed for the systems 321 and 322 were classified according to the decommissioning categories of equipment. The decommissioning category represents a typical equipment inventory item as for the material composition, size, thickness of the walls of the equipment and other physical parameters. Examples of decommissioning categories are pipes, valves, motors, elements of ventilation systems, etc., sub-classified according to dimensions. For each decommissioning category, the preferable decommissioning techniques are allocated. The main techniques considered in dismantling of systems 321 and 322 are the following:

- Hydraulic shears cutting for cutting of pipes with small dimensions, electrical cables, components of ventilation ducts and other equipments with thin walls;
- Plasma cutting for general equipments, preferably for stainless steel equipments;
- Mechanical cutting by mechanical saw or other mechanical cutting method which does not generate much heat, used in applications where lowered generation of aerosols is required, or for cutting for equipments with large wall thickness like reactor vessels;
- Oxygen - acetylene cutting of general equipment, preferably for those made for carbon steel;
- Manual dismantling using standard mechanical hand tools; and
- Grinding for cutting of equipments with medium wall thickness. The technique has relatively high cutting rate, but the release factors for radionuclides is high.

Selection of the above listed techniques was done automatically by OMEGA code used for calculation of decommissioning parameters, based on pre-selected optimal technology for individual decommissioning category. The current hazard detailed radiological accident analysis (given in Appendix III) does not analyze grinding techniques and indicates that grinding must not be used without further safety analysis. Techniques involving grinding were not considered in this test case calculation of decommissioning parameters.

Remote dismantling techniques are not considered for dismantling of the systems 321 and 322. These techniques will be used for dismantling of the reactor internals and reactor pressure vessels.

Decontamination techniques applied to room surfaces of systems 321 and 322 after removal of dismantled systems, are standard wiping decontamination techniques and the mechanical decontamination techniques. There are several techniques which mechanically remove the surface layer from the building surfaces like shaving. The thickness of the removed layer depends on depth of penetration of contamination and normally varies between 5 to 20 µm. In the NPP Test Case, the conservative approach was applied, which removes 20 µm of material.

It was also considered conservatively for rooms involved in the system 321, that all floors were mechanically decontaminated and all walls were chemically decontaminated. For rooms involved in the system 322, half of the areas of floors were decontaminated mechanically and half of areas of walls were decontaminated chemically.

3.6. WASTE MANAGEMENT

3.6.1. For the whole NPP

The radioactive waste at the site is being processed following the scheme, presented in Fig. 20.

(a) Sorting and size reduction of waste

Low-level waste will be transferred to the size reduction area from either the mixed waste sorting area or the potential clearance monitoring area. Size reduction will be carried out within an enclosed re-usable modular containment primarily using hand-held tools, see Fig. 20 (e.g. the waste handling area).

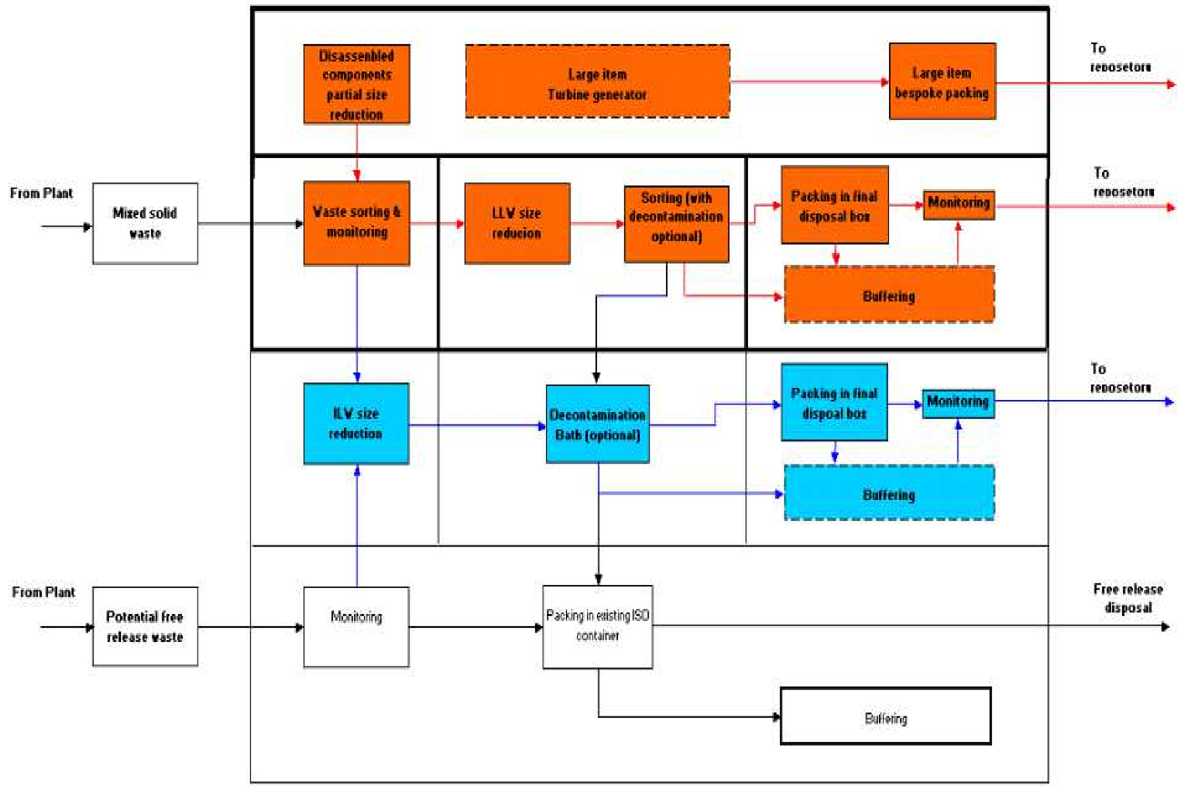

FIG. 20. NPP waste processing and package facility.

Thick line =	Denotes shielded cells equipped with remote handling equipment, saw table, vent plant, thermal cutting tools and effluent discharge tanks, etc.
Thin line =	Denotes conventional containment and route for “exempt” or “clearance” waste.
Red =	Waste route for “short-lived” ILW.
Blue =	Waste route for “short-lived” LLW.

The equipment includes both conventional mechanical size reduction equipment and more advanced techniques. Conventional equipment includes hand held power tools e.g.: hack saw, fret saw, band saw, bow saw, circular saw; shears; pipe cutters; diamond wire cutting rig; balers; and compactors.

The selection of the appropriate equipment will be largely driven by the nature of the object that is to be size reduced, although for some pieces of equipment such as compactors throughput economics will also be relevant. Many of the above techniques, particularly the saws, shears and pipe cutters, have the potential to be operated both manually and remotely.

(b) Decontamination

Following sorting and size reduction, there is the potential to decontaminate LLW arising down to clearance levels, in order to further reduce the volume of LLW generated. Whether this is cost effective will need to be assessed in a similar way to that of potentially decontaminating ILW. Factors such as the practicalities of decontamination in an area that may be subject to significant airborne contamination and the consequent disposal of decontamination waste will need to be considered further.

(c) Packaging and handling of waste

Dry waste will be transferred to the receipt area, see Fig. 20. (e.g. “size reduction” and “blasting and cleaning” area) using a small transfer container.

Waste will be removed from the transfer container, then monitored to determine its next destination. ILW material will continue along the ILW processing line, LLW and waste for potential clearance will be transferred to an adjacent size reduction area.

In some instances it may not be productive, or even necessary, to break large items of plant or equipment into components small enough to fit within a standard disposal container. Large items may therefore be contained within bespoke packaging suitable for final disposal of the category of waste concerned.

The long-lived waste mainly consists of the internals close to the core and is assumed to be transported and stored in 0.1 m thick steel containers with the outer dimensions 3.30 x 1.30 x 2.30 m. The inner volume is approximately 7.2 m^3 and the maximum weight, including 12 tonnes of waste, is 34 tonnes. The long-lived waste assumed to be deposited at the final repository for long-lived low and intermediate level waste.

LLW will be loaded into ISO-type containers. Loading will be carried out by primarily manually controlled techniques, using local lifting gear as appropriate. The dose rates could not exceed 2 mSv/h according to [10]. For the ILW that requires shielding a set of large transport containers have been developed. Each container carries loads of up to 20 tonnes and can take up to approximately 25 m^3 of waste. The container has thick shielding walls and can thus take waste packages with a high dose rate. The strongest container used at present can take packages with a dose rate of about 60 mSv/h, under the regulation of special arrangement it could take packages with dose rates up to 100 mSv/h. By using these shielding containers the waste can be concentrated and the volumes can be lowered. Received and outgoing goods are mainly transported by Lorries. Transport of radioactive waste for final disposal is carried out by sea. Lorries also do transport of other type of waste. Transport routes during decommissioning are planned to be as same as the ones used during operation.

Internal scanning and handling of the dismantled waste can be described as follows:

- The shaft for the main steam pipes, through the steam pipes wall entrance, are planned to use as transport way between the reactor and the turbine building.
- Large items of equipment will take a different route.
- Storage pools in the reactor hall are filled with water until dismantling of internals from the primary circuits has been finish.
- Wet-well (blow off-system pool) is planned as buffer storage for pipes from the containment.
- Equipment for dismantling and preparing for further handling and scanning are planned to take place in rooms were the high-pressure pre-heaters have been located, and rooms where the low-pressure pre-heaters have been placed are planned for buffer or intermediate storage.
- The turbine hall and condenser area are used for decontamination.
- Temporary buildings, directly connected to the turbine building, are planned to be used as buffer stores, scanning and packaging of intermediate and low-level waste and cleared material.

The waste processing facility will therefore need to have the capability to receive; segregate and process all decommissioning waste, from ILW and LLW to waste that is potentially acceptable for clearance. Once processed, LLW will be loaded into approved long-term disposal packages and dispatched to the final storage or disposal. ILW will be send to the intermediate decommissioning waste repository, while cleared waste will be routed through normal commercial channels.

(d) Clearance of material

Material, which is considered suitable for clearance is less than the clearance values described in Section 2 of this report [4, 6]. Material potential for clearance will therefore be monitored prior to packaging at a dedicated facility on the NPP site. The aim is to efficiently monitor the materials produced during decommissioning that are expected to be suitable for unrestricted release. This facility is equipped with appropriate automated scanning/monitor equipment and is located in an area of low background radiation.

Any material not complying with the clearance values will be isolated and transferred to the LLW size reduction area for further processing, storage or disposal.

Cleared material will be loaded into standard ISO-type freight containers for off-site transport. Loading will be carried out by manually controlled techniques, using local lifting gear as appropriate. These packages will be routed to an appropriate recycling or disposal facility.

3.6.2. Systems 321 and 322

The estimated mass of pipes and ducts and other structures for the systems 321 and 322, are presented in Fig. 21 below.

The expected waste categories from the decommissioning of systems 321 and 322 will be LLW and material with potential for clearance from regulatory control. The following approach to waste management generated during the decommissioning of systems 321 and 322 is planned to be implemented:

- All waste will undergo appropriate size reduction at the place of their generation in order to facilitate loading into a transfer container;
- Categories of waste will be initially determined at source (at systems 321 and 322 rooms) and will be confirmed during additional surveys;
- Some processing of mixed waste may be required at the waste processing facility at the NPP site;
- The material for potential clearance and LLW will be handled in standard ISO-type freight containers;
- Transfer of waste item to the waste processing and packaging area and then to the buffer storage on site; and
- Off-site transport of packaged decommissioning waste from the NPP site is out of the scope of this safety assessment.

	Removal of pipe 6.35 to 50.8 mm dia, meters	Removal of pipe >50.8 to 101.6 mm dia, meters	Removal of pipe >101.6 to 203.2 mm dia, meters	Removal of pipe >203.2 to 355.6 mm dia, meters	Removal of valves >101.6 to 203.2 mm dia,each	Removal of valves >203.2 to 355.6 mm dia,each	Removal of valves >203.2 to 355.6 mm dia,each	Removal of pipe fittings >101.6 to 203.2 mm dia,each	Removal of pipe fittings >203.2 to 355.6 mm dia,each	Pipe hangers for small bore piping, each	Pipe hangers for large bore piping, each	Removal of pumps, <135.9 kg, each	Removal of pumps, 453.1 - 4,531 kg, each	Removal of pump motors, 453.1 - 4,531 kg, each
321-Residual Heat Removal	105	-	129	157	8	24	2	5	13	3	51	-	2	-
322-Containment Spray-Wet Well Cooling	411	128	546	79	7	30	5	-	-	73	311	3	3	3

	Burial Volumes		Burial /
	LLW (m3)	ILW (m3)	Processed Wt., kg.
321-Residual Heat Removal	46	-	56 506
322-Containment Spray-Wet Well Cooling	95	-	116 530

FIG. 21. Overall estimate of pipe lengths, volumes, length and weights for dismantled system 321 and 322 (the upper values refers to length in meters).

A dedicated facility for management of all types of decommissioning waste, volume and rate of waste arising will be available during the decommissioning of systems 321 and 322. It will be a suitably sized at the NPP with:

- Good connections to the various work faces that will be producing radioactive waste;
- Sufficient space to allow the various processes of additional size reduction, and packing to be laid out efficiently;
- An active extract system, suitably rated to the flow of waste;
- Easy access to the outside for dispatch of loaded waste containers; and
- A suitably rated overhead crane, rated to shape and weight of the waste packages.

If adopting the fully engineered philosophy, there will be some segregation of wastes at the decommissioning workplace.

3.7. SUPPORTING FACILITIES

Supporting facilities are identified in the detailed engineering schedule presented in Section 6 and Section 7. In summary the planned supporting facilities and associated activities are the following:

- A temporary tent is planned to be constructed for the purposes of the decommissioning of systems 321 and 322. The accident conditions safety assessment in Appendix III shows that standards for the construction and testing of this temporary containment are important;
- Systems for personal dose meters, for their testing and control of issue;
- Building ventilation systems and associated filtration systems;
- Installed radiometric systems; and
- Systems for control of isolations.

3.8. END-POINTS

The end-point for the decommissioning of the two selected systems is that they are removed from the NPP building with the waste material taken beyond the boundary of this assessment to the waste management facility. Rooms where they were located will have been decontaminated as necessary and all equipment used for decommissioning will have been removed.

The end state of the facility after decommissioning will represent the site and the associated buildings passed out of regulatory oversight. The site owner then has the option to reuse the buildings or to demolish them.

4. HAZARD ANALYSIS: IDENTIFICATION AND SCREENING.

4.1. HAZARD IDENTIFICATION

Working techniques and methods were reviewed to identify potential hazards and initiating events that could arise during the decommissioning of systems 321 and 322. This was performed by a combination of:

- The checklist from Appendix VIII of Safety Assessment Methodology for Decommissioning of Facilities Using Radioactive Material (main report) and Appendix I [11]; and
- The HAZOP process.

The hazard identification and hazard assessment process was conducted by suitably qualified and experienced persons from a wide range of disciplines including plant operators and engineers, radiological protection specialists, safety engineers, human factors and criticality specialists and assisted by decommissioning workers. These hazards and initiating events were then screened to identify appropriate scenarios for further detailed analysis.

4.2. APPROACHES TO HAZARD IDENTIFICATION

A structured and systematic approach was adopted in order to identify a complete list of all reasonably foreseeable hazards, initiating events and scenarios that could lead to harm to the public, workers or the environment during the decommissioning. The process, which was undertaken iteratively with the detailed hazard analysis, included the following:

- A review of the operational history of the facility (see Section 3.4) including a detailed review of all significant events during the facility lifetime with interviews were conducted with operations personnel with specific facility history;

- The basis of the radiological inventory was the one measured prior to decontamination.
- Human factors/ergonomic walk down were performed;
- A checklist was used for preliminary identification of initiating events and hazards. Industrial hazards with no radiological consequences were not considered further in this safety assessment as prescribed by the methodology [11]. This forms the basis of the hazards presented in Section 4.3.1;
- Identification of event sequences was carried out primarily using HAZOP studies (both desktop and plant walk down). In addition, a “what-if” evaluation of potential failure modes, including initiating events such as power loss, fire and operator error, was applied;
- Construction of a hazard/event schedule for internal and external initiating event scenarios was based on the HAZOP. The identified hazards and initiating events were grouped logically to minimize the number of scenarios to be analysed. This forms the basis of the hazards presented in Section 4.3.2; and
- Input from the hazard analysis (including engineering assessment) to ensure that any new initiating events identified during the more detailed consideration of the scenarios and variations of the main scenarios (e.g. failure of protective measures) were taken into account.

4.3. PRELIMINARY HAZARD ASSESSMENT AND SCREENING

4.3.1. Use of a checklist

Preliminary hazards assessment as described above was applied to the planned decommissioning of the two systems 321 and 322. The results of this analysis are summarized in Table 16.

TABLE 16. HAZARDS FOR PLANNED AND FOR ACCIDENT CONDITIONS DURING DECOMMISSIONING OF SYSTEMS 321 AND 322

Hazards and Initiating Events	**Relevant for Planned Work**		**Relevant for Accidents**	
	System 321	System 322	System 321	System 322
INTERNAL INITIATING EVENTS				
Radiological Initiating Events				
Criticality [no fissile material present]	N	N	N	N
Spread of Contamination [system 322 is not significantly contaminated]		N		N
Loss of containment/barriers	Y		Y	
Dismantling of containment/barriers	Y		N	
Drop of radioactive materials, packages and waste	N		Y	
Cleanup of buildings (activated or contaminated) [cleanup is beyond the scope of this assessment]	N		N	
External exposure				
Activated materials and equipment	Y	N	Y	N
Direct radiation sources	Y	Y	Y	Y
Internal exposure		N		N
Physical and chemical state of the radioactive materials [Very small quantities of Pu are present]	Y		Y	
Contamination, corrosion, etc.		N		N
Pathways (inhalation, ingestion) [Inhalation relevant to system 321]	Y		Y	

Hazards and Initiating Events	Relevant for Planned Work		Relevant for Accidents	
	System 321	System 322	System 321	System 322
Spectrum, activity, emitters (presence of alpha emitters)	N		N	
Contaminated materials	Y		Y	
Gaseous Effluent	N		N	
Liquid Effluent [No wet cutting techniques are being considered]	N		N	
Non Radiological Initiating Events				
Fire				
Thermal cutting techniques (Zircalloy, etc.)	Y	Y	Y	Y
Decontamination process (chemical, mechanical, electrical methods or mixed methods to remove contamination from metals, concrete or others surface).	N	N	N	N
Accumulation of combustible materials and radioactive waste	Y	Y	Y	Y
Flammable gases and liquids [depends on the cutting solution finally selected]	Y	Y	Y	Y
Explosion				
Decontamination process.	N	N	N	N
Dust (graphite, Zircalloy, etc.) [not considered credible]	N	N	N	N
Radiolysis phenomena (radioactive waste storage, transport) [dose rates too low]	N	N	N	N
Compressed gases [depends on the cutting solution finally selected]	Y	Y	Y	Y
Explosive substances [depends on the cutting solution finally selected)	Y	Y	Y	Y
Flooding [no liquids present in room]	N	N	N	N
Leak of liquid storage				
Leak of pipes				
Toxic and hazardous materials				
Asbestos/glass wool in thermal insulation system [asbestos and other insulation has been removed. However, residual amounts could potentially remain. This needs to be taken into account during the control of normal operations.]	Y	Y	N	N
Lead in paint, shielding	N	N	N	N
Beryllium and other hazardous materials	N	N	N	N
Polychlorinated biphenyls (PCBs)	N	N	N	N
Oils	N	N	N	N
Pesticide use	N	N	N	N
Biohazards	N	N	N	N
Electrical hazards				
Loss of power supply [this will result in loss of ventilation and hence, potentially, a small dose]	Y	N	Y	N
High voltage	N	N	N	N
Non-ionizing radiation sources (lasers, …)	N	N	N	N

Hazards and Initiating Events	**Relevant for Planned Work**		**Relevant for Accidents**	
	System 321	System 322	System 321	System 322
Falling of heavy loads	Y	Y	Y	Y
Falling loads on SSCs important for safety	N	N	N	N
Falling loads on radioactive materials (packages)	Y	Y	Y	Y
Collapse of structure (due to ageing) [not credible]	N	N	N	N
Demolition activities [no demolition will take place during the decommissioning of 321 and 322]	N	N	N	N
Working at heights	Y	Y	Y	N
High noise area [this could be relevant in accident scenarios, but will not initiate any radiological accidents]	Y	Y	Y	Y
Excavations	N	N	N	N
Vehicle traffic	N	N	N	N
Pinch points, sharp objects	N	N	N	N
Obstruction of passageways or exits	Y	Y	Y	Y
Physical hazards				
Kinetic energy [no rotations, no movement]	N	N	N	N
Potential energy (springs, Wigner energy in graphite) [no springs, no graphite, pipework components not fixed in-situ]	N	N	N	N
Degraded or degrading structures, systems and components [systems are in a good state of repair]	N	N	N	N
Steam [no systems in operation needing steam]	N	N	N	N
Temperature extremes (high temperatures, hot surfaces, cryogenics)	N	N	N	N
High pressure (pressurized systems, compressed air)	N	N	N	N
Human and organizational initiating events				
Operator error/violation	Y	Y	Y	Y
Inadvertent entry into high-radiation areas	Y	Y	Y	Y
Misidentifications	Y	Y	Y	Y
Contractor and sub-contractor	Y	Y	Y	Y
Performing incompatible activities [no parallel activities are planned to be undertaken during the decommissioning]	N	N	N	N
Disabling services to other facilities [although there are electrical power cables in the room, these are remote from systems 321 and 322 and are clearly identifiable]	N	N	N	N
Poor ergonomic conditions	N	N	N	N
EXTERNAL INITIATING EVENTS **Note –** these factors are not relevant to the specific scope of this test case.				
Earthquake [these and the other external initiating events listed	N	N	N	N

Hazards and Initiating Events	Relevant for Planned Work		Relevant for Accidents	
	System 321	System 322	System 321	System 322
here are addressed in the baseline safety analysts' report.]				
External flooding [See above comment]	N	N	N	N
River				
Sea				
Infiltration of groundwater				
External fire (oil storage, etc.)	N	N	N	N
Extreme weather conditions (temperature, wind, snow, etc.) [See above comment]	N	N	N	N
Subsidence (formation of underground cavities (subsidence) from rain, waste degradation etc.) (This potential fault was identified by the NPP Test Case review team and is not found in the methodology)	N	N	N	N
Hazards due to industrial environment (explosion, etc.)	N	N	N	N
Airplane crash [See above comment]	N	N	N	N
OTHER INITIATING EVENTS				
High temperature and pressure	N	N	N	N
Corroded barriers	N	N	N	N
Unknown or unmarked materials	N	N	N	N

Note:

Comments found in brackets ( ) come from the table taken from Ref. [11].
Comments found in brackets [ ] have been added after the NPP Test Case review.

4.3.2. Separate identification of possible initiating events

Initiating events and hazards considered relevant for the planned decommissioning activities were used as an input to the detailed task analysis described in Section 5.

Initiating events and hazards considered relevant for accidents were further screened to identify those scenarios where further analysis was warranted.

- *Spread of contamination due to loss of containment/barriers (for system 321 only)*

Cutting of the system 321 pipework has the potential to result in a significant dose to the workers, and if filters fail, to the public. This hazard therefore requires detailed analysis (Sequence A1). It also requires an operating assumption that the systems 321 and 322 are proved to be drained and dry before decommissioning tasks start.

- *Spread of contamination due to drop of radioactive materials, packages and waste (for system 321 only)*

It is credible that pieces of cut contaminated pipework could be dropped, either as these are manoeuvred away from the workface, or while being transported from the room in containers. Such initiating events can be controlled through standard work place systems and procedures for normal handling. The anticipated size of the pieces of cut pipework/components means that detailed analysis of these initiating events is not warranted (Sequence B1).

- *External exposure due to activated materials and equipment (for system 321 only)*

Doses in credible accident sequences will be dominated by the surface contamination and will be largely unaffected by the fact that the components are also activated. In particular, no special precautions will need to be taken as a result of the activated nature of these components in any foreseeable accident scenario (though precautions will need to be taken in regard to normal operations). The presence of trace amounts of Pu does not give rise to neutron dose rates of any significance.

- *External exposure due to direct radiation sources (for systems 321 and 322)*

The decommissioning of systems 321 and 322 will be undertaken in a gamma radiation environment. Normal operational controls will therefore need to minimize doses (e.g. through limiting durations, provision of shielding, etc). Exposure to direct radiation in accident scenarios will arise if the scenario leads to increased proximity or elongated time in the radioactive environment. The relatively low dose rates, and credible evacuation times in these scenarios mean that detailed analysis in these cases is not warranted. Moreover, no special provisions are considered to be required to minimize the likelihood of initiating events beyond those in place for conventional safety.

- *Internal exposure due to physical and chemical states of the radioactive materials (for system 321 only)*

Very small quantities of Pu are present. This will need to be taken into account in the detailed hazard analysis (e.g. Sequence A1). However, the presence of Pu does not lead to any additional accident scenarios.

- *Contamination, corrosion, etc. – exposure pathways (inhalation, ingestion) (for system 321 only)* same as for previous entry.

- *Contamination, corrosion, etc. due to contaminated materials (for system 321 only)*

Addressed by Sequence A1 (see above).

- *Fire as a result of the use of thermal cutting techniques (Zircalloy, etc.) (for systems 321 and 322)*

The use of hot cutting techniques means that workplace fires are a credible scenario. Such accidents may be controlled/mitigated through good practice workplace systems and procedures (e.g. minimization of combustible material at the workface). Detailed analysis of these sequences would add little value (e.g. this would be unlikely to lead to any change in the safety measures provided) (Sequence B2).

- *Fire due to accumulation of combustible materials and radioactive waste (for systems 321 and 322)*

Same as for previous entry (Sequence B2).

- *Fire due to flammable gases and liquids (for systems 321 and 322)*

Same as for previous entry (Sequence B2).

- *Explosion as a result of compressed gases (for systems 321 and 322)*

Same as for previous entry, Sequence B2 also needs to address explosions.

- *Explosion due to explosive substances (for systems 321 and 322).*

Same as for previous entry (Sequence B2).

- *Electrical hazards due to loss of power supply (for system 321 only)*

Loss of power supplies will result in a loss of ventilation and hence, potentially, a small dose (since cutting operations will also (necessarily) have stopped). Good practice will mean that the operators will evacuate following the failure of the ventilation. Hence credible doses from this sequence will be small enough not to warrant additional detailed analysis (Sequence B3).

- *Decommissioning/workplace initiating events as a result of falling of heavy loads (for systems 321 and 322).*

Dropping of loads during the decommissioning of system 321 has been addressed above. Additionally, dropping of a piece of system 322 could conceivably lead to the levitation of contaminated material present below and hence to a dose to the operators. The consequences of such an accident for system 322 will be bounded by those for system 321. However, the scope of Sequence B1 needs to be expanded to include system 322.

- *Decommissioning/workplace initiating events – falling loads on radioactive materials (packages) (systems 321 and 322)*

Same as for previous entry (Sequence B1).

- *Decommissioning/workplace initiating events due to working at heights (for system 321 only)*

Same as for previous entry (Sequence B1).

- *Decommissioning/workplace initiating events due to high noise area (for systems 321 and 322)*

Although high noise is not a radiological initiating event, this aspect needs to be taken into account in the safety measures put in place (e.g. through the provision of audio-visual alarms). Accident scenarios that include operators failing to respond to audible alarms may be addressed by following good practice standards for working in high noise environments. Detailed analysis of these scenarios is not considered warranted (Sequence B4).

- *Decommissioning/workplace initiating events due to obstruction of passageways or exits (for systems 321 and 322).*

Although obstructed passageways etc. will not lead to a radiological initiating event, an inability to evacuate following an initiating event represents a scenario that need to be considered in the safety assessment. Such scenarios are however addressed by following good practice standards in regard to fire safety. Detailed analysis of these scenarios is not considered warranted (Sequence B5).

- *Human and organizational initiating events due to operator error/violation (for systems 321 and 322).*

Application of good ergonomic design of equipment and tasks, training and human factors review will be applied to minimize initiating events arising from operator errors and violations and to mitigate their consequences. Detailed analysis of scenarios of this type is not considered to be warranted. Similarly the variety of scenarios involving operator error/violation means that identifying individual scenarios would be of only limited value.

- *Human and organizational initiating events due to inadvertent entry into high-radiation areas (for systems 321 and 322).*

The location of systems 321 and 322 means that operators could inadvertently enter high radiation areas on their way to/from the workface. However, application of good practice signage, and access controls will be used to minimize the likelihood of such events. Detailed analysis of scenarios of this type is not considered to be warranted (Sequence B6).

- *Human and organizational initiating events due to misidentifications (for systems 321 and 322)*

Misidentification of plant and equipment is a form of operator error (see above).

- *Human and organizational initiating events due to use of contractor(s) and sub-contractor(s) (for systems 321 and 322).*

Application of appropriate training and work control need to ensure that tasks undertaken by contractors are carried out as reliably and to a similar standard as tasks undertaken by employees. Detailed analysis of scenarios of this type is not considered to be warranted. Similarly the variety of scenarios involved means that identifying individual scenarios would be of only limited value.

4.4. OUTCOME OF PRELIMINARY HAZARD ANALYSIS AND SCREENING

On the basis of the preliminary hazard analysis (see Section 4.3) the following accident sequences were identified for further detailed analysis:

Sequence	System(s) Affected	Hazard Type	Details
A1	321	Worker and public exposure due to inhalation	Failure to control the spread of contamination arising from the cutting of pipework and components

The following accident sequences was considered for detailed analysis and those that do not require detailed analysis (Table 17) but will need prevention, protection and/or mitigating measures derived from standard good practice workplace systems and procedures for normal operations:

TABLE 17. ACCIDENT SEQUENCES CONSIDERED IN THE HAZARD ANALYSIS

Sequence	System(s) Affected	Hazard Type	Details
B1	321 and 322	Worker inhalation	Dropping pieces of cut contaminated pipework, either as these are manoeuvred away from the workface, or while being transported from the room in containers
B2	321 and 322	Public and worker inhalation	Fire/explosion arising from hot cutting
B3	321	Worker inhalation	Power supply failure leading to a failure of the ventilation system
B4	321 and 322	Worker inhalation	Operators fail to respond to alarms due to high noise environment.
B5	321 and 322	Worker inhalation	Operators are unable to evacuate following an initiating event due to blocked emergency exits.
B6	321 and 322	Worker inhalation and direct radiation	Operators inadvertently enter areas of high radiation en-route to/from the workface

Nevertheless, depending on the national legal and regulatory framework in a Member State, the operator may consider appropriate or necessary to perform further detailed analysis of the sequences presented in Table 17.

5. HAZARD ANLYSIS EVALUATION

5.1. ANALYSIS OF NORMAL DECOMMISSIONING CONDITIONS

The scope of the normal scenario is the evaluation of safety of normal planned decommissioning activities for system 321 and 322, as presented in Section 3.5 and Tables 14. and 15. For evaluation of normal decommissioning activities (as planned), two approaches were applied:

(a) Evaluation of doses to workers resulting from normal decommissioning activities for the systems 321 and 322 in their full planned extent; and
(b) Evaluation of doses to public based on conservative approach taking into account the worst case for dismantling techniques from the point of view of all identifiable routes of exposure from the decommissioning activities in their full planned extent.

For the dismantling of items in systems 321 and 322 following techniques were considered:

— Hydraulic shears cutting for cutting of pipes with small dimensions, electrical cables, components of ventilation ducts and other equipments with thin walls;
— Plasma cutting for general equipments, preferably for stainless steel equipments;
— Mechanical cutting by mechanical saw or other mechanical cutting method which does not generate much heat, used in applications where lowered generation of aerosols is required, or for cutting for equipments with large wall thickness like reactor vessels;
— Oxygen - acetylene cutting of general equipment, preferably for those made for carbon steel;
— Manual dismantling using standard mechanical hand tools;
— Grinding for cutting of equipments with medium wall thickness. The technique has relatively high cutting rate, but the release factors for radionuclides is high.

Allocation of techniques to selected dismantling categories, as applied in the computer code OMEGA, is presented in Table 18. The table shows the default techniques (green colour) as selected by the code during generation of the calculation structure and alternative techniques, which can select the user.

TABLE 18. ALLOCATION OF TECHNIQUES TO SELECTED DISMANTLING CATEGORIES

Dismantling category	HDCT	COBO	PLSM	OCHC	MSW	OACT	PLHC	MNOC	MAND	MAPL	GROC	GRPL
Piping (SS), diameter =< D25 mm												
Piping (SS), diameter over 25 mm												
Piping (CS), diameter =< D25 mm												
Piping (CS), diameter over 25 mm												
Tanks (SS)												
Tanks and containers (CS)												
Heat exchangers (SS),												
Heat exchangers (CS) ,												
Pumps (SS, CS), mass <= 50 kg												
Pumps (SS), mass over 50 kg												
Pumps (CS), mass > 50 kg,												
Ventilators (SS, CS), mass <= 50 kg												
Ventilators (SS), mass > 50 kg,												
Ventilators (CS), mass > 50 kg,												
Valves (SS)												
Valves (CS)												
Electric motors, mass <= 50 kg												
Electric motors, mass > 50 kg												
Air conditioning components - piping (SS)												
Air conditioning systems others (SS)												
Air conditioning components - piping (CS),												
Air conditioning systems others (CS),												
Air conditioning systems, (Al)												
Electrical cables & conductors												
General electric equipment, (CS) mass <= 50 kg												
General electric equipment, (CS) mass > 50 kg												
Thermal insulations, non-metal covering												
Steel constructions, (CS)												
Small piece components, shielding (CS)												
Hoisting equipment (CS), electrical tackles												
Digestors, sampling boxes (CS)												
Piping feedthroughs, gulleys												
Hermetic and shielding doors (CS)												
Stainless steel linings, (SS)												
Carbon steel linings, (CS)												
Other general equipment												
Casing of technological equipment (CS),												
Casing of technological equipment (SS),												

Note:

HDCT	Hydraulic shears cutting
COBO	Core boring
PLSM	Plasma cutting
OCHC	Oxygen cutting - hydraulic cutting (combined technique)
MSAW	Mechanical cutting by saw
OACT	Oxygen cutting (oxygen - acetylene cutting)
PLHC	Plasma cutting - hydraulic cutting (combined technique)
MNOC	Manual dismantling - oxygen cutting (combined technique)
MAND	Manual dismantling (by tools)
MAPL	Manual dismantling - plasma cutting (combined technique)
GROC	Grinding - oxygen cutting (combined technique)
GRPL	Grinding - plasma cutting (combined technique)

Grinding is the alternative technique and was not applied in calculation of parameters for systems 321, 322. The release factors for aerosols for dismantling techniques are identified in the Table 17 by colour. The values are as follows:

PLSM	Release factor – 10%
OCHC	Release factor – 1%
HDCT	Release factor – 0.1%

5.1.1. Doses to workers

Other activities evaluated, such as the decontamination and radiation monitoring of building surfaces, are evaluated from the point of view of annual limit of 20 mSv for workers. During dismantling, the workers are occurring in the dose rate fields of the contaminated equipment to be dismantled and are exposed to risk of inhalation of radioactive aerosols generated during cutting. Technical, organizational and personnel protection means are used to minimize the exposure of workers.

Tables 15.and 16 give the detailed list of dismantling activities as proposed for the systems 321 and 322. The activities are organized according to the room-oriented approach (dismantling of equipment is organized room by room, see Appendix III) which involves:

(a) A set of preparation of works prior dismantling to prepare the working conditions and to support the dismantling in the room where equipment is going to be dismantled;

(b) The dismantling of the two systems and equipment according to the inventory content in the room;

(c) A set of finishing activities to remove all instruments, materials, supporting systems and for cleaning/decontaminating the room after dismantling; and

(d) All support activities for preparation for decommissioning tasks and for decontamination and radiological surveys to verify the condition of the rooms at the end of the decommissioning tasks.

As it is presented in Section 3, for the NPP Test Case it has been conservatively considered that the dismantling of each system will be performed by the same working group. The basic criterion for evaluation of safety of the dismantling activities is the annual limit 20 mSv for individual workers. The evaluation methodology applied is described in Section 5.3.2 and it is the evaluation of the individual effective dose to individuals in each decommissioning activity. This detailed calculation approach will ensure that already in planning phase it is possible to foresee the need of optimization of critical decommissioning activities from the safety point of view. The optimization tools like involving more personnel, application of remote controlled techniques or application of pre-dismantling decontamination can be then involved into planning in order to meet the 20 mSv criterion.

5.1.2. Doses to public

Public dose was calculated by the computer code DecDose [12] under the conditions of the decommissioning of system 321 as the most contaminated one. The additional inventory in system 322 could be seen not to add any significant contribution to the potential exposure of public. Because dismantling activities using water or liquid are not carried out, therefore, exposure doses to the public are potentially caused by atmospheric discharges of radionuclides and by radiation from radioactive wastes temporarily stored in the building. Primary assumptions to assess the normal public dose are as follows:

— Segmented pieces are put promptly into an ISO container. From the handling point of view, those pieces must be shorter than 1.5 m;
— Plasma arc cutting technique is applied to all components;
— No HEPA filter is installed on the contamination control enclosure (tent), while it is installed at the building ventilation system before stack;
— The NPP ceased power operation in April 2002; and
— Dismantling activity of system 321 starts on 1 July 2005 and ends on 31 March 2006; and

— The isotope inventory and characterization are as shown in Table 12.

5.2. ANALYSIS OF ACCIDENT SCENARIOS

This radiological accident analysis forms part of the safety assessment. Reference [11] gives the methodologies to be used for safety assessment of decommissioning of facilities using radioactive material.

Section 3 also defines a critical group for assessment of exposure from the normal consequences of decommissioning operations. For this accident analysis, the consequences identified are of a nature such that there is no need to redefine the critical group for the consequences of accident conditions. The same critical group as described in Section 3 thus applies to both analyses.

It is assumed that a baseline safety case for care and maintenance of the NPP already exists, so that this radiological accident analysis only needs to address new issues arising from the decommissioning operations. It is assumed that systems 321 and 322 are isolated from water feeds. In particular, system 321 is isolated from the primary circuit by blanks inserted between the twin isolation valves (numbers V1/V2 and V33/V34, supply and between V10/V32 and V9/V31 on the return side), as shown in Fig. 11. This then removes any interactions between the decommissioning of system 321 and the rest of the nuclear plant.

