SAFETY ASSESSMENT FOR DECOMMISSIONING

Annex II

Graded Approach to Safety Assessment for Decommissioning of Facilities Using Radioactive Material

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CONTENTS

FOREWORD

The purpose of Annex II is to present a summary of lessons learned and recommendations on the application of the graded approach to safety assessment for decommissioning of facilities using radioactive material, developed by the Graded Approach Working Group within the International Atomic Energy Agency's DeSa project (Evaluation and Demonstration of Safety for Decommissioning of Facilities Using Radioactive Material), that started with a meeting in October 2004. This Annex is meant as a supporting document to the main report on Safety Assessment for Decommissioning of Facilities Using Radioactive Material, the Application of the Safety Assessment Methodology to Specific Facilities (Annex I) and the Report on the Regulatory Review of Safety Assessment (Annex III) by providing in-depth information and highlighting additional issues and providing specific examples on the use of a graded approach. An additional aim is to describe experience on how a graded approach can be used in safety assessments for decommissioning by providing country-specific examples.

1. INTRODUCTION

1.1. BACKGROUND

According to the International Atomic Energy Agency (IAEA)'s Safety Guide [1], the role and objective of a safety assessment for a facility using radioactive material, also with respect to decommissioning, can be summarized as follows:

(a) To provide a systematic evaluation of the nature, magnitude and likelihood of consequences to workers, public or the environment during planned activities and in accident conditions;

(b) To quantify the systematic and progressive reduction in hazard potential that will be achieved through implementation of the decommissioning activities;

(c) To identify the limits, controls and conditions that will need to be applied to the decommissioning activities to ensure that the relevant safety standards are met and maintained throughout the decommissioning project;

(d) To demonstrate that the institutional controls after decommissioning are optimized without imposing undue burden to future generations;

(e) To provide input to on- and off-site emergency planning and to safety management arrangements;

(f) To provide an input to the identification of training needs for decommissioning and competencies of staff performing these activities;

(g) To document how the regulatory requirements and criteria are met for the authorization of the proposed decommissioning activities; and

(h) To demonstrate that the decommissioning plan is consistent with the decommissioning strategies and relevant requirements and criteria.

To fulfil these objectives, assessment of risks, doses or other consequences from decommissioning are necessary. Furthermore, the range of decommissioning activities for which such safety assessment is required is very broad, and the requirements relating to the scope, extent and level of detail of safety assessment vary across this range according to the level of hazard, to use a graded approach. The level of detail of the assessment needs to be adequate to obtain the result with the desired accuracy, with the effort spent for carrying out the analysis being commensurate with the hazard to be analyzed.

1.2. SCOPE

This Annex describes the application of the graded approach in safety assessment for facilities which are undergoing decommissioning or for which decommissioning activities are planned. It is meant as a supporting document to the main report “Safety Assessment Methodology for Decommissioning of Facilities Using Radioactive Material” which describes the results of the safety assessment methodology.

1.3. OBJECTIVES

The report aims to assist operators, regulatory bodies and supporting technical organizations and experts, involved in the development or review of safety assessment for decommissioning of different types of facilities that use radioactive material.

1.4. STRUCTURE OF THE REPORT

The report is structured as follows:

— Section 2 presents the concept of the graded approach, including a definition of the graded approach with respect to safety assessment for decommissioning.

— Section 3 presents the areas and steps in safety assessment for decommissioning in which a graded approach can be effectively applied, to direct the effort to the most relevant aspects of the safety assessment process and avoid using resources for low priority tasks. This section identifies five steps within safety assessment that are addressed in the following sections.

— Sections 4 to 8 discuss the application of the graded approach during the various steps of safety assessment as defined in Section 3, presenting the concept of graded approach in various contexts. These sections also provide general examples, which are further corroborated by examples provided in the Appendices.

— Section 9 presents the application of the graded approach to the three DeSa Test Cases. These test cases refer to three different types of facilities with significant differences in complexity and decommissioning work (a nuclear power plant - NPP, a research reactor, a nuclear laboratory), which is reflected in the complexity and approaches of the safety assessment.

— Section 10 provides the conclusions and recommendations.

— The Appendices contain additional country specific examples in relation to Sections 4 to 8 of the main part of this Annex.

2. THE CONCEPT OF THE GRADED APPROACH

The particular emphasis of this report is the use of the graded approach in those areas of safety assessment for decommissioning of facilities, where the means and procedures as well as the overall effort, can correspond to the hazard potential of the work to be assessed and the required level of detail of the data required for the assessment of facilities with different level of complexity and at different stage of their lifetime or decommissioning process. This includes:

— The graded approach in the description and characterization of the facility and in planning for decommissioning activities;

— The graded approach in carrying out safety assessment for decommissioning as planned;

- The graded approach in carrying out safety assessment for incidents or accidents occurring during decommissioning; and
- The graded approach in the (regulatory or independent) review of safety assessment.

The application of the graded approach aims at focusing the efforts of developers and reviewers to optimize resources and focus on the most important safety related aspects of decommissioning:

a) Without compromising safety; and
b) Being in compliance with safety requirements and criteria.

The graded approach concept is applied to all phases of the facility lifecycle – design, operation and decommissioning. The new international requirements on decommissioning of facilities using radioactive material [2] requires a graded approach to be applied to the development, content, review and updating of the decommissioning plans for facilities with different complexities and hazard potentials. It refers to the application of the safety requirements in a graded manner to the various types of facilities (NPPs, nuclear fuel cycle facilities, research reactors, laboratories, etc.), in accordance with the hazard potential.

There is no consistent definition of a "*graded approach*" to date. Examples for attempts to define the meaning of "graded approach" with respect to safety assessment can be found in various international documents, i.e.:

- *The <u>graded approach</u> is an application of the safety requirements that is commensurate with the characteristics of the practice or source and with the magnitude and likelihood of the exposures.* [2]
- *For a system of control, such as a regulatory system or a safety system, a process or method in which the stringency of the control measures and conditions to be applied is <u>commensurate</u>, to the extent practicable, with the likelihood and possible consequences of, and the level of risk associated with, a loss of control.*
- *An application of safety requirements that is commensurate with the characteristics of the practice or source and with the magnitude and likelihood of the exposures* [3].
- *<u>Graded Approach</u> means a process by which the level of analysis, documentation, and actions necessary to comply with a requirement in this part are commensurate with: (i) the relative importance to safety, safeguards, and security; (ii) the magnitude of any hazard involved; (iii) the life cycle stage of a facility; (iv) the programmatic mission of a facility; (v) the particular characteristics of a facility; and (vi) any other relevant factor* [4].

Taking into account this variety of pertinent definitions from other fields, the Graded Approach Working Group within the DeSa project suggested the following definition of a graded approach with respect to safety assessment for decommissioning:

<u>Graded Approach</u> with respect to safety assessment for a facility 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 legal requirements and safety requirements are commensurate with:

(a) The magnitude of any hazard (radiological or non-radiological) involved, associated with the facility or the work to be carried out;

(b) The particular characteristics of a facility:

- *Characteristics of a facility are the its type and size, the radioactive source term, consisting of the radionuclide vector, activity levels etc. as well as other unique features; and*
- *Other characteristics of a facility are events from the operational history, the level of ageing, the safety culture applied during operation and decommissioning etc.*

(c) Requirements/demands by the Regulatory Body;

(d) The step within the decommissioning process - The step within the decommissioning process depends on the decommissioning strategy (deferred – immediate decommissioning) and on the stage within this strategy, i.e. initial characterization, dismantling of peripheral systems, dismantling of highly contaminated or activated systems, final survey for release etc.; and

(e) The balance between radiological and non-radiological hazard(s).

Furthermore, the use of a graded approach in the implementation of safety assessments should be accompanied by a graded approach in the regulatory review process.