This radiological accident analysis considers the radiological effects of *potential* accidents that may occur during decommissioning of the shut down cooling and clean up system 321 and the containment spray system 322.

To summarize the scope:

— The operations involve cutting up Systems 321 and 322, transporting them to a size reduction facility, size reduction, placement in drums, lidding of drums and export to a waste management facility on site. Thus use of decontamination liquids is excluded from the scope of this analysis.

— This radiological accident analysis does not cover any expected doses from normal decommissioning operations. It does not cover chemotoxic hazards on-site or off-site, or industrial hazards. The impact of external hazards on the NPP has been judged not to be significantly affected by the operations included within the scope.

On the basis of the preliminary hazard analysis (see Section 4, see Table 19.) the following accident scenarios have been analysed:

(a) High external dose to a worker (Scenario 01);
(b) Accidents during cutting operations (Scenarion 02); and
(c) Dropped loads (Scenarion 03).

TABLE 19. SOURCE OF IDENTIFICATION OF HAZARDS COVERED BY RADIOLOGICAL ACCIDENT ANALYSIS FOR THE NPP TEST CASE

Sequence from Section 4 of the Safety Assessment	Scenario Description	Scenario Reference
Sequence A1	Failure to control the spread of contamination arising from the cutting of pipework and components	02 Accidents during cutting operations
Sequence B1	Dropping pieces of cut contaminated pipework, either as these are manoeuvred away from the workface, or while being transported from the room in containers	03 Dropped loads
Sequence B2	Fire/explosion arising from hot cutting	Detailed analysis not required (see pre-amble to table)
Sequence B3	Power supply failure leading to a failure of the ventilation system	02 Accidents during cutting operations
Sequence B4	Operators fail to respond to alarms due to high noise environment.	02 Accidents during cutting operations (specifically a performance requirement within Table 42)
Sequence B5	Operators are unable to evacuate following an initiating event due to blocked emergency exits.	Detailed analysis not required (see pre-amble to table)
Sequence B6	Operators inadvertently enter areas of high radiation en-route to/from the workface	01 High external dose to worker

The basic approach adopted (and presented in Appendix III) for each accident scenario is a graded approach based on the consequences of accident scenarios without any mitigation, as follows:

— Assessment of the unmitigated consequences of the accident scenarios;
— Comparison of the number of independent and complete safety measures with the criteria (see Section 2, Table 1);
— Identification of those safety controls that make up the required independent and complete safety measures; and
— Consideration on whether the risk is As Low As Reasonably Achievable (ALARA).

Appendix III contains a detailed analysis of these scenarios, i.e.:

— Initiating event;
— Description of the potential consequences;
— Defence-in-depth;
— Safety measures/safety controls; and
— Conclusion on risk.

Even for this most significant scenario, the risk is not considered to be high because three independent low likelihood accidents would have to occur. As a result, no numerical frequency or probabilistic analysis has been carried out for any of the three scenarios analysed. The engineering analysis (and any other section of the safety assessment) has not required any allocation to frequency bands, so this has not been necessary, thus simplifying the radiological accident analysis.

Section 7 summarizes the key outputs from the whole radiological accident analysis, including:

— Procedural safety controls;
— Engineered safety controls (see also Section 6);
— Shortfalls and recommendations; and
— Outstanding issues.

All scenarios are judged to have an acceptable level of defence-in-depth and to present an acceptable risk that is ALARA, subject to resolution of shortfalls and outstanding issues.

Because of the conservative approach taken to this radiological accident analysis, no sensitivity analysis is carried out within this document, as there is a high level of confidence in the parameters used.

5.3. MODELLING AND CALCULATION OF CONSEQUENCES

5.3.1. General input data

(a) Contamination and radionuclide composition of the systems 321 and 322

The contamination levels of the inner surfaces of the main components of the system 321 as individual radionuclides are presented in the Table 12. The reference date is 1 July 2005. These data were conservatively used for all components of the system 321 for evaluation of the dose to public and doubled values in evaluation of accident scenarios.

The decommissioning inventory database was developed for the systems 321 and 322, containing the physical data and radiological data for individual components of the systems, based on:

— The constructional data of the NPP

— The radiological data based on site measured dose rate data; and

— The results of calculation modelling.

The procedure for developing the database is presented in Section 5.3.2. The physical and radiological data in the database were used for evaluation of the doses to workers for normal decommissioning activities and the physical data were used for evaluation of doses to the public.

5.3.2. Doses to workers

Worker dose was calculated by using the computer code OMEGA [13] and addressing the full scope of the decommissioning of systems 321 and 322. This Section describes the application of this detailed analytical computer code for applications in the evaluation of dose to individual workers.

(a) Introduction

The computer code OMEGA was applied for calculation of dose to workers for normal decommissioning activities (as planned). The main features of the code are described in Appendix I and Ref. [13]. The methodology implemented in the code is based on calculation modelling of the decommissioning process including waste management as described in Ref. [14]. The code applies a bottom-up approach which is based on calculation and evaluation of data for each discrete decommissioning activity. This principle is recommended as the most accurate method for evaluating of decommissioning parameters as set out in Ref. [15]. It has the following features:

— The calculation structure implements the standardized cost items structure for decommissioning, issued by OECD/NEA, EC and IAEA in 1999 [16]. The calculated data are transparent, traceable and comparable with other decommissioning programmes and activities.
— The calculation process is sequentially linked-up in such a way that it simulates real decommissioning process flow and relevant material/radioactivity flow. The calculation items are linked to the material/radiological data of the inventory database and to the database of interim material/radiological items created during calculation, so the calculation uses data for material and radioactivity inventories that come from verifiable measurements.
— The calculation process is radionuclide specific and incorporates the radioactive decay of individual radionuclides. This allows the use of radionuclide resolved limits for treatment/conditioning/disposal/release of materials within the material flow and enables to study the effects of deferred decommissioning. The decommissioning infrastructure is simulated by various scenarios for management of radioactive waste. The scenarios include decommissioning activities linked from dismantling up to the disposal of conditioned radioactive waste or release of materials.
— The calculation structure of the code is standardized for all applications. The tool was developed for generating of the option specific work breakdown structure (WBS) by user defined grouping/linking of the items of the standardized calculation structure to the WBS items. The resulting WBS is transferred to the MS Project software for on-line optimization (tasks linking, critical path definition, period dependent activities adjustment, deferred decommissioning phases definition, etc.) and after optimization it is transferred back to the code for recalculation of decommissioning parameters according to the optimized WBS. The process can be repeated in an iterative way.

An additional aspect of this advanced methodology is the possibility to perform the sensitivity analysis which can reveal the margins of decommissioning parameters by considering various levels of contamination, various radionuclide composition (effects of alphas), application of various decommissioning technologies, various durations of deferred decommissioning phases, etc. This is achieved due to due to the internal linking of the calculation process and to the compactness of the calculation structure. The OMEGA code is able to estimate the following:

— Exposure of personnel;
— Duration of specific activities, phases and overall project duration;
— Waste data including the source terms for gaseous effluents and liquid discharges;
— Manpower;
— Costs; and
— Other planning data like number and professions of the personnel, requirements on materials, energy, technical media, equipment, etc.

The code was proposed for calculation of the planned normal decommissioning activities for the two systems 321 and 322 because it can calculate the dose received by the individual workers involved for the full extent of planned decommissioning activities related to decommissioning of these systems. The use of the code is facilitated by automatic generation of the standardized calculation structure and user friendly oriented modules. A more detailed description of the code and methods of its application are presented in Appendix I.

(b) Key assumptions

The inventory database

The key data set for application of the code OMEGA is the decommissioning inventory database with the structure of data prescribed for application of the code. The inventory database was developed for the systems 321 and 322 in the level of detail enabling the application of the bottom-up approach for generating the standardize calculation structure with discrete decommissioning activities for application of room oriented approach for systems 321 and 322. The procedure for development of the inventory database is presented in Appendix I.

Calculation of dose for internal exposure

The procedure for calculation of the internal dose presumes that the workers will use the protective means as allocated in the calculation procedure, see Section (c) below. The calculated internal data will be valid only under these assumptions. It will therefore be necessary that operational measures are put in place to ensure that that this assumption will met as decommissioning work is performed.

Duration of the process

The duration of the evaluated decommissioning activities needs to be analysed using the approach as implemented in the OMEGA code, i.e. the decommissioning time schedule in Microsoft Project software. The duration of the dismantling is used for comparison of calculated dose relevant for duration of dismantling, with the annual dose limit for individual.

Conservative approach in evaluating the dose to individuals

In order to evaluate the effective dose to individuals conservatively, the dismantling of the systems 321 and 322 was assumed to be performed by the same team of workers.

(c) Modelling approaches

Calculating of the external dose for professions of the working group

The dose uptake during performing the planned decommissioning activities is evaluated for each individual decommissioning activity and has following components:

- Dose uptake by the dose rate 0.5 m from the equipment to be dismantled;
- Dose uptake by the average dose rate in the room where the dismantling is performed; and
- Dose uptake caused by the average dose rate in the background in the controlled area.

The working time structure for dismantling involves the productive working time needed for dismantling of the equipment and the non-productive time components. The manpower needed for dismantling is normally calculated using the principle of categorization of equipment which is grouping of the types of equipment with similar physical properties into categories for which the decommissioning unit factors are defined. Calculation of the productive manpower for dismantling is based on the mass of the equipment and the manpower unit factors for individual categories of equipment. In this way, the time needed for calculation of the dose uptake during dismantling is obtained.

Various multiplying factors can be applied when calculating the productive manpower, to account for inefficiencies resulting from work in radioactive controlled areas, work on scaffolding, work in congested areas, work on complicated tasks, etc. These factors can be significant from the point of productive manpower calculation. This multiplying factor depends on the level of dose rate from dismantled equipment, and will have a graded character corresponding to levels of dose rate.

The manpower calculated for performing the dismantling is the basis for the calculation of the manpower for non-productive working time components which also contribute to the total exposure of the personnel, see Fig. 22. The dose uptake for non-productive time components are calculated taking into account the dose rate of the background of the controlled zone. The dose rate relevant for calculation of the dose uptake during performing the preparatory and finishing activities is the average dose rate in the room. In the case of finishing activities the dose rate in the room is multiplied by the conservative factor of 0.1 to allow for time spent in lower dose rates.

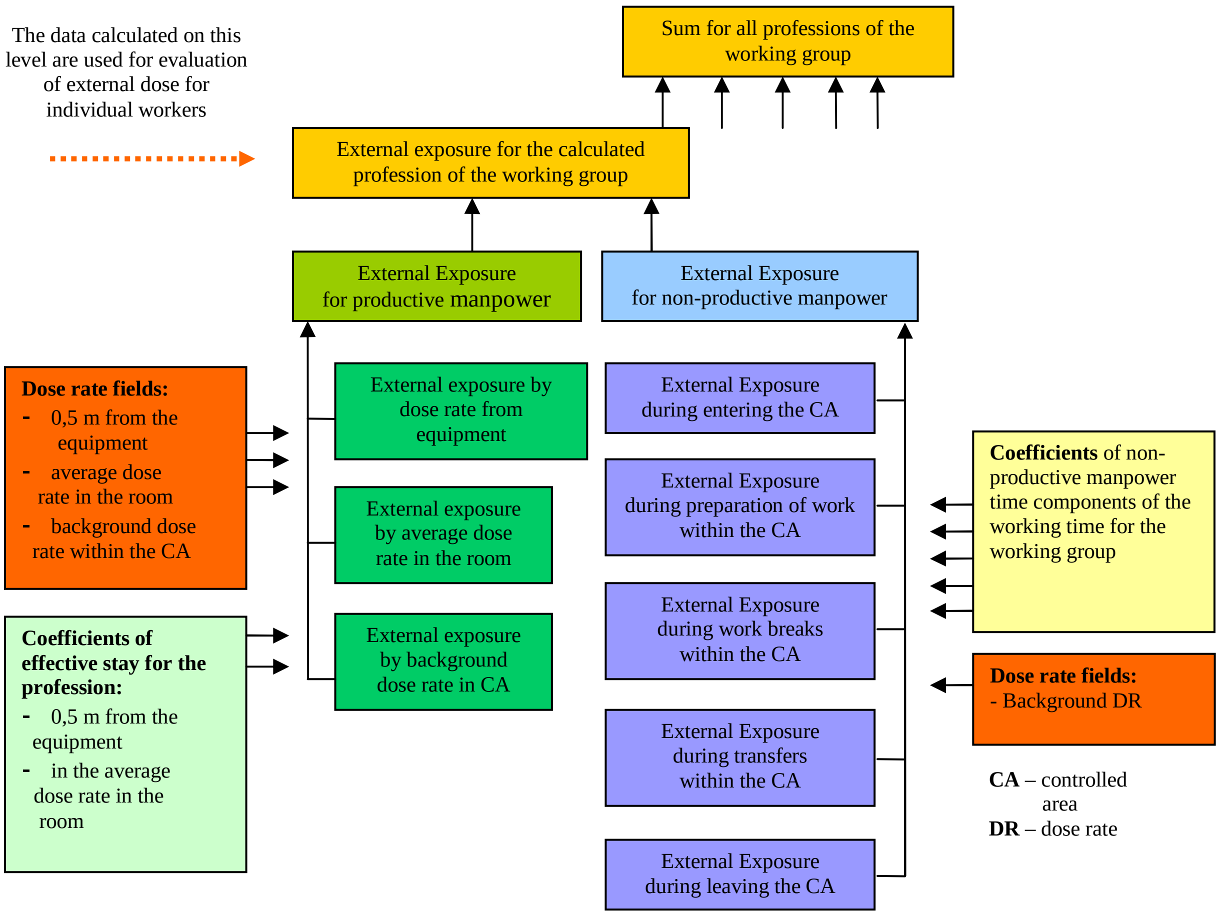

FIG. 22. Concept of calculation of the external dose uptake for individual dismantling activities.

The calculation of the dose uptake is organized subsequently to calculation of manpower components for individual dismantling activities according to the room oriented approach as described in Section 5.1.1. The calculated manpower items are the basis for calculation of the dose uptake. The manpower data are calculated according to the professions of the working groups which perform the preparatory, decommissioning and finishing activities.

The individual professions of workers are exposed in different way in accordance with the type of work they perform. The most exposed professions are those who directly perform the cutting and are most exposed to the dose rate of the dismantled equipment. For other professions the average dose in the room is dominant. For the rest of the working time, the dose rate in the background of the controlled zone is applied. These conditions are taken into account in the calculation of the dose uptake for individual professions of the working group and they are expressed by coefficients of effective stay in the working distance from the equipment and coefficients of effective stay in the average dose rate in the room.

The dose uptake is calculated for individual decommissioning activities as a sum of dose items for individual professions of the working group for individual productive and non-productive components of their working time. The calculation is performed for each preparatory and finishing activity according to the rooms involved and for each inventory item in the rooms as introduced into the inventory database. The principle of calculation of dose uptake for dismantling activities is schematically presented on the Fig. 22.

Normally, the calculation of the dose uptake in relation to the average dose rate in the room is performed conservatively. It means that the dose rate in the room is applied for dismantling of all items in the room. A methodology was developed in the OMEGA code to calculate the dose items related to the dose rate in the room more realistically, by taking into account the subsequent decrease of the average dose rate in the room during dismantling. This methodology of optimized calculation of the dose uptake was applied in the NPP Test Case.

Calculation of the internal dose for different professions of workers

A similar approach, based on calculated productive and non-productive manpower components, is applied also in calculation of the dose from internal exposure. Other data needed for calculation are the breathing data, conversion factors for individual radionuclides [Sv/Bq] and retention factors of protective means. Volume activity of aerosols at the working place is calculated using the release factors of radionuclides from cutting for the volume of 10 m^3. The code evaluate first what would be the internal dose for the worker if he/she has no protective means and based on calculated hypothetical value, the code allocate the protective means in order to keep the dose as low as reasonable achievable, see Section 5.3.2.(d). The calculation of the effective dose from internal exposure is the performed under the assumption that the worker has the allocated protective means. Average volume activity of aerosols in the room and average volume activity of aerosols in the background of the controlled area are estimated values. The principle is presented on the Fig. 23.

As with external dose, there are assumptions made in this code that need to be implemented in the decommissioning practices adopted, if the safety assessment is approved. In this case, it is necessary to ensure that workers wear respiratory protective equipment that meets the performance criteria assumed in the modelling code. This needs to be recorded in the assumptions part of a real safety assessment.

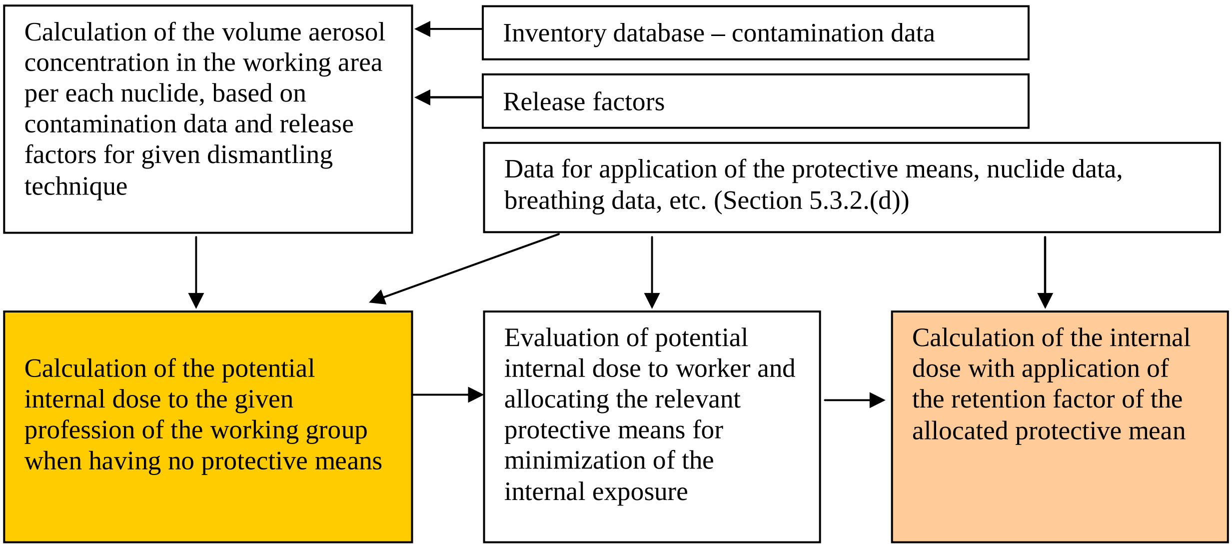

FIG. 23. Principle of calculation of internal dose during dismantling.

<u>Calculating of dose for individual workers</u>

The sum of the external and internal doses for individual professions of workers is summed and presented as a total dose for individual professions and after summing over all professions of the working group is presented as the overall dose for the discrete decommissioning activity. According to the general procedures for protection of workers, the dose for individual workers must be evaluated and controlled in order to be ALARA and in any case, to be lower then the annual limit of 20 mSv in a year per individual.

The bottom-up approach used in the calculation of decommissioning parameters, as applied in the OMEGA code, enables to evaluate the dose to individuals during the discrete decommissioning activities. By summing the data over the duration of the given decommissioning phase, like dismantling the system 321 or 322, and comparing the duration of the phase with one year duration, it is possible to evaluate whether the annual limit of 20 mSv per year was met. The principle applied is the following:

— For each decommissioning activity, the dose is calculated for each profession of the workers separately. The calculation is dependent on the profession that results in coefficients of stay in main components of the dose rates. The doses involve all productive and non-productive time components, spent by each member of the profession in the controlled area. This approach corresponds with the real organization of the working time within the controlled area and recording of dose data for individuals. The model working time structure is presented on Fig. 24.
— Each profession of workers can have in principle several members. The manpower and dose calculated for the profession as whole, is distributed to each individual of the profession according to the number of workers in the given profession of the working group.
— The manpower allocated to an individual, represents the real duration of the discrete activity within the controlled area. When dividing the dose allocated to an individual by this manpower, the normalized dose rate for the decommissioning activity is calculated. This dose

rate represents the averaged level of risk for the individual of the profession involved in the working group.

— A table is constructed, having on one axis the normalized dose rate in selected intervals (for example 2 μSv/h) and other axis the data of manpower components which fits with the given interval of the normalized dose rate as picked up from the database of calculated data. The individual effective dose components can be calculated as the product of manpower component in the given interval of the normalized dose rate and the middle value of the interval.

— By summing the data over the whole range of the normalized dose rate, the total effective dose for individual can be calculated. The data needs then to be compared with the duration of the evaluated process.

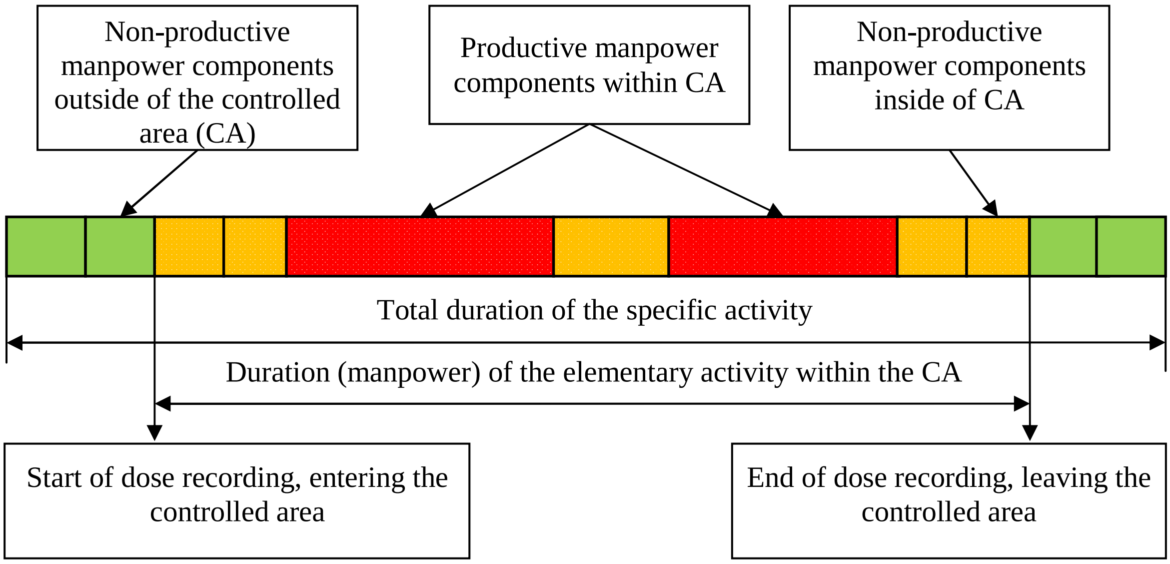

FIG. 24. Model working time structure for calculating the dose to workers.

A manpower spectrum can then be constructed for the decommissioning activities or for selected group of decommissioning activities which represents the distribution of manpower components versus the normalized dose rate. Parallel to the spectrum of manpower, the individual effective dose components spectrum can be reconstructed. This resulting graph represents the distribution of the overall exposure risk for members of individuals of the professions involved in the planned decommissioning activities or its stages. In the NPP Test Case, the spectrum was constructed for dismantling of the systems 321 and 322. The shape of the spectrum shows the distribution of exposure risk in evaluated decommissioning activities. The distribution is facility specific, related to the radiological situation in facility systems and structures.

(d) Parameter values

The main data used in the calculation of the individual doses to workers is the following:

— Inventory database for systems 321 and 322 and for relevant rooms;
— Dismantling categories allocated to equipments of the systems 321 and 322 to be dismantled;

- Cutting techniques for dismantling categories applied and aerosols release factors for techniques;
- Increase factors for calculating the manpower;
- Composition of the working groups;
- Coefficients of stay of professions of the working group for individual dose components;
- Retention factors of the personnel protective means used;
- Non-productive time components; and
- Dose rate of the background of the controlled zone.

(e) Treatment of uncertainties

Uncertainties of parameters used in the calculation of the doses to workers are the following:

- The equipment of systems 321 and 322 were introduced into the inventory database in full extent according to the facility documentation. All relevant physical data for individual items of the database were collected or estimated based on current knowledge of the facility.
- The radiological data for the equipment and the rooms of systems 321 and 322 were developed based on the existing facility data, on model calculation and selected data were checked by on-site measurements. The radiological data were developed in full extent needed for the calculation. The uncertainties of radiological data are estimated to be approximately 30 %.
- The extent of preparatory and finishing activities was involved into the calculation case according to the Tables 14 and 15 in Section 3. The extent is representative and is considered as sufficient for the test case.
- Unit factors, coefficients of stay for workers, aerosols release factors and other data used for the calculation in the OMEGA code were applied in several other decommissioning projects up to now. The data are subject of continuous periodic updating, and comparison with real results. The uncertainties of these data are estimated to be approximately 20 %.

(f) Results

The main calculation data for both systems 321 and 322 is presented in Table 20. It is evident that the dismantling of systems is the critical operation from the point of safety of planned activities. The decontamination of building surfaces will be bounded by the dismantling assessment and the annual limit of 20 mSv is not reached.

TABLE 20. MANPOWER AND COLLECTIVE DOSE FOR THE DECOMMISSIONING OF SYSTEMS 321 AND 322

Decommissioning Activity	**Manpower** [man-hours]	**Collective dose** [man.µSv]
Sum 321	19 363	183 581
Dismantling, 321	7 048	180 597
Decontamination of rooms surfaces, 321	9 666	2 656
Radiation survey of rooms surfaces, 321	2 648	328
Sum 322	49 825	95 842
Dismantling, 322	12 886	87 718
Decontamination of rooms surfaces, 322	29 075	7 127
Radiation survey of rooms surfaces, 322	7 863	997

The values of individual effective dose to workers during decommissioning, evaluated according to the procedure presented in Section 5.3.2. (c) above are presented in the Table 21. The effect of delaying the dismantling was also evaluated and the results are presented in Appendix I. As an example, the individual dose for most exposed profession B, when dismantling in year 2010, is 14 954 µSv compared to 29 032 µSv for prompt dismantling.

TABLE 21. MANPOWER AND INDIVIDUAL EFFECTIVE DOSE FOR THE DISMANTLING OF 321 AND 322 SYSTEMS

	Profession	**A**	**B**	**C**	**D**	**E**	**F**
Manpower, System 321	[manhours]	928	841	797	212	903	571
Dose, System 321	[µSv]	21 185	29 032	28 013	3 385	14 059	10 895
Manpower, System 322	[manhours]	1 644	1 437	1 361	508	1 637	1 155
Dose, System 322	[µSv]	10 105	14 920	14 150	1 209	6 704	5 264

The duration of dismantling evaluated using the MS Project schedule, as generated by the OMEGA code, is 8 months for the system 321 and 14 months for the system 322. The duration is rounded for whole months. The schedules are also presented in Appendix I.

The manpower spectrum for decommissioning of systems 321 and 322 is also presented in Fig. 25 and the individual dose spectrum in Fig. 26. It is evident from the figures and from the Table 20 that the dismantling of system 321 needs to be optimized and this is discussed in more detail in Section 7.

As for decommissioning of the system 322, most of manpower components are located in “safe” range of normalized dose rate below approximately 14 µSv/h. Under this value, the annual limit of 20 mSv is not expected to be reached. Some manpower components of the system 322, in the range of 30-40 μSv/h, are related to dismantling of the heat exchangers.

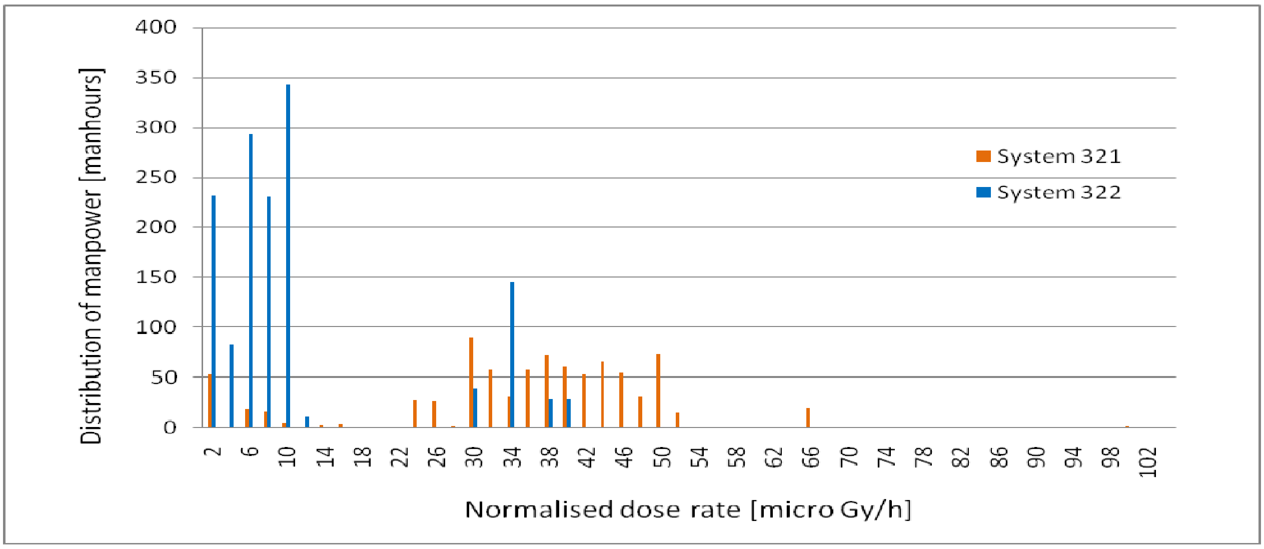

FIG. 25. The manpower spectrum for dismantling of system 321 and 322.

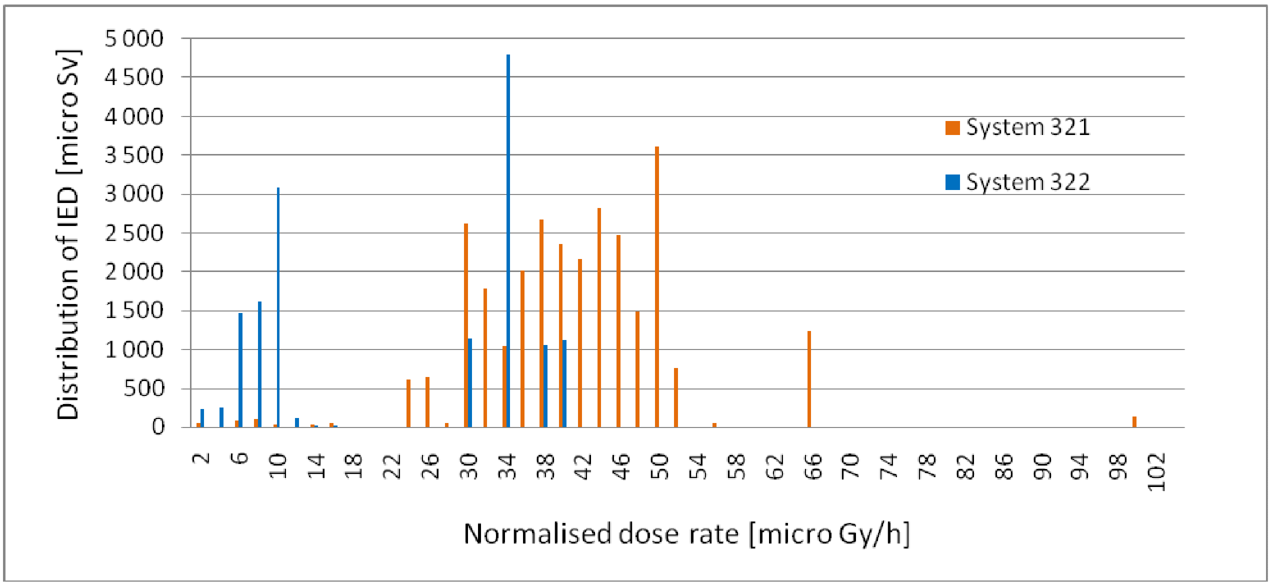

FIG. 26. The individual effective dose spectrum for dismantling of system 321 and 322.

5.3.3. Doses to public

(a) Introduction

The DecDose code [12] was applied for the estimation of the public dose in the normal situation during the decommissioning of systems 321 and 322. DecDose assesses the annual public dose from radioactive gas and airborne particles discharged into the environment through various pathways from the nuclear facility, where decommissioning activities, such as cutting and decontamination are conducted (see Fig. 27). The amount of radionuclides released into the environment is calculated according to the dismantling conditions relating to the assumptions on cutting tools, contamination control enclosures and filters. A cutting model shown in Fig. 28 was proposed for the calculation of the amount of radionuclides dispersed into working environments such as workshop and enclosure.

The annual public dose is evaluated from the year the dismantling activity started to the year it is completed. The public doses before and after dismantling activity are beyond the scope of this assessment.

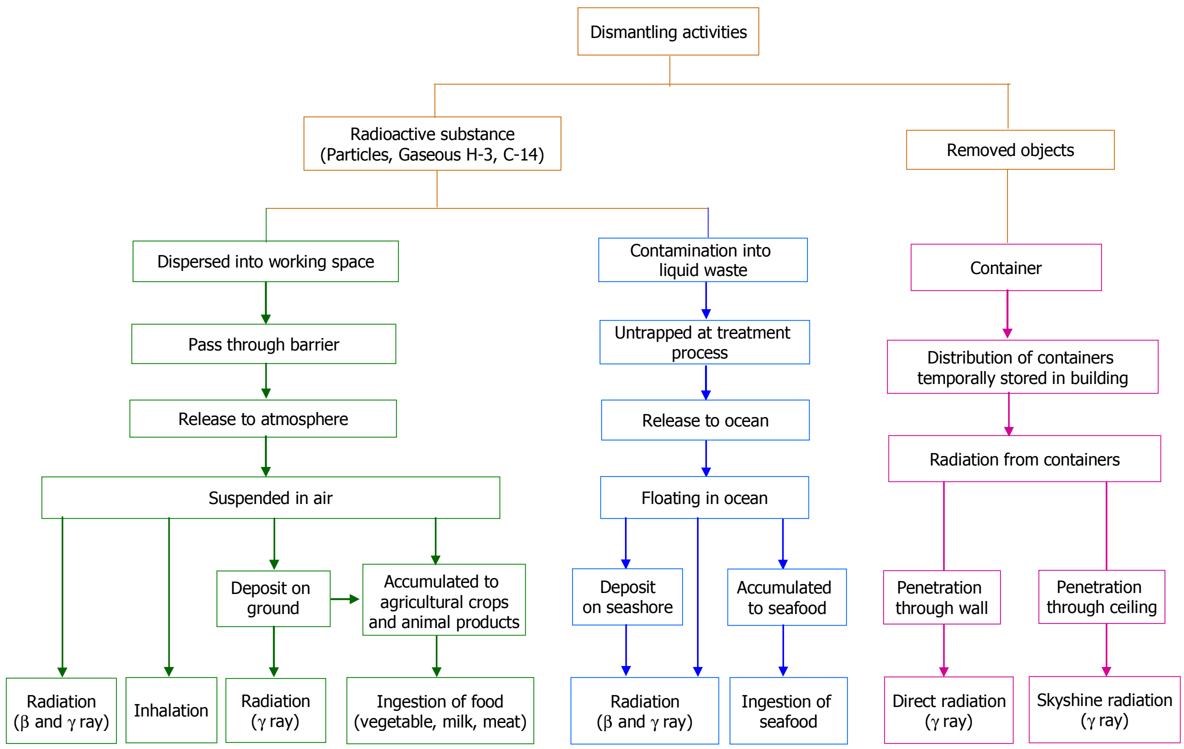

FIG. 27. Pathways for public dose evaluation in DecDose.

(b) Key assumptions

All of the components and structures in the system 321 are assumed to be contaminated uniformly. Radioactive decay during one-year decommissioning activity can be neglected and is not taken into account in the dose assessment.

(c)Modelling approaches

<u>Discharge of radioactive gas and airborne particles from the stack</u>

The public dose focused on discharge of radioactive substances from the surface contaminated materials into the atmosphere. No data on activated material is provided for the system 321.

The release of radioactive substances from the facility into the atmosphere was modelled based on the pathways shown in Fig. 27, according to the decommissioning plan which includes working schedule, cutting techniques and contamination control conditions. DecDose is capable of dealing with up to fifty five radionuclides including gaseous radionuclides of H-3, C-14 and their decay products. For accurate evaluation of the radioactivity emitted from the contaminated materials, the kerf area in cutting activities is required to be determined as shown in Fig. 28. Kerf width, *w*, actually depends on the cutting tool applied and thickness of the component, and kerf length, *L*, depends on the dimensions of the container in which the component is stored, and the shapes of components such as piping and ducts. Therefore cutting models also depend on the component shape in evaluating the kerf area. It is considered that cutting for size reduction is conducted to radioactive material which is already cut out of their originally installed positions in different places.

Radioactive contaminants usually exist on the inner and/or outer surface of the components and structures, except percolation. A simple plate is taken for a typical example as already shown in Fig. 28. The surface density of radioactive contamination is assumed to be constant on the plate. The quantity of radioactive substances emitted from the object to working space is expressed by multiplying the cutting length L by kerf width w_j by surface density f_i of radionuclide i, which is obtained using radionuclide composition ratio for the object. If the contaminant exists on the both sides, it must be evaluated in consideration with the surface densities for each side. The quantity A_{ij} is expressed by the following equation using emission rates to air b_{ij} for a single side contaminated plate.

$$A_{ij} = L*w_j*b_{ij}*f_i \qquad (2)$$

where the emission rate for surface contaminated material, b_{ij}, is defined as the ratio of radioactivity dispersed into the working space as airborne particles excluding those deposited on the floor to the surface radioactivity contained in the kerf.