It is clear that the graded approach with respect to safety assessment for decommissioning is more a general concept than a strict prescription how to perform such analyses. This concept and the above definition are illustrated by examples in this report.

According to Ref. [1] and the main report, the application of the graded approach needs to take into account:

— 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-points of the decommissioning process and the end-state of the facility (e.g. 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 safety assessment 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 (shutdown after normal operation, or shutdown 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); and
- Availability of applicable safety assessment for this or other similar facilities or novelty of proposed decommissioning activities.

3 GRADED APPROACH PROCESS FOR DECOMMISSIONING SAFETY ASSESSMENT

3.1. OVERVIEW

Safety assessment for the decommissioning of facilities using radioactive material requires complex considerations and consumes considerable effort. It is therefore important to direct this effort to those areas where the highest dose/risk potential prevails and where the assessments serve the best purpose – protection of workers, public and the environment.

Figure 3.1 illustrates that the level of detail and the complexity in which a safety assessment needs to be carried out not only depends on the type and hazard potential of the facility, prior to the start-up of decommissioning, but also on the specific decommissioning activities. The figure lists main types of facilities of decreasing overall hazard potential. Decommissioning of each type of facility is assumed to be divided into various phases – e.g. decontamination dismantling, removal of material release of site from regulatory control.

The coloured blocks indicate the required level of detail in which the safety assessment is carried out on a qualitative basis. In a very simplified example, decommissioning of a NPP or, more general, of any complex and large facility, may be structured as follows:

(a) Dismantling and decontamination of contaminated components, as well as a facility of special equipment needed for the next phase (b), like remote controlled segmenting techniques etc.;

(b) Dismantling of the reactor pressure vessel and the bioshield;

(c) Dismantling of any remaining activated and contaminated systems and structures;

(d) Decontamination of the buildings; and

(e) Release of the buildings and the site from radiological control, including possibly conventional dismantling of the buildings.

According to the idea of the graded approach, the safety assessment for phases a) and b) would need to be the most complex, as during phase a) an overall assessment of the safety for the entire project would be needed, and phase b) is related to work at the systems with the highest activity. The red colour indicates that the safety assessment for these phases would usually require the highest effort. The safety assessment for phases c) and d) could be kept less complex, indicated by colours orange and yellow. Finally, the safety assessment for phase e) would correspond to a very simple analysis of the least complexity, indicated by the colour green. Similar considerations apply to other types of facilities, which are indicated in Fig. 3.1 with medium and low complexity. This figure illustrates how the level of detail and the complexity of safety assessment depend on the hazard potential of the facility and the planned decommissioning work.

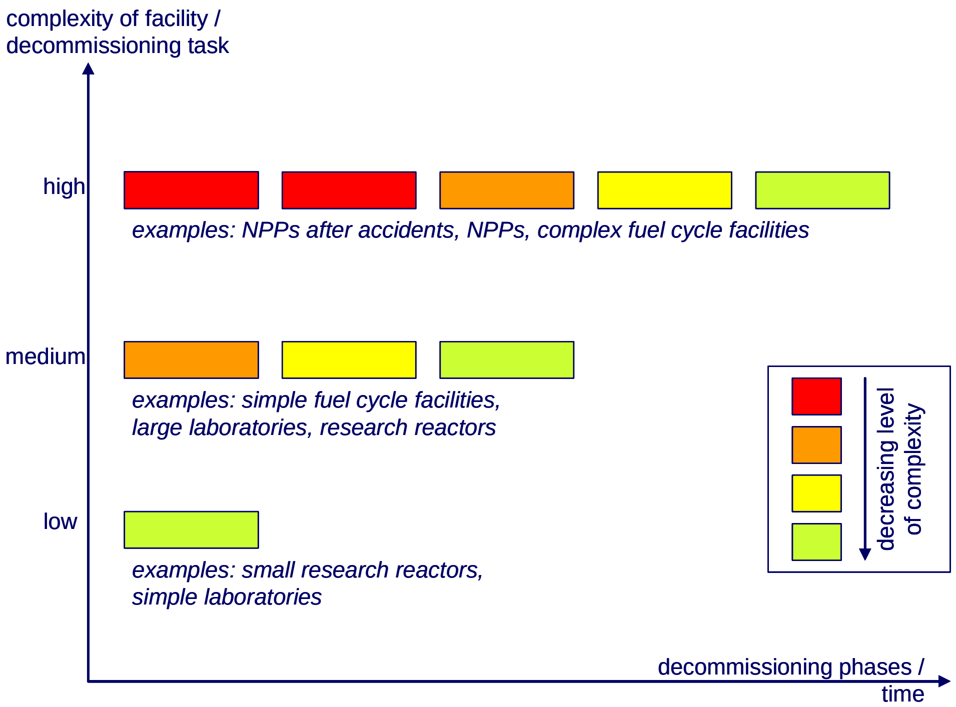

FIG. 1. Influence of the hazard potential of the facility and the planned work on the level of detail and the complexity of safety assessment.

The change of the risk profile with time, as addressed in Fig. 1, is further illustrated in Fig. 2, which is taken from the DeSa NPP Test Case (see Annex I). This figure illustrates that the overall risk and especially the radiological risk profile will be significantly reduced when the operational phase of the NPP has ended. A further reduction is associated with defueling. The dismantling phase is initially connected with a slight increase in radiological risk, e.g. when systems are opened and the authorized releases may temporarily fluctuate. In the long run, the risk during the dismantling phase will also decrease gradually until it will drop to zero when the site has reached free release conditions. Figure 2 also contains a profile of the industrial risk. It is indicated that the industrial risk will temporarily increase at times when new activities commence, like at the start of defueling, at the start of dismantling etc. During the dismantling phase, the industrial risk will initially even be higher than during operation, as dismantling involves totally new work activities. The industrial risk will, however, also decrease with the decommissioning progress in the long run.

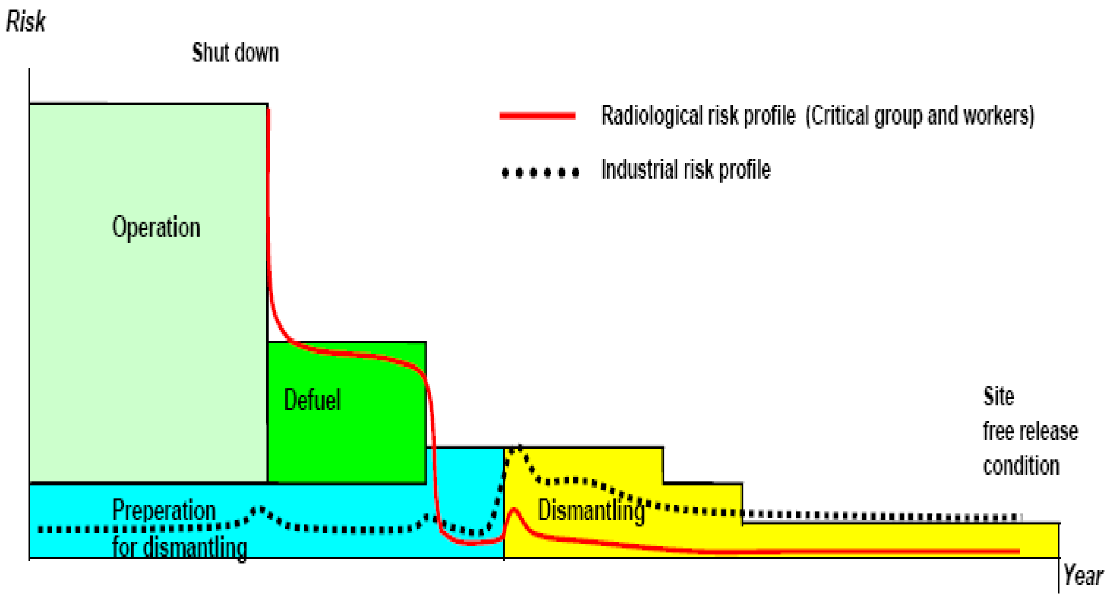

FIG. 2. Illustration of the changing risk profile for an NPP (Annex I, part A of this report).

3.2. GRADED APPROACH PROCESS

Based on the review of countries experience, five main steps for the application of the graded approach can be identified (see also Fig. 3):

— Step 1 – Definition of regulatory requirements and criteria that form the boundary for the safety assessment;

— Step 2 – Preliminary analysis of the facility, including radiological characterization, possible release pathways etc.;

— Step 3 – Hazard categorization and preliminary hazard assessment of the facility and its systems, structures and components, in order to gain a first overview of the hazard potential and to assign the facility/the decommissioning work into the appropriate hazard category;

— Step 4 – Performance of safety assessment for decommissioning; and

— Step 5 – Implementation of the safety assessment results.

1. Identification of requirements for the safety assessment by the regulatory framework
   Where can / may a graded approach be used?

2. Preliminary analysis of the facility
   What are the key features of the facility (activity inventory, possible release pathways etc.)?

3. Hazard categorization and preliminary hazard assessment of the facility and its systems, structures and components
   What is the overall hazard potential? What are the most relevant parts of the facility?

4. Safety assessment for decommissioning
   How can the SA be carried out commensurate to the overall hazard potential, i.e. with sufficient level of detail, yet avoiding wasting effort at irrelevant issues?

5. Implementation of safety assessment results
   How are the results implemented? How does the intended use of the results influence the decommissioning?

FIG. 3. Overview of steps in a safety assessment where a graded approach can be applied.

These steps are introduced briefly below:

(a) Identification of requirements for the safety assessment by the legal framework (Step 1)

Safety assessment has to be carried out according to the legal framework of the particular country in which the facility is located. The regulations of a particular country may contain requirements for:

— Categorization of facilities and grading of the relevant safety related documents (e.g. US Nuclear Regulatory Commission's categorization of facilities [9]); and

— Safety assessment, in particular how they are to be carried out; or others.

In some cases, these requirements may differ according to the hazard potential or type of facility. Section 4 outlines to which parts of the safety assessment these requirements may pertain and gives examples how grading may be implemented in the regulatory framework.

(b) Preliminary analysis of the facility (Step 2)

The preliminary analysis of the facility pursues the aim of providing an initial estimate of hazards that the facility and the foreseen decommissioning activities may pose. The way in which this preliminary hazard assessment may be carried out may depend on the type of the facility, the existing knowledge of its operation history and on other factors. A graded approach for performing this preliminary hazard assessment may be used depending on these factors. Section 5 describes the approach and gives examples.

(c) Hazard categorization and preliminary hazard assessment of the facility and its systems, structures and components (Step 3)

It is common practice to assign the facility and the parts of the facility, mainly the systems and rooms, to certain categories, usually on the basis of the radioactivity contents (e.g. contamination, activation), the type of radionuclides present, the possibility for mobilizing contamination, etc. The way in which this categorization is performed may vary according to the complexity of the situation, ranging from simple conclusions drawn from the facility's history to a detailed radiological sampling and evaluation programme. Section 6 describes the approaches and the methods that can be used for the categorization of the facility and its systems depending on the characteristics of the facility and the situation to be assessed.

(d) Performance of safety assessment for decommissioning: evaluation of hazard consequences for decommissioning as planned and for incident or accident situations (Step 4)

The numerical evaluation of hazard consequences in terms of doses or risk to public and workers needs to address:

— Planned and abnormal decommissioning conditions for workers;

— Planned decommissioning conditions to critical groups of the public; and

— Incident or accident conditions during decommissioning to critical groups of the public,

has the largest potential for using a graded approach, i.e. assessment methods that are commensurate in terms of complexity and effort with the risk level or hazard potential that is to be assessed. First indications where such levels are to be expected can be derived from the preliminary analysis of the facility (see point b above). Section 7 describes at which points of the assessments grading is possible and provides examples. Grading can be also based on the consequences from the normal and accidental scenarios; e.g. (i) off-site impacts; (ii) on-site impacts; and (iii) impacts within the facility premises.

(e) Implementation of the safety assessment results (documentation, review, training of personnel, qualification of procedures, oversight, etc.) (Step 5)

Execution of the safety assessment in various levels of details may also have an influence on the application of the results. Therefore, it has to be considered how to apply the results when the safety assessment is revised. In particular, the required documentation, the way in which the safety assessment will be reviewed, the identification of the need for training, aspects of regulatory surveillance and other aspects will have an influence on how to carry out safety assessment. Section 8 outlines these interdependencies and provides examples.

4. THE GRADED APPROACH IN THE IDENTIFICATION OF REQUIREMENTS FOR SAFETY ASSESSMENT (STEP 1)

4.1. OVERVIEW

Safety assessment for decommissioning needs to be carried out in accordance with the relevant regulations in the particular country. This means that requirements may exist concerning which scenarios or pathways are to be taken into account or which calculation methods need to be used, regardless whether their application may be justified by the complexity and risk potential of the facility. Examples for such requirements are given in Section 4.2.

Even though there may be legal requirements prescribing certain approaches, the use of a graded approach may still be possible. For example, the use of site-specific data and approaches may be allowed so that the level of detail of the analysis can correspond to the site characteristics.

A graded approach in the legal framework as described here usually only pertains to the way in which the safety assessment is to be carried out. It usually does not affect the criteria (e.g. dose limits, limits for risk, etc.) against which the results of the safety assessment have to be evaluated.

4.2. LEGALLY BOUNDING REQUIREMENTS FOR SAFETY ASSESSMENT

Requirements for safety assessment may comprise of, e.g.:

(a) The scenarios to be taken into account (e.g. airplane crash etc., depending on the national requirements);

(b) Certain parameter values to be used, e.g. for certain critical groups (e.g. prescribed data on dietary habits, default exposure time etc.);

(c) Certain calculation models/prescriptions (e.g. the use of calculation models laid down in the regulatory framework of a particular country);

(d) Certain computer models which need to be used (e.g. certain computer codes like RESRAD [5]); and

(e) Optimization requirements (e.g. requirement to minimize doses even below dose constraints, as is required in certain countries).

Furthermore, the risk limits or dose constraints/dose limits according to which the safety assessment has to be carried out will usually be prescribed in the regulatory framework.

4.3. COUNTRY SPECIFIC EXAMPLES FOR A GRADED APPROACH IN REQUIREMENTS FOR SAFETY ASSESSMENT

Country specific examples that illustrate the prescription of the use of certain calculation models for dose assessments are provided in Appendix I.

4.3.1. Dose assessments calculations via air and water pathways (Germany)

The way in which dose assessments for nuclear facilities, both in operation and in the decommissioning, are to be carried out is prescribed by the German Radiation Protection Ordinance in combination with a General Administrative Regulation. Grading exists with respect to the calculation: A detailed model with complex calculations for the dispersion of radionuclides via airborne and water pathways in the environment and from there via various food chains and water consumption to man has to be used for large nuclear facilities, i.e. those requiring a licence according to the German Atomic Energy Act. For smaller facilities (i.e. those not requiring such a licence) pre-defined default values for concentrations in air and water can be directly applied, while no calculations are required. This illustrates that the type of facility and thus the hazard potential determines whether a detailed and complex analysis is necessary or whether simply default concentration values may be used. – In this case, grading consists of the use of a complex approach (full model calculations) vs. the use of a simple approach (pre-defined threshold values) for different types of facilities.