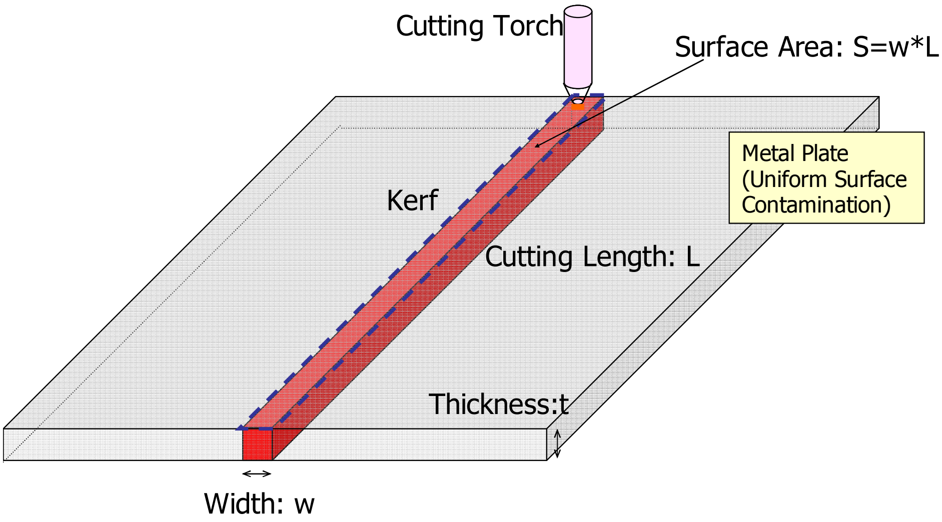

FIG. 28. Dispersed model during cutting activity for surface contaminated plate in DecDose.

Radionuclides dispersed into the working space are discharged to the atmosphere through one of the following three routes as shown in Fig. 28. In this regard, airborne particles are assumed not to be deposited on the inner surface of ducts and building walls for conservative evaluation.

i) Route passing through both the enclosure and the building ventilation (discharge at higher position)

Among the quantity A_{ij} of radionuclides dispersed into the working space using method j, the quantity B_{1ij} of radionuclide i discharged to the atmosphere through this route is expressed as follows:

$$B_{1ij} = (1-p)(1-q_i)(1-s_i)A_{ij} \qquad (3)$$

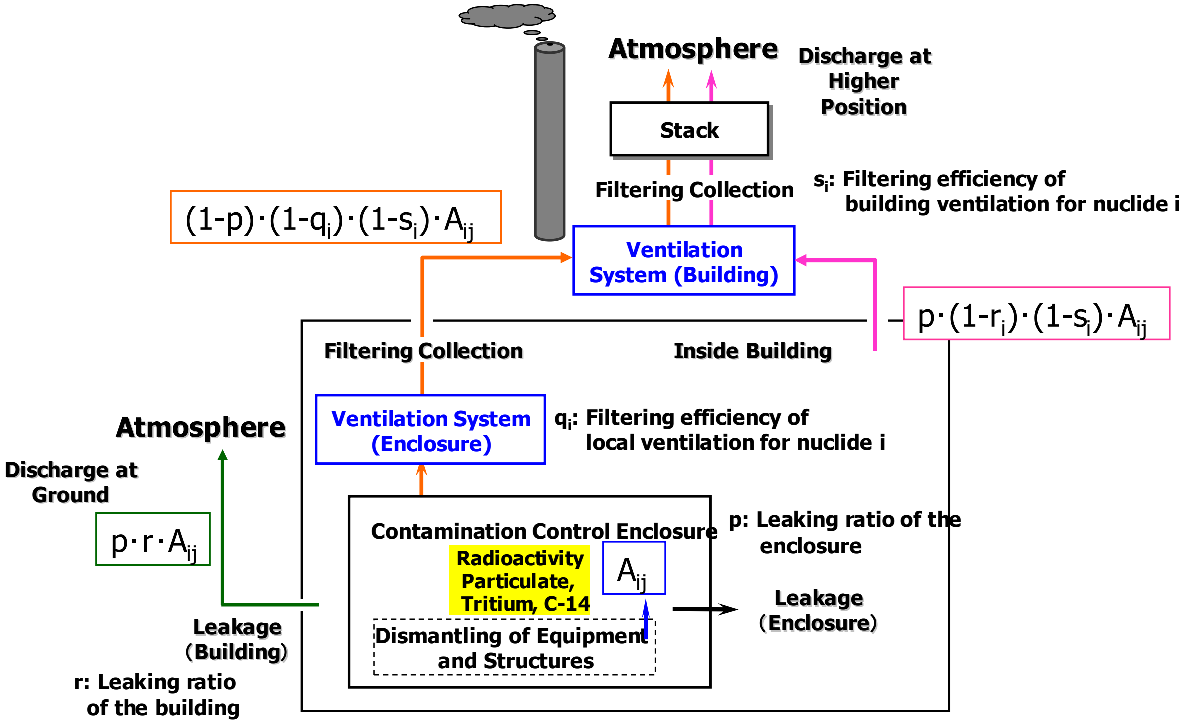

FIG. 29. Discharge routes from enclosure to the atmosphere in DecDose.

ii) Route passing through building ventilation filter after leakage from the enclosure (discharge at higher position)

The quantity B_{2ij} of radionuclide i discharged into the environment is expressed as follows:

$$B_{2ij} = p\,(1\text{-}r_i)\,(1\text{-}s_i)\,A_{ij} \qquad (4)$$

iii) Route of leakage from both the enclosure and building containment (ground level discharge)

The quantity B_{3ij} of radionuclide i discharged into environment is expressed as follows:

$$B_{3ij} = p{*}r_i{*}A_{ij} \qquad (5)$$

Segmenting methods for piping depend on the diameter of the piping, in order for densely packaging in waster containers. Vertical cuttings are needed for piping with larger diameters in addition to circumferential cutting as shown in Fig. 29.

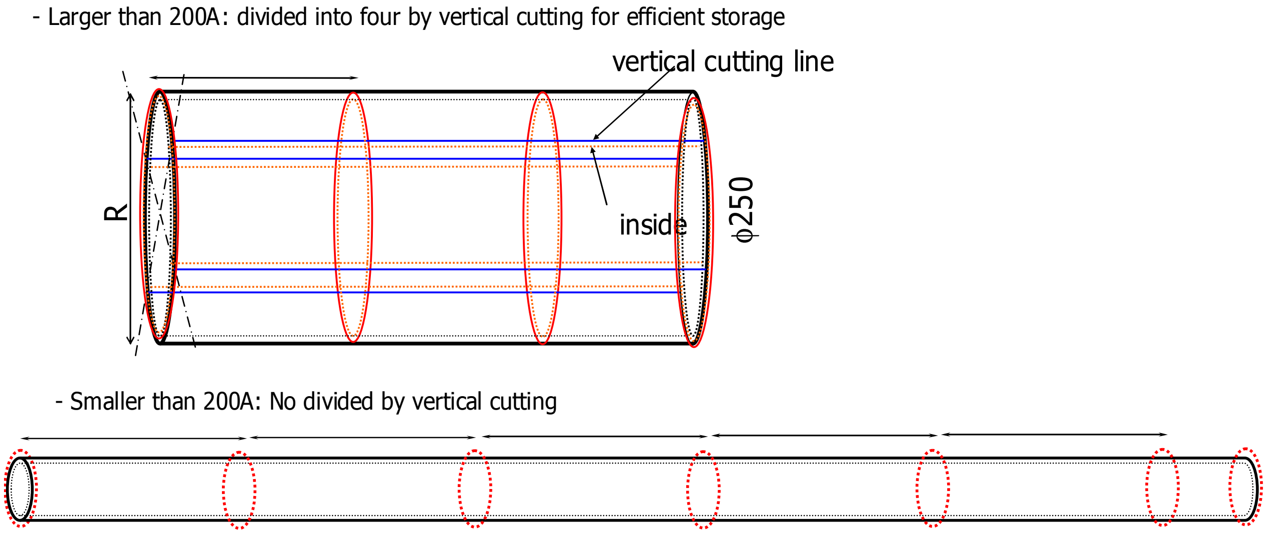

FIG. 30. Segmentation model for piping in DecDose.

Radioactive gas and airborne particles discharged from the stack are advected and diffused in the atmosphere. Equations for advection and diffusion applied here are the same as those in the operational phase using latest meteorological data [17]. Models for surface deposition of radionuclides and transfer to vegetable and livestock are also the same in the operating stage.

<u>Radiation from dismantled waste temporally stored in buildings</u>

In the evaluation of the public dose from direct and skyshine radiations in DecDose, the radiation attenuation in walls and ceilings of containers and buildings is taken into account in addition to self shielding of dismantled objects in the waste container. Packaging efficiency for each container and type of solid material such as metal or concrete is also used to determine the amount of radioactivity in the container. Distribution of containers in the building is assumed so that it gives the maximum public dose at the NPP site boundary for conservative evaluation.

(d) Parameter values

<u>Components in system 321</u>

The surface contamination density of components of system 321 is shown in Table 22. At the reference date of 1 July 2005 the total contamination density of the inner surface of the inner surfaces of pipes, T-junctions, valves, pumps and heat exchangers is 9.76 x 10^5 Bq/cm^2 and that of the outer density is 0.6 Bq/cm^2.

<u>General parameters</u>

Dismantling of the system 321 is assumed to be completed in nine months. The activity for system 321 starts at 1 July 2005 and ends at 31 March 2006.

TABLE 22. COMPONENTS IN SYSTEM 321

<table>
<tr><th colspan="2">Component</th><th>Number</th><th>Total Weight [ton]</th><th>Inner Surface Density [Bq/cm^2]</th><th>Outer Surface Density [Bq/cm^2]</th></tr>
<tr><td rowspan="3">Piping, T-junction</td><td>20 mmϕ</td><td>34</td><td rowspan="3">16.4</td><td rowspan="6">9.76 x 10^5</td><td rowspan="7">0.6</td></tr>
<tr><td>80 mmϕ</td><td>6</td></tr>
<tr><td>250 mmϕ</td><td>59</td></tr>
<tr><td colspan="2">Valve</td><td>197</td><td>14.6</td></tr>
<tr><td colspan="2">Pump</td><td>2</td><td>5.4</td></tr>
<tr><td colspan="2">Heat exchanger</td><td>2</td><td>7.0</td></tr>
<tr><td colspan="2">Motor, etc.</td><td>75</td><td>6.8</td><td>0</td></tr>
</table>

<u>Parameters for the calculation of the radionuclides amount</u>

Plasma arc cutting whose kerf width and emission rate are generally larger than any other cutting tools is assumed to be applied to all of the components of system 321 for conservative evaluation. The emission rate, b_{ij}, of radionuclide i from a surface contaminated metal plate up to 70% was obtained for plasma arc cutting in the experiment at the nuclear reactor facility. Another experiment showed the lower emission rate by a factor of eight for piping shape components than that of plate components. Therefore, the emission rate of plasma arc cutting for the NPP Test Case is assumed to be 10% or 70%.

Cutting length depends on the dimension of the ISO container which is 5.7 m x 2.3 m x 2.2 m inside. Considering the actual dismantling activities in the facility, however, piping of 1.5 m or shorter in length are better for handling in the working space. kerf width by plasma arc cutting in air is assumed to be 1 cm.

HEPA filters are not installed at the local contamination control enclosure where cuttings are carried out, and only the filtration at the building ventilation is taken into account. The filter efficiency of 99.0% or of 99.97% was assumed to cases as shown in Table 23.

TABLE 23. CALCULATION CONDITIONS FOR EACH CASE ON EMISSION RATE AND FILTRATING EFFICIENCY

	Emission Rate of Plasma Cutting for Contaminated	Building Filter
Case 1	10%	Yes (99.97%)
Case 2	70%	Yes (99.97%)
Case 3	70%	Yes (99.0%)

Parameters for calculation of public dose

— *Physical parameters*

The critical group is assumed to reside at the NPP site boundary on the ground level. The effective height of stack is the same as the actual stack height of 110 m. The distance between the site boundary and the stack is assumed to be 500 m at the height of 0 m. Based on the meteorological data shown in Table 4, dilution factor, χ/Q at the boundary is calculated to be 1.46 x 10^{-14} s/cm^3, which is actually the maximum value at approximately 4 km from the boundary of the site.

— *Social parameters*

As described in Section 3.1.3 (Definition of critical group), members of a critical group live on the site boundary and ingest foods such as vegetables, meats, and milk and its products which were cultivated at the same place. The amount of food ingested by adults according to the NPP Safety Report is shown in Table 24.

TABLE 24. FOOD CONSUMPTION FOR INDIVIDUAL ADULT MEMBER OF CRITICAL GROUP

Food		Quantity (g/d)
Vegetables	Leaf	310
	Root	400
Meat		250
Milk and its products		1000

(e) Uncertainties

It must be noted that some of the parameter values assumed in 5.3.3(d) range widely and associated with uncertainties. For example, kerf width is considered to depend not only on the cutting technique applied, but on the skill of the cutting worker, and the uncertainty associated with worker skill is not taken into account. Most of parameter values were selected as conservative as possible. In this assessment, three cases with different values of emission rate and filtering efficiency were carried out as the sensitivity analysis.

(f) Results

The calculated results of the amount of radionuclides discharged into the atmosphere and on this basis the estimated public dose for each pathway are summarized in Tables 25 and 26. In the Case 1, where the emission rate for surface contaminated materials is 10% , the total radioactivity of 6.27 x 10^5 Bq is estimated to be discharged into the atmosphere resulting in the total public dose of 2.22 x 10^{-7} μSv/a.

In the Case 2, where the emission rate is 70%, the estimated total radioactivity discharged into the atmosphere increases to 4.39 x 10^6 Bq and the total public dose increases accordingly to 1.55 x 10^{-6} μSv/a.

In the worst case evaluation with filtrating efficiency of 99.0% (Case 3), instead of 99.97%, the estimated total radioactivity that is discharged into the atmosphere is 1.46 x 10^8 Bq and the dose to the public is 5.17 x 10^{-5} μSv/a.

Compared with the safety criteria for the public (see Section 2), however, these evaluated value is six orders of magnitudes less than the public dose limit of 1 mSv/a in the normal situation and less than the dose constraint for the site (o.3 mSv/y). The pathway of ground surface deposition, which causes external exposure dose to public of more than 60% of entire annual public dose, is the dominant for all cases.

TABLE 25. CALCULATED RESULTS OF RADIONUCLIDES DISCHARGED INTO ATMOSPHERE

Radionuclides	Activity (Bq) at the Start (7/1/2005)	Case 1 (Bq)	Case 2 (Bq)	Case 3 (Bq)
Total	4.03×10^{12}	6.27×10^{5}	4.39×10^{6}	1.46×10^{8}
Mn-54	2.38×10^{10}	3.71×10^{3}	2.60×10^{4}	8.66×10^{5}
Fe-55	3.48×10^{12}	5.42×10^{5}	3.80×10^{6}	1.27×10^{8}
Co-60	3.12E $\times 10^{11}$	4.85×10^{4}	3.40×10^{5}	1.13×10^{7}
Ni-59	1.37×10^{9}	2.14×10^{2}	1.50×10^{3}	4.99×10^{4}
Ni-63	1.92×10^{11}	3.00×10^{4}	2.10×10^{5}	6.99×10^{6}
Tc-99	8.98×10^{4}	1.40×10^{-2}	9.79×10^{-2}	3.26
Sb-125	1.65×10^{10}	2.57×10^{3}	1.80×10^{4}	6.00×10^{5}
Pu-238	1.37×10^{5}	2.14×10^{-2}	1.50×10^{-1}	4.99
Pu-239	1.56×10^{4}	2.43×10^{-3}	1.70×10^{-2}	5.66×10^{-1}
Pu-240	2.47×10^{4}	3.85×10^{-2}	2.69×10^{-2}	8.98×10^{-1}
Pu-241	5.76×10^{6}	8.96×10^{-1}	6.28	2.09×10^{2}
Am-241	1.10×10^{4}	1.71×10^{-3}	1.20×10^{-2}	3.99×10^{-1}
Cm-244	1.74×10^{5}	2.71×10^{-2}	1.90×10^{-1}	6.32

TABLE 26. CALCULATED RESULTS OF PUBLIC DOSE FOR EACH PATHWAY AND TOTAL (μSv/y)

Pathway			Case 1	Case 2	Case 3
External	Cloudshine		7.58×10^{-9}	5.30×10^{-8}	1.77×10^{-6}
	Ground surface deposition		1.50×10^{-7}	1.05×10^{-6}	3.50×10^{-5}
	Direct gamma radiation		1.55×10^{-11}	1.54×10^{-11}	1.54×10^{-11}
	Skyshine radiation		2.88×10^{-15}	2.87×10^{-15}	2.87×10^{-15}
Internal	Inhalation	Adult	7.49×10^{-9}	5.24×10^{-8}	1.75×10^{-6}
	Agricultural crops	Leaf	2.31×10^{-8}	1.61×10^{-7}	5.38×10^{-6}
		Root	1.94×10^{-9}	1.36×10^{-8}	4.54×10^{-7}
	Livestock	Milk	4.66×10^{-9}	3.26×10^{-8}	1.09×10^{-6}
		Meat	2.70×10^{-8}	1.89×10^{-7}	6.30×10^{-6}
Total			2.22×10^{-7}	1.55×10^{-6}	5.17×10^{-5}

5.3.4. Doses in accident scenarios

(a) Introduction

The purpose of this Section is to explain how the doses in accident scenarios have been calculated.

(b) Key assumptions

There are a number of assumptions within the modelling approaches and parameter values that were used in the evaluation, as described below. These are either justified in Appendix III.

(c) Modelling approaches

The modelling approach is based on the dose rates and contamination levels for the systems 321 and 322. For external doses, dose rates are combined with times of exposure. For internal doses, contamination levels are combined with release fractions, airborne release modelling, decontamination factors, breathing rates, dose release factors etc. In all cases, a conservative approach is used.

(d) Parameter values

Key parameters are presented in Table 41 of Appendix III. The surface contamination levels used in the radiological accident analysis are assumed to be twice those given in Section 5.3.1 above, in order to ensure a conservative approach.

(e) Uncertainties

There are a number of uncertainties in parameters used, but these are addressed by using conservative values, ensuring that the assessed dose is conservative. Although it might be possible to reduce the level of conservatism in the accident dose assessment, the radiological analysis does not need this reduction, so the level of conservatism is considered acceptable.

(f) Results

The highest dose calculated for an accident scenario is 96 mSv to a worker, for accidents during cutting operations. Although this is above the 20 mSv level in Ref. [4], Section 5.2 above shows that the risk is low. Nevertheless optimization can be applied to reduce the potential exposure as discussed in Appendix III.

6. ENGINEERING ASSESSMENT

6.1. ENGINEERING ASSESSMENT METHODOLOGY

The process for identifying engineered control measures (safety related systems, structures and components - SSCs) has been outlined in Section 2.6.3 and Section 5.2 above. As part of the safety assessment, a safety assessor needs to specify the necessary safety related functions, and any performance requirements, of each SSC. Then engineering evaluation needs to be performed to demonstrate that the safety and performance requirements assumed by the safety assessor will be provided by each SSC as expected.

It is normal practice to categorize SSCs in accordance with the importance of the safety function that they are be required to provide. This allows a graded approach so that engineering expertise and effort can be applied in proportion to the safety significance of the SSCs. The operator may develop his/her own engineering assessment process, as there is no universal international standard in this area, with national arrangements being driven by national regulators and their specific requirements.

The NPP Test Case has arrived at an assessment that is not strongly based on a risk based approach but instead takes a more deterministic approach. As such, SSC classes were found not to be needed, but some discussion of risk classes is made in the other test cases. An example is given below for information and consideration (see also main report):

SSC Category 1 – Those SSCs that are principle means for the prevention/mitigation of significant public exposure and major worker exposure. Typically this is applied for Risk Class I accident scenarios. Category 1 SSCs are not usually to be expected in a decommissioning safety assessment.

Requirement – *Engineering assessment to be supported by detailed engineering investigations and calculations, assessment against national engineering codes and standards, review of operational experience, specification of surveillance programme requirements and a demonstration of fitness for purpose in meeting functional requirements under accident conditions.*

SSC Category 2 – Those SSCs that make a significant contribution to the prevention/mitigation of decommissioning worker exposure, other workers on the site but a lesser public risk, where the risk is commensurate with Risk Class II accident scenarios. Category 2 SSCs may be required in decommissioning safety assessments, but will not be commonly found in decommissioning applications.

Requirement – *The requirement is similar to SSC Category 1 items, but with an appropriately lesser level of detail in the engineering assessment.*

SSC Category 3 – Those that have only a minor contribution in the prevention/mitigation of worker exposure. Typically, this is applied to Risk Class III accident scenarios. This will be the category of SSC often found in decommissioning safety assessments.

Requirement – *The requirement will be to demonstrate adequate functionality and performance only based on records or/and a structured plant walkdown to demonstrate that the facility is in good condition and in accordance with engineering drawings.*

SSC Category 4 – Those that make only slight contribution to the prevention/mitigation of worker exposure. Category 4 SSCs may be applied in Risk Class IV accident scenarios.

Requirement – *The only requirement is to register the SSCs in the facility surveillance programme, and may only be required to be considered for response when they become non-functional.*

If a SSC is provided by new facility engineering, assessment by the operator is not needed. Instead, the design documentation needs to be in accordance with the appropriate national engineering codes or standards, together with a demonstration that the safety and functional requirements of the SSC specified in the safety assessment are satisfied. SSCs required during decommissioning must be subject to an appropriate Ageing Management Programme (AMP), such as that shown in Fig. 31, to ensure that the functional requirements continue to be met. IAEA Safety Reports Series No. 15 gives details [18]. The detail in the engineering assessment, demonstrating compliance with functional and performance requirements need to be proportionate to its SSC Category.

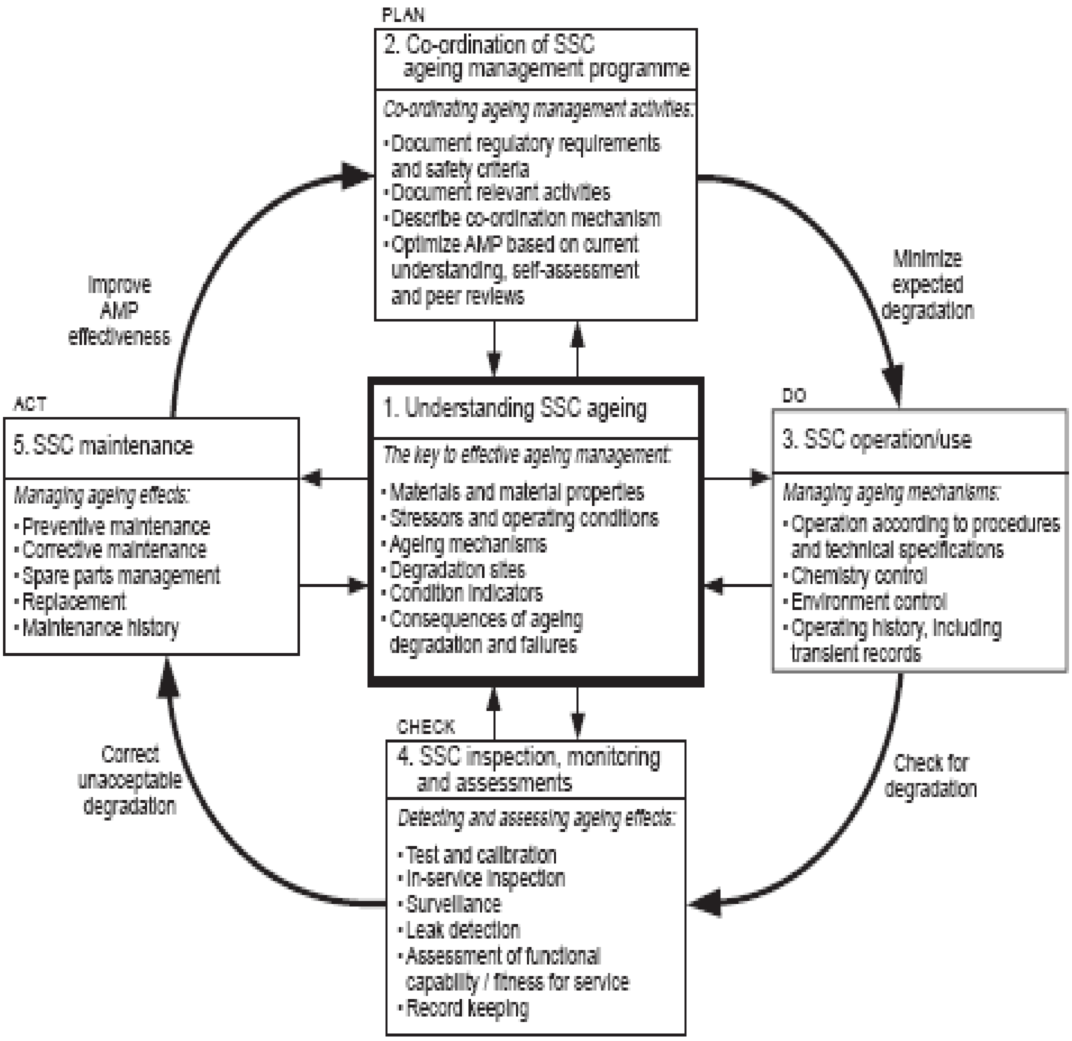

FIG. 31. Features of a pragmatic ageing management programme.

The principle of defence in depth is required to be applied during decommissioning and the compliance with relevant international engineering codes. This requires hierarchical deployment of different levels of equipment and procedures in order to maintain the effectiveness of physical barriers placed between radioactive materials and workers, the public or the environment, in normal operation, anticipated operational occurrences and, for some barriers, in accidents at the facility [19].

6.2. ENGINEERING MEASURES DERIVED FROM THE SAFETY ASESSMENT OF SYSTEMS 321 AND 322

The results of the safety assessment are presented in Sections 5 and 7 in a series of summary tables - one for each significant accident scenario selected for assessment. Table 26. identifies the engineering measures necessary to ensure that the radiological consequences of each accident scenario are within the requirements of the accident risk criteria (see Section 2.6) and are also ALARA. The full analysis is presented in Appendix III. On the basis of the results shown in Appendix III, the identified SSCs for decommissioning of Systems 321 and 322 are summarized in Table 26.

A 'desktop review' was then carried out by an expert group that included the facility operator, appropriate engineering staff, the safety assessment engineer and the decommissioning engineer. The specific functional and performance requirements of each SSC were discussed to confirm that they could be met or identify measures necessary to resolve any shortfall. This was followed by a walkdown of systems 321 and 322 to provide a visual inspection of the condition and environment of the SSCs. The walkdown was also used to consider ergonomic and human factors aspects of the work area and planned decommissioning operations, with any concerns being recorded. A record of significant findings during the walkdown was made on a standard proforma and any improvement or corrective actions identified on the proforma. The schedule of SSCs was then updated following the desktop review and walkdown into a form that is sometimes called the 'Engineering Schedule'. This schedule identifies the SSCs, their safety categories, their functional and performance requirements and any actions necessary to deal with shortfalls. The Engineering Schedule for systems 321 and 322 is attached as Table 26.

Additional investigation of ALARA measures was performed and identified the following two enhancements for consideration over and above those specified in the risk assessments and by the facility Safety Management Arrangements. These are:

— During the set-up of the room containing the systems, shielding has to be fitted to the pipework of system 321 to minimize the dose to operators during set-up. The effectiveness of this needs to be confirmed by radiation survey; and
— During the post system dismantling of the room containing the systems, the operators must wear respiratory protection to further reduce the possibility of internal dose uptake.

6.3. APPLICATION OF A CATEGORIZATION SCHEME

The accident analysis conducted for the NPP Test Case did not set out a categorization scheme and then seek to apply protection systems or mitigation systems until a desired low category was achieved. Such an approach takes the safety assessment towards probabilistic risk assessment, which is not necessarily the appropriate method for all decommissioning assessments.

Instead, the accident analysis in this test case took the view that deterministic safety measures were preferable, and set out criteria for the numbers of protective barriers to be available. The test case working group did not want to try to assign probabilities to accidents that might occur in non-routine tasks that would only be performed once.

Another DeSa test case has used a more probabilistic approach, and has set out a categorization scheme (see Part C of Annex I of this report). This alternative categorization scheme shows:

— Category 1 – An engineered barrier that provides mitigation of potential consequences of > 250 mSv to workers or >10 mSv to the public.
— Category 2 – An engineered barrier that provides mitigation of potential consequences in the range 20 – 250 mSv to workers or 0.1 – 10 mSv to the public.
— Category 3 – An engineered barrier that provides mitigation against potential consequences in the range 2-20 mSv to worker or 0.01 - 0.1 mSv to the public, i.e. minor consequences.

— Category 4 – An engineered barrier that provides mitigation against potential consequences of < 2 mSv to worker or <0.01 mSv to the public, i.e. insignificant consequences.

If this alternative scheme was to be used to categorize the NPP Test Case output, then it is evident that none of the SSCs would be in Category 1. Two SSCs would be in Category 2, these being the ventilation failure alarm and the respiratory protective equipment shown on lines 3 and 4 of Table 27.

TABLE 27. ENGINEERING SCHEDULE FOR THE SYSTEMS 321 AND 322 DECOMMISSIONING

Description of SSC	SSC Safety Class	SSC Safety and Performance Requirement and Identified Shortfalls	Action to Address Shortfall
1. Derived from the Radiological Accident Analysis (see Appendix III)			
Personal dose meters that incorporate an alarm on dose	3	To alarm when the dose reaches the alarm level Alarm level is controllable to pre-defined levels Operator training required	Provide adequate training
Ventilation extract for local enclosures, fitted with fans	3	To minimize spread of contamination from the enclosure Extract to exhaust into the building ventilation system. Fans to provide a flow rate of air extracted from the enclosure exceeding $20 m^3/min$. A fan needs to be procured	Procure fan to deliver safety and performance requirements
Alarm for failure of local ventilation extract	3	To warn a worker within the tented area or size reduction facility that the local ventilation extract has failed To alarm on loss of depression in local ventilation extract To alert the worker above the noise of cutting operations Alarms need to be confirmed as operational.	Carry out routine testing and maintenance of alarms before work commences.
Respiratory protection equipment	3	To mitigate worker dose when there is airborne contamination present To provide a filtration efficiency of > 99% for particulate material.	None identified.
Filters on building ventilation system	3	To clean up the ventilation extract To provide a filtration efficiency of > 99% for particulate material Filters need to be confirmed as operational.	Carry out routine testing and maintenance of filters before work commences.

TABLE 27. ENGINEERING SCHEDULE FOR THE SYSTEMS 321 AND 322 DECOMMISSIONING (CONT.)

2. Additional SSCs Identified from Analysis of Normal Operations. These are required to support normal operations of systems 321 and 322 dismantling and the overall decommissioning programme (see footnote)			
Ventilation stack	3	To be at the height specified in the analysis of normal operations so that extract dispersion supports public dose targets. To be available for the discharge of aerial effluents. Stack height and integrity to be confirmed.	Confirm stack height and perform periodic engineering review to confirm integrity.
Ventilation ductwork	3	To be sealed and connected. Integrity to be confirmed.	Perform engineering review to confirm integrity.
Ventilation stack radiometrics	3	To ensure that aerial discharges are monitored. Availability and calibration to be confirmed.	Carry out routine testing and maintenance of stack radiometrics before work commences and periodically thereafter.
Building ventilation fans	3	To adequately extract the building to mitigate normal worker exposure. Confirm correct functionality of the fans.	Carry out routine testing and maintenance of fans before work commences and periodically thereafter.
Process isolation equipment (blinds, spades etc)	3	To prevent liquor seepage into work area. Confirm that all isolations are correctly fitted.	Carry out review of isolations and plant walkdown.
Waste container/transport trolley	3	To contain and transport waste materials safely from work area to waste handling facility. Confirm correct functionality of transport trolley and integrity of waste container.	Carry out routine inspection and maintenance of transport trolley and integrity of waste container before work commences and periodically thereafter.

Footnote – as given in the NPP Safety Assessment Report [20]

7. EVALUATION OF RESULTS AND SAFETY MEASURES

7.1. COMPARISON OF ANALYSIS RESULTS WITH CRITERIA

7.1.1. Summary of criteria

The criteria for this safety assessment are set out in Section 2, Section 2.6. The results of the safety assessment are presented in in Section 5. These results show that for both normal planned activities and accident conditions, mitigation is necessary to achieve compliance with criteria.

For normal planned activities, the mitigation consists of rotation of the workforce to limit exposure of individuals.

For the accident analysis, the NPP Test Case did not adopt *a priori* criteria for risk, probability or consequence, as to demonstrate compliance would then have needed a sophisticated probabilistic analysis. Instead the accident analysis looked at the unmitigated consequences of an initiating event and derived criteria about the numbers of barriers that needed to be in place to prevent the event leading to consequences.

Identified engineered safety control measures, i.e. systems, structures and components (SSCs) were subject to engineering assessment, as described in Section 6, to demonstrate that the selected SSCs can deliver their specified functional and performance requirements. The results of the assessment are summarized in Table 27, the Engineering Schedule, in Section 6. Once the recommendations in the schedule are completed the engineered and administrative control measures will be included in the plant maintenance schedule that supports the decommissioning plan.

(a) Normal decommissioning operations

Worker dose limit

This has been identified in Section 2.6 as effectively being 20 mSv/y; this being the interpretation of 100 mSv averaged over 5 years.

Public dose limit/dose constraint

This has been identified in Section 2.6 as effectively being the dose constraint of 0.15 mSv/y.

(b) Accident conditions

Consequences - worker dose and public dose

The accident analysis has been conducted with the intent of identifying accidents that would exceed the normal decommissioning dose limits and constraints identified for normal decommissioning operations. This test case has not sought to use categorization schemes, though other DeSa test cases have. A provisional categorization for these decommissioning activities would be Category 3, using the categorization scheme set out in Section 6.

Frequency

No criteria have been set or identified as being appropriate. The accident analysis does however identify some improvements necessary to both engineered controls and operational controls to ensure compliance with this assumption.

7.1.2. Safety assessment results

(a) Normal conditions

Worker dose limit

A maximum dose of 29 mSv in one year is identified in the assessment of the worker dose from normal decommissioning activities. As discussed above, mitigation measures are thus necessary to ensure compliance with the safety criteria. This leads to the assumption that a two team rotation approach will be used to perform the decommissioning tasks identified, and that monitoring of workers doses will be done to ensure compliance with the assumptions and criteria set out in the safety assessment.

The worker dose is dependent on the decommissioning techniques selected. The assessment here presents its lowest dose outcome which comes from the selection of thermal cutting techniques. Selection of other cutting techniques would result in a higher worker dose.

This then would lead to a worker dose of 14.5 mSv in one year which complies with the criteria.

Public dose limit/dose constraint

The public dose is dependent to some extent on the decommissioning techniques selected. The assessment here presents its most conservative outcome which comes from the selection of thermal cutting techniques. Selection of other cutting techniques would result in a lower public dose.

The conservatively estimated dose to the public is approximately 5.10^{-6} mSv/y to the member of the critical group. This is below the criterion, and it can be deduced that compliance with the public dose criterion is not dependent on the choice of cutting techniques. It also needs to be borne in mind that the dose constraint of 0.3 mSv/a applies to the whole NPP site and not only to the two systems 321 and 322.

(b) Accident conditions

Consequences - worker dose and public dose

The unmitigated consequences of those accident scenarios selected for analysis were that a worker dose of 96 mSv was a possibility. To make this outcome unlikely, requirements were identified for engineered measures to give defence in depth. The engineered measures are supplemented by robust administrative measures. Collectively, all these measures ensure safety. A further positive search for additional control measures was made. Where these additional control measures were found to be justified as being ALARA, then they were introduced.

The worker dose from unplanned events (accidents) is dependent on the decommissioning techniques selected. The assessment here presents its lowest dose outcome which comes from the selection of thermal cutting techniques. Selection of other cutting techniques would result in a higher worker dose.

Some assumptions were made in the accident analysis, and to comply with the outcome of the safety assessment, it will be necessary to ensure that the assumptions are translated into the practice. This would be a key requirement for the overarching decommissioning and its implementation instructions.

Frequency

As explained in Sections 2 and 6, a numerical consideration of frequency has been found not to be necessary. It has thus been possible to limit probabilistic aspects in making this safety assessment, and the case is made almost entirely on deterministic grounds in each of the three components.

The results of the safety assessment are summarized in Table 28.

TABLE 28. SUMMARY OF CRITERIA USED AND OUTCOME OF ASSESSMENT

Criterion	Value	Assessment outcome
Normal decommissioning operations		
Worker dose	20 mSv/a	14 mSv/a.maximum identified
Public dose (constraint)	0.15 mSv/a	Less than 2 10^{-6} mSv/a
Accident conditions		
Worker dose	20 mSv/a	Maximum of 90 mSv per event, if unmitigated. When mitigated - insignificant
Public dose	None set	Insignificant
Accident conditions		
Consequences – defence in depth requirement		
Worker protection against accidents – higher consequences	2 layers specified (i.e. number of independent complete safety measures)	3 layers identified in the most demanding scenario. This bounds the other scenarios discussed in detail in Appendix III.
Worker protection against accidents – significant consequences	1 layer specified	3 layers identified in the most demanding scenario. This bounds the other scenarios discussed in detail in Appendix III.
Worker protection against accidents – insignificant consequences	0 layers specified	3 layers identified in the most demanding scenario. This bounds the other scenarios discussed in detail in Appendix III.
Public	No requirement	Not applicable
Risk		Not quantified because of low consequences

7.2. TYPES OF AND TREATMENT OF ASSUMPTIONS AND UNCERTAINTIES

Due to the limited activities associated with the scope of the decommissioning of systems 321 and 322 (i.e., the removal and size reduction of specified systems from the NPP primary circuit and containment structure), the uncertainties in the safety assessment are limited. In addition, the NPP has had a good and well known operational history with modifications being well controlled and documented. Significant effort has gone into characterizing the remaining radioactivity and quantifying the waste inventories. As a result, the considerations described in the DeSa guidance, such as uncertainty about the physical facility, construction and facility aging, models/codes used and waste/waste streams are not pertinent.

The amount and type of information available about the radiological condition of the selected systems 321 and 322 is very detailed and complete, due to the post-operational clean-up activities and the comprehensive characterization activities. In addition, the safety controls and procedures were developed using the “worst case” input parameters discussed in Appendix III, and the release model used is a conservative one. There is confidence in the reliability and performance of key systems such as the building ventilation system and its filters. Uncertainties lie in the adequacy of implementation of the decommissioning plan requirements and in the quality of the workforce and its supervision during operations.

7.3. SAFETY CONTROL MEASURES

The safety control measures derived from the safety assessment for the decommissioning of the selected NPP systems are summarized in Tables 29 and 30 below.

TABLE 29. SUMMARY OF SAFETY CONTROL MEASURES

Control Measure Number	Engineered Safety Control Measures (SSCs)	Associated Administrative Safety Control Measures
1	Personal dose meters that incorporate an alarm on dose	— System of calibration; — System of control of issue and recording of results; — System to relate recorded results to approved dosimetry records, and to engineering work packages; — Adequate training to support the above; — Adequate training to wearers of personal dose meters.
2	Ventilation extract for local enclosures, fitted with fans Temporary enclosure for cutting operations.	Standard for construction and testing of tented enclosures.
3	Alarm for failure of local ventilation extract	— Carry out routine testing and maintenance of alarm before each work period; — System of calibration.
4	Respiratory protection equipment	— System of testing before issue; — System of calibration for test equipment; — System to recover, clean and reassemble prior to testing; — Wearers to be individually tested for protective factors obtained on a periodic basis.
5	Filters on building ventilation system	Design standard. Periodic testing to confirm performance.
6	Ventilation Stack	Periodic testing of flow and integrity.
7	Ventilation ductwork	Periodic testing of flow and integrity.
8	Ventilation stack radiometrics	System of calibration. Periodic testing and calibration.
9	Building ventilation fans	— Periodic testing of flow; — Routine maintenance.
10	Process isolation equipment (blinds, spades, etc.)	— Plant configuration control; — Design standards; — Plant procedures to drain and confirm drained the primary circuit.
11	Waste container/transport trolley	Routine maintenance and inspection.

TABLE 30. SPECIFIED SAFETY CONTROLS

Description	Safety Function	Internal Reference
Procedures to be Implemented		
Systems 321 and 322 are isolated from water feeds. In particular, System 321 is isolated from the primary circuit by blanks inserted between the twin isolation valves (numbers xx1 and xx2).	This removes any interactions between the decommissioning of System 321 and the rest of the nuclear plant.	Section 1.2 and Section 5
High dose rate jobs are planned such that a target dose for each job must be defined.	To define the boundary of intended operations with respect to dose.	Scenario 01 (Appendix III)
Workers must be issued with personal dose meters that incorporate an alarm on dose, with the alarm level set at a lower dose than the target dose for the job	To enable workers to evacuate when the dose approaches the limits of intended operation.	Scenario 01 (Appendix III)
Workers must evacuate the area if their personal dose meter alarm is activated.	To ensure that workers do not significantly exceed the dose alarm level.	Scenario 01 (Appendix III)
Cutting operations for System 321 metalwork must be carried out in an enclosure that is provided with a local ventilation extract. The enclosure must be well enclosed, even if temporary in nature. For a tent, ‘well enclosed’ means that the tenting must be visibly under depression when the local extract is working.	To minimize the spread of contamination from the area local to the cutting site, and thereby minimize doses to workers outside the immediate area.	Scenario 02 (Appendix III)
System 321 metalwork must be cut by a plasma torch, mechanical shear, a reciprocating saw, a band saw, or an oxy-acetylene torch, but must not be cut by a grinder.	To minimize the amount of airborne activity created during cutting operations.	Scenario 02 (Appendix III)
At the start of a System 321 cutting operation with a worker present, the local ventilation extract must be working. If it fails during the cutting operation, the worker must cease cutting operations and evacuate from the enclosure.	To minimize dose to the worker and minimize spread of contamination.	Scenario 02 (Appendix III)
During System 321 cutting operations, the worker present must wear respiratory protection.	To mitigate the dose to a worker arising from airborne contamination created during cutting operations.	Scenario 02 (Appendix III)
Where achievable, cutting operations on System 321 must be carried out remotely.	To minimize the maximum dose that a worker could receive.	Scenario 02 (Appendix III)
Parameters		

<table>
<tr><th>Description</th><th>Safety Function</th><th>Internal Reference</th></tr>
<tr><td>Whole body external dose rates in working areas for decommissioning System 321 & 322 are less than 2 mSv/h.</td><td>The radiological accident analysis has assumed a maximum dose rate equal to this value. If it were higher, the potential external doses would be higher, and this radiological accident analysis would not be valid.</td><td>Scenario 01 (Appendix III)</td></tr>
<tr><td>There is no significant contamination of System 322.</td><td>The radiological accident analysis has not assessed any potential for internal doses from cutting up System 322.</td><td>Scenario 02 (Appendix III)</td></tr>
<tr><td>The maximum length of cut during a work period is 1.5m.</td><td>To limit the amount of airborne contamination that can be released during a work period.</td><td>Scenario 02 (Appendix III)</td></tr>
<tr><td>The volume of an enclosure in which System 321 metalwork is cut is greater than 8 m^3.</td><td>The radiological accident analysis has assumed a minimum volume equal to this value. If it were lower, the potential internal doses would be higher, and this radiological accident analysis would not be valid.</td><td>Scenario 02 (Appendix III)</td></tr>
<tr><td>Surface contamination levels on the inside of System 321 (averaged over the area of any single cut) do not exceed:
<table>
<tr><th></th><th>Activity [Bq/cm^2] System 321</th></tr>
<tr><td>Mn-54</td><td>5.2×10^{4}</td></tr>
<tr><td>Fe-55</td><td>7.6×10^{5}</td></tr>
<tr><td>Co-60</td><td>6.8×10^{5}</td></tr>
<tr><td>Ni-59</td><td>3.0×10^{3}</td></tr>
<tr><td>Ni-63</td><td>4.2×10^{5}</td></tr>
<tr><td>Tc-99</td><td>2.0×10^{-1}</td></tr>
<tr><td>Sb-125</td><td>3.6×10^{4}</td></tr>
<tr><td>Pu-238</td><td>3.0×10^{-1}</td></tr>
<tr><td>Pu-239</td><td>3.4×10^{-2}</td></tr>
<tr><td>Pu-240</td><td>5.4×10^{-2}</td></tr>
<tr><td>Pu-241</td><td>1.3×10^{1}</td></tr>
<tr><td>Am-241</td><td>2.4×10^{-2}</td></tr>
<tr><td>Cm-244</td><td>3.8×10^{-1}</td></tr>
</table></td><td>The radiological accident analysis has assumed a surface contamination level equal to this value. If it were higher, the potential internal doses would be higher, and this radiological accident analysis would not be valid.</td><td>Scenarios 02 and 03 (Appendix III)</td></tr>
</table>

7.4. SHORTFALLS, RECOMMENDATIONS AND OUTSTANDING ISSUES

Mitigation measures have been identified that are necessary to ensure that the identified criteria are met. Implementation of these in practice will be necessary before decommissioning starts.

8. GRADED APPROACH

8.1. INTRODUCTION

The graded approach with respect to safety assessments for facilities undergoing decommissioning means a process by which the level of detail of the analysis, the complexity of the approach, the documentation, and other issues necessary to demonstrate compliance with safety requirements are commensurate with:

(i) The legal requirements;
(ii) The magnitude of any hazard involved;
(iii) The particular characteristics of a facility;
(iv) The step within the decommissioning process; and
(v) Any other relevant factor.

According to [11] and the main part of this report, an application of the graded approach needs to take into account the following factors:

— The purpose of the safety assessment (e.g. preliminary and final decommissioning plan, the phase of the decommissioning process);

— The scope of the assessment (e.g. a part of a facility, a single facility at a multi-facility site or the whole site, handling of spent fuel);

— The end-state of the facility (unrestricted or restricted use);

— The radiological hazard potential (source term) – e.g. activity inventory of the facility (surface, bulk contamination); radiological characteristics (short or long-lived radionuclides, presence of alpha emitting radionuclides); the chemical and physical state of the radioactive material (solid, liquid, gaseous; sealed sources);

— The radiological criteria with which the results will be compared;

— The size and type of the facility (including its complexity);

— Site characteristics (seismic risks, flooding, influence from or dependence on any neighbouring facilities);

— The presence and type of initiating events for incident/accident sequences (e.g., chemicals, temperature, fire, etc.);

— Likelihood and consequences of hazards;

— The physical state of the facility at the start of the decommissioning work (shut down after normal operation, or shut down after an incident; longer period of poor maintenance; uncertainty on the state of the facility);

— Complexity of decommissioning activities (e.g. the situation after a planned shutdown of the facility vs. the situation when shutdown occurred as the consequence of an incident or accident);

— Availability of applicable safety assessments for this or other similar facilities or novelty of the proposed decommissioning activities.

The NPP Test Case addresses these factors, but does so by the limited scope of considering the dismantling of two of the whole set of reactor systems.

The application of the graded approach in the NPP Test Case is illustrated in Fig. 32.

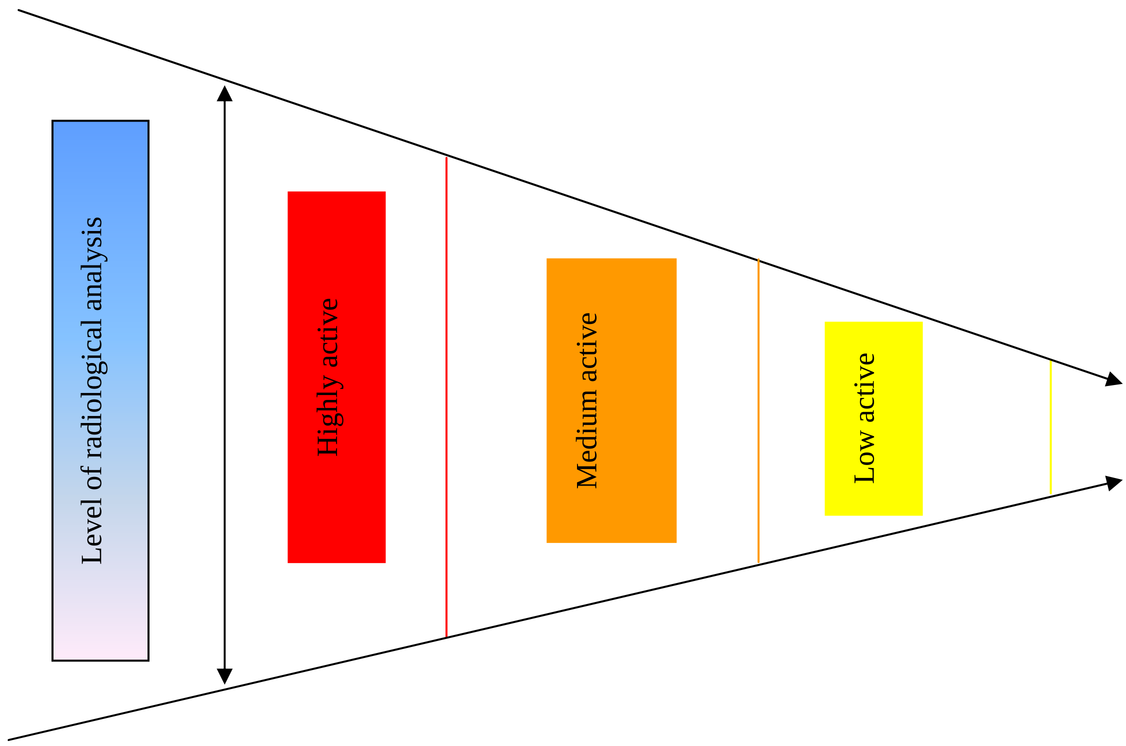

FIG. 32. Schematic of risk reduction profile.

The application of the graded approach to the development of the safety assessment for the unit1 (in particular systems 321 and 322) was also related to the decommissioning plan as shown in Fig. 33 below.

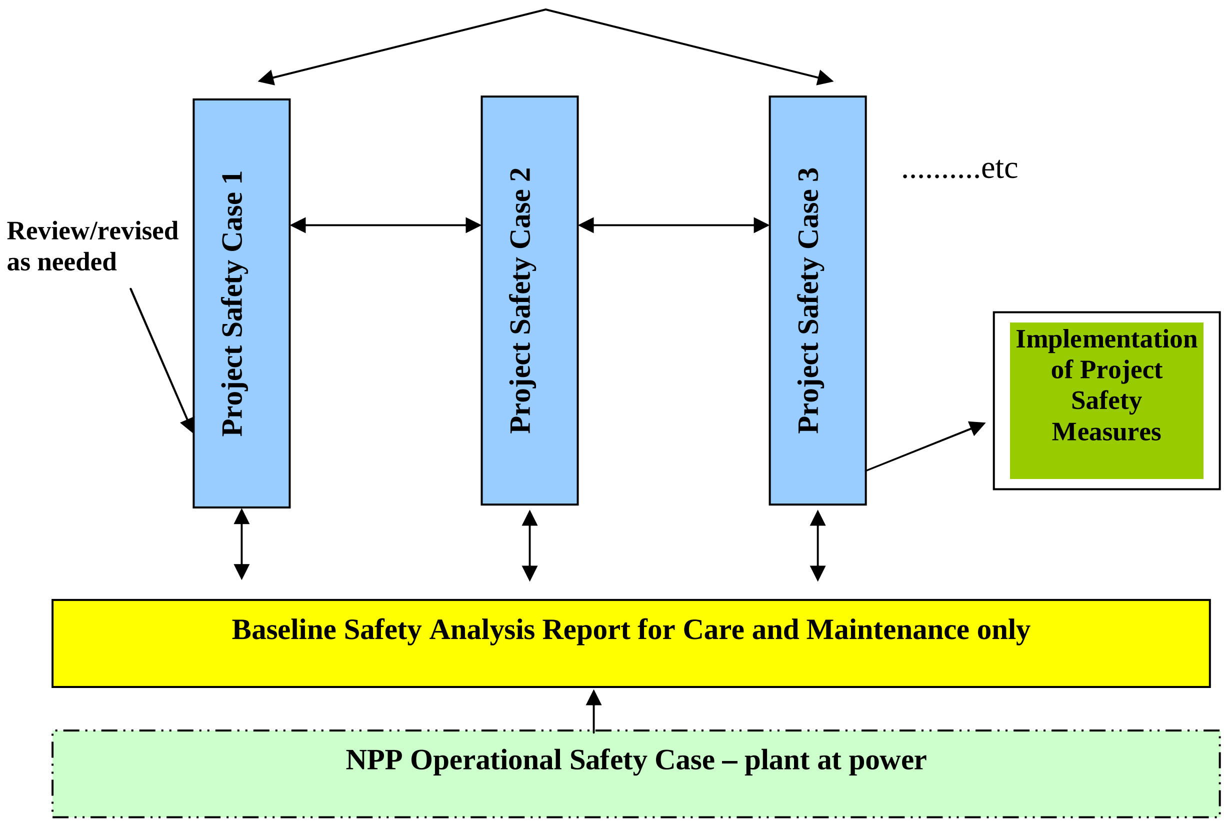

FIG. 33. Example of a decommissioning plan (safety case) approach.

8.2. LEVEL OF DETAIL FOR SAFETY ASSESSMENTS AND DOCUMENTATION

The graded approach report (see Annex II of this report) illustrates schematically the steps that a number of facilities can pass through, linking complexity of the facility and risks from decommissioning to stages of assessment. The NPP Test case aligns with the category of "nuclear power plant", but not with the most severe category of "nuclear power plant shutdown after accident". The number of blocks in Fig. 34 on the NPP line is not meant to be a fixed number, rather the length of the line indicates the number of phases relative to other types of plants.

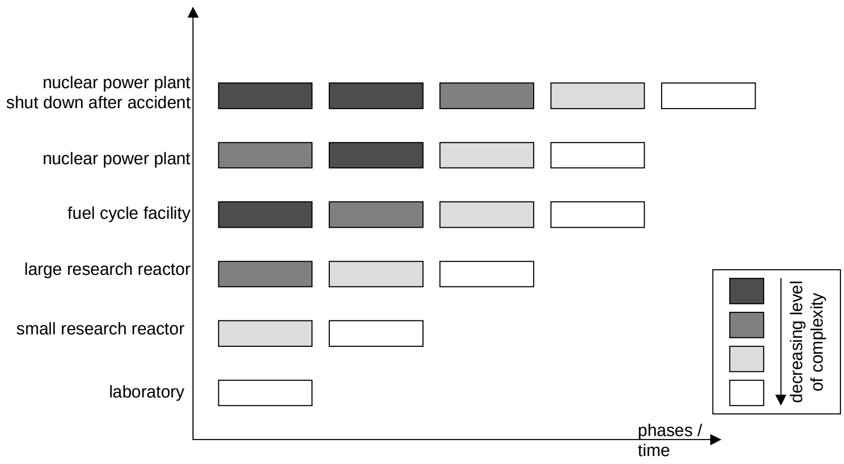

FIG. 34. Illustration of the application of the graded approach to various types of facilities (see Volume III).

The DeSa methodology also shows a relationship between different types of administrative control measures, as presented in the main report. This can be summarized by the flow chart below in Fig. 35.

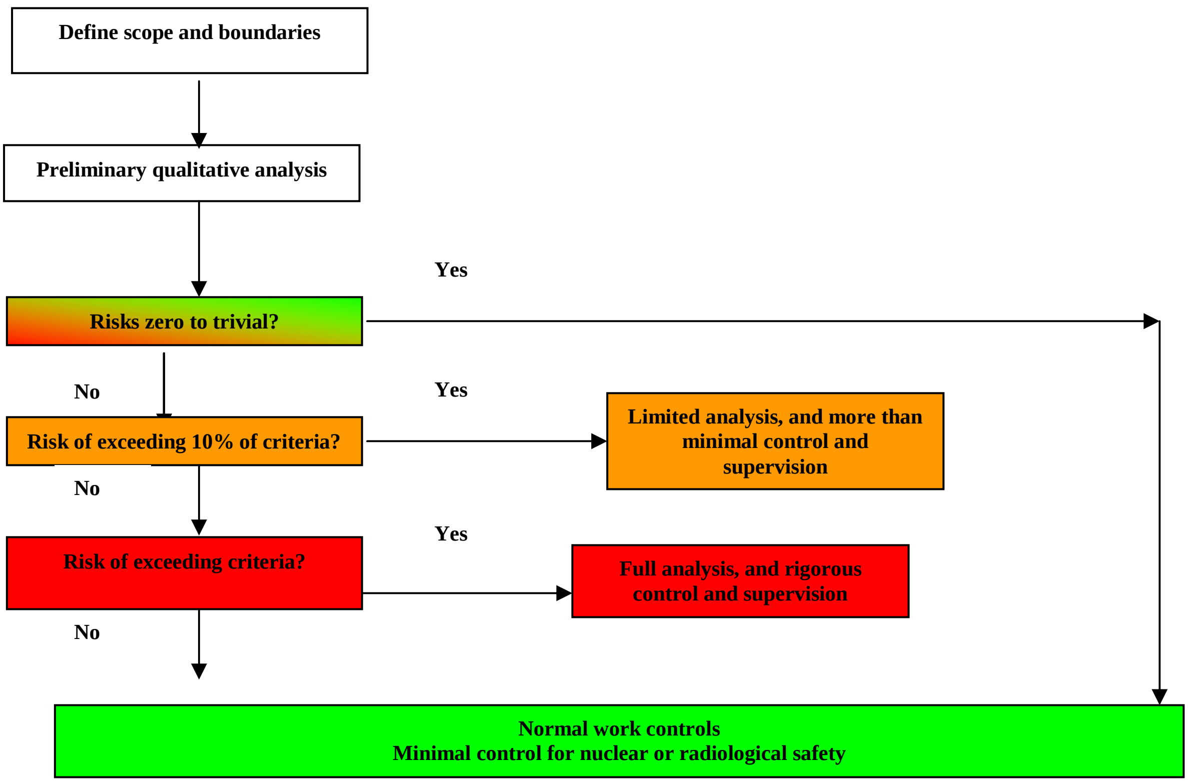

FIG. 35. Relationship between levels of control and safety category.

8.3. THE GRADED APPROACH IN THE RADIOLOGICAL CHARACTERIZATION AND DATA ACQUISITION OF THE FACILITY

The NPP had an excellent operational record with complete continuity of ownership and management. Similarly, the transition from power generation, through defueling and plant decontamination was well managed.

There is therefore good knowledge of the plant and the history of operations, modifications and preparation for decommissioning.

For the systems selected to form the basis of the NPP Test Case, facility characterization by analysis is appropriate because of the uncertain science of trying to theoretically deduce the transfer of contamination around the primary circuit.

8.3.1. Review of historic documents

The design, construction, operational life and modifications to the NPP were all well documented. These were reviewed for the NPP’s Safety Analysis Report for Care and Maintenance, and were reviewed again for the NPP’s Safety Assessment Report.

8.3.2. Characterization

Characterization has been carried out by sampling and by direct measurements, as described in Section 3.

8.3.3. Calculation of activation

This is not relevant to the systems chosen for the two systems 321 and 322, but would be for other NPP systems and structures. Possible examples would be activation of the core barrel, or activation of the civil structure of the building to assess whether it can be removed from regulatory control.

8.3.4. Preparation of the sampling and characterization plan

This activity was graded according to the radioactive inventory (i.e. hazard). System 321 has been extensively characterized by radiochemical analysis and by radiation survey. For system 322, the analysis has been confined to the extent necessary, to show that the system has no contamination or activation products within it.

8.3.5. Performing the direct measurements

This is adequately addressed in the text above.

8.3.6. Determination of radionuclides to for analysis

In theory, the analysis of radionuclides can be restricted or selected according to the hazard that each one poses. In practice, there are some caviats in this argument:

The selection for grading can only be made in the light of the knowledge of the radionuclides. A comprehensive survey is therefore necessary for a selection to be made.

Characterization for waste management and disposal purposes requires anyway a very comprehensive characterization.

8.3.7. Correlation method for measurement of hard to detect radionuclides

This is applicable to the systems 321 and 322 selected for the NPP Test Case, because of the presence of hard to detect activation products, such as Ni-63.

8.4. THE GRADED APPROACH IN CARRYING OUT THE SAFETY ASSESSMENT

8.4.1. Screening and grouping of hazards

The identification of hazards was described in Section 4.2. The HAZOP process applied in the NPP Test Case relies on the quality of the information laid in front of the HAZOP team and also on their expertise and judgement about the relevance of a potential hazard. In deciding that some potential hazards were judged to be important enough to be analysed in detail, while others were not, the HAZOP team was applying a form of a graded approach.

8.4.2. The complexity of the approaches and calculation methods

Detailed analysis has been performed of the two systems of the unit 1. One of these holds a significant inventory of radioactive material and the other has none. By using the contaminated system as a “bounding case”, it was possible to then easily state that the non-radioactive system needed no limits and conditions associated with it uniquely.

The calculation methods were:

(a) A detailed database of decommissioning tasks and dose rates from rooms and plant items – to produce worker doses from normal operations.

The analysis tool used to calculate normal consequences is quite detailed. This is because of the broad need to be aware of and to control worker dose in detail. This need arises not so much from the level of hazard, but from the requirements to plan work and effectively manage resources. As for the application of the graded approach, the difference in the nature of the systems 321 and 322 is reflected in the level of detail of the assessment. For system 321 dose rates at several points per object are needed. For system 322, only room dose rates are needed.

(b) A detailed database of decommissioning tasks and contamination levels – to estimate public doses from normal operations.

This also arises automatically from the tools needed to plan decommissioning work, and is embedded in the OMEGA code. Again, the difference in the nature of systems 321 and 322 is shown in the level of analysis. For system 322, the non-active system, release fractions are simply set at 0.00 and no activity is released.

(c) Accident conditions analysis

The safety assessment identified those reasonably foreseeable accident conditions that could occur during planned decommissioning activities. The evaluation then grouped these accidents into categories to assess the maximum unmitigated radiological exposure that could result. The graded approach adopted here for the safety assessment is to keep the assessment as simple as possible to limit assessment effort, while at the same time ensuring that the assessment results are sufficient to evaluate risk and identify safety measures that will ensure risk to workers and public are optimized and ALARA. It is important to ensure that by application of the graded approach that it does not compromise safety and or compliance with the relevant safety requirements and criteria.

8.5. ALTERNATIVE APPROACHES FOR DEMONSTRATION OF SAFETY

In the safety assessment presented in this Annex, the radiological inventories and other hazards were well characterized, thus avoiding the need for overly conservative assumptions. The identification of accident grouping was also used to reduce the extent of safety assessment by grouping accidents with similar initiating events. A deterministic approach was followed rather than set targets for frequency of accidents or for risk outcome, as would be the case with an operational power reactor. If the NPP Test Case had followed a more probabilistic based approach, then it would have allocated categories of consequence. For example, the initiating event frequency and consequence evaluation allocated accident scenarios into four risk classes. The risk classes are read from the matrix as depicted in Table 31.

— Class 1 off site consequences;
— Class 2 significant on site consequences;
— Class 3 minor consequences in facility; and
— Class 4 insignificant consequences.

TABLE 31. CATEGORIZATION OF RISK CONSEQUENCES ACCORDING TO THE INITIATING EVENT FREQUENCIES

Consequence Level	**Beyond Extremely and Unlikely < 10^{-6} per year**	**Extremely Unlikely 10^{-4} to 10^{-6} per year**	**Unlikely 10^{-3} to 10^{-4} per year**	**Anticipated 10^{-1} to 10^{-3} per year**
High Consequence	3	2 SAR	1 SAR	1 SAR
Moderate Consequence	4	3	2 SAR	1 SAR
Low Consequence	4	4	3	3

Note: SAR= Safety Analysis Report

The potential consequences could be divided in 3 categories as follows:

— High consequence for public (100 mSv to 1000 mSv) and for workers (> 1000 mSv);
— Moderate consequence for public (10 mSv to 100 mSv) and for workers (100 mSv to 1000 mSv); and
— Low consequence for public (1 mSv to 10 mSv) and for workers (10 mSv to 100 mSv);

The initiating event frequencies are graded as follows (see Table 31):

— Anticipated (1 x 10^{-1} to 1 x 10^{-2} per year);
— Unlikely (1 x 10^{-2} to 1 x 10^{-4} per year);

— Extremely unlikely (1 x 10^{-4} to 1 x 10^{-6} per year); and

— Beyond extremely unlikely (<1 x 10^{-6} per year).

The output from such an assessment could then be used to drive further design work and the addition of further active mitigation systems to reduce risk to a member of a critical group during decommissioning.

For example, a high consequence event which is anticipated at beyond extremely low frequencies would be classified as a risk Class 3. Similarly, a moderate consequence event at extremely low frequencies would be classified as a risk Class 4. For risk Class 4 no further assessments is required, since the Safety Management Programme (SMP) is regarded as adequate to optimize and control the risk to be as reasonably low as achievable, taken cost into consideration. For Class 1 and 2 events, a detailed safety assessment is required.

During the safety assessment process engineering and administrative control measures are identified and their mitigation effects taken into consideration in the mitigated accident dose assessment. To reduce the risk class, more mitigation measures are added and the effects are recalculated. This process is repeated until the resulting Risk Category is Class 3 or 4 and the activities are thereby optimizing the process to reduce the effects of radiological exposure to a minimum, taking cost into consideration.

9. CONFIDENCE BUILDING IN THE SAFETY ASSESSMENT

9.1. QUALITY MANAGEMENT

The management system is not specifically discussed in this document. The management system applied during normal conditions would be maintained during decommissioning, which has to comply with the requirements of the Regulatory Body. Such a system would typically make provision for organizational and management responsibilities, the appointment of suitably qualified and experienced persons a document configuration and control measures, the control of the all activities, keeping of records, checks and balances, traceability requirements and a non-conformance management measures.

Furthermore, management system need to ensure that all activities are performed in sufficient quality and in accordance with the legal and regulatory criteria – e.g. authorization of activities and modification of licences, design control, justification of release of land from regulatory control, clearance of material, material accounting, waste management and minimization, safeguards, waste minimization programme, radiation protection programme, environmental monitoring, security, access control, transport of radioactive material, in-service inspection, maintenance, care and maintenance (for the period of institutional control after the completion of decommissioning), staffing and training, emergency preparedness and response, fire protection, etc. Some of these activities have been mentioned but not included in detail in the NPP Test Case. Nevertheless, the management system applied to the development of the safety assessment for the decommissioning of the two systems 321 and 322 was followed and is presented in this Section.

In accordance with the NPP decommissioning plan a safety assessment team was comprised of qualified operational personnel and safety experts. It was assembled to plan and evaluate the safety of the proposed decommissioning activities for the two systems. Facility characterization was included in the preparatory work in the form of a radiological survey, material sampling, and review of operational history. International practice and recommendations were followed [1, 2, 4, 5, 9, 11] and an independent review of the NPP Test Case report was conducted by the Regulatory Review Working Group and the Graded Approach Working Group of the DeSa project.

9.2. INDEPENDENT REVIEW AND APPROVAL PROCESS

As part of the approval process, the safety assessment would normally be subject to an independent review by an independent party to ensure that the assessment addresses all safety aspects adequately. Therefore the operator has to demonstrate that:

(a) The input data and assumptions are valid;

(b) The assessment reflects the actual state of the facility and the decommissioning activities;

(c) The limits and conditions derived from the safety assessment are adequate to the decommissioning activity; and

(d) The safety assessment is kept updated to reflect the evolution of the facility and of knowledge and understanding about it.

This would normally include a review of the whole safety assessment, which would amongst others include the review of the methods used for identification of the initiating events, verification of calculations, review of the adequacy of the derived engineering measures, administrative measures and the safety management programmes to be applied during decommissioning of the facility in accordance with the predefined end states and within the confines of the authorization criteria of the Regulatory Body. In some countries the independent review by the Regulator Body is regarded as adequate.

In the case of the DeSa project the NPP Test Case was developed by a group of experts with various and broad experience. As mentioned earlier, the NPP Test Case report was reviewed by the Regulatory Review and Graded Approach Working Groups for independent scrutiny in order to verify completeness but also to ensure consistency in approach with the other test cases. The comments received have been incorporated into this document.

Decommissioning and closing of nuclear facilities, in particular when there is a release of the site for other applications, is often of concern to local and regional authorities and to the surrounding population. Such concerns are particularly likely if the site contains a large quantity of low level waste. Therefore, stakeholder participation needs to be pursued in these cases [21].

Implementing the three pillars of trust – safety, participation and local development – is the key to successful decommissioning projects. In general, there is limited international [2, 5, 22] or legal requirement to involve interested parties directly in decommissioning decisions; though (at least in some cases) there can be substantial consequences for local communities in terms of decreasing employment rate and an eventual reduction of revenues for the host municipality. On the other hand, interested parties do generally have the legal right to be involved in the consequential decision about the strategy for decommissioning the shut down plant – i.e. the actions taken to facilitate the end of regulatory oversight of the facility – typically through participation in an environmental impact assessment process. As the decision process moves from issues concerned with the shutdown of the plant to strategies for its dismantling, the importance of purely local interests becomes greater. For this reason, it is necessary to develop dialogue and co-operation among regulators, implementers, and local interested parties as early as practicable. The host municipalities for nuclear facilities tend to focus their attention on the day-to-day issues arising from the activities at the plant and, as regards decommissioning, will generally favour the early reuse of the site for economic or cultural purposes. As in other phases of the nuclear facility life cycle, it is necessary to develop trust among interested parties in decommissioning and dismantling projects. This may be accomplished through involving local and regional actors in decision-making, but also in monitoring activities, so as to have a better grip on the continuous changes taking place at the site. Transparency is needed indecision-making and in the respective roles played by regulators, implementers and local authorities. At all times, proactive information, and efforts to "translate" technical information into language meaningful to the chosen audience, will contribute to building mutual understanding and trust. Partnership arrangements, by which institutions enter into structured relationships with local communities, have been found beneficial. Decommissioning in both nuclear and non-nuclear areas may be viewed as an opportunity to improve the sustainability of the host community. The creation of added cultural or economic value can contribute to increasing quality of life over the years. More recent designs integrating reflection on the end use of the facility and site, or technical provisions for quick transitions to other types of facilities, provide better assurance to the host community that there will be flexibility in future planning capacity. There is an increasing recognition that, although there is a gradual convergence in terms of the technical approaches to decommissioning, and in the overall decommissioning objectives, there is also a need to retain a degree of flexibility as these are implemented, in order that local considerations can be adequately accommodated. For this reason, actual practices will necessarily differ from context to context [23].

The NPP Test Case has not engaged in consultation with other interested parties; expect the independent review as described above. In real decommissioning projects, the involvement of interested parties will be essential for the successful implementation and completion of the project.

10. SUMMARY AND LESSONS LEARNED

The development of the NPP Test Case was performed with the following four main objectives. The first aim was to illustrate the application of the safety assessment methodology developed as part of the phase 1 of the DeSa project to an NPP, and in doing so to illustrate the application of and benefits of the graded approach. The second aim was to present an assessment of proposed decommissioning activities at the NPP to give confidence that the DeSa safety assessment methodology can be used to prepare a safety assessment that can be submitted to a Regulatory Body, showing that the activities described can be conducted safely and thus that the regulatory body could issue a licence for this work. Thirdly, it aimed to provide an illustration of the application of the graded approach and its benefits. Finally, and specific to the NPP Test Case, it aimed to illustrate that phasing can be applied within a large decommissioning programme where it is not possible or desirable to carry out a detailed safety assessment of all the proposed decommissioning activities at once in order to be able to start the work.

Taking into account the DeSa project constraints, the test case used for the illustration of the assessment methodology two specific systems 321 and 322 from the unit 1 of the volunteered NPP. The NPP Test Case covers immediate dismantling of the specified systems, with the aim for their removal and the decontamination of the rooms they were in so as to be able to satisfy the end-point objective. This in turn contributes to the achievement of the overall and state objective which is that the building structures that remain after decommissioning can be released from regulatory control. Demolition or re-use of the existing civil structures can be a choice made later by the owner without any constraints remaining from the nuclear activities that were formerly carried out on the site.

The methodology developed in the DeSa project for the assessment and evaluation process was followed in this test case. It was quantitatively demonstrated what the effects of the application of individual safety significant components, administrative measures and limiting conditions of operation on the resulting effective dose would be. This was done using a purely deterministic approach to the safety assessment and there was no attempt to assign risk categories to tasks or plant systems. It was also quantitatively demonstrated how the implementation of the identified control measures reduced consequences and risk to workers and the public to acceptable levels. Mitigation was assumed for both the consequences to workers of both normal planned activities and of accident conditions. The results are illustrated in the assessment in terms of unmitigated and mitigated dose to both workers and the public, both for normal planned activities and for accident conditions.

The NPP Test Case report has a structure that follows the structure set out in the DeSa safety assessment methodology (see main report). This was found to be appropriate. By following this structure an independent regulator is taken systematically through the stages of the safety assessment process, relevant criteria are identified and compliance with them is shown. The necessary conditions and assumptions for ensuring this compliance are identified and can form the basis of inspection by a Regulatory Body once the decommissioning has started.

It was demonstrated how through the application of the graded approach, effort applied for analyzing the consequences associated with decommissioning could be minimized when the bounding criteria are well established and applied. As an example, this test case has a thorough evaluation of the public dose consequences which shows that the public dose below the criterion by many orders of magnitude. Since the system 321 represents a bounding case for loose contamination in the NPP systems, it will not be necessary to repeat the public dose calculation for any of the other systems. The practical use and benefits of the graded approach was clearly demonstrated through:

- The application of the of the safety assessment to the accident conditions with the highest consequence only. It was clearly possible to dismiss some initiating events very early in the process of hazard identification.
- Not using a probabilistic risk assessment approach. Early quantification of the unmitigated consequences was used to set a boundary, and from this it was clear that a simple approach of barriers and defence in depth was adequate for this safety analysis.
- It was possible to avoid risk categorization, as this requires a calculation of risk so that a category can be assigned to the decommissioning activities.
- The selective approach to plant characterization that is linked to the hazards emerging from the particular systems being decommissioning.

The depth of safety assessment required depends on the complexity of the plant and the hazards associated with the decommissioning activities. In the case of the NPP, the complexity of the decommissioning programme resulted in a safety assessment that considers only one part of the overall decommissioning plan.

On the basis of the NPP Test Case, some additional points can be made:

- The importance of input data was demonstrated in each scenario. On the one hand it is important to use conservative input data in order not to overestimate the results in calculations in order to demonstrate confidence in uncertainties. On the other hand, unrealistic and over conservative data can lead to unnecessary effort in the amount of assessment work required to demonstrate compliance with the basic safety criteria. This was illustrated by the analysis of various layers of mitigation. The end-point was identification of a number of safety measures that gave a good margin on compliance with criteria, but were not onerous or disproportionately expensive to implement.

— The amount and type of information available about the radiological condition of the NPP systems and the rooms that they are in is very detailed and complete, due to the post-operational activities for sampling and characterization. In addition, the safety measures and procedures were developed using conservative input parameters.

— This test case did not deal with materials management as this was beyond its scope, other than recognizing the existing waste acceptance criteria and stating that the waste produced was added to existing site waste inventories. The existing waste acceptance criteria did determine decommissioning activities, as plant systems had to be cut into pieces that could be handled within the waste management system at the NPP.

The safety assessment demonstrates that the decommissioning of the NPP systems does not impose unacceptable hazards (e.g. leading to effective doses in excess of relevant constraints, criteria and limits) or undue burdens on future generations, if additional safety measures are implemented. For this specific facility, the waste is entered into the existing waste management system of the NPP, which in turn is linked to a national infrastructure. The end state of the overall decommissioning project is that buildings are left which are outside of regulatory control and the site owner is free to determine the future use of the site.

The applied method demonstrated that various engineering measures, administrative measures and safety management programmes can be applied, and indeed must be applied to ensure that the conditions and assumptions set out in the safety assessment are to be met in the conduct of the decommissioning activities. The test case demonstrates that, through the application of the methodology, the most appropriate and effective mitigating factors could be identified and implemented, thereby optimizing the amount of decommissioning activities and the associated effects. The assessment demonstrated that the DeSa methodology could be applied effectively to facilities of various types, sizes and complexities to identify safety significant components and structures, to evaluate safety measures and demonstrate compliance with specific regulatory requirements.

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APPENDIX I: ASSESSMENT OF WORKERS' DOSE IN PLANNED DECOMMISSIONING ACTIVITIES

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I.1. INTRODUCTION

This appendix presents the methods and results of the evaluation of exposure of workers during the planned decommissioning activities of the NPP Test Case. The scope of the evaluation and optimization are the planned decommissioning activities for dismantling of systems 321 and 322 as set out in Section 3. The computer code OMEGA was used for these purposes.