4.3.2. Derivation of release levels with the RESRAD code (USA)

The RESRAD code uses a computer model designed to estimate doses and risks from RESidual RADioactive materials [5], issued by the Environmental Assessment Division of Argonne National Laboratory (USA). There are model versions for materials, buildings and sites. The computer codes for these models are used in dose/risk assessments for nuclear facilities in the USA as a kind of standard. A graded approach with these computer codes is possible to illustrate as follows: the RESRAD models incorporate all necessary (default) parameter values and exposure pathways that would cover any generic exposure condition. The user can apply these data and assumptions for deriving suitable exposure assessments and release criteria. It is, however, also possible to use site-specific parameters that are derived from evaluations of the conditions prevailing at that particular facility and/or to exclude certain exposure pathways on the basis of site-specific evaluations. In this case, the graded approach applies to the complexity of the input data for the RESRAD model calculations (generic vs. site-specific), as well as to the clearance levels that are calculated by this model.

4.3.3. Dose assessments for NORM facilities according to the MR AGIS (The Netherlands)

The Radiation Protection Ordinance of The Netherlands, the *Besluit Stralingsbescherming* [6], defines when activities, i.e. activities dealing with naturally occurring radioactive materials (NORM), fall under reporting or authorization requirements. More detailed regulation regarding activities and standard forms for reporting or license applications are given in the ministerial guideline mr-NABIS [7]. Guidelines and methods for risk assessments are in another ministerial guideline, the mr-AGIS. This legislation is applicable to all operations dealing with NORM including decommissioning. Grading has been introduced in the way in which risk assessments for work activities falling under reporting requirements have to be carried out. As a first step, it is possible to use conservative rough estimates based on e.g. extrapolation of dose rate measurements or gamma dose constants. Only if specific dose limits are exceeded, more detailed calculations are required. The calculation of the dose to the public by external irradiation can be performed by conservative yet realistic estimation in the first instance. If the result of this rough estimate is less than 10 µSv/a, a more precise assessment is not required, and no stringent requirements are prescribed with regard to the implementation of the ALARA (As Low as Reasonably Achievable) principle. Otherwise, more complex calculations are required. If the total activity to be discharged is lower than the exemption values for discharges specified in the *Besluit Stralingsbescherming [6]*, the discharges are exempted and no site-specific risk estimate is required. Discharges above the exemption level fall under authorization regime. In this case, a site-specific risk estimate is required. In this case, grading relates to the complexity of the required assessments depending on the activity concentration in the material.

4.3.4. Identification of requirements and criteria for the safety assessment (Cuba)

While the main requirements for the decommissioning of facilities and for the conduction of safety assessment are outlined in the National Basic Safety Standards and other regulations such as the “Regulation for the Authorization of Practices associated with the use of Nuclear Energy”, no national regulation specifically addressing decommissioning activities exists in Cuba. Requirements for safety assessment exist, but leave flexibility. In the case of decommissioning of a brachytherapy facility located at the National Institute of Oncology and Radiobiology (INOR) contaminated with Cs-137 from leaking disused sources, the initial requirements established by the Regulatory Body for decommissioning could not be met. This flexibility in the regulations was subsequently used to derive radiological criteria for clearance on the basis that the annual dose received by members of the public (after the facility is release from regulatory control for non-nuclear use) must not exceed 0.3 mSv above the natural background in the worst-case scenario. This was then translated into operational reference levels in term of dose rate and specific activity that were used during decontamination and decommissioning. In this case, grading related to the derivation of clearance criteria in accordance with the characteristics of the situation to be analyzed, based on the flexibility allowed by the regulatory framework.

5. THE GRADED APPROACH IN THE PRELIMINARY ANALYSIS OF THE FACILITY (STEP 2)

5.1. OVERVIEW

A preliminary analysis of the facility regarding safety during decommissioning may already be carried out during the design and operational phase of the facility (long before the planned date for final shutdown) and in the development of the safety assessment. During this preliminary analysis of the facility, the basic data needed to carry out the safety assessment are compiled and preliminary assessments are carried out, from which the overall hazard potential becomes apparent. The knowledge of the radiological situation in the facility is an essential prerequisite for carrying out meaningful safety assessment. Data on the activity levels, the radionuclide composition of the contamination, the distribution of dose rates and other aspects are necessary inputs for all types of safety assessments for worker safety, for calculation of doses to the environment and the public, etc. In many cases, the variability of these conditions within a single facility is substantial.

The level of detail with which the radiological characterization is performed must therefore correspond to the hazard potential of the facility or the work to be carried out. In smaller facilities it may be enough to know the radiological conditions with a lower level of detail than in facilities with high dose rate areas or complex contamination situations where it cannot be seen *a priori* what radiological or safety implications the performance of a certain task may have.

5.2. EXAMPLES FOR A GRADED APPROACH IN THE PRELIMINARY ANALYSIS OF A FACILITY

The preliminary analysis of the facility with respect to safety assessment for decommissioning concentrates on the following:

(a) What is the activity inventory and the type of radionuclides in the facility? This determines the kind of scenarios to be used and gives an indication of the maximum exposure conditions. This is an important prerequisite for changing the right complexity of the safety assessment. For example:

- General activity levels: activity concentrations (Bq/g) – e.g. less than the values specified in RS-G-1.7 [8]; up to 10 times these values; up to 100 times these values; higher values;
- Surface contamination (Bq/cm^2) – e.g. less than 100; from 100 to 1 000; from 1 000 to 10 000; and more than 10 000;
- Maximum doser rate (mSv/h): e.g. 0.01 or less, up to 0.1; up to 1, up to 10, more than 10;
- Radionuclides present – e.g. naturally occurring radionuclides, fission products, nuclear material, fissile material; and
- Form of radionuclide – e.g. fixed contamination, sealed sources, loose contamination, spent fuel (damaged or not); short- or long-lived radionuclides.

Example 1: An NPP or a research reactor where no fuel failures have occurred, the contamination is comprised mainly of beta/gamma emitting radionuclides like Co-60 and Cs-137. In this case, the safety assessment needs to take into account primarily exposure by external irradiation, not neglecting of course other exposure pathways, like the possibility that during opening of systems as well as during decontamination and size-reduction processes air-borne particles or droplets can be formed, and internal exposure can also be relevant. In a fuel cycle facility where alpha contamination has built up during the operating period, the safety assessment needs to concentrate on exposure from inhalation and ingestion.

Example 2: In an NPP or a large research reactor with a great variety of rooms all with different contamination levels and in some rooms activation present, the safety assessment has to take into account the varying conditions. As a consequence, the analysis e.g. of the dose uptake during decommissioning, has to be carried out in an appropriate level of detail. In a small facility where only a few rooms exist and the contamination situation (and thus the exposure conditions) are more or less uniformly low or can be approximated by enveloping assumptions, the effort for characterization and the level of detail in which the safety assessment is carried out can be much smaller.

(b) What types of scenarios have to be considered during normal operation and during incidents/accidents, and which types of scenarios can safely be excluded as a consequence of the type of the facility and its status?

Example 3: While in an operating NPP criticality has to be taken into account in safety assessments, this is no longer the case in a shutdown NPP where the fuel has been removed. In an irradiation facility handling of the source during decommissioning and the risk of damaging the shielding has to be taken into account.

Example 4: This is an example of change of status arises with the decommissioning of the UK's gas cooled Magnox reactors. When these reactors were operating, some key safety limits and conditions were set for the maximum temperature of the primary circuit coolant gas, and its ability to transfer heat from the reactor core to the heat exchangers. Sometimes it happens that after shutdown of the reactor, it is not possible to defuel it promptly. In this circumstance, the decay heat from the fuel elements gradually diminishes. There has been one case where the fuel lost so much decay heat that it was necessary to inject heat into the primary circuit coolant gas so as to prevent condensation. The key safety limits and conditions now are related to excluding moisture from the primary circuit and not letting the circuit temperature fall below a minimum that would allow condensation to form, as this would promote corrosion in the fuel cladding material.