The decommissioning parameters related to safety assessment must be incorporated in the set of calculated and optimized parameters. The purpose of the calculation and optimization of safety related decommissioning parameters is to demonstrate that the planned decommissioning activities can be carried out safely, i.e. the dose to a worker will be lower than the authorized limits 20 mSv per year.

The selection and implementation of decommissioning strategies for nuclear facilities requires calculation, evaluation and optimization of decommissioning parameters with a sufficient level of detail for the actual stage of planning. The accuracy increases from the preliminary stages of feasibility studies up to the final detailed decommissioning plan. General data needed for planning of decommissioning activities are:

- Costs;
- Manpower;
- Dose for personnel;
- Dose for public;
- Personnel needed for performing the decommissioning activities (professions, amount per profession);
- Waste resulted from decommissioning (types, amounts, radiological data);
- Consumable items (materials, electricity, other technical media, etc.); and
- Equipment needed for performing the decommissioning activities.

The cost of decommissioning is the central and most important decommissioning parameter into which all input data, having influence on decommissioning process are transformed. Manpower and personnel data (professions and amounts) are "neutral" parameters, significantly independent of facility, site and national conditions. Waste data are used for planning and organization of waste management, consumable items and equipment data, needed for performing of decommissioning activities, are all used for planning of support activities for the decommissioning tasks. From the safety point of view, the following parameters are needed for an evaluation of the safety of decommissioning activities:

- Doses to workers;
- Gaseous effluents and dose to public; and
- Liquid discharges (waste water from the site) and dose to public.

Doses for workers are calculated based on calculated duration of individual planned decommissioning activities and on the radiological conditions at the working place, both for external and internal exposure. Individual radiation protection means are taken into account. Optimization of the dose uptake can be accomplished by managing of number of personnel for performing the work, by pre-dismantling decontamination of systems or by application of remote techniques. The calculated data are used for demonstration that the planned decommissioning activities can be performed within the limited values for exposure of individual personnel.

The data for gaseous effluents are calculated in order to present that the influence of planned decommissioning activities on the critical group of public is within the limited value, based on existing exposure pathways for evaluation of migration of radionuclides to the critical group under local conditions. The gaseous effluents are calculated as radioactivity per individual radionuclides at the discharging point of the central ventilation chimney of the nuclear facility. These data can be compared with the authorized limits of gaseous effluents for the site. Alternatively the data of gaseous effluents can be used for calculation of dose to public using other pathways than those from the central ventilation chimney. This NPP Test Case uses a separate code to assess doses to the public, as shown in Appendix II. The role of the OMEGA code has been to provide a set of data that are common between the two assessments given in Appendices A and B.

The data for liquid discharges can be calculated as volumes of discharged waste waters from decommissioning activities. The discharging of waste waters is limited according to individual radionuclides. The values of limits are derived based on facility specific scenarios for the critical group for radionuclide intake from water-based scenarios. If it is demonstrated that the specific radioactivity of discharging waters is under the limited value, also the dose uptake of the public is under the limited value. The calculated data for liquid discharges can be used also for evaluation of other exposure scenarios.

For the NPP Test Case, the computer code OMEGA was used for the evaluation of potential exposure of personnel during normal (planned) decommissioning activities. Safety aspects of hazardous situations were evaluated separately and the dose to public, based on gaseous and liquid effluents was evaluated using the DecDose computer code [12]. This is shown in Appendix II. Appendix I presents the general properties of the OMEGA code, methods for evaluation of collective dose for personnel and methods for evaluation of individual doses in order to demonstrate that the individual effective dose for each worker involved will meet the safety criteria for individual workers.

I.2. THE OMEGA COMPUTER CODE

The basis of the OMEGA code is that the standardized structure [16] is in principle the complete list of decommissioning activities which make up the scope being assessed. This is the base for the standardized calculation core which is in internal structure equal for all decommissioning options. What is specific for the individual decommissioning option is the work breakdown structure (WBS). The WBS can be defined individually for each option by linking the WBS items to the grouped or non-grouped calculation items of the standardized calculation structure, including allocating of calculated data. A tool (WBS interface) was developed for transforming the WBS into the Gantt chart of the decommissioning option in MS Project software and for allocating the calculated data to the Gantt chart. Optimization of the decommissioning options and management of time is accomplished by defining the time structure in the Gantt chart. From the point of view of evaluating the safety of workers during performing the decommissioning activities, this optimization tool can be used effectively for optimization of number of working groups, number of working shifts in order to lower or keep the limits of exposure for individual workers or in adjusting the duration of deferred dismantling.

The top level structure of the OMEGA code is presented in Fig. 36.

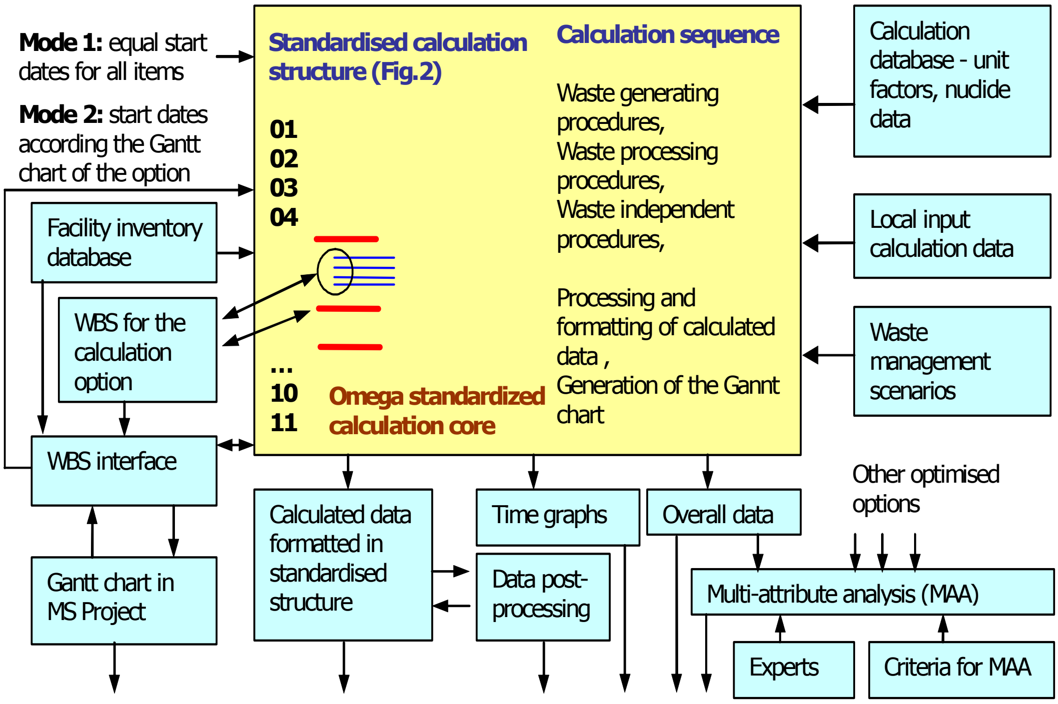

FIG. 36. Principal scheme of computer code OMEGA.

The costing methodology as implemented in the computer code OMEGA, is based on calculation modelling of the decommissioning process including the waste management [Ref. 14]. The main features of the code are:

— The calculation structure implements the standardized cost items structure for decommissioning, issued jointly by OECD/NEA, EC and IAEA, 1999 [Ref. 16]. The calculated costs are transparent, traceable and comparable with other decommissioning projects. The calculation structure is one compact package and includes also the waste management.

— The calculation process is sequentially linked-up in such a way that it simulates real decommissioning process flow and relevant material/radioactivity flow. The calculation items are linked to the material and radiological data of the inventory database and to the database of interim material/radiological items created during calculation, so the calculation use actual material and radiological parameters of waste.

— The calculation process is radionuclide-resolved and respects the radioactive decay of individual radionuclides. This enables to use the nuclide resolved limits for treatment/conditioning/disposal/release of materials within the material flow and enables to study the effects of deferred decommissioning. The decommissioning infrastructure is simulated by various calculation scenarios for management of radioactive waste. The scenarios include decommissioning activities linked from dismantling up to the disposal of conditioned radioactive waste or release of materials.

The methodology has “multiple options” capability, meaning that several decommissioning options can be defined for a decommissioning project in order to evaluate all possible scenarios of decommissioning in the frame of the project. Each decommissioning option is analysed, optimized and evaluated individually and the optimal option can be selected based on multi-attribute analysis.

An additional aspect of the advanced costing methodology, due to internal linking of the calculation process and due to compactness of the calculation structure, is the possibility to perform the sensitivity analysis which can define the margins of decommissioning costs by considering various levels of contamination, various radionuclide composition (effects of alphas), application of various decommissioning technologies, various durations of deferred decommissioning phases, etc.

The most recent presentation of the properties of the code and lessons learned from application of the code is in [13]. The principles of decommissioning costing implemented into the OMEGA code are:

- **What to do**: configuration of decommissioning activities of a decommissioning option in the standardized format using the templates of the standardized structure; computer generation of calculation structures and management of these standardized hierarchical calculation structures, which correspond to the facility structure of buildings – floors – rooms/cells - inventory items in rooms/cells.
- **How to do**: allocation of calculation procedures into the standardized calculation structure, definition of the conditions for calculation, computer evaluation of the radiological condition at the planned time of execution of individual activities in order to select manual/remote operations and to allocate relevant personnel protection; generation/editing of relevant calculation data and correction factors for manpower calculation.
- **In what sequence**: implementation of the concept of material and radioactivity flow modelling of the decommissioning process based on data linking of the calculation process and on definition of the calculation sequence which correspond to real primary and secondary waste flow in waste generation decommissioning activities (e.g. dismantling) and in waste management activities (e.g. sorting of waste) including the final waste disposal and material release and effluents from the process into environment.
- **At what time**: implementation of the concept of on-line optimization of the project work breakdown structure by direct data linking of the computer code to standard MS Project software.

One of the most important features of the code is the compactness of the standardized calculation structure which includes all activities of the decommissioning option and the activities for management of waste resulting from decommissioning. The compactness, internal linking and sequencing of the standardized calculation structure enables the calculation of costs and other decommissioning parameters for a decommissioning option within one calculation run including processing of the calculated data and generating of output data formats and the decommissioning schedule of the option.

The templates of the standardized calculation structure were developed for computer generation of the standardized calculation structures and for generation of the default values of input calculation data. These values are derived from the inventory data of the systems and structures and from the inventory radiological data updated by the code for the planned start dates of decommissioning activities.

The concept of radionuclide vectors was used in definition of nuclide composition of contamination, activation, mass/volume activity and dose rate. The radionuclide vectors are stored with the date of the definition and prior to application in the calculation; the radionuclide composition is recalculated for the decay of individual radionuclides. The recalculated radionuclide vectors are then used for generation of contents of individual radionuclides, used in radionuclide resolved calculation process. The effect of time and radionuclide composition can be thus evaluated directly. The concept is applied in calculation of exposure of personnel, effluents, in selecting the manual or remote operations and in waste management.

For management of material and radioactivity flow in the calculation process the code uses the principle of initial mathematical material and radiological partitioning of the individual inventory items of the facility inventory database into one–material components and the principle of linking and sequencing of individual activities of waste management. The one–material components are applied in relevant calculation items of waste management. The flow of interim waste forms, secondary waste and final waste forms can be monitored. The calculation process is radionuclide resolved and the radionuclide resolved limits for individual technological equipment for waste processing, acceptance limits for low level and intermediate level repositories and limits for release of materials into environment, can be applied. The gaseous effluents and waste waters discharges are included into the material and radioactivity flow evaluation.

I.3. THE DECOMMISSIONING ACTIVITIES FOR EVALUATION

I.3.1. THE STRUCTURE OF DECOMMISSIONING ACTIVITIES

The scope of evaluation of safety of normal (planned) decommissioning activities is the dismantling of system 321 and 322, as the critical operations from the safety point view. Other activities evaluated, such as the decontamination and radiation monitoring of building surfaces, are evaluated as safe from the point of view of annual limit of 20 mSv per worker. During dismantling, the workers are occurring in the dose rate fields of the contaminated equipment to be dismantled and are exposed to risk of inhalation of radioactive aerosols generated during cutting. Technical, organizational and personnel protection means are used to minimize the exposure of workers. The calculation process needs to take this into account these aspects. The accuracy of calculation on decommissioning parameters, including the safety related parameters, depends on the level of details of decommissioning activities evaluated. Best results can be achieved when the evaluated activities are the decommissioning activities like dismantling of a single inventory item, e.g. a valve. Individual radiological data (dose rate, contamination, radionuclide composition, etc.) related to this valve, could then be taken into account. This principle is generally described as bottom-up approach.

The bottom-up approach is considered as the most accurate method for evaluation of decommissioning parameters [Ref. 15]. The data are calculated at the lowest level of details of decommissioning activities and the results are consequently grouped up to the level of overall results. The breakdown of the dismantling activities can be organized based on following principles:

(a) Room oriented structure of dismantling activities;
(b) System oriented structure of dismantling activities; and
(c) Manually composed structure of dismantling activities.

(a) The room oriented approach for organizing the dismantling activities room by room involves:

— A set of preparatory activities prior to dismantling to prepare the working conditions and to support the dismantling in the room where equipment are going to be dismantled;
— Dismantling of the equipment according to the inventory content in the room; and

- A set of finishing activities to remove all instruments, materials, supporting systems and for cleaning the room after dismantling.

At the end, the room is ready for carrying out consecutive set of decommissioning activities, like decontamination of building surfaces. The list of preparatory activities for dismantling, as applied in the NPP Test Case is as follows:

- Survey of radiological situation in the room for confirmation of the data used in the planning of dismantling activities;
- Covering of the floor with protective foils to inhibit the contamination of the floor;
- Installation of local ventilation to suppress the aerosols from dismantling;
- Installation of scaffolding for dismantling activities;
- Installation of temporary electrical connections and media for dismantling;
- Delineation of cuts on equipment;
- Transport of dismantling tools to the room;
- Isolation of equipment from electrical connection or operating media;
- Preparation of dismantling tools for the work;
- Installation of protective tenting for suppressing the spreading of aerosols;
- Preparation (instructions) of the working group for the work; and
- Transport of containers for dismantled materials.

Carrying out the dismantling activities is organized based on the inventory content of the room. For example, in the case of dismantling, for each item in the inventory database (individual pipes, valves, motors, etc.) a separate item for calculation of decommissioning parameter is generated. After performing calculations for all inventory items, a set of finishing activities will follow, such as:

- Removal of protective foils on floors;
- Removal of local ventilation;
- Removal of scaffolding;
- Removal of temporary electrical connections and media for dismantling;
- Removal of protective tenting;
- Removal of dismantling tools for the work;
- Transport of containers; and
- Final cleaning of the room after dismantling.

A comparable set of preparatory and finishing activities is defined for other typical room oriented decommissioning activities like dismantling of embedded elements, dismantling of asbestos, contaminate or activated concrete. The extent of preparatory and finishing activities for building surface decontamination and radiation survey of building surfaces as applied in the NPP Test Case is presented in the Section 5 of the main report.

(b) System oriented approach for organizing the decommissioning activities is applied mostly for the components with large dimensions and complicated structure, like reactors, refuelling machines, large components of the primary circuit, etc. The procedures are specific for each component and normally the dismantling is the procedure inverse to construction. A typical structure of decommissioning activities is organized according to the individual construction sub-assemblies of the dismantled systems:

- Set of general preparatory activities;
- Dismantling of the construction sub-assembly No. 1:

- o Set of specific preparatory activities for the construction sub-assembly No.1;
- o Dismantling of individual components of the construction sub-assembly No.1;
- o Set of finishing activities for the construction sub-assembly No.1.

— Dismantling of the construction sub-assembly No. 2:

- o Set of specific preparatory activities for the construction sub-assembly No.2;
- o Dismantling of individual components of the construction sub-assembly No.2;
- o Set of finishing activities for the construction sub-assembly No.2.

And so on, until**…….**

— Dismantling of the construction sub-assembly No. N

- o Set of specific preparatory activities for the construction sub-assembly No.N;
- o Dismantling of components of the construction sub-assembly No.N;
- o Set of finishing preparatory activities for the construction sub-assembly No.1.

— Set of general finishing activities;
— Set of continuous supporting activities like radiological monitoring, waste removal; maintenance of dismantling equipment, etc.

(c) Manually composed structure of decommissioning activities is the definition of the specific decommissioning activities specific case by case.

I.3.2 THE DEVELOPMENT OF CALCULATION STRUCTURES

The calculation structure is a composition of above presented types of decommissioning activities. In order to harmonize the structure of decommissioning activities, it is recommended to implement the general structure of decommissioning activities as they are defined in the document [16]. The document contains the comprehensive list of decommissioning activities which can be identified in any decommissioning project. This principle was fully implemented in the OMEGA code.

The preparation of the calculation structure with implementation of the bottom-up approach requires the generation of large number of individual calculation items. This is facilitated effectively in OMEGA code by the implementation of methods for computer generation of the calculation structures based on inventory database and the templates of the standardized structure. The structures with a total number of calculation items on the level of 10^5 items can be generated within several hours.

From the point of view of evaluating the safety parameters for decommissioning, it is important, that they are calculated on level of above listed items and for individual professions involved in the working groups for dismantling. The individual professions of the workers will differ in working conditions during the decommissioning work, so the exposure of workers is profession dependent. These aspects are considered in the OMEGA code, so it is possible to go down in evaluating of the exposure up to the level of individual workers as members of individual professions.

The user can configure the standardized calculation structure in three steps using the templates which facilitates significantly the work of the user. The base for this work is the general standardized template which covers the decommissioning activities as defined in Ref. [16]:

(a) In the first step the user can develop the master template which is specific for a type of a nuclear facility.

(b) In the second step the user can adapt the selected master template to the standardized structure specific to the decommissioning option to be calculated. In this step the user can define as much calculation options as required for the evaluation within the decommissioning project. The option “specific standardized structure of decommissioning activities” involves also the prescriptions for generation of lower levels of calculation items, for allocating the calculation procedures and definition of calculation sequence.

(c) The third step is the computer generation of the standardized calculation structure for the calculation case. The typical feature of this structure is that it has the hierarchical structure of the buildings - floors - rooms/cells - inventory items in the room/cell in selected sections of the standardized structure, as required in basic definition of decommissioning activities in Ref. [16].

The generated structure contains also input calculation data with default values. After the generation, the user can review/edit the generated calculated structure and the generated default values of the calculation data and can define the extent of calculation by clicking in the individual calculation items. An example of the executive calculation is shown in Fig. 37.

This three-stage style of the work enables flexibility in developing the standardized calculation structures for any nuclear facility. The precondition is the inventory database for the nuclear facility with relevant structure and data needed for application of standardized structure. The methods for development of the inventory database with these properties were developed.

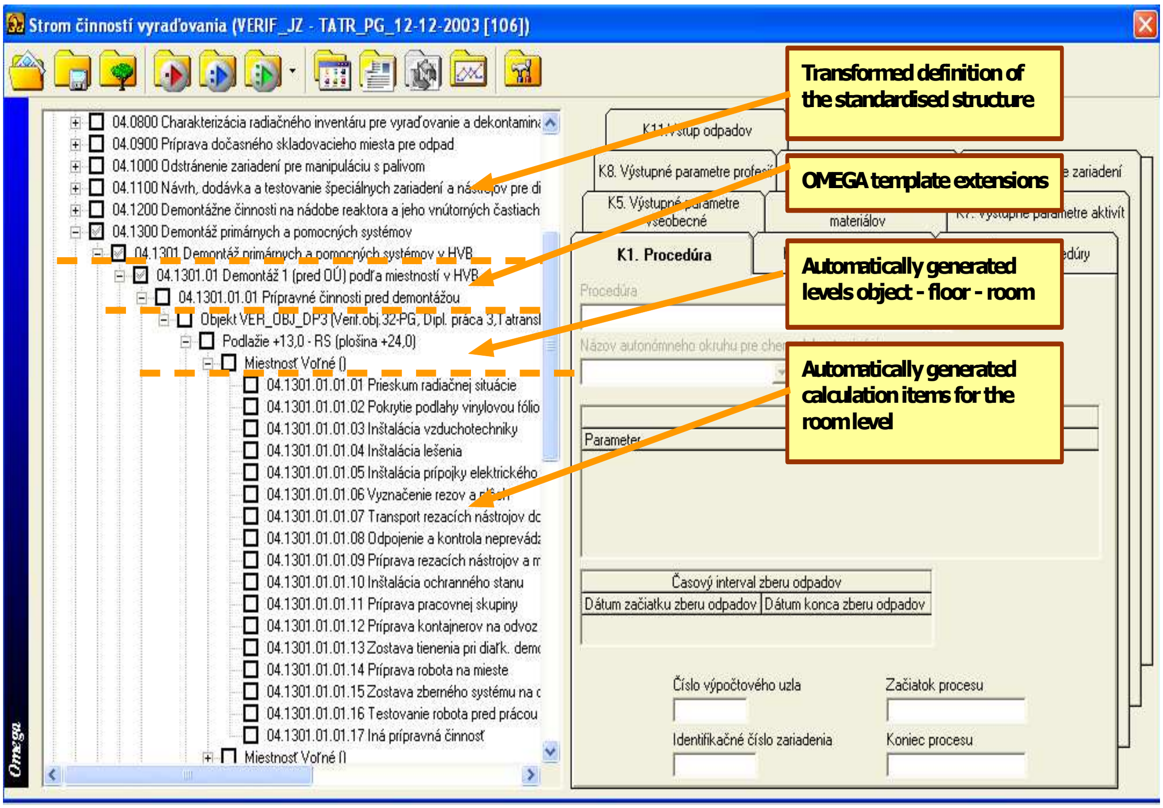

FIG. 37. Example of a standardized calculation structure.

I.3.3. THE INVENTORY DATABASE

The databases for systems 321 and 322 were developed in a hierarchical structure starting with objects of nuclear facility (only one object – the reactor building was used for these cases) through individual floors, rooms up to items of plant equipment in each room. The inventory database has two main components:

(a) The physical inventory; and

(b) The radiological parameters.

The physical inventory normally refers to identification of the inventory item in the frame object - floor - room system and to parameters like mass, surfaces, volume, categories of equipment while the radiological parameters refer to surface contamination of inner and outer surfaces, volume of induced activity, dose rates (all of them radionuclide resolved), dates of definition of radionuclide vectors and dates of definition radiological parameters and theirs radionuclide vectors.

Based on the description of the two systems (see Sections 2 and 3), the inventory data for the 321 and 322 systems were developed. The documentation involved the following data and information:

— Drawings of the central part of the NPP;
— Inventory data of equipment relevant for both systems which include the list of equipment, names, locations, identification of types, masses, surfaces, etc.;
— List of rooms, dimensions of some rooms and allocation of equipment to individual rooms;
— Radionuclide vectors of dose rates and contamination; and
— Dose rates at selected points at the equipment.

Some parameters were not present in the data available at first and were derived specifically for this test case application. After analysis of the drawings, the missing dimensions of room were introduced to the room database. The missing data for inner surface contamination of technological items was calculated (based on the delivered dose rates) in Microshield code for model types of equipment (pipes, valves, tanks etc.). Then the approximation of the contamination data for all inventory items was done. The dose rates in 0.5 m. distance from the equipment item was approximated based on dose rates measured on-site in selected points. The completion of technological item database was done by identification of equipment categories for all individual items of the systems.

The average dose rates for rooms of the system 321 were calculated as the average dose rate in 2 m. distance from all equipment in the room (calculated by the Microshield code). The average dose rates for room of the system 322 were calculated for rooms with heat exchangers, other data were estimated.

At the end of database preparation process, completed database tables (objects, floors, rooms, technological items) were imported to Oracle software in order to be able to generate the standardized calculation structure for the 321 and 322 systems.

I.4. THE CALCULATION OF DOSES TO WORKERS

I.4.1. PRINCIPLES OF DOSE CALCULATION FOR WORKERS

The principle of calculation of manpower used during decommissioning is the calculation of manpower components for individual professions of the working group in the first step and calculation of exposure of individual professions for specific manpower components. The procedure is the following:

- Calculation of manpower for specific decommissioning activities;
- Distribution of calculated manpower to individual professions of the working group;
- Extending the manpower for non-productive working time components; and
- Calculation of external and internal exposure based on local radiological conditions, protective means and manpower components.

Calculation of manpower and exposure is different for groups of decommissioning activities. There are three main types of decommissioning activities regarding to the exposure of workers:

(a) Hands-on decontamination and dismantling. The exposure is dominated by dose rates in working distances to the equipment to be dismantled. The exposure can be controlled partially by duration of stay of workers in the vicinity of the dismantled equipment, by application of more personnel or by application of remote dismantling or pre-dismantling decontamination of equipment to be dismantled.

(b) Work at technological facilities for radioactive waste handling and processing. The exposure is controlled by appropriate technical means at the working place for the workers, like shielding and can kept normally below annual exposure limits.

(c) Period dependent activities like surveillance, maintenance, management, technical support. The exposure is controlled by organization of the working time and can normally be kept below annual exposure limits.

The critical decommissioning activities from the point of view of calculation of exposure are the dismantling activities, where the personnel during carrying out the decommissioning activity, is present close to the contaminated equipment. The dose uptake during performing decommissioning activities is calculated as external exposure and internal exposure of workers. External exposure is calculated based on duration of activities and dose rates at working places and the internal exposure based on concentration of aerosols generated during decommissioning activities and theirs conversion factors, personnel protection means applied and breathing data.

The bottom-up approach discussed in Section 5, enables to calculate the exposure data on the level of a specific decommissioning activity. Each hands-on decommissioning activity is decomposed into specific productive and non-productive manpower components and for each manpower component, the relevant radiological data are allocated based on inventory data of the facility. The model time structure of a specific decommissioning activity, as defined in the calculation model, is presented in Fig. 38. The selected approach is the attempt to model the real sequence and content of decommissioning activities of this type.

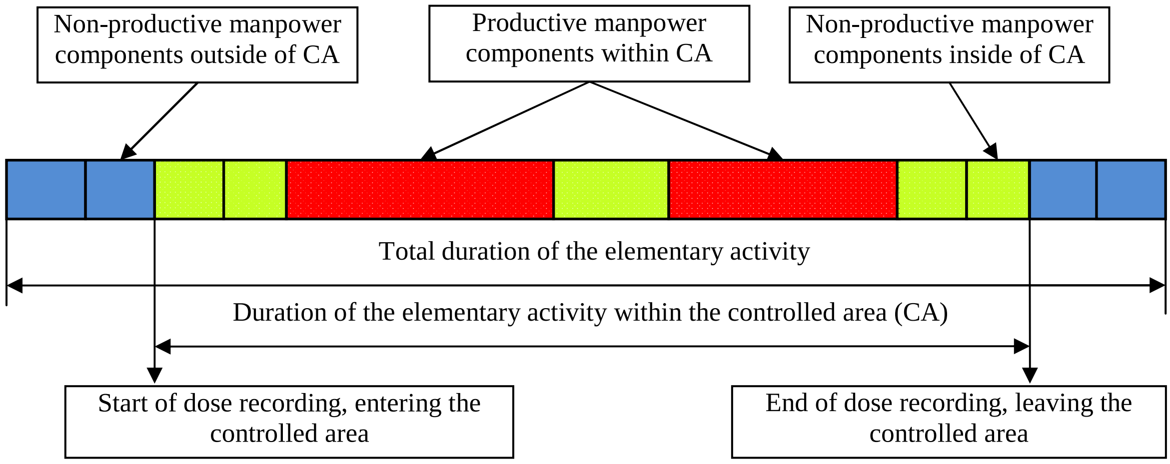

FIG. 38. Model working time structure of a specific decommissioning activity.

The first step is a calculation of productive manpower components. All other non-productive manpower components are calculated based on coefficients which increase the basic productive manpower. The productive manpower components, depending on working conditions, can be increased due to difference between the ideal and local working conditions, like working in ambient with ionizing radiation, etc.

Another aspects which is taken into account when calculating the exposure of workers, are the different working conditions for different professions of the working group. For each decommissioning activity, a working group is defined - professions needed and number of workers per professions. Various professions of the working group are exposed differently from the contaminated equipment, from the average dose rates in the rooms. In the case of the internal exposure the OMEGA code allocate a personal protection means, depending on local conditions, in order to decrease the amount of inhaled aerosols. Therefore the exposure of personnel is calculated on the level of the individual professions of the working group.

As the summary the principal scheme for calculation of exposure of personnel is presented in Fig. 39. The main input data needed for calculation of exposure of personnel are:

— Manpower components resolved according to individual professions of the working group;
— Composition of the working group – professions and number of workers per professions;
— Dose rate data – dose rate 0.5 m from the equipment (average working distance), average dose rate in individual rooms of the facility, average background dose rate of the facility; and
— Concentration of aerosols in working places, depending on release factors of cutting techniques, local and facility ventilation, personnel protection means allocated.

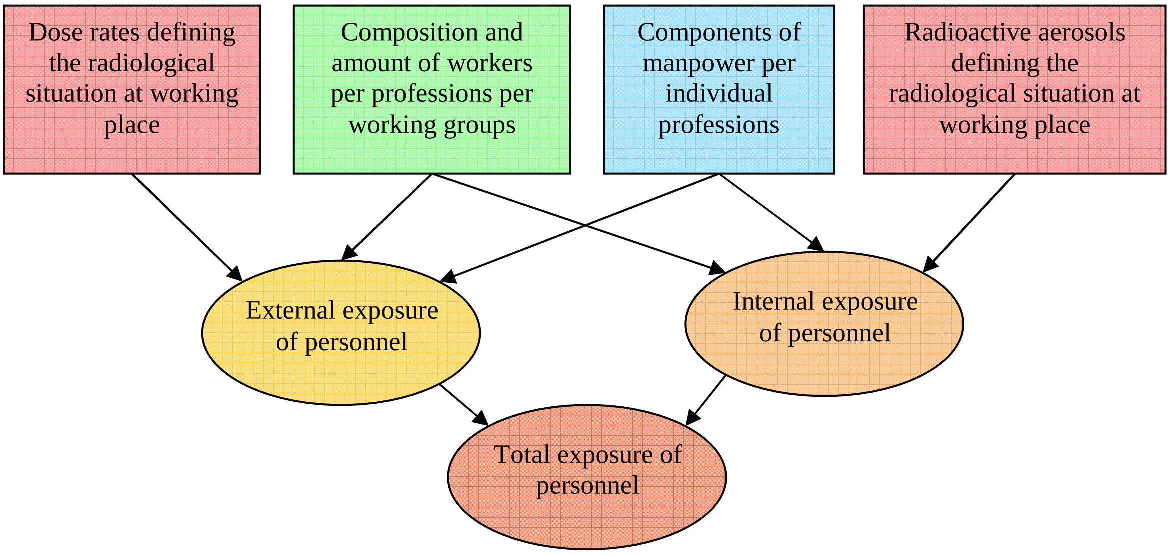

FIG. 39. Principal scheme for calculation of exposure of personnel.

I.4.2. THE CALCULATION OF MANPOWER FOR DECOMMISSIONING ACTIVITIES

The productive manpower is calculated based on principle of categorization of equipment to be dismantled. The decommissioning categories are typical representative equipment for a group of equipment similar as for size, material composition, thickness as other the main physical properties. Several thousand types of equipment, which can be identified in a typical NPP, are grouped into several tens of decommissioning categories based on similar physical properties. As the unit factor approach is applied in the OMEGA code, the unit factors for calculating manpower and other decommissioning parameters are defined for individual decommissioning categories. Calculation of the productive manpower for dismantling is then based on the mass of the equipment and the manpower unit factors for individual categories of equipment.

In this way, the duration needed for calculation of the dose uptake during dismantling is obtained. Various increase factors can be applied when calculating the productive manpower, like increase factors for work in dose rates, work on scaffolding, work in congested areas, work on complicated tasks, etc. Increase factor for the work in dose rate fields is significant from the point of productive manpower calculation. This increase factors depends on the level of dose rate from dismantled equipment and average dose rate in room where dismantling is being carried out. Increase factor has graded character corresponding to levels of dose rate. The default values of this factor used are listed in Table 32.

TABLE 32. DEFAULT VALUES OF MANPOWER INCREASE FACTOR FOR WORK IN DOSE RATE FIELDS USED IN OMEGA CODE

Dose rate range [μSv/h]	**Increase factor**
0 – 5	1
5 – 20	1.2
20 – 50	1.4
50 – 200	1.6
> 200	2.4

The manpower calculated as total manpower for the specific decommissioning activity is distributed down to the level of individual professions. These profession resolved manpower components are then used for calculation of exposure of individual professions of the working group. The principal scheme for calculation of the specific manpower components per profession is presented in the Fig. 40.

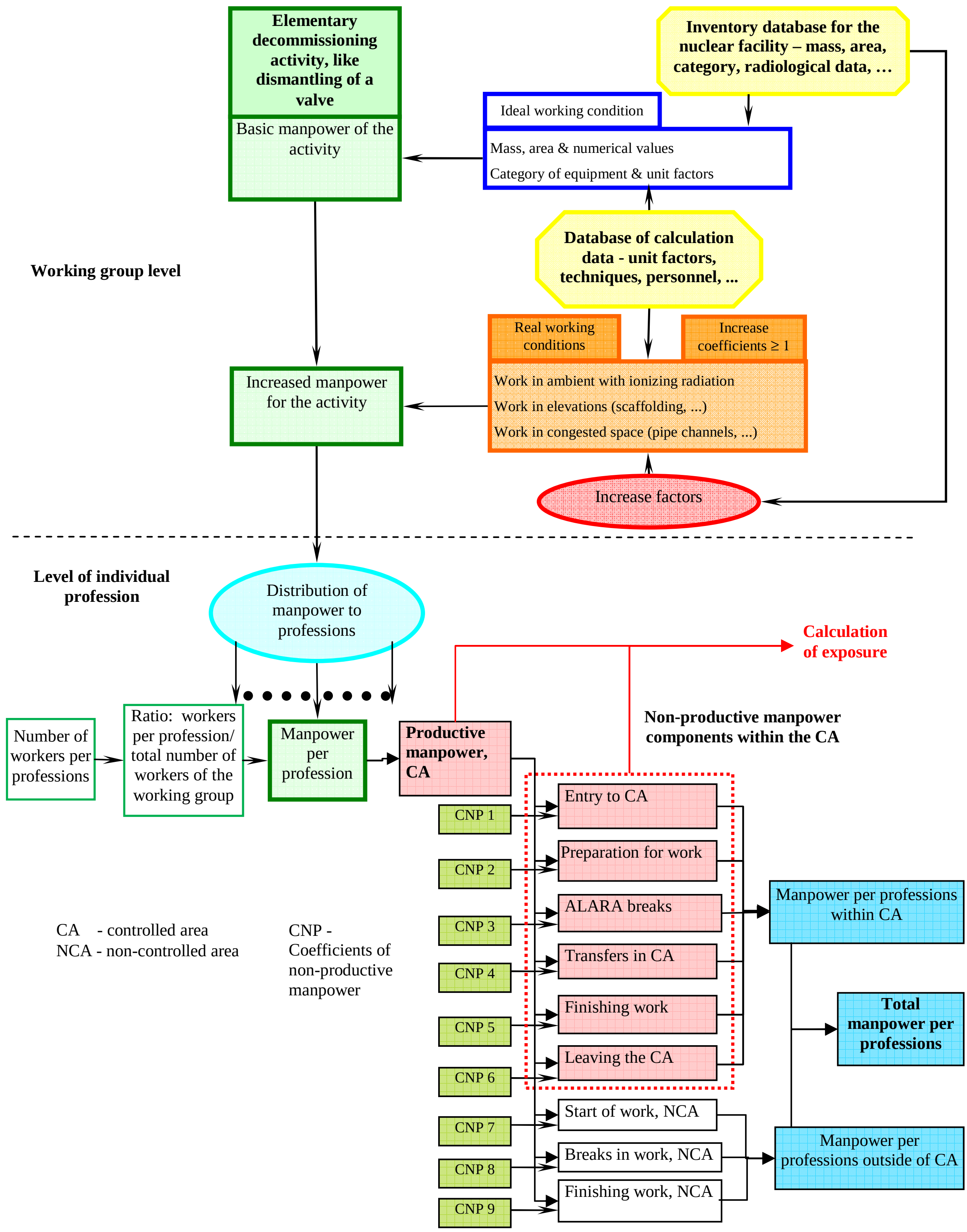

FIG. 40. The principal scheme for calculation of specific manpower components.

Normally for calculations, the remote controlled dismantling operations are selected for dose rates over 200 microSv/h for dismantling with longer duration. The manpower calculated for performing the dismantling is the base for calculation of manpower for non-productive working time components which also contributes to the total exposure of the personnel. The dose uptake for non-productive time components are calculated taking into account the dose rate of the background of the controlled zone. The coefficients of non-productive manpower as applied in the OMEGA code are presented in the Table 33.

TABLE 33. DEFAULT VALUES OF NON-PRODUCTIVE MANPOWER COMPONENTS

Non-productive Time Components	**Ratio to Productive Time Components [%]**
Entry to non-controlled area	3
Work preparation in non-controlled area	2
Work breaks in non-controlled area	6
Moving within non-controlled area	2
Entry to controlled area	3
Work preparation in controlled area	3
ALARA breaks	6
Work breaks in controlled area	4
Moving within controlled area	4
Work finishing in controlled area	3
Exit from controlled area	5
Exit from non-controlled area	3

I.4.3. THE CALCULATION OF EXTERNAL EXPOSURE

The dose uptake during planned decommissioning activities is evaluated for each of individual decommissioning activities and has components as follows:

- Dose uptake caused by the dose rate 0.5 m from the equipment (average working distance) to be dismantled during carrying out the dismantling;
- Dose uptake caused by the average dose rate in the room where the equipment are dismantled; and
- Dose uptake caused by the average dose rate in the background of the controlled area of facility.

The dose rate relevant for the calculation of dose uptake during the preparatory and finishing activities, is the average dose rate in the room and in the background of the controlled area.

The individual professions of workers are exposed in different way in accordance with the type of work they perform. The most exposed professions are those who directly perform the dismantling and are most exposed to the dose rate of the dismantled equipment. For other professions the average dose in the room is dominant. For the rest of the working time, the dose rate in the background of the controlled zone is applied. These conditions are taken into account in calculation of the dose uptake for individual professions of the working group and they are expressed by coefficients of effective stay in the working distance from the equipment and coefficients of effective stay in the average dose rate in the room.

These coefficients for dose rate from equipment ranges from 0.1 to 0.5, the values of 0.5 are allocated to workers performing the cutting and dismantling of the equipment, the lowest values are allocated to foreman of the working group. The coefficients for average dose rate from the room ranges also from 0.1 to 0.5, the lowest values are allocated to dismantlers, the highest for workers supporting the dismantling.

The dose uptake is calculated for individual decommissioning activities as a sum of dose items for individual professions of the working group for individual productive and non-productive components of their working time. The calculation is performed for each preparatory and finishing activity according to the rooms involved and for each inventory item in the rooms as recorded in the inventory database. The principle of calculation of dose uptake for dismantling activities is schematically presented in the Fig. 41.

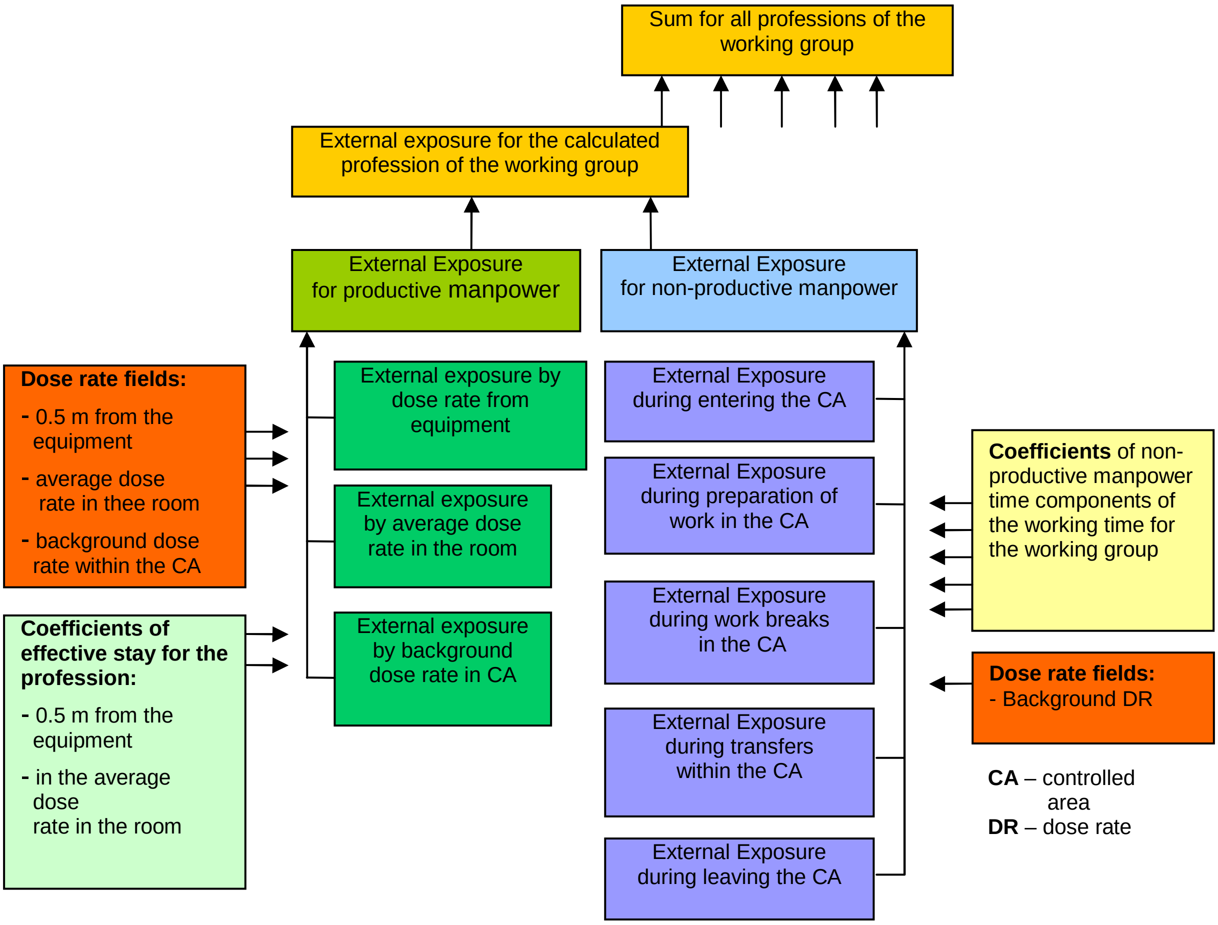

FIG. 41. Concept of calculation of the dose uptake for individual dismantling activities.

Normally, the calculation of the dose uptake is performed conservatively. It means that the dose rate in the room is used for dismantling of all items in the room. A methodology was developed in the OMEGA code to calculate to dose items, relevant for the room, more realistically by taking into account the decrease of the average dose rate in the room during dismantling of the equipment in the room.

I.4.4. THE CALCULATION OF INTERNAL EXPOSURE

Calculation of internal exposure is based on the following input data:

- Productive and non-productive manpower components;
- Volume concentration of aerosols at the working place;
- Average volume concentration of aerosols in rooms;
- Background volume concentration of aerosols in the controlled area;
- The breathing data;
- Conversion factors for individual radionuclides [Sv/Bq]; and
- Retention factors of protective means.

The manpower components are the same as in the case of external exposure. Volume activities of aerosols at the working place are calculated using the release factors of individual radionuclides from cutting, Section 2.2. Other data are the data from the database of calculation parameters. The retention factor for respiration protection using simple respirators is 0.9, for portable breathing air sets the factor is 0.999 and for whole body pressurised suits with external delivery of air it is 0.9999.

The code evaluates first what would be the internal dose for the worker if he would not have no protective means and based on calculated hypothetical value, the code allocate the protective means in order to keep the dose as low as reasonable achievable. The calculation of the effective dose from internal exposure is performed under the assumption that the worker uses the allocated protective means. Average volume activity of aerosols in the room and average volume activity of aerosols in the background of the controlled area are estimated values. The principle of calculation is presented in the Fig. 42.

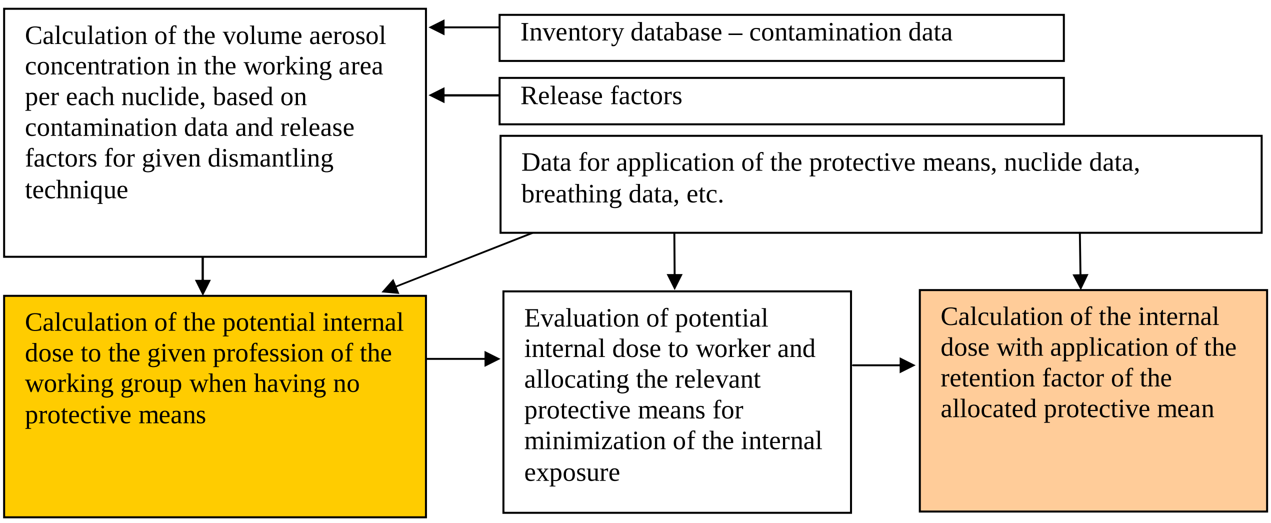

FIG. 42. Principle of calculation of internal dose during dismantling.

The process of calculation is repeated for each radionuclide in the nuclide vector, as is presented in Fig. 43.

Internal exposure per professions of the working group

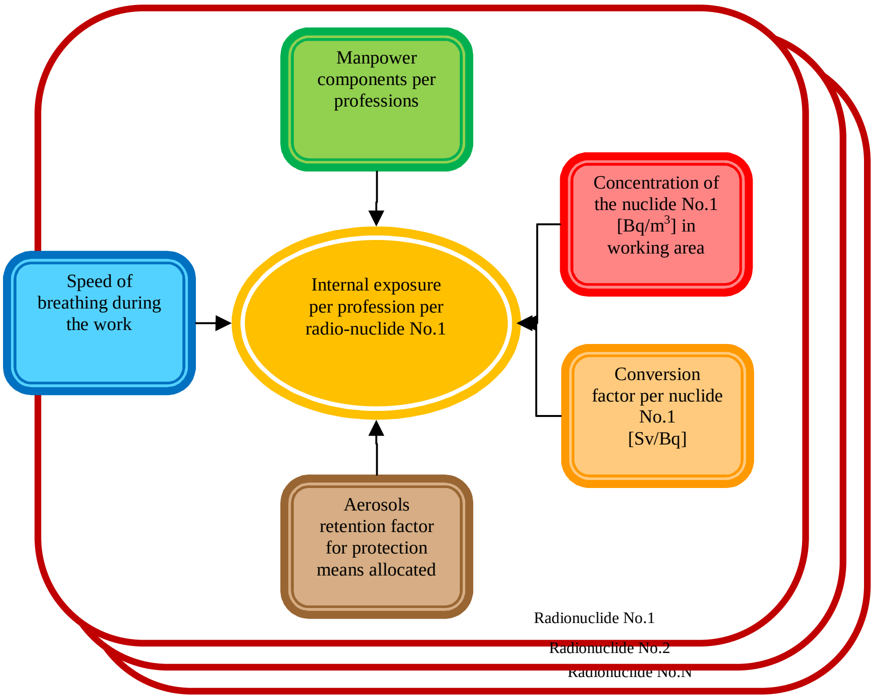

FIG. 43. Calculation procedure for internal exposure.

I.4.5. OPTIMIZATION OF DOSE CALCULATION

As discussed in Section 3.3, the individual professions of the working group are exposed in different way in accordance with the type of work they perform. The most exposed professions are those who directly perform the dismantling and are most exposed to the dose rate of the dismantled equipment. For other professions the average dose in the room is dominant. For the rest of the working time, the dose rate in the background of the controlled zone is applied. These conditions are in calculation of the dose uptake for individual professions of the working group expressed by coefficients of effective stay in the working distance from the equipment and coefficients of effective stay in the average dose rate in the room.

Normally, the calculation of the dose uptake is performed conservatively and the dose rate in the room is used for dismantling of all items in the room. A methodology was developed in the OMEGA code to calculate to dose items, relevant for the room, more realistically by taking into account the decrease of the average dose rate in the room during dismantling of the equipment in the room. The method corresponds to application of the ALARA principle when the equipment with the highest dose rate is dismantled as the first in order to decrease the resulting average dose rate in the room. The effect is presented on the Fig. 44. This procedure was applied also for calculations for systems 321 and 322.

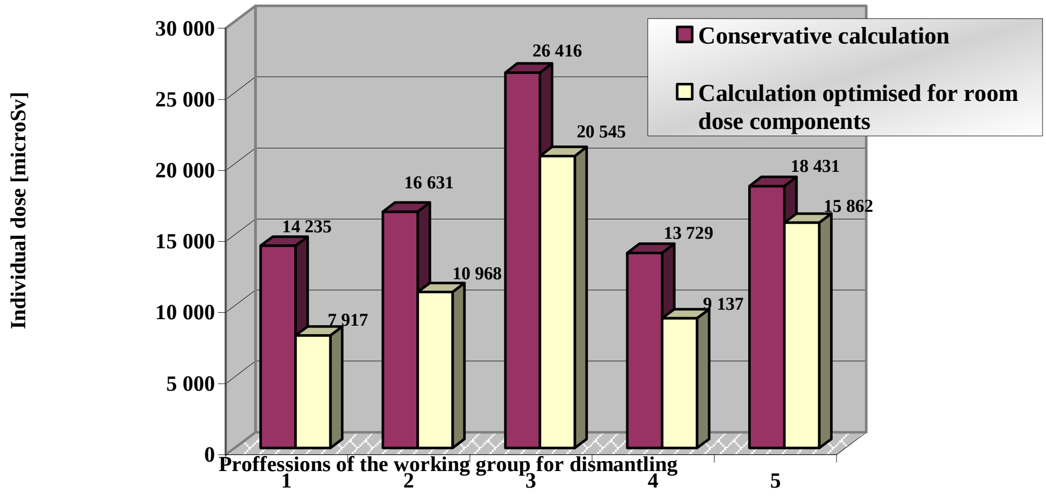

FIG. 44. Demonstration of optimization of dose uptake calculation versus conservative calculation.

I.4.6. THE APPLICATION OF REMOTE DISMANTLING

One of the methods for reducing the exposure of personnel during dismantling is the application of remote dismantling. In this case, the OMEGA code selects the remote dismantling methods automatically, based on the actual dose rate in the vicinity of the equipment to be dismantled and based on the pre-selected value of the dose rate for application of the remote dismantling techniques. Implementation of the remote dismantling decreases the exposure of personnel due to fact that the personnel is located in shielded working places, but the manpower needed for performing the decommissioning activities is significantly higher (approximately 5-10 times) and the costs for the work are also higher, in the rate similar to manpower.

Optimization of the level of the dose rate at the equipment for implementing the remote dismantling can be performed in the computer code OMEGA effectively when all inventory data for the whole NPP are available. The optimization is NPP specific, depending on the real radiological state of the NPP. As an example, the case of A1 NPP in Slovakia is presented in this Section for demonstration of application of this procedure. The methodology of calculation, as applied in the OMEGA code, enables to select automatically by the code the application of manual or remote dismantling technique based on:

- Actual dose rate at the equipment to be dismantled (recovered for the date of dismantling); and
- Dose rate limit for application of the remote dismantling – defined by the user.

This can be used for evaluation of the optimal level for application of remote dismantling and for cost benefit analysis of the type costs and manpower versus dose uptake during dismantling. Model calculations were performed for the primary circuit of the A1 NPP (Slovak Republic) and the results are presented in Fig. 45. The results show that the optimal level for application of remote dismantling is in the interval between 100 – 200 μSv/h. The individual dose uptake for each member of the working group can be optimized also by varying the number of working groups in order to meet the annual limit value 20 mSv, depending on duration of the process of dismantling, as it is seen in Fig. 45.

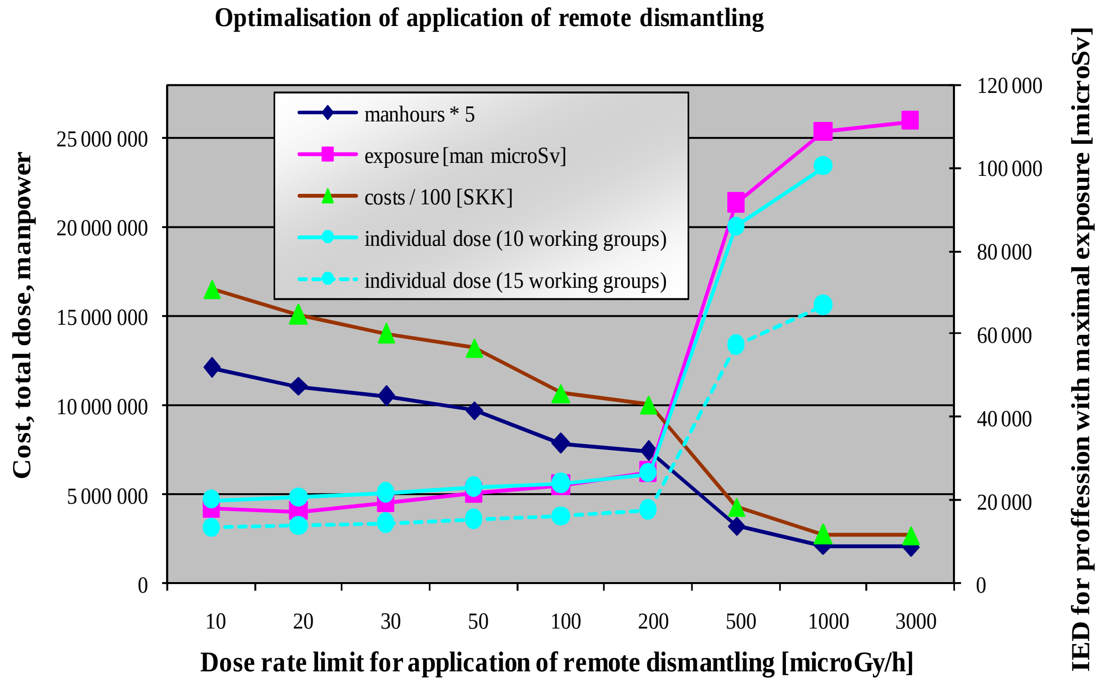

FIG. 45. Evaluation of costs for dismantling of the primary circuit of A1 NPP versus dose uptake by varying the limit for application of remote dismantling.

Remote dismantling was conservatively not applied in calculation for systems 321 and 322.

I.4. THE EVALUATION OF INDIVIDUAL EFFECTIVE DOSE

The sum of the external and internal doses for individual professions of workers was summed and presented as total collective dose for individual professions and after summing over all professions of the working group is presented as the overall dose for the discrete decommissioning activity. According to the general procedures for protection of workers, the dose for individual workers needs to be evaluated and controlled according to the ALARA principle and in any case needs to be lower than the annual limit of 20 mSv per individuals.

The bottom-up approach and profession resolved approach implemented in calculation of decommissioning parameters in the OMEGA code, enables to evaluate analytically the dose to individuals during performing the discrete decommissioning activities. By summing the data over the duration of the given decommissioning phase, like dismantling the system 321 or system 322 and comparing the duration of the phase with one year duration, it is possible to evaluate whether the annual limit of 20 mSv per year was met. The principles applied are:

— For each decommissioning activity, the dose is calculated for each profession of the working group separately. The calculation is dependent on profession resolved coefficients of stay in main components of the dose rates as described in this Appendix, Section I.5. above. The dosed involves all productive and non-productive time components, spent by each member of the profession in the controlled area. This approach corresponds with the real organization of the working time within the controlled area and recording of dose data for individuals. The model working time structure is presented in Section 3.1.

- Each profession can have in principle several members. The manpower within the controlled area and dose calculated for the profession as whole is distributed to each individual of the profession according to the number of workers for the given profession of the working group. The dose calculated at this level has already the character of the individual effective dose.
- This manpower allocated to an individual, represents the real duration of the discrete activity per individual within the controlled area. When dividing the dose allocated to an individual by this manpower, the normalized dose rate for the decommissioning activity is calculated. The normalization in this case means the relation to overall duration of a specific activity within the controlled area. This dose rate represents the averaged level of radiological risk for the individual of the given profession of the working group.
- A table of manpower spectrum is constructed, having on horizontal axis the normalized dose rate in selected intervals (for example 2μSv/h) and on vertical axis the data of manpower components which fits with the given interval of the normalized dose rate as picked up from the database of calculated data. Parallel to spectrum of manpower, the individual effective dose spectrum can be reconstructed. The individual effective dose components are calculated as the product of manpower component in the given interval of the normalized dose rate and the middle value of the interval.
- By summing the data over the whole range of the normalized dose rate scale, the total effective dose for an individual can be calculated. The calculated effective dose is then divided by the duration (unit are years) of the evaluated decommissioning phase, like dismantling of the system 321 and the result is compared with the annual limit of 20 mSv for individuals.

Manpower spectrum and the individual effective dose spectrum can be developed for the decommissioning project or for selected group of decommissioning activities. The shape of the spectrum shows the distribution of exposure risk specific for individuals in evaluated decommissioning activities or its sub-phases. The distribution is facility or system specific, related to the radiological situation in facility systems and structures. The manpower spectrum for the systems 321 and 322 is presented in Fig. 46.

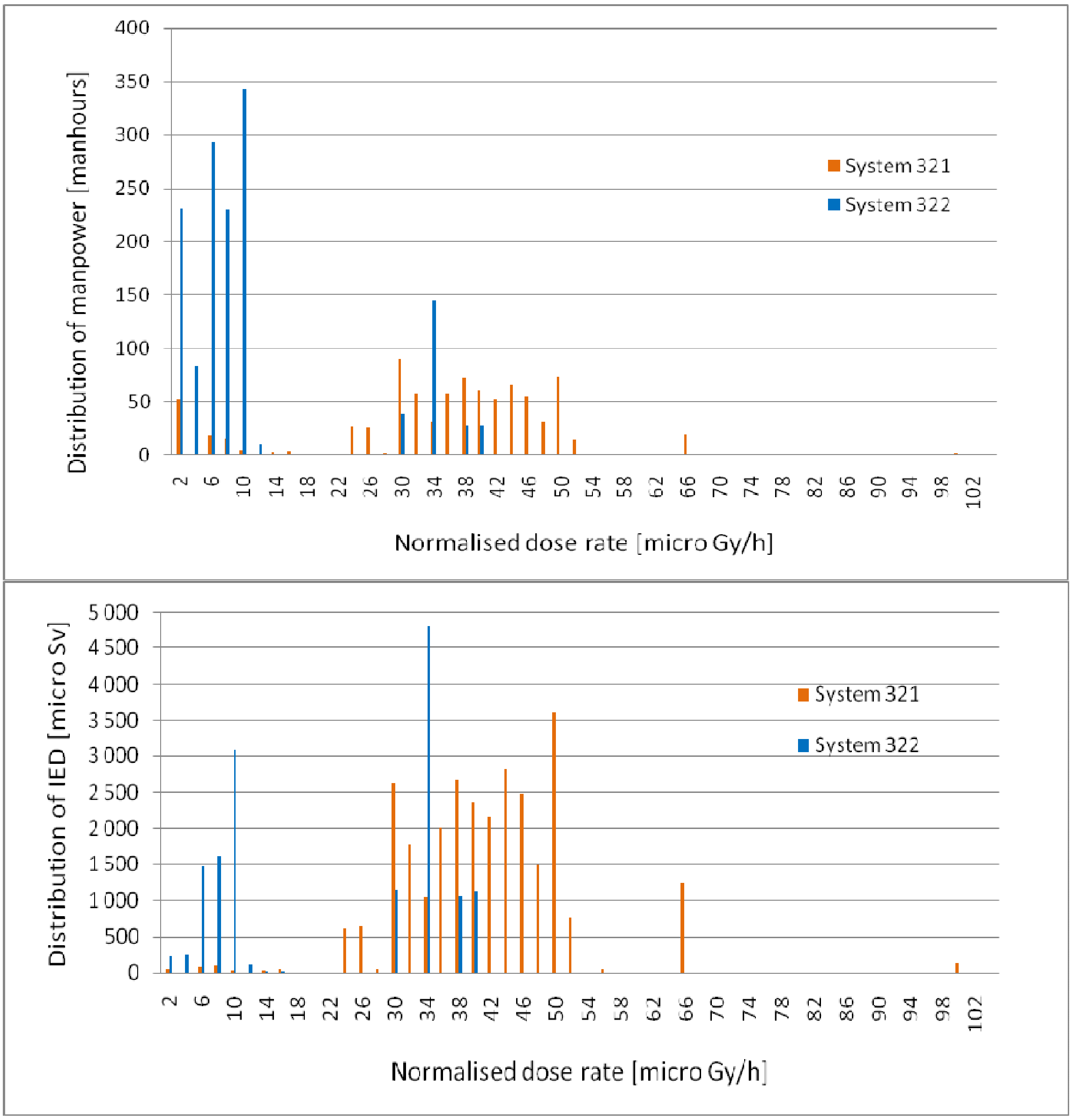

FIG. 46. Distribution of manpower and individual dose for dismantling of systems 321 and 322 for the most exposed profession.

The above presented procedure creates the data for optimization of exposure for individual professions involved in the decommissioning project. Depending on duration of the decommissioning project or its phases under evaluation, the distribution of individual effective dose can be managed by following measures:

- Application of pre-dismantling decontamination;
- Involving more identical working groups or prolonging the critical decommissioning activities (“diluting” the manpower in time). The minimum number of personnel for critical operations can be optimized;
- Managing the performing decommissioning activities – for example by mixing the activities performed under higher exposure risk with the activities with low exposure risk;

— Implementation of remote controlled operations (Section 3.6). The ratio cost versus "saved Sieverts" can be evaluated; and
— Deferring the dismantling. The time point, when the individual dose in under the annual limit for all profession, can be found. The duration and the extent of safe enclosure phases in deferred dismantling can be thus justified analytically.

Demonstration of the above discussed procedure is presented in Fig 47. where the dismantling was delayed for 5 years. The green colour is dismantling in 2005 and the red is dismantling in 2010.

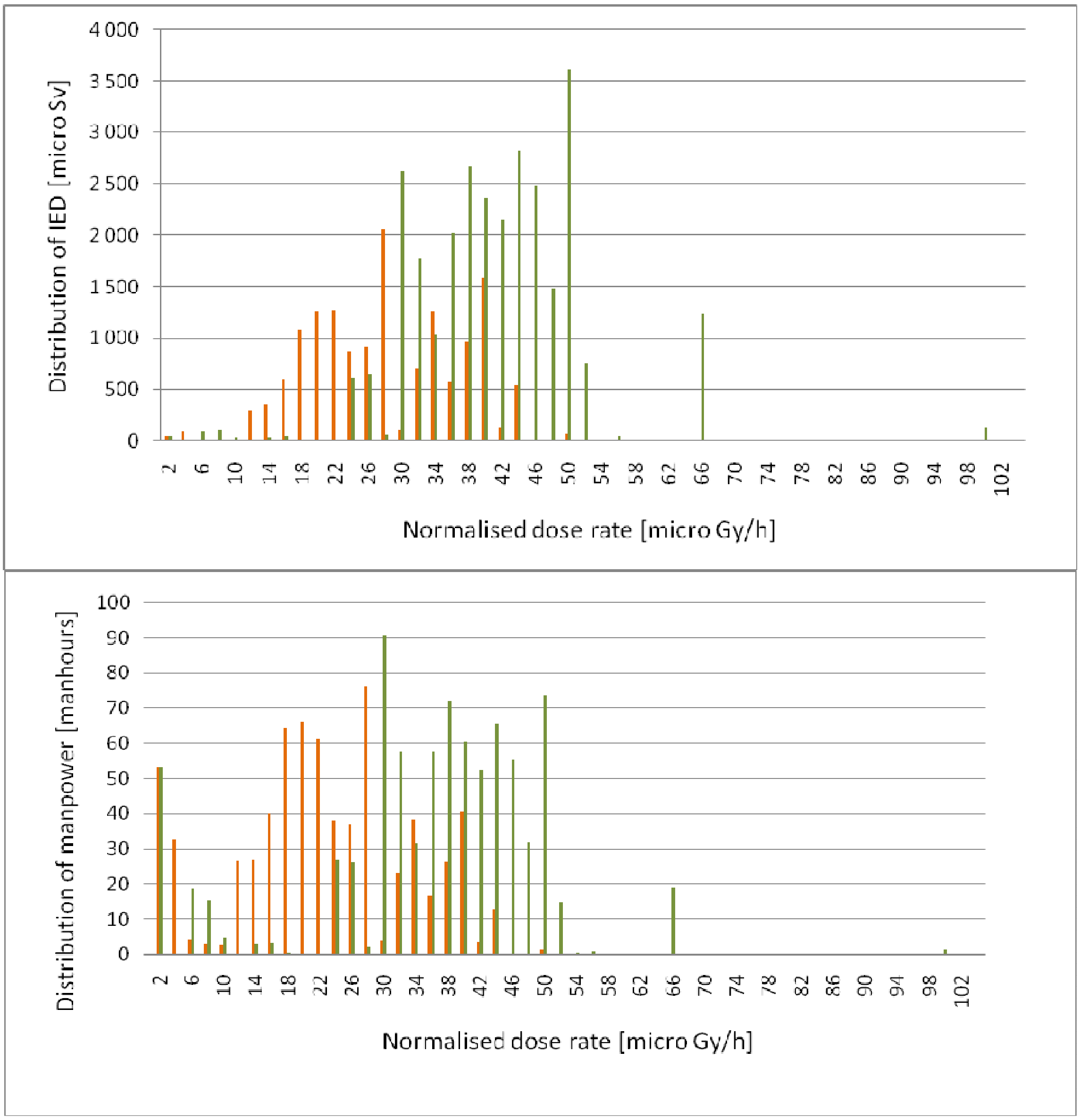

FIG. 47. Distribution of manpower and individual dose for dismantling of systems 321 and 322 for the most exposed profession – effect of delaying the dismantling.

The duration of dismantling or other decommissioning phases is evaluated by construction of the critical path in Microsoft Project software. The schedules for dismantling of systems 321 and 322 are presented in Figures 48 and 49 below.

OMEGA — *System 321 Decommissioning Schedule* — DeSa Test Case NPP

ID	Task Name	Duration	Start
1	**Test Case NPP model decommissioning schedule**	**154,3 days**	**Fri 1.7.05**
2	**Decommissioning of main reactor building, Unit 1**	**154,3 days**	**Fri 1.7.05**
3	**Dismantling of system 321**	**154,3 days**	**Fri 1.7.05**
4	**Floor +103,00**	**0,04 days**	**Fri 1.7.05**
5	Room 1R1.57PL3 (1R1.57PL3)	0,25 hrs	Fri 1.7.05
6	Room 1R3.31 (1R3.31)	0,08 hrs	Fri 1.7.05
7	**Floor +111,50**	**54,37 days**	**Fri 1.7.05**
8	Room 1R1.57PL4 (1R1.57PL4)	3,18 hrs	Fri 1.7.05
9	Room 1R4.31 (1R4.31)	336,58 hrs	Fri 1.7.05
10	Room 1R4.60 (1R4.60 - Contaiment)	95,18 hrs	Tue 30.8.05
11	**Floor +116,35**	**12,71 days**	**Thu 15.9.05**
12	Room 1R4.60.51 (1R4.60.51 - Contaiment)	101,66 hrs	Thu 15.9.05
13	**Floor +116,50**	**14,64 days**	**Tue 4.10.05**
14	Room 1R1.57PL5 (1R1.57PL5)	3,15 hrs	Tue 4.10.05
15	Room 1R5.31 (1R5.31)	113,99 hrs	Tue 4.10.05
16	**Floor +118,80**	**0,11 days**	**Mon 24.10.05**
17	Room 1R5.66 (1R5.66 - Contaiment)	0,84 hrs	Mon 24.10.05
18	**Floor +121,10**	**1,32 days**	**Mon 24.10.05**
19	Room 1R4.60.61 (1R4.60.61 - Contaiment)	0,56 hrs	Mon 24.10.05
20	Room 1R4.60.62 (1R4.60.62 - Contaiment)	1,57 hrs	Mon 24.10.05
21	Room 1R4.60.63 (1R4.60.63 - Contaiment)	0,37 hrs	Tue 25.10.05
22	Room 1R4.60.64 (1R4.60.64 - Contaiment)	2,25 hrs	Tue 25.10.05
23	Room 1R4.60.65 (1R4.60.65 - Contaiment)	3,43 hrs	Tue 25.10.05
24	Room 1R4.60.66 (1R4.60.66 - Contaiment)	2,25 hrs	Tue 25.10.05
25	Room 1R4.60.67 (1R4.60.67 - Contaiment)	0,1 hrs	Wed 26.10.05
26	**Floor +121,50**	**2,61 days**	**Wed 26.10.05**
27	Room 1R1.57PL6 (1R1.57PL6)	20,9 hrs	Wed 26.10.05
28	**Floor +126,50**	**20,64 days**	**Fri 28.10.05**
29	Room 1R7.47 (1R7.47)	165,08 hrs	Fri 28.10.05
30	**Floor +91,10**	**27,3 days**	**Mon 28.11.05**
31	Room 1R1.31 (1R1.31)	106,67 hrs	Mon 28.11.05
32	Room 1R1.32 (1R1.32)	111,76 hrs	Thu 15.12.05
33	**Floor +91,10/+92,60**	**2,68 days**	**Wed 4.1.06**
34	Room 1R1.30 (1R1.30)	21,45 hrs	Wed 4.1.06
35	**Floor +92,60**	**17,8 days**	**Mon 9.1.06**
36	Room 1R1.57 (1R1.57)	142,38 hrs	Mon 9.1.06
37	**Floor +97,00**	**0,09 days**	**Thu 2.2.06**
38	Room 1R2.36 (1R2.36)	0,75 hrs	Thu 2.2.06

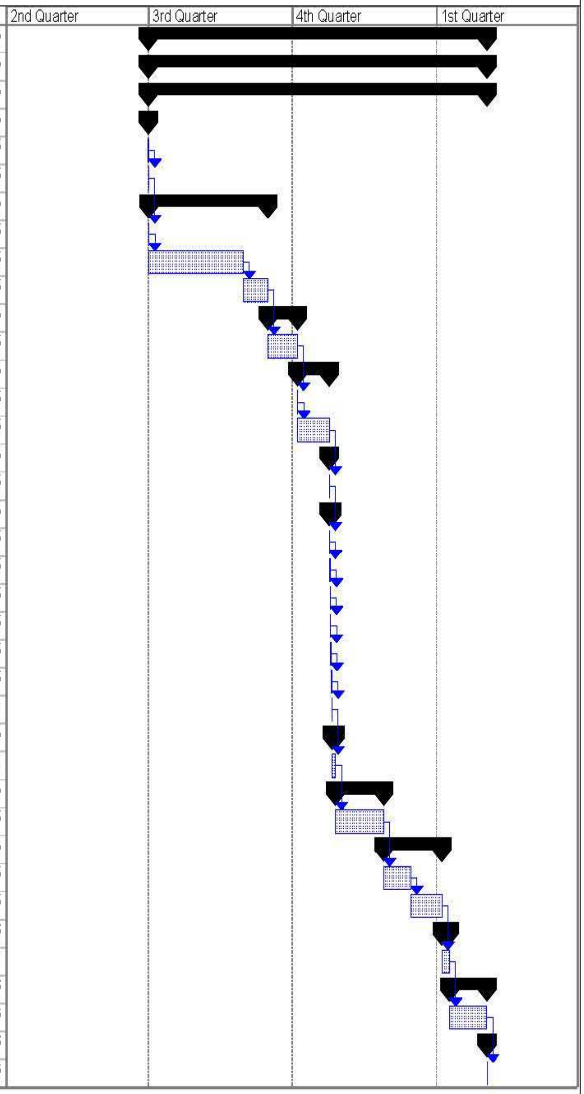

FIG. 48. Dismantling schedule for the system 321.