Example 5: Fuel cycle facilities, like Pu-laboratories or fuel processing facilities, were designed in a safe way, so that criticality or other critical situations during operation were avoided. During decommissioning, however, consideration must be given to situations that may arise from removal of shielding, accumulation of residual material etc., not having been of any concern during operation.

(c) How is the activity inventory distributed, and how is it bound to its place (e.g. as part of activation inside metal or concrete structures, as loose surface contamination, or as fixed surface contamination that cannot be easily removed)? This determines how and to which percentage the activity inventory may become available for mobilization during incidents and accidents as well as during handling or dismantling of equipment, etc.

Example 6: The largest part of the activity inventory in an NPP after removal of the fuel elements is in the activation within the metallic parts of the reactor pressure vessel and its interiors as well as in the inner parts of concrete of the biological shield. However, this activity is not or only to a small part available for mobilization in cases of fire, explosions or other initiating events. Similar considerations apply to the high contamination levels on interior surfaces of pipes and components that would be available for mobilization also only to a minor extent.

(d) What are the possible initiating events for incidents or accidents?

Example 7: A first screening of external initiating events is already possible by the nature of the site and its surroundings; e.g. the possibility of flooding, of an earthquake or a major explosion from industrial installations or from transports of explosives in the vicinity of the nuclear site. The safety assessment may then concentrate on hazard analysis starting with the remaining initiating events.

(e) What are the possible exposure pathways for members of the public from discharges? The way in which land around the nuclear site is used gives an indication of those exposure pathways over which uptake of radionuclides is possible and of those situations where people may be exposed to external irradiation.

Example 8: If the decommissioning activities do not give rise to liquid or to airborne discharges, the corresponding exposure pathways need not be taken into account. This might be the case with very small facilities or even with NPPs in safe enclosure where usually no liquid discharges occur.

Other aspects which may help decide on the complexity of more detailed safety assessment and for which data may be gathered during the preliminary assessment of the facility or that may be known from the history of the facility and previous assessments are the following:

— Exposure pathways to the environment;
— Chemical contamination;
— Structures physically unsafe;
— Closest public exposure;
— Site security;
— Distance to groundwater;
— Distance to and contamination of surface water; and
— Surrounding land use.

The effort required to collect the necessary information and data on the facility also depends on the availability of records and the extent to which the facility has previously been characterized. Unknown facilities for which no data are available may initially require higher effort for characterization in order to obtain a sufficient database for preliminary hazard assessments. If may then turn out that the required level of detail may be decreased again if the hazard potential is sufficiently low.

5.3. COUNTRY SPECIFIC EXAMPLES FOR A GRADED APPROACH IN RADIOLOGICAL CHARACTERIZATION OF FACILITIES

Country specific examples for radiological characterizations of facilities using radioactive material are provided in Appendix II. These examples illustrate how grading was applied to various types of nuclear facilities, with the aim of reducing the overall effort for the characterization by making use e.g. of correlation factors between key radionuclides and hard-to-measure radionuclides and by adjusting the number of samples to the hazard potential of the facility.

5.3.1. Assessment of the contamination of the reactor and auxiliary systems at the Caorso NPP (Italy)

This example presents the approach taken during a campaign for a preliminary estimation of the radiological contamination deposited on the equipment of reactor drywell, auxiliary and turbine buildings in the Caorso NPP. This procedure included direct measurement of dose rate in contact with the components, the application of suitable conversion factors between dose rate and surface β/γ contamination, properly derived by experimental procedures; and the application of scaling factors, in order to estimate the total (α, β, β-γ, X) activity of deposited contamination.

In this case, a graded approach was exerted in the complexity of measurements in the relation to the requirements of the radiological survey. As the primary aim was to provide an overview of the radiological situation with respect to contamination, simple and swift dose rate measurements were used, while the radionuclide composition was determined from separate and much fewer measurements.

5.3.2. Computational model for the assessment of contamination on the primary circuit of the Trino NPP (Italy)

As the pre-decommissioning radiological characterization of piping and equipment in shut down nuclear reactors represents a major effort in the general framework of the activities preceding removal and dismantling operations, the use of computer codes for the assessment of contamination on components and piping can be used to simplify this task. A suitable computer code taking into account characteristics of a particular NPP, as well as other parameters such as time elapsed between operation and the data for which the characterization is required has been used for the characterization of the primary circuit of the Trino NPP. The application of this computer code has been validated and calibrated by a limited number of sample analyses.

In this case, the graded approach consisted in the fact that the accuracy with which the computer code can predict the activity present in the primary circuit of an NPP was regarded to be sufficient for the purposes of the initial characterization. Thus, it has been possible to safe significant effort for sampling and measurement while still obtaining meaningful characterization results.

5.3.3. Graded approach concerning the number of samples from the research reactor in Sofia (Bulgaria)

The sampling programme described in this example for reconstruction of the research reactor IRT at Sofia was adjusted to the aim of the work for which a safety assessment had to be carried out. As the endpoint of the intended work was to bring the facility into a status suitable for reconstruction and not for release for conventional purposes, the requirements concerning performance of samplings, measurements and analyses were appropriately reduced, and statistical approaches for the evaluation of measurement results were applied.

In this case, the graded approach applied to the characteristics of the sampling and measurement programme, which were chosen in agreement with the purposes for which this programme was carried out.

5.3.4. A semi-empirical model for assessment of hard-to-detect radionuclide levels and their significance in decommissioning waste from accidentally shut down NPP Bohunice A1 (Slovak Republic)

As the NPP Bohunice A1 was shut down after an accident, the characterization of the radiological situation in this facility requires special effort. This example describes the approaches taken to identify and characterize the hard-to-detect radionuclides (HD-RN, α and pure β-emitters) for which direct measurements would require substantial effort. Therefore, a combined theoretical-empirical approach was developed, utilizing a calculated radionuclide inventory in spent fuel and a developed model with effective empirical release coefficients (ERC) relative to relative Cs-137, describing the released fraction of HD-RN from the spent fuel. This model evaluates the radiological importance (relative dose contribution according to concentration limits of Republic Waste Repository Mochovce) of 18 prescribed HD-RN in the NPP A1 operational radioactive waste system and takes into account all historically available data from NPP A1 acquired in 1992-93. This semi-empirical method enabled an effective evaluation of radiological significance of HD-RN as well as their content declaration in waste streams. The graded approach in this case can be seen in the reduction of effort required for sampling and measurement of hard-to-detect radionuclides by the use of a dedicated model, which is commensurate with the accuracy required for these purposes.

6. GRADED APPROACH IN THE HAZARD CATEGORIZATION, THE PRELIMINARY HAZARD ASSESSMENT OF THE FACILITY AND ITS SYSTEMS, STRUCTURES AND COMPONENTS (STEP 3)

6.1. OVERVIEW

As stated in Section 5, the radiological characterization of a facility planned to be decommissioned is an essential prerequisite for the safety assessment. With respect to the categorization of parts of the facility, it is a good approach to establish the boundaries or zones in the facility or between systems with similar characteristics. The approach which is referred to as “zoning” in some countries is to establish a boundary between “(potentially) contaminated” and “uncontaminated” parts of the facility. Other more specific zones may be established within the contaminated area with respect to the radionuclide vector characteristics, e.g. to identify those systems and rooms where the contamination and/or dose rate is very high or where significant alpha contamination is present or may be present. A further classification of the facility may be done on the basis of dose rate values and/or the presence of loose contamination. This information is generally closely linked to the preliminary hazard analysis during which the hazard potential in the zones of the facility is roughly estimated and the zones or work packages for decommissioning may be assigned to certain hazard categories.

6.2. THE USE OF A GRADED APPROACH

A graded approach in establishing zones with respect to a safety assessment for decommissioning aims at directing effort for the safety assessment to the type of hazards that are really present in the respective zone thus and optimizing the work of operators and Regulatory Bodies. A typical categorization of a facility that is planned to be or is undergoing decommissioning might consist of the following zones or areas (see discussion of additional factors in Section 4.2):

(a) Areas/zones that are definitely not contaminated;

(b) Areas/zones that may be contaminated;

(c) Areas/zones with (loose or fixed) contamination (e.g. with significant content of alpha emitters or other radionuclides with high inhalation/ingestion dose coefficients);

(d) Areas/zones with high dose rates where restrictions for access and for work apply; and

(f) Areas/zones with high dose rates where direct access is not possible.

While the position and extent of the latter areas with high dose rates is usually known from the operating history, the extent of the spread of contamination and especially its radionuclide composition may not be known with the desired accuracy. In addition, knowledge of the contamination in systems and components is essential when planning and assessing dismantling work where inner surfaces are opened and contamination could be spread. Therefore, it can significantly simplify the safety assessment for workers exposures when the type of contamination in a given system and in a particular room is known. This bears the great advantage that the safety assessment can be tailored specifically to the conditions prevailing in each area or zone, allowing to adapt the effort for performing evaluation of hazards and their consequences in each area or zone accordingly.

Further consideration needs to be given to applying grading over time. This means that the reduction of the activity inventory, of the remaining decommissioning work etc. within a certain area of a facility during the progress of decommissioning work will automatically lead to changes in the assessment of the situation. This can lead to re-assessment of the situation to reflect new conditions at the facility and possibly to reduce the required effort for subsequent safety assessment.

6.3. METHODS FOR A GRADED APPROACH

It is possible to combine various methods to establish reasonable boundaries for the areas or zones described above. Those methods could be grouped in the following way:

- Use of detailed facility history (i.e. the history of every system and component during operation, outage and abnormal conditions, including leakages, spills relevant to safety, etc.) in combination with the analysis of the systems (purpose and operation of each component, connection between systems in normal operation conditions, during maintenance, etc.) require little effort in comparison to a detailed sampling programme. It greatly simplifies the establishment of boundaries of the facility, e.g. for those areas of the facility where alpha emitters may be present.
- Taking samples from the facility for confirmation of the conclusions drawn from the evaluation of the facility history and system analysis may be regarded as a second step. In this case, samples at selected places of the facility (e.g. before and after a presumed boundary of an area) would be taken to see if the relevant features (e.g. extent of alpha contamination) are present to this area of the facility as assumed.
- If the assumptions on the radionuclide composition in the contamination are reasonably accurate, effort can be saved by performing simpler measurements, e.g. dose rate measurements instead of gamma spectrometry of material samples or smear tests, or gamma spectrometric measurements by which hard to measure radionuclides (like Sr-90, alpha emitters, etc.) are derived from key radionuclides (Co-60, Cs-137, Am-241) via scaling factors instead of performing laboratory analyses with full radiochemical separation and beta and/or alpha spectrometry.
- A much more complex approach would be to perform a complete sampling and measurement programme without making assumptions on the type and level of contamination or activation that is to be expected.

The methods being applied may thus vary according to the required level of characterization of the facility, ranging from analysis of the facility history to complex sampling and measurement programmes. Grading needs to be applied based on the risks associated with the decommissioning work and hazards, as well as the situation that is to be analyzed.

6.4. COUNTRY SPECIFIC EXAMPLES FOR THE GRADED APPROACH IN INITIAL CATEGORIZATION OF FACILITIES

Country specific examples of the use of a graded approach in categorization schemes for nuclear facilities are provided below and also in Appendix III.

6.4.1 Categorization and zoning of facilities using radioactive material (Germany)

An example for the categorization and zoning of a facility is the TRIGA-type reactor HD-1 at the German Cancer Research Center (DKFZ) in Germany. There is no distinctive categorization scheme as described for the USA and the UK below. The effort required for certain steps within the safety assessment, like the determination of the characteristics of the facility through radiological characterization, were proposed by the operator and are fixed in the licensing procedure or in the surveillance phase by the Regulatory Body. This took into account the recommendations of independent experts acting on behalf of the authority. In this example, the graded approach taken with respect to the categorization and zoning consisted of the use of plausibility arguments for restricting the characterization programme only to certain parts of the facility. This approach was to a large extent built on the trust that the Regulatory Body had developed during the good progress of the work.

6.4.2 Categorization scheme for the activities at a facility under decommissioning (UK)

An example of a graded approach is the establishment of a categorization scheme for decommissioning activities that has been applied at a facility in the UK. Decommissioning tasks can be categorized with respect of the safety significance of the risks that may pose to workers, the public and the environment. Radiological characterization can be based on the potential nuclear or radiological consequences of the decommissioning task with respect to workers, the public and the environment. A safety assessment would be performed which will provide a basis on which to make this decision. This safety assessment may be complex and supported by hazard identification techniques such as HAZOP[1] and HAZAN[2]; or it may be as simple as the preparation of a dose budget for the task or just as completing a checklist. The categorization of the decommissioning tasks is then based on the most significant potential consequences. The following safety categorization can be followed, i.e. decommissioning activities with:

(a) A major impact on the public;

(b) Minor or significant radiological safety significance;

(c) Minor radiological safety significance; and

(d) No radiological safety implication.

6.4.3 Initial hazard categorization (USA)

The United States Department of Energy (US DOE) has introduced Threshold Quantities (TQs) in the DOE Standard 1027 [9] against which facility radioactive material inventory needs to be compared for *initial* hazard categorization. Initial hazard categorization is a simple screening step that does not involve detailed calculations (not including consideration of material form, location, dispersibility and interaction with available energy sources called for in final hazard categorization). The purpose of the *final* hazard categorization is to ensure that the facility and the accident specific factors that could:

— Either change the fraction of material released in an accident, or

— Change the amount of the total inventory of material subject to an accident are addressed to ensure the facility is properly categorized.

[1] HAZOP stands for Hazard and Operability Analysis Technique

[2] HAZAN stands for Hazard Analysis Technique

This example illustrates the categorization scheme by which a facility is assigned to Category 1, 2, 3 or even below Category 3. This decision tree is mainly based on the radionuclide specific activity contained in the facility, as well as on its type.

This procedure has been illustrated by the hazard assessment for the ETR Facility Hazard Categorization, Idaho, where the actual radioactive inventory of the main parts of this engineering test reactor facility have been evaluated and compared to the TQs referred to above.

7. GRADED APPROACH IN PERFORMING SAFETY ASSESSMENT FOR DECOMMISSIONING (STEP 4)

The aim of a safety assessment for the workers, the public and for the environment from normal and accident decommissioning conditions is to demonstrate that the potential exposure via any relevant pathway (e.g. either direct external irradiation from the facility or inhalation, ingestion or external irradiation from radionuclides that have been released from the facility) is below the relevant safety criteria. Such a safety assessment consists of a number of steps:

(a) Determination of the source term;

(b) Dispersion of radionuclides over environmental pathways; and

(c) Dose assessment for uptake of radionuclides via environmental pathways (food, water, etc.).

Each of these steps may require significant effort, e.g. require data collection, modelling and calculation. Therefore, a graded approach can be used with benefit:

— By using enveloping approaches and methods by which compliance with limits is demonstrated using simple approaches which overestimate the real situation; and

— By limiting the effort in the assessment to those parts which are most relevant and on which the result will depend to the largest extent.

7.1. ANALYSIS OF SOURCE TERMS

The first step in a graded approach is to determine the source term only on the basis of the enveloping scenario. More refined approaches could take account of various scenarios and thus of various source terms.