OMEGA — ***System 322 Decommissioning Schedule*** — DeSa Test Case NPP

ID	Task Name	Duration	Start
1	**Test Case NPP model decommissioning schedule**	**301,97 days**	**Wed 1.3.06**
2	**Decommissioning of main reactor building, Unit 1**	**301,97 days**	**Wed 1.3.06**
3	**Dismantling of system 322**	**301,97 days**	**Wed 1.3.06**
4	**Floor +103,00**	**9,71 days**	**Wed 1.3.06**
5	Room 322_1R3.08 (322_1R3.08)	1,01 hrs	Wed 1.3.06
6	Room 322_1R3.10 (322_1R3.10)	76,67 hrs	Wed 1.3.06
7	**Floor +111,50**	**29,54 days**	**Tue 14.3.06**
8	Room 322_1R4.51 (322_1R4.51)	0,37 hrs	Tue 14.3.06
9	Room 322_1R4.60 (322_1R4.60)	233,02 hrs	Tue 14.3.06
10	Room 322_1R4.60.42 (322_1R4.60.42)	0,37 hrs	Mon 24.4.06
11	Room 322_1R4.60.43 (322_1R4.60.43)	0,37 hrs	Mon 24.4.06
12	Room 322_1R4.60.44 (322_1R4.60.44)	0,37 hrs	Mon 24.4.06
13	Room 322_1R4.60.45 (322_1R4.60.45)	0,37 hrs	Tue 25.4.06
14	Room 322_1R4.60.62 (322_1R4.60.62)	0,37 hrs	Tue 25.4.06
15	Room 322_1R4.60.63 (322_1R4.60.63)	0,37 hrs	Tue 25.4.06
16	Room 322_1R4.60.64 (322_1R4.60.64)	0,37 hrs	Tue 25.4.06
17	Room 322_1R4.60.65 (322_1R4.60.65)	0,37 hrs	Tue 25.4.06
18	**Floor +116,50**	**0,04 days**	**Tue 25.4.06**
19	Room 322_1R5.41 (322_1R5.41)	0,29 hrs	Tue 25.4.06
20	**Floor +126,50**	**0,15 days**	**Tue 25.4.06**
21	Room 322_1R7.47 (322_1R7.47)	1,19 hrs	Tue 25.4.06
22	**Floor +91,10/+92,60**	**153,36 days**	**Tue 25.4.06**
23	Room 322_Virtual room (322_Virtual room)	39,66 hrs	Tue 25.4.06
24	Room 322_1R1.08 (322_1R1.08)	125,6 hrs	Tue 2.5.06
25	Room 322_1R1.09 (322_1R1.09)	152,31 hrs	Wed 24.5.06
26	Room 322_1R1.28 (322_1R1.28)	1,55 hrs	Tue 20.6.06
27	Room 322_1R1.40 (322_1R1.40)	142,17 hrs	Tue 20.6.06
28	Room 322_1R1.40PL1 (322_1R1.40PL1)	0,55 hrs	Fri 14.7.06
29	Room 322_1R1.40PL2 (322_1R1.40PL2)	93,52 hrs	Fri 14.7.06
30	Room 322_1R1.40PL7 (322_1R1.40PL7)	4,67 hrs	Mon 31.7.06
31	Room 322_1R1.47 (322_1R1.47)	98,33 hrs	Tue 1.8.06
32	Room 322_1R1.47PL1 (322_1R1.47PL1)	1,46 hrs	Thu 17.8.06
33	Room 322_1R1.47PL2 (322_1R1.47PL2)	9,93 hrs	Thu 17.8.06
34	Room 322_1R1.47PL5 (322_1R1.47PL5)	4,6 hrs	Mon 21.8.06
35	Room 322_1R1.50 (322_1R1.50)	109,91 hrs	Mon 21.8.06
36	Room 322_1R1.50PL2 (322_1R1.50PL2)	3,94 hrs	Fri 8.9.06
37	Room 322_1R1.50PL7 (322_1R1.50PL7)	4,6 hrs	Fri 8.9.06
38	Room 322_1R1.57 (322_1R1.57)	110,59 hrs	Mon 11.9.06
39	Room 322_1R1.57PL4 (322_1R1.57PL4)	1,54 hrs	Fri 29.9.06
40	Room 322_1R1.57PL5 (322_1R1.57PL5)	4,6 hrs	Fri 29.9.06
41	Room 322_1R1.60 (322_1R1.60)	298,94 hrs	Mon 2.10.06
42	Room 322_1R1.60.04 (322_1R1.60.04)	9,2 hrs	Wed 22.11.06
43	Room 322_1R1.60.05 (322_1R1.60.05)	9,2 hrs	Thu 23.11.06
44	**Floor +97,00 (+96,00/+97,00)**	**109,17 days**	**Fri 24.11.06**
45	Room 322_1R2.08 (322_1R2.08)	134,81 hrs	Fri 24.11.06
46	Room 322_1R2.09 (322_1R2.09)	143,79 hrs	Tue 19.12.06
47	Room 322_1R2.10 (322_1R2.10)	52,42 hrs	Fri 12.1.07
48	Room 322_1R2.28 (322_1R2.28)	70,79 hrs	Tue 23.1.07
49	Room 322_1R2.30 (322_1R2.30)	42,95 hrs	Mon 5.2.07
50	Room 322_1R2.31 (322_1R2.31)	174,99 hrs	Mon 12.2.07
51	Room 322_1R2.35 (322_1R2.35)	252,89 hrs	Wed 14.3.07
52	Room 322_1R2.36 (322_1R2.36)	0,73 hrs	Thu 26.4.07

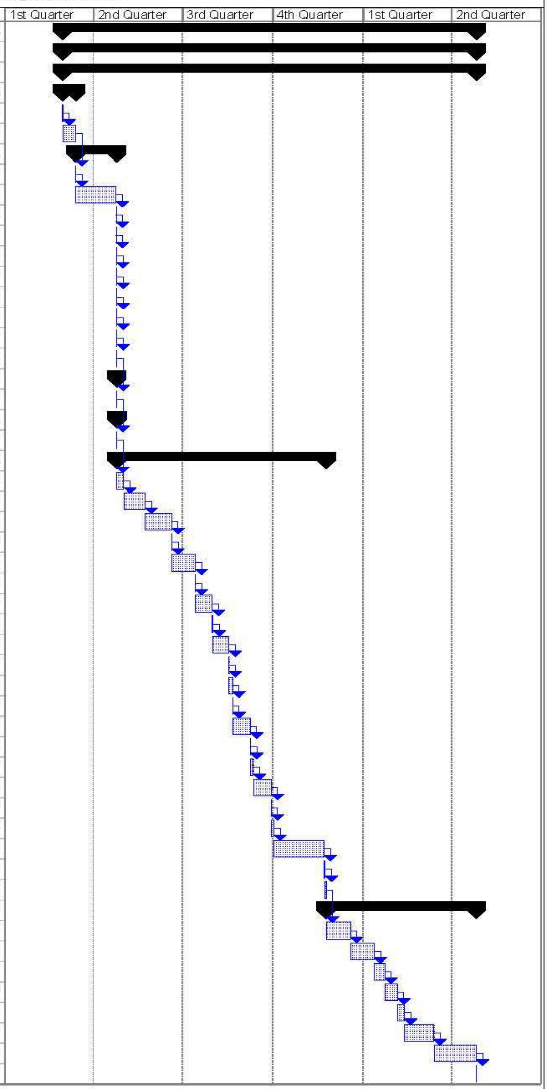

FIG. 49. Dismantling schedule for the system 322.

I.5. CONCLUSIONS

The methodology for the evaluation of exposure of personnel, as presented in this Appendix is implemented by the OMEGA code for evaluating of planned decommissioning activities. The level of evaluation is at the discrete decommissioning activities. The input data used for calculations are derived from calculated specific manpower components of working time and the radiological data as recorded in the facility inventory database. The radiological data are updated for decay before calculation of dose.

The dose data are calculated at the level of individual professions, so the dose evaluation and optimization can be performed at this level. The individual effective dose is evaluated at the level of individuals of the professions of the working group. For evaluation of duration of evaluated phases of a decommissioning project, the Microsoft Project software is used which organizes the evaluated decommissioning activities in a sequence in correspondence with real sequence during performing the planned activities. In this way, the individual effective dose per individual per evaluated duration can be compared with the annual dose limit of 20 mSv per individual.

The methodology implemented enables also effective optimization of the individual effective dose for most exposed professions by involvement of more working groups, working shifts or evaluating the effect of deferring the dismantling or application of remote dismantling.

APPENDIX II: DETAILED MODEL DESCRIPTION OF PUBLIC DOSE ASSESSMENT CODE FOR DECOMMISSIONING OF NUCLEAR REACTORS (DECDOSE) AND ITS APPLICATION TO NPP TEST CASE

II.1. INTRODUCTION

Radioactive substances are dispersed as dusts and/or a gaseous state to working spaces and liquid wastes during dismantling activities such as cutting of radioactive components. This condition is different from that in operating phase of nuclear reactors. Radioactive substances pass through filtration and then are discharged to environment. And a large amount of radioactive waste temporarily stored in the facility emits radiation to the environment through the building wall. They may cause radiation exposure to public.

In order to evaluate the potential exposure of people surrounding the NPP in decommissioning stage, a public dose assessment code for decommissioning of nuclear reactors (DecDose) was developed. Further details can be found in Ref. [12]. This appendix describes evaluation methods for public dose during decommissioning, modelling of cutting of objects, ventilation containment and the distribution of containers in buildings, and example calculations.

II.2. CHARACTERIZATION OF DECOMMISSIONING ACTIVITIES

Characterization of decommissioning was carried out from the viewpoint of public exposure during dismantling activities in comparison with operating situation. While constant quantity of radionuclides is assumed to be discharged in safety assessments for operation of nuclear reactors, the quantity of radioactivity discharged to environment depends on the dismantling schedule in a decommissioning stage. For instance, in case of a ten-year decommissioning plan, relatively larger quantity of radionuclides may be discharged in only a few years when highly activated components such as core shroud are dismantled. In addition, while I-131 and radioactive noble gas are focused on in the safety assessment for the operation due to fuel failures, radionuclides of Co-60 and Cs-137 are main radioactivity in decommissioning stage, because materials activated and contaminated during operation are dispersed by cutting works.

II.3. EXPOSURE PATHWAYS FOR PUBLIC

Figure 50 shows exposure routes to public around the NPP from radiation and radioactive materials as modelled in DecDose. Two kinds of radioactive materials are regarded as the radiation source of public dose:

(a) One is the radioactive substance dispersed to working spaces during the activities of cutting and decontamination; and

(b) The other is the radioactive component and structure removed from the original position, and is temporarily stored before shipping to the treatment facility.

II.3.1. Release of radioactive materials into environment

Radioactive substances of airborne particles and gas are dispersed to working space and liquid during activities of cutting and decontamination, and a part of them may be discharged from the stack to the atmosphere and from the outlet to the ocean. In these discharges, following pathways need to be considered in public dose evaluation.

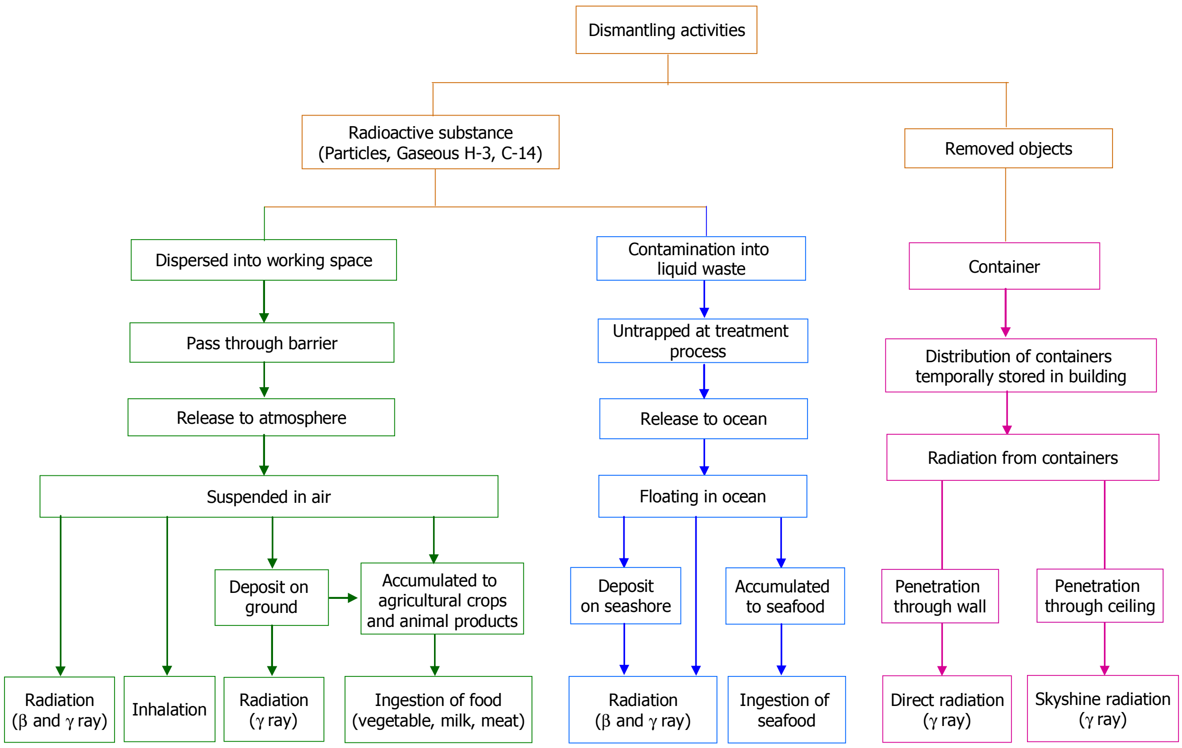

FIG. 50. Exposure pathways for public dose.

(a) External exposure by gamma ray:

- — Radioactive plume;
- — Ground deposition;
- — Fishing net;
- — Swimming;
- — Activity at shore;
- — Fishing activity;
- — Direct radiation; and
- — Skyshine radiation.

(b) Internal exposure:

- — Inhalation; and
- — Ingestion:
  - Agricultural crops (vegetables, cereals and edible root);
  - Animal products (milk and its products, meat);
  - Seafood (fish, shellfish, marine vegetation).

II.3.2. Radiation from dismantled waste temporally stored in buildings

It is assumed that the radioactive components and structures arising from dismantling activities are temporally stored with containers. Since radiation sources are fixed in the building, only gamma ray from the wastes is considered in this evaluation.

II.4. EVALUATION METHODS

II.4.1. Release of radioactive materials into environment

The process for releasing radioactive substances from the facility into the atmosphere and ocean was modelled. DecDose is capable of dealing with up to fifty five radionuclides including gaseous radionuclides of H-3, C-14 and their decays in accordance with a dismantling plan which includes working schedule, cutting and contamination control conditions. It is very important, for accurate evaluation, to evaluate kerf volume and area in cutting activities because radioactive substances exist in the material in activated components, and they exist on the surface of the material in contaminated materials. Kerf width actually depends on the cutting tool applied, while the depth depends on the component. Kerf length of the component depends on the type of the container in which the component will be stored. Shapes of components such as piping and ducts also affect kerf lengths. Therefore cutting models for various shapes were developed to evaluate kerf volume and area. It is considered that fine cutting may be applied to the piece which is already cut in the different place that is rough cutting, to carry out the activity efficiently in the enough space. Though evaluation methods on activated and contaminated objects are discussed separately in the following part, for both activated and contaminated objects, the sum of radioactivity on both surfaces is regarded as the quantity dispersed to working space.

II.4.1.1. Activated objects

A simple plate is taken as a typical example to show cutting activities as shown in Fig. 51. Objects to be dismantled are assumed to be uniformly activated. When the plate is divided into two pieces at the centre line by using cutting method such as plasma arc and band saw cutting, the volume which is formed by the length l on a side, the thickness d and the cutting width w is removed from the plate. A part of those removed drops down to the floor as a dross, and the balance is dispersed into working space. When a emission rate to air is defined as the ratio of the weight dispersed into working space as aerosols to the one removed from the object, the weight dispersed into working space is expressed by the product of kerf weight and emission rate. Then the radioactivity is calculated by multiplying the weight dispersed by radioactive concentration of the object. The emission rate to air depends on the dismantling method applied and the type of material of the object.

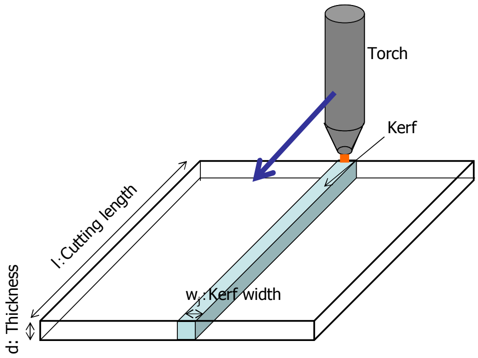

FIG. 51. Model for cutting of plate with activation and contamination.

Since emission rates are important parameters for evaluating the quantity of radionuclides dispersed into air and liquid, a number of experimental data obtained by laboratories such as JAEA, CRIEPI and NUPEC in Japan was collected to use in DecDose.

The quantity A_{ij} of radionuclide i dispersed into air with cutting method j is expressed by using emission rate to air a_{jj} as follows.

$$A_{ij} = l*d*w_j*a_{ij}*c_i \quad (6)$$

The quantity A_{ij} of radionuclide i dispersed into liquid with cutting method j using water is expressed by using emission rate to liquid g_{ij} as follows.

$$A_{ij} = l*d*w_j*g_{ij}*c_i \quad (7)$$

In case of dismantling without a cutting line, a whole object with the volume V_0 and radioactivity Q may be dismantled as the controlled blasting method. The quantity A_{ij} with dismantling method j is expressed by using emission rate to air a_{ij} and by using emission rate to liquid g_{ij} as follows.

$$A_{ij} = a_{ij}*V_0*c_i \quad (8)$$

$$A_{ij} = g_{ij}*V_0*c_i \quad (9)$$

$$c_i= Q*z_i/V_0 \quad (10)$$

II.4.1.2. Contaminated objects

Radioactive contaminant usually exists on the inner and/or outer surface of components and structures except percolation. A simple plate is also taken for a typical example similar to the activated object as already shown in Fig. 51. Surface density of radioactive contamination is assumed to be constant in the plate. While the same volume as activated one is removed in the cutting, the quantity of radioactive substances removed from the object space is expressed by multiplying the cutting length l by kerf width w_j by surface density because the contamination exists on the surface of the plate. If the contamination exists on the both sides, it has to be evaluated in consideration with surface densities and radionuclide composition ratio for each side. The quantity A_{ij} is expressed using emission rates to air b_{ij} and to liquid h_{ij} for contaminated objects by the following equation.

$$A_{ij} = l*w_j*b_{ij}*f_i \quad (11)$$

$$A_{ij} = l*w_j*h_{ij}*f_i \quad (12)$$

where the emission rate is defined as the ratio of radioactivity dispersed to working space without dropping the removed radioactivity to floor.

II.4.1.3. Decontamination for building surface

Radioactive contaminations usually exist on the surface of components and structures. When there is decontamination for building surfaces such as floors, walls and ceilings, it is impossible to deal with it without new registration of contamination information in addition to object data. Actually, information on the area to be decontaminated which has evidence of radioactivity, shaved depth and types of radionuclide composition ratio has to be registered in advance of the evaluation. The quantity A_{ij} of radionuclide i is expressed by the following equation in consideration with activation.

$$A_{ij} = S*d*c_i*a_{ij} + S*f_i*b_{ij} \quad (13)$$

II.4.1.4. Radioactivity mixed into liquid waste

Radioactive substances from hand wash water and laundry water are considered as liquid waste to be discharged to the ocean. The quantity A_{iex} of hand wash and laundry of radionuclide i is expressed by the following equation.

$$A_{iex} = A_{ihd}*N_w+A_{ild}*N_w \quad (14)$$

where N_w means the number of workers exit from the activities, A_{ihd} radioactivity per handwash, and A_{ild} radioactivity per cloth at the exit. Some parameters above are empirically obtained in operating phase.

II.4.2. Quantity of radionuclides discharged to environment through the barriers

II.4.2.1. Atmospheric discharge

Radionuclides dispersed into working space pass through one of following three routes as shown in Fig. 52, until they are discharged to the environment. In this regard, aerosol particles are assumed not to be deposited on inner surface of ducts and building walls for conservative evaluation.

(a) Route passing through both local ventilation and building filters [discharge at higher position].

Among the quantity A_{ij} of radionuclides dispersed into working space using method j, the quantity $B1_{ij}$ of radionuclide i discharged to environment through this route is expressed as follows.

$$B1_{ij} = (1-p)(1-q_i)(1-s_i)A_{ij} \quad (15)$$

(b) Route passing through building ventilation filter after leakage from the enclosure [discharge at higher position]

The quantity $B2_{ij}$ of radionuclide i discharged into environment is expressed as follows.

$$B2_{ij} = p\ (1-r_i)\ (1-s_i)\ A_{ij} \quad (16)$$

(c) Route of leakage from both the enclosure and building containment [discharge at ground level]

The quantity $B3_{ij}$ of radionuclide i discharged into environment is expressed as follows.

$$B3_{ij} = p*r_i*A_{ij} \quad (17)$$

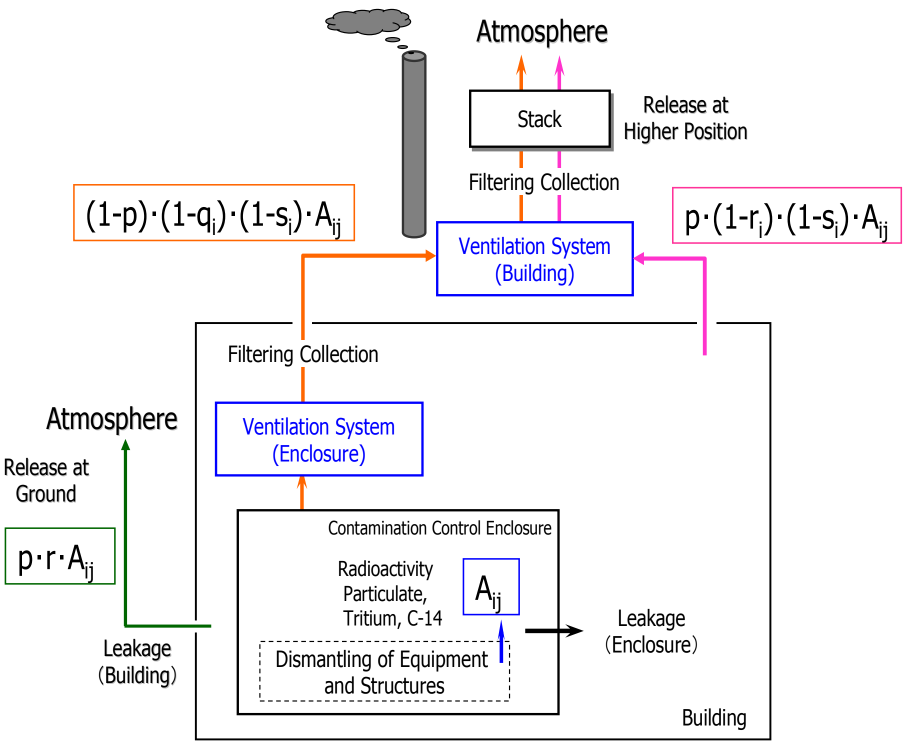

FIG. 52. Discharge routes to the atmosphere.

II.4.2.2. Discharge of liquid waste to the ocean

Liquid waste from activities is processed by liquid treatment systems existing and/or temporally installed. Liquid from handwash and laundry of which concentration is relatively low is considered to be treated only through existing system. For liquid from activities for dismantling highly activated components and decontamination of which concentration is relatively high, a special treatment system is usually installed. In Japan Power Demonstration Reactor (JPDR) dismantling activities, the cleaning device was applied for water during underwater cutting for reactor internals inside the reactor pressure vessel. The quantity of radioactive substance discharged to the ocean depends on the performance of those treatment devices, that is, their decontamination factors. It is necessary to pay attention to those factors which depend on the concentration of original liquid before treatment.

For this test case however, there are discharges of liquid effluents, and so this particular factor is not carried forward into the DecDose calculation.

(a) Liquid waste from hand wash and laundry

Liquid waste arising from dismantling activities similar to that in operating phase is usually discharged to the ocean only through existing processing system for liquid waste. When the decontamination factor of existing processing system for liquid waste is defined as D_{ai}, the quantity B_{iex} of radioactive substances about radionuclide i discharged to the ocean is expressed as follows, among the quantity A_{iex} of radionuclide of liquid waste from hand wash and laundry.

$$B_{iex} = A_{iex} / D_{ai} \quad (18)$$

(b) Cooling water and liquid from the decontamination

Temporal treatment devices are usually applied to special activities in which highly activated materials are cut or highly contaminated materials are decontaminated. Liquid waste arising from these activities is discharged to the ocean through both temporal and existing treatment devices. With decontamination factor D_{ai} of existing processing system and D_{bi} of processing system temporally installed, the quantity B_{idis} to be discharged to the ocean about radionuclide i in the quantity A_{ij} in the liquid waste from Equations (2) and (6) is expressed by following equation.

$$B_{idis}= A_{ij}/ D_{ai} / D_{bi} \quad (19)$$

II.4.3. Radiation from dismantled waste temporally stored in buildings

When direct and skyshine radiations are evaluated in DecDose, the radiation attenuation at walls and ceilings of containers and buildings is taken into account in addition to self shielding of dismantled objects in the container. In storing the objects in the container, containing efficiency for each container and material is used for calculation of the number of container and the quantity of radioactivity arising from the activity. Distribution of containers in the building is assumed that gives maximum doses for public at the boundary of the reactor facility for conservative evaluation.

II.5. MODELLING WORK

Based on the described above, several models for those evaluations were developed. Basic concepts of those models are described as follows.

II.5.1. Cutting

Shapes of components such as piping and ducts also affect kerf lengths. Both rough and fine cuttings in different places to put in a container efficiently are also taken into account. Cutting models for each shape were developed to evaluate kerf volume and area. As the typical model, a cutting model for circular tubes is described here. Circular tubes include piping, shroud and reactor pressure vessels. A circular tube with outer diameter of 2R and radial thickness of T is supposed.

Rough cutting is defined that the tube is cut in length direction by the value which is calculated by container inner width a_D by multiple number of m. And fine cutting is defined that the length direction after rough cutting is cut by the value container inner width a, and is cut vertically by the segment number k which depends on the diameter of the tube. For instance, Fig. 53 shows the segment number for vertical cutting. Length of circular tube L is calculated by dividing V_0 by the cross section of the circular tube S_a.

$$V_0 = W_0/D \quad (20)$$

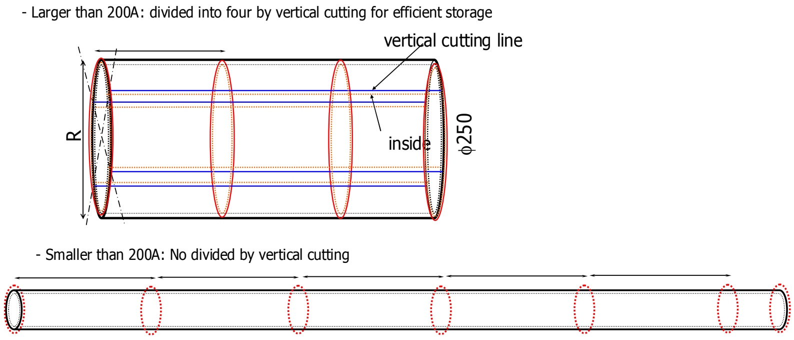

FIG. 53. Segmentation model for piping in DecDose.

$$S_a = \pi\left(R^2 - (R-T)^2\right), \; L = V_0/S_a \tag{21}$$

$$cutL_1 = \mathrm{int}\left(\frac{L}{a_D \cdot m_W + t_1 + t_2(m_W - 1)}\right) \tag{22}$$

$$cutL_2 = \mathrm{int}\left(\frac{L\%(a_D \cdot m_W + t_1 + t_2(m_W - 1))}{a_D + t_2}\right) \tag{23}$$

$$V_{11} = S_a \cdot t_1 \cdot cutL_1 \tag{24}$$

$$\begin{aligned} V_{12} &= S_a \cdot t_2((m_W - 1)cutL_1 + cutL_2) \\ &\quad + T \cdot t_2 \cdot k(L - (t_1 + t_2(m_W - 1))cutL_1 - t_2 \cdot cutL_2) \end{aligned} \tag{25}$$

$$S_{11i} = 2\pi(R-T)t_1 \cdot cutL_1 \tag{26}$$

$$S_{11o} = 2\pi \cdot R \cdot t_1 \cdot cutL_1 \tag{27}$$

$$\begin{aligned} S_{12i} &= 2\pi(R-T)t_2((m_W - 1)cutL_1 + cutL_2) \\ &\quad + t_2 \cdot k(L - (t_1 + t_2(m_W - 1))cutL_1 - t_2 \cdot cutL_2) \end{aligned} \tag{28}$$

$$\begin{aligned} S_{12o} &= 2\pi \cdot R \cdot t_2((m_W - 1)cutL_1 + cutL_2) \\ &\quad + t_2 \cdot k(L - (t_1 + t_2(m_W - 1))cutL_1 - t_2 \cdot cutL_2) \end{aligned} \tag{29}$$

II.5.2. Storage in containers

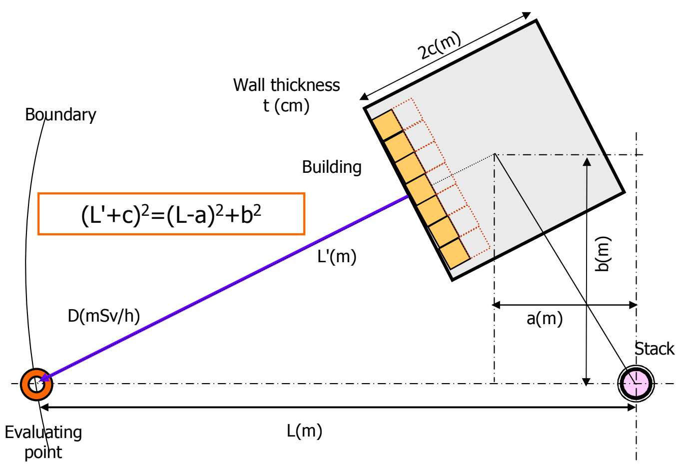

FIG. 54.Concept of the modelling for direct radiation from wastes.

Components and structures from which the kerf Volume V1 was already removed are categorized by the type of container and material. The total volume $V2_{sum}$ of sum of the objects in the same category is allocated to the container by the product of the volume V_c of the inner capacity of the container and containing efficiency f_{con} of the container for the material of the object. This concept is different from that of cutting in which dimensions of the cutting piece were considered.

The number K of containers necessary for total volume of V2 is expressed by following equation.

$$K= -\mathrm{int}(-V2_{sum}/V_c/f_{con}) \tag{30}$$

It is also possible that the criterion for transportation of stored radioactivity is also satisfied simultaneously by operating their storage efficiency automatically.

Libraries for the direct radiation and skyshine radiation were prepared by calculating distribution of dose rate from each type of container with the unit radiation source using QAD, DORT and so on to reduce the calculation time.

II.5.3. Distribution of containers in buildings

Figure 54 shows the concept of the modelling for direct radiation. Shapes of buildings are regarded as square and buildings are assumed to be just opposite to evaluating points in all of 16 directions around the site. Containers are also assumed to be allocated at three columns vertically for any types of containers. Containers are taken until sum of the length on a side or the diameter of containers reaches the value of three times of the length on a side of the building. The first container extracted is put on the floor at the centre line of the building. Next two containers are situated on the first container. It means three columns in vertical direction. The fourth container is assigned next to the first container on the left side on the floor. And then the fifth and sixth were put on the forth container. The seventh is next to the first on the right side. Following containers are put in a similar way. Radiation emitted from subsequent containers located behind the first line is neglected because most of radiation is shielded by containers at first row.

While actual wall thickness is used for the evaluation of the direct radiation from the container on the centre line, for any containers else, the effective distance and the wall thickness are applied to the evaluation with correction because the angle between the container and the evaluation point is not rectangular. Actual distance to the evaluation point L'_{eff} is expressed as follows.

$$L'_{eff} =\{(L'-c)^2+y^2\}^{1/2} \tag{31}$$

where y means the width from the centre line to the position of the centre of the container along the wall, which depends on the place of the container stored in the building. Effective wall thickness t_{weff} is expressed as follows:

$$t_{weff} = t(L'_{eff} /(L-c)) \tag{32}$$

II.5.4. Diffusion of radioactive substance at ocean

Radioactive concentration at outlet is regarded as that of sea water for the safety evaluation in operating stage of nuclear power reactors. When the same way is applied to decommissioning stage where sea water of cooling is not taken for the condenser, relatively higher value of the concentration is calculated, although the quantity of radioactive substances discharged to the ocean is much smaller than in operation. It is required for reasonable evaluation in decommissioning to consider the diffusion effects at the ocean. IAEA method [17] for diffusion in the ocean was installed. The equations for radioactive concentration of sea water are given by Equations (33) and (34).

- concentration for the region where fish and shellfish live

$$C_{w,tot} = \frac{962 \cdot U^{0.17} \cdot Q_i}{D \cdot x^{1.17}} \tag{33}$$

Q_i: discharge rate (Bq/sec),

x: distance in a longitudinal direction between the outlet and the potential receptor (m) [along the tidal stream]

- concentration along the seashore line

$$C_{w,tot} = \frac{962 \cdot U^{0.17} \cdot Q_i}{D \cdot x^{1.17}} \exp\left(-\frac{\left(-7.28\times10^{5}\right)U^{2.34} \cdot y_0^2}{x^{2.34}}\right) \tag{34}$$

y_0: distance from outlet to seashore (m)

II.5.5. Assignment of radioactive substance for each year

The public exposure dose is evaluated for each year according to work schedule in this code. If a working unit may run the period over a year, the quantity of radioactive substance arising from this activity needs to be distributed for each year. When the quantity may be distributed simply in proportion to the duration for each year, the evaluated dose is not always conservative. A working unit consists of preparation, dismantling and containing, and cleaning after removal. However, it is difficult to specify the duration within the unit. Therefore, it is assumed that the weight to be removed is proportional to the duration.

II.5.6. Discharge of radioactive substances to environment

When components and structures are specified in the order of the value which is calculated by radioactive concentration discharged and dose coefficients in advance, it is possible to always carry out conservative evaluation by selecting the components and structures in the order of the value from the highest every year during the working unit. However, sum of doses for each year throughout the unit obtained by above way is too conservative to indicate representative dose because the sum of the quantity of radioactive substances evaluated always exceeds the whole of quantity discharged to the environment. Therefore, careful attention must be paid to treat those results.

II.5.7. Temporary storage

Basic concepts of stored components in containers for conservative evaluation are that they are stored in order of the highest to the lowest of radioactive concentration of objects according to storage efficiency in the container. When containers are specified in the order of the value which is calculated by radioactivity corresponding to Co-60 and gamma ray dose coefficients and linear attenuation coefficient at the container’s wall, it is also possible to always carry out conservative evaluation by selecting the container in the order of the value from the highest in the similar way.

II.6. STRUCTURE OF DECDOSE AND DATA

II.6.1. Introduction

DecDose consists of three evaluation parts: (1) “Release” which calculates the amount of radionuclides released into the atmosphere and ocean; (2) “Andose” which calculates relative concentration and relative dose rate at evaluation points in 16 directions; and (3) “Edose” which calculates public exposure doses for each year as shown in Fig. 55 which also shows data and its flow used in DecDose. Andose which meets the requirement of the meteorological guideline published by NSC had been already developed for operating reactor in JAERI. Object data are fundamental in this evaluation, and users must prepare these data before evaluations. Scheduling and detailed working conditions are configured by users on the screen of DecDose. Gaseous radionuclides of H-3 and C-14 and 53 particulate radionuclides such as Co-60 and Cs-137 are dealt with as the standard setting. The radionuclide composition ratio is set for each material and for each component and structure. Further it is possible to insert new radionuclides other than those present or to delete radionuclides.

II.6.1.1. Object data

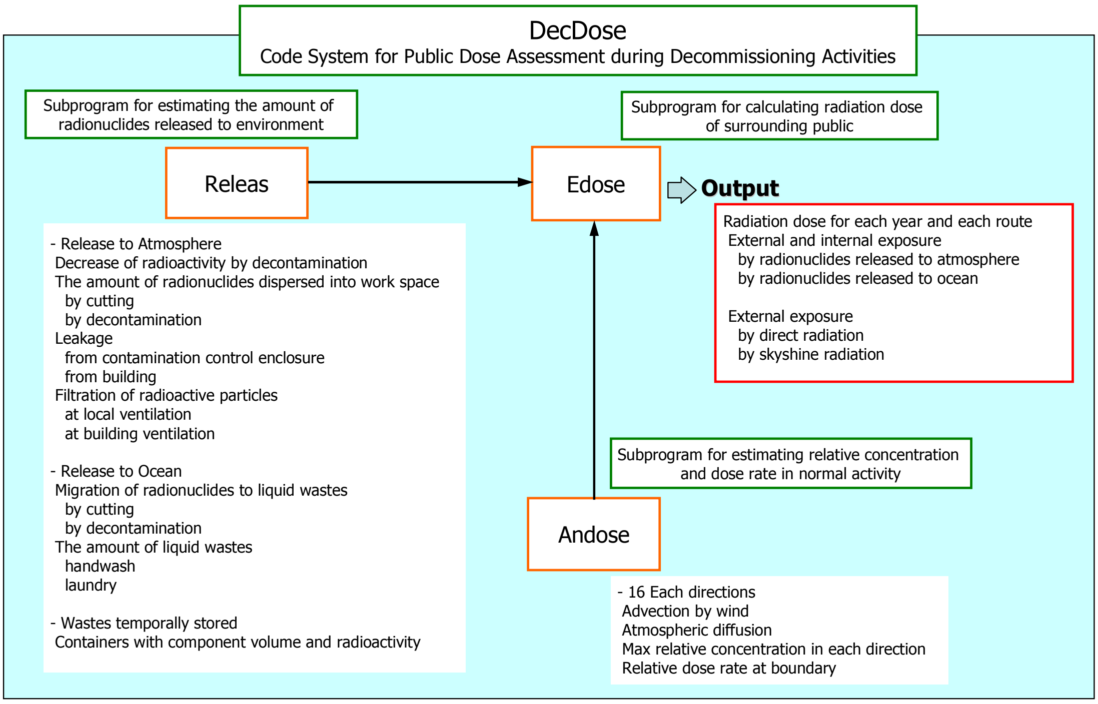

FIG. 55. Structure of DecDose and data for the evaluation.

Object data is information on equipment and structures in the plant of interest. The data fields consist of building, floor, area, component, shape, line, volume, material, radioactivity and surface contamination density and so on. The data definitions are in connection with keywords, independently of the order. Composition ratios of radionuclides for each material such as stainless steels, carbon steels, concretes and contaminations are included in this category. Fifty-five radionuclides are dealt with in DecDose as default, and it is possible to add or reduce them. These data are based on the reference date described in object data file. In addition, users can add materials so that different composition ratios will be registered to the same materials and contaminations.

Object data also include area data for decontamination of surface concrete in the room, and dimension and surface contamination density are registered for each face of floors, walls and ceilings. Since these areas are regarded as objects to be dismantled, tools and working conditions need to be also selected. If there is a large gradient of radioactivity concentration in the object, it is necessary to divide the object into some pieces as each piece has almost uniform distribution of the concentration, in advance.

II.6.1.2. Scheduling and detailed working condition

Dismantling activities are divided into some divisions and starting and terminating date for each category are input by users according to the decommissioning plan applied to the regulatory body. In addition, components and structures are set for each division in consideration of areas and the line. Working conditions such as dismantling tools and with or without a contamination control enclosure are also set for each component and structure. Multiple activities for one component can be set as rough cutting and fine cutting are conducted in a different place.

II.6.1.3. Database for safety assessment

Many parameters are necessary for evaluating public doses. Parameters are divided into two categories, plant dependence and independence. For example, social parameters are plant dependent. On the other hand, exposure parameters such as dose conversion ratios are constant, that is plant independent. Though data on tool and ventilation are treated as the category of plant independence, those can be changed by users according to actual data of the plant.

II.6.2. EXAMPLE EVALUATION

II.6.2.1. Calculation Conditions

II.6.2.1.1. Components in system 321

The surface contamination density of components of system 321 is shown in Table 34. At the reference date of July 1, 2005, the total contamination density of the inner surface of the inner surfaces of pipes, T-junctions, valves, pumps and heat exchangers is 9.76 x 10^5 Bq/cm^2 and that of the outer density is 0.6 Bq/cm^2.

TABLE 34. COMPONENTS IN SYSTEM 321

<table>
<tr><th colspan="2">Component</th><th>Number</th><th>Total Weight</th><th>Inner Surface Density</th><th>Outer Surface Density</th></tr>
<tr><td rowspan="3">Piping, T-junction</td><td>20 mmϕ</td><td>34</td><td rowspan="3">16.4 ton</td><td rowspan="6">9.76 x 10^5 (Bq/cm^2)</td><td rowspan="7">0.6 (Bq/cm^2)</td></tr>
<tr><td>80 mmϕ</td><td>6</td></tr>
<tr><td>250 mmϕ</td><td>59</td></tr>
<tr><td colspan="2">Valve</td><td>197</td><td>14.6 ton</td></tr>
<tr><td colspan="2">Pump</td><td>2</td><td>5.4 ton</td></tr>
<tr><td colspan="2">Heat exchanger</td><td>2</td><td>7.0 ton</td></tr>
<tr><td colspan="2">Motor etc.</td><td>75</td><td>6.8 ton</td><td>0</td></tr>
</table>

II.6.2.1.2 General parameters

Dismantling activity of the system 321 is assumed to be completed in nine months. The activity for system 321 starts at July 1, 2005 and ends at March 31, 2006.