The determination of the source term for releases into the facility and to the environment as a result of decommissioning is closely related to the initiating events and the scenario (Section 7.3.2). The source term can be calculated using an enveloping scenario as proposed above and applying this to the contamination situation in the facility. It must, however, be noted that incident or accident situations could affect different components or parts of the facility than identified for normal decommissioning activities so that the source term would differ accordingly.

7.1.1. Determination of dose rates

Apart from release of radionuclides from the facility, external irradiation could also lead to exposure of the workers and the public (in the vicinity of facilities). Here, the assumptions pertaining to the operational phase of the facility are usually a good estimate also for the decommissioning phase, until the radioactive inventory has significantly decreased. A first step in a graded approach could therefore be to use the same values for the facility contamination as during operation, unless specific information for the shutdown stage is available.

The safety assessment will need also to consider that a local increase in dose rates may occur during decommissioning if highly radioactive parts of the facility are dismantled, handled conditioned as radioactive waste for further storage (e.g. in an on-site waste storage building). The gradual increase of radioactive waste inventory in this waste storage building may lead to an increase of dose rate to public at the perimeter of the site, making a re-assessment of dose rates to public necessary.

7.1.2. Determination of source terms for gaseous and liquid releases

The source terms for airborne gaseous and liquid releases from a facility under decommissioning will be different from the operational phase. The details depend significantly on the type of facility and the operational history. For example, in NPPs, where no nuclear fission takes place any more, a large part of the radionuclide spectrum like short-lived fission products or some noble gases will not occur any longer or in significantly reduced quantities. On the other hand, certain decommissioning activities will lead to an increase in releases when dismantling or decontamination techniques give rise to additional gaseous or liquid discharges, when opening of systems make inner surfaces accessible, etc. The source term in the decommissioning phase therefore differs in many respects from the operational phase. This also applies to fuel cycle facilities or other types of facilities.

However, the source term that has been applied for the assessment during operation is a bounding case also for the decommissioning phase. For the case of NPPs, a first step in a graded approach to the determination of the source term for decommissioning (after defueling) might therefore be to use the same source term as for operation without those radionuclides that can no longer be present due to physical reasons, like noble gases and certain very short-lived radionuclides. Experience has shown that a source term, which is determined in such a way, is usually far too large, especially for NPPs, as the source term for the operational phase has been designed to also cover a certain range of operating conditions. Therefore, a second step could be to reduce the radionuclide concentrations by a certain factor (e.g. to 20 % of the values used during operation).

A more realistic approach would be to derive completely updated source terms for airborne and liquid releases of radionuclides which is based upon the analysis of the contamination situation, the filters and the actual dismantling and decontamination work which is to be carried out. This, however, requires the largest effort and also requires a good radiological characterization of the facility, resulting in the facility inventory as discussed in Section 7.1.1.

The detailed and more realistic source terms for safety assessment for decommissioning can be derived by evaluation of individual radionuclides, for individual decommissioning activities according the detailed decommissioning plan. The individual source terms are summarized to define the overall source term for the decommissioning project (application of bottom up principle). The source term is then radionuclide resolved and time resolved according the time distribution of tasks in the decommissioning plan.

7.2. GRADED APPROACH IN SAFETY ASSESSMENT OF WORKERS EXPOSURE DURING DECOMMISSIONING

It is the aim of a safety assessment for worker exposure during decommissioning to assess the consequences from normal decommissioning, as well as from abnormal (accident) at the facility. Hazards to workers during decommissioning may arise from:

— External irradiation caused by contamination or activation in the structures (components, buildings, surfaces, etc.);
— Airborne releases during the application of cutting decommissioning techniques (mainly thermal, but also mechanical techniques);
— Aerosols originating from chemical decontamination (e.g. baths) or the application of mechanical decontamination techniques; and
— Other sources, e.g. background radiation, radioactive waste storage, cleanup operations after application of dismantling techniques, etc.

An important tool for hazard identification at workplaces is a map of dose rates and a map of contamination conditions established at least for the accessible parts of the facility. Such maps may differ from the operational phase as for example there may be spots where contamination accumulates which have not been there in the operation phase and which constitute new sources of contamination and exposure, such as ventilation filters.

A first step in a graded approach is to concentrate on characterization measurements (e.g. dose rate and contamination) to those areas of the facility, where the highest levels may be expected. If dose rates are low and the contamination does not include radionuclides with high inhalation or ingestion dose coefficients, the values obtained may be regarded as enveloping for the entire facility.

For dose assessments concerning the level of detail in the planning of decommissioning activities, the data on the facility inventory, dose rate, the level of contamination and activation, including the radionuclide vectors, need to be developed for systems and structures of the facility located in the controlled area (see main report).

In facilities with high dose rates or more complex contamination, it may be necessary to perform more detailed measurements and to create more refined maps in two dimensional or three dimensional formats including the models of rooms and equipment to be decommissioned (see Volume II of this report).

7.2.1. Dose assessments for decommissioning activities as planned

An overview of dose rates as described in Section 7.1.1 allows a swift and reliable assessment of potential doses to workers from external irradiation. The expected individual doses for a particular work package can be estimated from the required time that the workers will spend at particular locations and the dose rates at these locations. Collective doses can then be derived by appropriate summation over the workers involved in that work package. Doses from internal exposures are usually kept as low as reasonably achievable by prescribing the use of respiratory protection in any circumstances where the risk of inhaling radionuclides with high inhalation dose coefficients exists.

Because of the simplicity of this type of dose assessment, the use of a graded approach is possible only to a lesser extent than for dose assessments for the public as described in Section 7.3. Grading may be applied to the level of detail into which the entire decommissioning or certain work packages are broken down. A very cursory approach may be to simply multiply the estimated man-hours per worker with the maximum dose rate and compare this value against prescribed limits. If the result is far below the limit values, this may suffice as a demonstration that dose limits will be complied with. This approach is, however, usually not sufficient for a detailed resource scheduling. Detailed examples for these approaches are provided in the safety assessments within the DeSa Test Cases in Annex I of this report.

On the level of the detailed planning of decommissioning activities, the schedule of decommissioning activities need to be analyzed in order to identify to the individual time components (i.e. productive and non-productive) and relevant dose rate in facility premises, as defined in the facility inventory database, need to be allocated to these time components. Examples of these values are the dose rates at the working distance from the equipment to be dismantled, averaged dose rates in individual rooms, or dose rate at working places for stabile waste management equipment, as defined in safety documentation for the equipment. When taking into account the correction coefficient for individual professions of the working group, which express the efficient stay of individual professions in the relevant dose rate component, the dose estimates could be evaluated on the level of individual professions, using the calculated duration of the activities. The collective doses can then be again derived by summation over the professions of the working group and over individual tasks of the project.

The estimation during the planning phase of the activity volume concentration in those rooms, where the decontamination and dismantling activities are to be performed, can be accomplished by application of releasing factors which are specific for individual decommissioning techniques, e.g. cutting. The portion of the contamination as defined in the facility inventory database, can be then calculated and based on parameters of the rooms and ventilation system, the volume concentration can be estimated. The calculated volume concentration can be used for identification of appropriate personal protections means needed for performing the decommissioning activities under given radiological conditions.

For safety assessment of the decommissioning activities, which will be performed in radiologically complicated situations, appropriate software tools can be used that evaluate the doses for the workers based on two or three dimensional modelling of the movement of the workers within the radiation fields during execution of the tasks. This modelling can be combined with training of personnel using the mock-ups in order to reduce the dose to minimum value during execution phase of the task.

7.2.2. Dose assessments for accident conditions

In addition to the hazards which may occur from the envisaged routine decommissioning activities (see Section 7.2.1), a number of hazards may arise only in abnormal decommissioning conditions and these need to be identified. Those may include leakages or spillages of process fluids still being in the systems or of other liquids, lack of shielding, inadvertent entry to places with high dose rates, failure of protective measures, especially those against inhalation, like rupture of tents, failure of ventilation, etc. and many others.