II.6.2.1.3 . Parameters for calculation of the amount of radionuclides

Plasma arc cutting whose kerf width and emission rate are generally larger than any other cutting tools is assumed to be applied to all of the components of system 321 for conservative evaluation. The emission rate, b_{ij}, of radionuclide i from surface contaminated metal plate up to 70% was obtained for plasma arc cutting in the experiment at the nuclear reactor facility. Another experiment showed the lower emission rate by a factor of eight for piping shape components than that of plate components. Therefore, the emission rate of plasma arc cutting is assumed to be 10% or 70% in this evaluation. As sensitivity analyzes, emission ratios for neutron induced material of carbon steel and stainless steel were applied.

Cutting length depends on the dimension of the ISO container which is 5.7 m x 2.3 m x 2.2 m inside. Considering of actual dismantling activities in the facility, however, piping of 1.5 m or shorter in length are better for handling in the working space. 1m3 container widely used was also applied for sensitivity analysis.

Kerf width by plasma arc cutting in air is assumed to be 1cm for both materials, 0.3 cm for carbon steel, or 0.5 cm for stainless steel.

HEPA filters are not installed at the local contamination control enclosure where cuttings are carried out, and only the filtration at the building ventilation is taken into account. The filter efficiency of 99.0% or of 99.97% was assumed to cases as shown in Table 35. For sensitivity analyzes, some more cases with local ventilation filters whose filtering ratio were 99.9% were calculated.

TABLE 35. CALCULATION CONDITIONS FOR EACH CASE ON EMISSION RATE AND FILTRATING EFFICIENCY

	container	kerf width	Emission ratio	GH (filtering ratio)	Building Filter
case 1	1m^3 (1.0)	0.3cm(CS)/ 0.5cm(SUS)	1.3%(CS)/ 2.4%(SUS)	Yes (99.9%)	Yes (99.9%)
case 2	ISO	0.3cm(CS)/ 0.5cm(SUS)	1.3%(CS)/ 2.4%(SUS)	Yes (99.9%)	Yes (99.9%)
case 3	ISO	0.3cm(CS)/ 0.5cm(SUS)	1.3%(CS)/ 2.4%(SUS)	No	Yes (99.97%)
case 4	1m^3 (0.8)	0.3cm(CS)/ 0.5cm(SUS)	1.3%(CS)/ 2.4%(SUS)	No	Yes (99.97%)
case 5	1m^3 (0.8)	0.3cm(CS)/ 0.5cm(SUS)	10%	No	Yes (99.97%)
case 6	ISO	0.3cm(CS)/ 0.5cm(SUS)	10%	No	Yes (99.97%)
case 7	ISO	1.0cm (CS,SUS)	10%	No	Yes (99.97%)
case 8	1m^3 (0.8)	1.0cm (CS,SUS)	10%	No	Yes (99.97%)
case 9	1m^3 (0.8)	1.0cm (CS,SUS)	70%	No	Yes (99.0%)

II.6.2.1.4 . Parameters for calculation of public dose

(a) Physical parameters

The critical group is assumed to reside at the site boundary at ground level. The effective height of stack is the same as the actual stack height of 110 m. Distance between site boundary and stack is assumed to be 500 m at the height of 0m. Based on the meteorological data shown in Table 34, dilution factor, χ/Q at the boundary is calculated to be 1.46 x 10^{-14} sec/cm^3, which is actually the maximum value at approximately 4km from the boundary of the site.

(b) Social parameters

As described in Section 3.1.3 (Definition of critical group), members of critical group live on the site boundary and consume (by ingestion) foods such as vegetable, meats, and milk and milk products which were cultivated at the same place. The amount of food ingested by adult according to the NPP Safety Report is shown in Table 36.

TABLE 36. THE AMOUNT OF FOOD INGESTION FOR CRITICAL GROUP

Food		Quantity (g/d)
Vegetables	Leaf	310
	Root	400
Meat		250
Milk and its products		1000

II.6.2.2. Uncertainties

It must be noted that some of parameter values assumed in (1) range widely and associated with uncertainties. For example, kerf width is considered to depend not only on the cutting technique applied, but on the skill of the cutting worker, and the uncertainty associated with worker skill is not taken account. Most of parameter values were selected as conservative as possible. In this assessment, three cases with different values of emission rate and filtering efficiency as sensitivity analysis.

II.6.2.3. Results of evaluation

The calculated results of the amount of radionuclides discharged into the atmosphere and the public dose for each pathway are summarized in Tables 37 and 38. In cases using emission rate for activated materials (case 1 to 4), the small radioactivity discharged into the atmosphere and the total public doses calculated lower than 4.4 x 10^{-8} μSv/a were obtained. In case with emission rate of 1% (case 2) for surface contaminated materials, total radioactivity discharged into the atmosphere increases to 2.5 x 10^{6} Bq and the total public dose accordingly increases to 9.0 x 10^{-7} μSv/a. In the worst case evaluation with filtrating efficiency of 99.0% (case 9) instead of 99.97%, the total radioactivity as high as 5.9 x 10^{8} Bq is discharged into the atmosphere and the public exposes 2.1 x 10^{-4} μSv/a. Compared with the criteria for the public, however, these evaluated value is five orders of magnitudes less than the public dose criterion of 0.15 mSv/a in the normal situation. The pathway of ground surface deposition which causes external exposure dose to public of more than 60% of entire annual public dose is dominant for all cases.

TABLE 37. CALCULATED RESULTS OF RADIONUCLIDES DISCHARGED INTO THE ATMOSPHERE

Nuclide	Residual Activity (July 1 2005)	Radioactivity released into atmosphere (Bq)								
		Case 1	Case 2	Case 3	Case 4	Case 5	Case 6	Case 7	Case 8	Case 9
Mn-54	2.9E+10	2.0E+01	2.9E+00	1.8E+02	7.4E+02	5.6E+03	1.4E+03	4.5E+03	1.5E+04	3.5E+06
Fe-55	4.0E+12	2.8E+03	4.3E+02	2.6E+04	1.1E+05	8.1E+05	2.0E+05	6.6E+05	2.2E+06	5.1E+08
Co-60	3.5E+11	2.5E+02	3.8E+01	2.3E+03	9.7E+03	7.3E+04	1.8E+04	5.9E+04	2.0E+05	4.6E+07
Ni-59	1.5E+09	1.1E+00	1.7E-01	1.0E+01	4.3E+01	3.2E+02	7.9E+01	2.6E+02	8.6E+02	2.0E+05
Ni-63	2.2E+11	1.5E+02	2.4E+01	1.4E+03	6.0E+03	4.5E+04	1.1E+04	3.7E+04	1.2E+05	2.8E+07
Tc-99	1.0E+05	7.0E-05	1.1E-05	6.7E-04	2.8E-03	2.1E-02	5.1E-03	1.7E-02	5.7E-02	1.3E+01
Sb-125	1.9E+10	1.3E+01	2.0E+00	1.2E+02	5.2E+02	3.8E+03	9.5E+02	3.1E+03	1.0E+04	2.4E+06
Pu-238	1.5E+05	1.1E-04	1.7E-05	1.0E-03	4.3E-03	3.2E-02	7.9E-03	2.6E-02	8.6E-02	2.0E+01
Pu-239	1.7E+04	1.2E-05	1.9E-06	1.2E-04	4.9E-04	3.6E-03	8.9E-04	3.0E-03	9.8E-03	2.3E+00
Pu-240	2.8E+04	1.9E-05	3.0E-06	1.8E-04	7.7E-04	5.8E-03	1.4E-03	4.7E-03	1.6E-02	3.6E+00
Pu-241	6.5E+06	4.5E-03	7.1E-04	4.3E-02	1.8E-01	1.3E+00	3.3E-01	1.1E+00	3.6E+00	8.5E+02
Am-241	1.2E+04	8.6E-06	1.3E-06	8.2E-05	3.4E-04	2.6E-03	6.3E-04	2.1E-03	6.9E-03	1.6E+00
Cm-244	2.0E+05	1.4E-04	2.1E-05	1.3E-03	5.4E-03	4.1E-02	1.0E-02	3.3E-02	1.1E-01	2.6E+01

TABLE 38. CALCULATED RESULTS OF PUBLIC DOSE FOR EACH PATHWAY AND TOTAL (μSV/A)

Pathway			Case 1	Case 2	Case 3	Case 4	Case 5
External	Cloudshine		3.8E-11	6.0E-12	3.6E-10	1.5E-09	1.1E-08
	Ground surface deposition		7.6E-10	1.2E-10	7.2E-09	3.0E-08	2.2E-07
	Direct gamma radiation		1.2E-11	1.6E-11	1.6E-11	1.2E-11	1.2E-11
	Skyshine radiation		4.4E-15	2.9E-15	2.9E-15	4.3E-15	4.3E-15
Internal	inhalation	Adult	3.8E-11	5.9E-12	3.6E-10	1.5E-09	1.1E-08
	Agricultural crops	Leaf	1.1E-10	1.8E-11	1.1E-09	4.6E-09	3.5E-08
		Root	9.9E-12	1.5E-12	9.3E-11	3.9E-10	2.9E-09
	Livestock	Milk	2.4E-11	7.3E-13	2.2E-10	9.4E-10	7.0E-09
		Meat	1.4E-10	5.1E-13	1.3E-09	5.4E-09	4.0E-08
Total			1.1E-09	1.7E-10	1.1E-08	4.4E-08	3.3E-07

Pathway			Case 6	Case 7	Case 8	Case 9
External	Cloudshine		2.8E-09	9.3E-09	3.1E-08	7.2E-06
	Ground surface deposition		5.5E-08	1.8E-07	6.1E-07	1.4E-04
	Direct gamma radiation		1.5E-11	1.5E-11	1.2E-11	1.1E-11
	Skyshine radiation		2.9E-15	2.9E-15	4.3E-15	4.2E-15
Internal	inhalation	Adult	2.8E-09	9.2E-09	3.0E-08	7.1E-06
	Agricultural crops	Leaf	8.5E-09	2.8E-08	9.3E-08	2.2E-05
		Root	7.1E-10	2.4E-09	7.9E-09	1.8E-06
	Livestock	Milk	1.7E-09	5.7E-09	1.9E-08	4.4E-06
		Meat	9.9E-09	3.3E-08	1.1E-07	2.5E-05
Total			8.1E-08	2.7E-07	9.0E-07	2.1E-04

APPENDIX III: RADIOLOGICAL ACCIDENT ANALYSIS FOR DECOMMISSIONING OF SYSTEMS 321 AND 322

III.1. SUMMARY OF RADIOLOGICAL ACCIDENT ANALYSIS

The radiological accident analysis has been divided into a number of accident scenarios, whose titles are given in the contents page. The analysis of each scenario identified in Table 39 is undertaken in section II.2. of Appendix III.

The basic approach adopted in section II.2. for each scenario is a graded approach, based on unmitigated consequences:

— Assess the unmitigated consequences of the accident scenarios;
— Compare the number of independent complete safety measures with the defence-in-depth criteria given in Section 2, Table 1 of this safety assessment;
— Identify the safety controls that make up the required independent complete safety measures; and
— For all scenarios, consider if the risk is As Low As Reasonably Achievable (ALARA).

Different formats are adopted in III.2. to reflect the above graded approach.

Only one scenario has consequences that are significant, namely Scenario 02, which considers accidents during cutting operations. Fig. 56 shows how the key scenario could develop. This radiological accident analysis shows that there are no major concerns with this scenario.

Table 40 provides a scenario schedule for hazards considered in this radiological accident analysis, the purpose of which is to provide a relatively brief but comprehensive listing of scenarios arising from internally initiated events, along with the Basket Safety Measures that are relevant to each initiating event. It can be seen that the criteria for defence-in-depth are satisfied, in terms of numbers of safety measures, for all scenarios.

Conclusions

Most of this radiological accident analysis has addressed System 321, since the hazard associated with System 322 is low.

Table 30 lists the important procedural safety controls identified in this radiological accident analysis, including procedures and parameters, along with their safety functions. Table 41 summarizes each safety important structure, system or component taken into account in this radiological accident analysis along with its safety function and performance requirements. The safety functions and performance requirements should be confirmed before decommissioning starts. If at any stage these safety functions and performance requirements are not met, or not known to be met, decommissioning has to stop or not be started, until this radiological accident analysis has been reviewed and any required changes implemented within the facility.

All scenarios have been subject to an ALARA review in Part 2. Two shortfalls have been identified, together with recommendations to address the shortfalls – see Table 42 This radiological accident analysis is still valid even if the recommendations are not completed – the recommendations relate to what may be considered to be good radiological practice. All that is required is for facility management to consider the shortfalls as part of the ‘Overall ALARA’ process. Subject to appropriate consideration/implementation of the shortfall/recommendation, it is judged from the perspective of radiological accident analysis that the risk will be ALARA.

Outstanding issues have been identified in Table 39. Unless otherwise indicated, any outstanding issues are to be resolved well before implementation of the safety case.

Provided that the shortfall/recommendation and the outstanding issues are addressed (including implementation of all the safety controls), it is considered that the radiological risks from accidents associated with decommissioning systems 321 and 322 are acceptable.

TABLE 39. SOURCE OF IDENTIFICATION OF HAZARDS COVERED BY THIS RADIOLOGICAL ACCIDENT ANALYSIS

Sequence from Section 4 of the Safety Assessment	Scenario Description	Scenario Reference
Sequence A1	Failure to control the spread of contamination arising from the cutting of pipework and components	02 Accidents during cutting operations
Sequence B1	Dropping pieces of cut contaminated pipework, either as these are manoeuvred away from the workface, or while being transported from the room in containers	03 Dropped loads
Sequence B2	Fire/explosion arising from hot cutting	Detailed analysis not required (see pre-amble to table)
Sequence B3	Power supply failure leading to a failure of the ventilation system	02 Accidents during cutting operations
Sequence B4	Operators fail to respond to alarms due to high noise environment.	02 Accidents during cutting operations (specifically a performance requirement within Table 1.4)
Sequence B5	Operators are unable to evacuate following an initiating event due to blocked emergency exits.	Detailed analysis not required (see pre-amble to table)
Sequence B6	Operators inadvertently enter areas of high radiation en-route to/from the workface	01 High external dose to worker

Section 4 of this Safety Assessment has screened the hazards and identified a number of ‘sequences’. It divided ‘sequences’ into those requiring detailed analysis (only one, Sequence A1) and others that ‘**do not require detailed analysis** but will need prevention, protection and/or mitigating measures derived from standard good practice workplace systems and procedures for normal operations’ (Sequences B1 to B6). This radiological accident analysis therefore only needs to address Sequence A1, but some other ‘sequences’ have been addressed anyway, as detailed below.

The ‘scenario reference’ column refers to the number of the scenario within Section III.2. of this radiological accident analysis.

TABLE 40. SCENARIO SCHEDULE

The detailed radiological accident analysis presented in III.2 of this Appendix is summarized in the following tables.

SCENARIO No. 01: High External Dose to Worker
Table III.1.2.01.1. Introduction

Description of potential consequences: Highest consequence threshold exceeded: Safety controls**:**	Worker: High external dose to worker Public: Not applicable Worker: 2 mSv Public: Not applicable See schedule.

Table III.1.2.01.2. Schedule

Initiating Event Description	Defence-in-depth Requirement	Safety Measures	Other safety controls
Extended time spent in high radiation area unintentionally. Procedure: Planning doses.	Worker: One independent complete safety measure Public: None	*Mitigating Systems (Worker)* **1. Evacuation** Procedure on issuing dose meters Equipment: Dose meter Evacuation following an alarm Defence-in-depth summary: One independent complete safety measure is provided – this meets requirements.	Parameter: whole body dose rates.
Conclusion on risk: provided that the relevant safety controls are implemented, it is considered that the radiological risk for this scenario is acceptable.			

SCENARIO No. 02: Accidents during Cutting Operations
Table III.1.2.02.1. Introduction

Description of potential consequences:	None for System 322. For System 321: Worker: High internal dose from airborne contamination Public: Dose from aerial release from reactor building
Highest consequence threshold exceeded:	Worker: 20 mSv Public: None
Safety controls:	See schedule

Table III.1.2.02.2. Schedule

Initiating Event Description	Defence-in-depth Requirement	Safety Measures	Other safety controls
Local ventilation extract fails. Equipment: Local ventilation extract Procedure: Starting cutting Procedure: Remote cutting	Worker: Two independent complete safety measures Public: None	*Mitigating Systems (Worker)* **1: Evacuation upon failure of local ventilation extract** Equipment: Alarm on failure of extract Procedure: Stopping cutting and evacuating **2: Respiratory protection** Procedure: Wearing respiratory protection Equipment: Respiratory protection equipment Defence-in-depth summary: Two independent complete safety measures are provided – this meets requirements.	Procedures: cutting methods and requiring an enclosure. Parameters: surface contamination levels, maximum length of cut during a work period, and volume of enclosure.
Conclusion on risk: provided that the shortfalls/recommendations identified are adequately addressed, and all the relevant safety controls are implemented, it is considered that the radiological risk for this scenario is acceptable.			

SCENARIO No. 03: Dropped Loads
Table III1.2.03.1. Introduction

Description of potential consequences: Highest consequence threshold exceeded: Safety controls:	None for System 322. For System 321: Worker: High internal dose from airborne contamination Public: Dose from aerial release from reactor building Worker: 2 mSv Public: None Parameter: Surface contamination.
Conclusion on risk: provided that the relevant safety control is implemented, it is considered that the radiological risk for this scenario is acceptable.	

The assumptions made and the intended procedures and the values of certain parameters used in this assessment are presented in Table 30

The ‘safety function’ column describes the function that is to be achieved by the procedure or the parameter for safety purposes. There may be other non-safety functions achieved by the procedure or parameter, which are not relevant to this radiological accident analysis.

Table 41 lists the assumptions made in this assessment about equipment, which may be structures, systems or components.

The ‘safety function’ column describes the function that is to be achieved by the structure, system or component; there may be other non-safety functions achieved by the structure, system or component, which are not relevant to this radiological accident analysis. The safety function is then amplified, if appropriate, by performance requirements, which represent minimum standards required by this radiological accident analysis. The ‘scenario reference’ column refers to the number of the Scenario within the detailed radiological accident analysis.

TABLE 41. SPECIFIC ENGINEERED SAFETY CONTROLS

Structure, System or Component	Safety Function(s)	Performance Requirements	Scenario Ref
Personal dose meters that incorporate an alarm on dose	To alarm when the dose reaches the alarm level	Alarm level is controllable to pre-defined levels	01
Ventilation extract for local enclosures, fitted with fans	To minimize spread of contamination from the enclosure	Extract to exhaust into the building ventilation system. Fans to provide a flow rate of air extracted from the enclosure exceeding 20 m^3/min.	02
Alarm for failure of local ventilation extract	To warn a worker within the tented area or size reduction facility that the local ventilation extract has failed	To alarm on loss of depression in local ventilation extract To alert the worker above the noise of cutting operations	02
Respiratory protection equipment	To mitigate worker dose when there is airborne contamination present	To provide a decontamination factor of 100 for particulate material	02
Filters on building ventilation system	To clean up the ventilation extract	To provide a decontamination factor of 100 for particulate material	02

Table 42. lists the shortfalls that need to be considered by facility management, as part of the overall ALARA process, before submission of the safety case. For each shortfall, the associated recommendation is a suggested improvement which would remove or significantly reduce the shortfall. If the recommendation is not considered practicable to implement, other improvements need to be considered, i.e. it is primarily the shortfall (rather than the recommendation) that must be addressed.

This radiological accident analysis is still valid even if the recommendations are not completed – the recommendations relate to what may be considered to be good radiological practice. All that is required is for facility management to consider the shortfalls as part of the 'Overall ALARA' process. Subject to appropriate consideration/implementation of the shortfall/recommendation, it is judged from the perspective of radiological accident analysis that the risk will be ALARA.

TABLE 42. SUMMARY OF SHORTFALLS AND RECOMMENDATIONS

No.	Scenario No.	Shortfall in this Scenario	Recommendation
1	02	There is no warning to workers of any release of airborne contamination from enclosures.	Provide airborne activity monitoring to warn of any airborne activity release from an enclosure, enabling evacuation if an alarm is activated.
2	02	Surface contamination levels on the inside of System 321 have not yet been reduced as far as reasonably achievable.	Before decommissioning operations begin, a decontamination process of System 321 has to be undertaken.

The outstanding issues, identified in this document, which need to be resolved are presented in Table 42. Unless otherwise indicated, these outstanding issues are to be resolved well before implementation of the safety case.

TABLE 43. SUMMARY OF OUTSTANDING ISSUES

No.	Outstanding Issue	Significance of Outstanding Issue	Internal Reference
1	Identified safety controls (procedures, parameters and equipment) given in earlier tables are confirmed by facility operators to provide the identified safety functions and performance requirements. Where this is not practicable until the time of the operation (e.g. ventilation flow rates into each tent constructed), there must be confidence that the safety control will be capable at the time of such provision. Furthermore, any degradation of safety controls during decommissioning operations must result in operations being stopped until this radiological accident analysis has been reviewed, and any changes in safety controls have been implemented.	Without this confirmation, the radiological accident analysis is not valid.	Tables 1.3 and 1.4
2	This radiological accident analysis needs to be implemented in accordance with robust site procedures covering training, supervision etc., within an appropriate safety culture. In particular, this requires the adoption of the general, specific and task-specific administrative controls defined in Reference 2.	Without this implementation, the radiological accident analysis is not valid.	Section 1.1

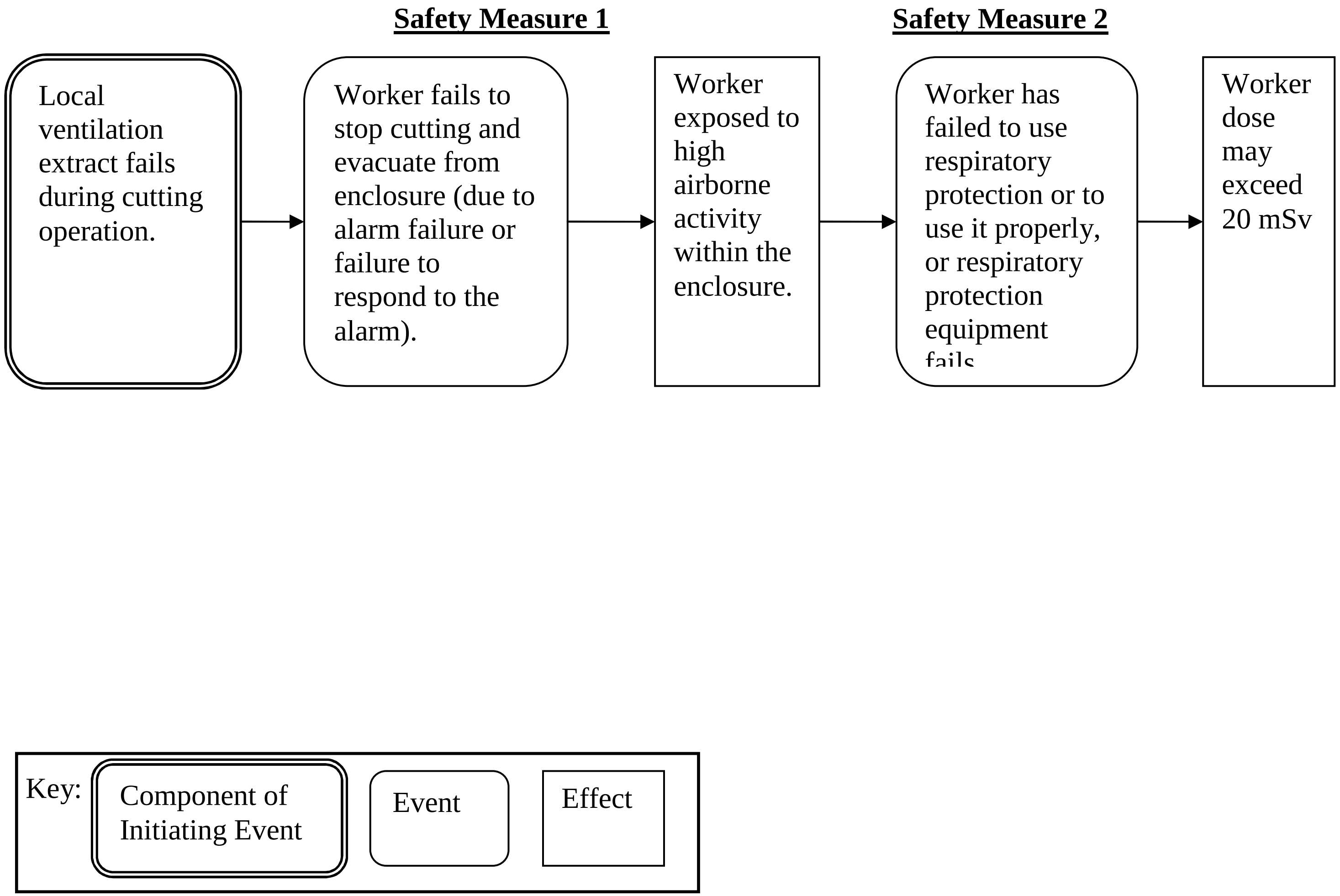

FIG. 56. Scenario progression diagram for faults during cutting operations.

II.2 DETAILED RADIOLOGICAL ACCIDENT ANALYSIS

Further to Section 5, more detailed information about the analysis of the accident scenarios for the decommissioning of the two systems 321 and 322 are provided below.

Where safety controls (procedures, parameters and equipment) are identified within the scenario, the safety function of each safety control is available in Table C1.3 or C1.4, and is not repeated within this Section.

(A) SCENARIO 01 HIGH EXTERNAL DOSE TO A WORKER

This is an example of a scenario for which the potential consequences of an accident are significant but not 'higher'.

Scenario No: 01	**Scenario Title:** **High external dose to a worker**
Qualitative Description of Initiating Event: Extended time spent in high radiation area unintentionally. **Procedure: High dose rate jobs are planned such that a target dose for each job must be defined.**	
Qualitative Description of Potential Consequences: High abnormal external radiation dose.	
Justification that Consequences are not in the Higher Category: **Parameter: Whole body external dose rates in working areas for decommissioning Systems 321 and 322 are less than 2 mSv/h.** Over an 8 hour shift (taken to be the absolute maximum period of exposure), the maximum dose that a worker could receive is 16 mSv, which is less than 20 mSv.	
Safety Measures: There is one operational mitigating system, made up of the following safety controls: **Procedure: Workers must be issued with personal dose meters that incorporate an alarm on dose, with the alarm level set at a lower dose than the target dose for the job.** **Equipment: Personal dose meters that incorporate an alarm on dose** **Procedure: Workers must evacuate the area if their personal dose meter alarm is activated.**	
ALARA Considerations: Although the safety measure is operational, and is only mitigating, there is high confidence that the safety measure will be effective. There are no other obvious safety measures that could be implemented that would significantly reduce the risk.	
Conclusion on Risk: Provided that all the above safety controls are implemented, it is considered that the radiological risk for this scenario is acceptable.	

(B) SCENARIO 02: ACCIDENTS DURING CUTTING OPERATIONS

This is an example of a scenario for which the potential consequences of an accident are 'higher'

There are no internal dose issues from cutting up system 322 metalwork, so this is not considered further here.

— **Parameter:** There is no significant contamination of System 322**.**

Abnormal external dose issues are addressed in Scenario 01.

Cutting operations on system 321 metalwork have the potential to create large amounts of airborne contamination, which must not be allowed to spread beyond the local area. In order to ensure that contamination does not spread, the enclosure (even if very temporary) must be ventilated to an appropriate standard:

— **Procedure:** Cutting operations for System 321 metalwork must be carried out in an enclosed area that is provided with a local ventilation extract. The enclosure must be well enclosed, even if temporary in nature.

— **Equipment: Ventilation extract for local enclosures, fitted with fans.**

Sequence B1 from Section 4 (power supply failure leading to failure of the ventilation system) is just one way in which the ventilation extract could fail, and is deemed to be incorporated within the above equipment safety control.

— **Description of Scenario**

This section includes a description of the initiating events and scenario progression, but first the potential consequences are assessed, since on a graded approach only the scenarios with higher consequences require more detailed analysis.

(i) Assessment of consequences

Consequences are assessed for workers (both directly involved workers and nearby workers), and for the public. In both cases, it is necessary to understand how much is normally released in the cutting operation.

In order to calculate a dose the following data is required:

- A release fraction of 0.1 Bq/cm of cut per Bq cm^{-2} is adopted based on working experience. To support this release fraction, the following procedure is required:
  Procedure: System 321 metalwork may be cut by a plasma torch, mechanical shear, a reciprocating saw, a band saw, or by an oxy-acetylene torch, but must not be cut by a grinder.
- The length of cut during a work period. Information provided by NPP personnel indicates a maximum of 1.5 m:
  Parameter: The maximum length of cut during a work period is 1.5 m.
- The level of contamination on the inner surface of the metalwork. It is understood that the activation products are there as a result of deposition of activity on the inside of the system, not as a result of direct irradiation of the system. Thus all the radionuclides, both those arising from spent fuel and those arising from activation, are within a thin layer on the inside surface of the pipework etc. Reference [7] provides information on typical surface contamination levels, and a factor of two between the worst location and a typical location is also allowed; it is presented as a parameter in case evidence to the contrary arises during decommissioning:

Parameter: Surface contamination levels on the inside of System 321 (averaged over the area of any single cut) do not exceed:

	Activity [Bq/cm^2]
	System 321
Mn-54	5.2×10^4
Fe-55	7.6×10^5
Co-60	6.8×10^5
Ni-59	3.0×10^3
Ni-63	4.2×10^5
Tc-99	2.0×10^{-1}
Sb-125	3.6×10^4
Pu-238	3.0×10^{-1}
Pu-239	3.4×10^{-2}
Pu-240	5.4×10^{-2}
Pu-241	1.3×10^1
Am-241	2.4×10^{-2}
Cm-244	3.8×10^{-1}

Note that no allowance is made for radioactive decay (which would reduce the above activities), thus placing no constraints on a start date for dismantling operations.

Unmitigated Worker Dose

In order to calculate the potential worker dose, the above data has been combined with other relevant data within a computer model. This gives a dose of 0.64 mSv per cm of cut for maximum contamination levels, conservatively assuming:

- The worker is present in the enclosure throughout the cutting operation;
- No ventilation working;
- No activity dropping out of the air during the cutting operation; and
- An enclosure size as small as 8 m^3.

  Parameter: The volume of an enclosure in which system 321 metalwork is cut is greater than 8 m^3.

The computer model output also showed that Co-60 was by far the dominant radionuclide in terms of worker dose.

For worst contamination levels and the maximum length of cut, this gives a dose of:

$$0.64 \text{ mSv/cm} \times 150 \text{ cm} = 96 \text{ mSv} \qquad (35)$$

This exceeds the 20 mSv consequence threshold, and therefore this scenario requires analysis against defence-in-depth criteria.

Mitigated Worker Dose

Worker dose is mitigated by two systems:

- The local ventilation extract of the enclosure; and
- Respiratory protection worn by the worker

When the local ventilation extract is working, the airborne concentration is reduced from the assumptions in the computer model calculation. Calculations show that this reduces the dose by factor of more than 50, assuming a flow rate of air through the enclosure of 20 m^3 per minute. Failure of this extract means that cutting operations do not start or are discontinued:

Safety Measure (Mitigating System): Cutting operations only occur while the local ventilation extract is working. This consists of the following safety controls:

Procedure: At the start of a System 321 cutting operation with a worker present, the local ventilation extract must be working. If it fails during the cutting operation, the worker must cease cutting operations and evacuate from the enclosure.

Equipment: Alarm for failure of local ventilation extract.

Respiratory protection must be capable of reducing doses to the worker by a further factor of 100:

Safety Measure: Respiratory protection. This consists of the following safety controls:

Procedure: During System 321 cutting operations, the worker present must wear respiratory protection.

Equipment: Respiratory protection equipment

The expected dose during a large cutting operation is therefore less than 96 mSv/ (100 x 50) that is less than 0.02 mSv. If just one of the mitigating systems is working, the maximum dose would be less than 0.2 mSv, which is significantly less than the 20 mSv threshold used in this analysis.

Nearby workers

Nearby workers (outside the enclosure) would not be expected to wear respiratory protection, so they are at greater risk in this respect. Nearby workers could be at risk either from:

- The exhaust from the local extract, or
- Activity escaping from the enclosure.

With regard to the exhaust from the local extract, this pathway can be eliminated by routing the exhaust directly into the building ventilation system. Thus a performance requirement has been added to Table III.1.3.

With regard to activity escaping from the enclosure:

- The decontamination factor provided by an enclosure that is well enclosed and ventilated is so large that doses to these workers would be negligible.
- If the local ventilation extract fails, the decontamination factor provided by the enclosure would reduce, but would not be lower than that provided by a respirator.

Thus there is no need to separately consider the nearby worker, as the worker doing the cutting represents the bounding case.

Public

The unmitigated consequence assumes that the airborne activity created during cutting operations is fed directly outside the building with no filtration. However, the dose to the public arising from this is very significantly lower than 10 μSv, so no safety measures are required. Nevertheless, it seems appropriate to minimize aerial releases as far as reasonably achievable:

Equipment: Filters on building ventilation system

A nominal decontamination factor of 100 is assigned for the filters within Table III.1.3., although in practice much greater decontamination factors must be achievable.

Initiating Event

The main initiating event is failure of the ventilation extract during cutting operations.

Another initiating event may be postulated, namely starting cutting operations without the ventilation extract working. However, given that the alarm must be activated at this time, this initiating event appears to be too remote to warrant detailed consideration.

Scenario Progression

Figure 56 illustrates the scenario progression.

(ii) Comparison with Criteria

Comparison with Defence-in-Depth Criteria

The number of independent complete safety measures required for the public is zero, based on insignificant consequences. This places no requirements on the analysis of the scenario.

The number of independent complete safety measures required for the worker is conservatively assigned as two. Two independent complete safety measures have been defined above, so the criteria are met. However, they do not consist entirely of engineered protection. The implications of this are explored in the ALARA section below.

Comparison with Probabilistic Risk Criteria

A quantitative analysis has not been carried out. It is considered qualitatively that the risk for this scenario is low enough to avoid quantitative analysis for the following reasons:

- There are two independent safety measures available (acknowledging that both require human action);
- The safety measures are simple to carry out, and must have an intrinsically high reliability; and
- Both safety measures are fully independent from the initiating event.

Thus it is considered that little would be gained from a numerical analysis. Furthermore, provided that the shortfall/recommendation identified below is adequately addressed, and all the above safety controls are implemented, it is considered that the radiological risk for this scenario is acceptable.

(iii) ALARA considerations for this scenario

The number of safety measures satisfies the defence-in-depth criteria, and the risk is assessed to be low, but the safety measures do not consist entirely of engineered protection.

Consideration has been given to options for improving the type of safety measure; however, no obvious worthwhile improvements have been identified. Nevertheless, the following procedure is identified to help minimize risk as far as reasonably achievable:

> **Procedure: Where achievable, cutting operations on System 321 need to be carried out remotely.**

The analysis has not assumed that any activity-in-air monitors are present, since it is understood that none are available. The main purpose of such a monitor is to warn workers in areas where they do not have to wear respiratory protection that airborne activity is present; there is thus little advantage in providing any such monitor within an enclosure, as airborne contamination is to be expected for cutting up System 321, and respiratory protection will be worn by workers when cutting anyway. The following is considered to be a shortfall:

> **Shortfall: There is no warning to workers of any release of airborne contamination from enclosures.**

The following associated recommendation is made:

> **Recommendation: Provide airborne activity monitoring to warn of any airborne activity release from an enclosure, enabling evacuation if an alarm is activated.**

If this is not provided, this radiological accident analysis is not undermined, but it is still considered appropriate for facility management to consider this shortfall/ recommendation.

The analysis has been performed using current data on surface contamination levels, but it is understood that System 321 will be subject to decontamination prior to the start of decommissioning operations. The following is considered to be a shortfall based on the current situation:

> **Shortfall: Surface contamination levels on the inside of System 321 have not yet been reduced as far as reasonably achievable.**

The following associated recommendation is made:

> **Recommendation: Before decommissioning operations begin, a decontamination process of System 321 needs to be undertaken.**

If this is not provided, this radiological accident analysis is not undermined, but it is still considered appropriate for facility management to consider this shortfall/ recommendation. Note that this analysis does not cover this decontamination process, which if it is decided to be carried out, needs to be subject to a separate suitable safety assessment.

(C) SCENARIO 03 Dropped Loads

This is an example of a scenario for which the potential consequences of an accident are negligible.

Scenario No: 03	**Scenario Title:** **Dropped Loads**
Qualitative Description of Potential Consequences: For System 322, there are no radiological hazards. For System 321, apart from any industrial hazards, a small amount of contamination held on the inside of metalwork from System 321 could be released.	
Justification of Negligible Consequences: In comparison to Scenario 02, the area of surface contamination that is involved in the release is much higher. However, the following bullet points all indicate a lower release in practice: • The contamination is well fixed to the inside surface of the metalwork, following previous decontamination processes; • The ‘challenge’ of a drop is **far lower** than that of the allowed cutting mechanisms, giving release fractions orders of magnitude lower; • Any airborne activity released inside lengths of piping needs to get to the end of the pipe to become a hazard; • The likely low respirability of any contamination made airborne. A quick calculation (based on surface contamination levels as defined in Scenario 02, and assuming a 10 m drop height and a long length of pipework being dropped) has indicated a consequence to the worker well below 2 mSv, so the consequences are in the negligible consequence region. It is acknowledged that greater drop heights are feasible, but this would not be enough to make the consequences significant. **[Parameter: Surface contamination levels as in Scenario 02]** From a comparison with Scenario 02, it follows that the public consequences are negligible.	
ALARA Considerations: There are no obvious improvements for implementation.	
Conclusion on Risk: Provided that the above safety control is implemented, it is considered that the radiological risk for this scenario is acceptable.	

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