At the beginning of the application of the graded approach to dose assessments for abnormal working conditions, it is necessary to analyse the most adverse scenario leading to a significant increase in dose rates or to a significant release of radionuclides into the working environment in the facility. This part of the analysis is often performed without taking into account any mitigation like existing shielding, filters etc. ("unmitigated consequences") to get an idea of the most adverse consequences. The analysis may then be repeated with appropriate consideration of mitigation measures.

An example of a scenario is a fire with subsequent mobilization of surface contamination or a drop of a container or a waste package that ruptures and releases part of its contents, rupture of components for retrieval, transporting or storing the radioactive liquids, drops of contaminated dismantled segments, etc. A further simplification is to assume that no protective measures are taken and that the personnel is exposed for a long time. If doses calculated from such a conservative scenario still are sufficiently below prescribed limits, e.g. volume concentration of individual radionuclides, limits for dose uptake, this approach may be adequate for demonstrating compliance in incident or accident situations. If these simple calculations would result in limit values to be reached or exceeded, a more refined assessment would be needed.

Graded approach in such cases may include the designing the technical and organizational means for mitigation of consequences of the analyzed event, so that the situation is manageable by the operating personnel. Example of such technical means could be the additional auxiliary systems for collecting and safe management of the spills in critical locations, avoiding the presence of personnel in critical locations in critical working phases, precautionary installation of additional local ventilation, etc. The proposed technical and organizational means can be evaluated, if necessary, using the software codes as discussed in Section 7.2.1.

7.3. GRADED APPROACH IN SAFETY ASSESSMENT FOR PUBLIC EXPOSURE DURING DECOMMISSIONING

7.3.1. Dose assessments for decommissioning activities as planned

Dispersion of radionuclides over environmental pathways

The dispersion of radionuclides in the environment via air, water and food chains can be modelled in different levels of detail depending on the purpose of the safety assessment. Examples can be found in the Reference [10] where several approaches of differing complexity are provided for each step of the assessment. That report therefore provides a good example of a graded approach in environmental modelling.

A graded approach needs to be oriented to the overall hazard potential (source term) that is to be analyzed. If this potential is low, then an enveloping approach and a simple model could be used, e.g. the assumptions of simple and stable mixing conditions in water or air or for transport via groundwater, followed by simple methods for dose assessments (see Section 7.2.). If the resulting doses to members of the public are well below the defined safety limits and criteria, this approach may suffice as it shows that compliance with these criteria can even be demonstrated with simple, enveloping assumptions. If, however, the resulting doses are close to or even above the prescribed safety limits and criteria, more detailed models have to be used. For example, dispersion in air or water can then be taken into account, site-specific parameters could be analyzed, certain pathways could be excluded or modified, etc.

Dose assessment via pathways to members of critical group

Following the analysis of the dispersion of radionuclides in the environment, a number of pathways lead to exposure of public, like food chains and water consumption. Models with different complexity are also presented in [10]. They are comprised of intake of radionuclides with vegetables and fruit, with milk and meat, with fish and other aquatic biota and with drinking water, both from surface water and from wells fed from groundwater. The way in which these food chains are modelled may differ considerably.

In applying a graded approach it will be useful to start the dose assessment by using default parameters and the whole set of scenarios and pathways that may be prescribed for a particular situation. If the resulting dose to the member of the public is below the prescribed dose constraint, e.g. 0.3 mSv/a, this approach may be sufficient. Otherwise, a more sophisticated approach is needed, taking into account site- and the facility-specific parameters, a more refined analysis of the critical groups and other factors.

External exposure to public originates from the decommissioning activities at the facility itself (as already addressed above), as well as from the radionuclides deposited on the ground or on sediments of the rivers near the facility. The deposition of radionuclides on the ground or on sediments follows from environmental pathways discussed above.

Similar considerations for the use of a graded approach apply in this case as in the previous sections.

7.3.2. Dose assessments for accident conditions

Similar to the examples for assessments for as planned decommissioning activities presented in Section 7.2, examples for graded approaches can also be found for incident and accident conditions during decommissioning. The reason is that there are many potential initiating events and scenarios that selection of the most relevant for decommissioning is necessary for the purposes of the safety assessment. For small facilities and hazards, the analysis may be very simple focussing on just one bounding scenario, while for large and complex facilities and decommissioning work with a high hazard potential, a larger number and a more in-depth analysis may need to be performed. However, depending on the resulting dose/risk and the legal and regulatory basis in the country, even evaluation of incidents and accidents at larger facilities may be treated with an enveloping approach.

Analysis of external and internal initiating events and hazards

Identification of initiating events by which activity releases from the facility may subsequently be caused is the prerequisite for performing evaluation of incidents and accident conditions. At least for large facilities undergoing decommissioning, the spectrum of such events will differ considerably from the operational phase, as for example, the spent fuel and most of the high-level waste from operation would have been removed from the facility. External hazards may include natural hazards like adverse meteorological conditions (e.g. wind, snow, rain, ice, temperature, flood, and lightning) or earthquakes, which may usually be similar to those considered for safety assessment during operation. Man-made hazards may include airplane crashes, blasts, fires, gas clouds, as well as intrusion (mainly in cases where the facility is in a safe enclosure status), and are generally specific to decommissioning.

It is the aim of a graded approach to identify only those external and internal initiating events and hazards that could cause the significant harm and thus could lead to the largest activity release and exposure to the public. The aim of this approach is to demonstrate that even with this most adverse possible scenario the dose limits relevant for incidents and accidents are complied with.

An example of this approach can be found in [11] where scenarios have been analyzed for NPPs. It has been found that a large fire in an NPP with light-water reactor would lead to the incident with the largest radiological consequences. Such a fire could start from combustible material that has been introduced in the NPP in the course of decommissioning work and could affect contamination on outer surfaces as well as certain amount of inner surfaces in systems, which are assumed to have been opened during dismantling. Starting from this initiating event, an enveloping scenario has been created. More details on such an enveloping scenario are given in Appendix IV.

A graded approach in the analysis of initiating events and hazards could therefore lead to using a scenario that has been justified and could lead to the most adverse conditions and is still realistic for the particular facility.

Analysis of scenarios leading to releases of radionuclides

Scenarios leading to the release of radionuclides following initiating events as identified in Section 7.2 may include failure of filters in case of fires so that radioactivity is released over the stack without filtering or even damage to the building. The level of detail of scenarios leading to releases of radionuclides need to correspond to the probability of the initiating events as well as to the overall hazard potential of the facility. In many cases, the analysis of a single enveloping scenario may be fully sufficient, as has been demonstrated in [11]. Thus, the number of scenarios and the level of detail of the whole analysis can be considerably reduced, particularly for cases where the overall hazard potential also in incident and accident situations is comparatively low. Nevertheless, in a complex facilities and decommissioning activities, more than one accidental scenario (e.g. flooding, fire) will need to be assessed.

At the beginning the only the enveloping case will need to be analysed. For the case of an NPP undergoing decommissioning and assuming a fire as the initiating event, such an enveloping scenario might be based on the assumptions that:

— The whole contamination on outer surfaces in that area, as well as a certain percentage of the activity on inner surfaces, is mobilized by the heat and draft; and

— A part of this activity will condense again on colder surfaces and that the filters have been congested by the aerosols generated by the fire so that the activity will leave the NPP unfiltered.

In many cases a scenario of this kind which is relatively easy to analyze is fully enveloping for other scenarios which are possible during decommissioning. More refined approaches for accident conditions during decommissioning could take account of other scenarios.

Dispersion of Radionuclides over Environmental Pathways and Dose Calculations

The dispersion of radionuclides in the environment via air, water and food chains can be modelled in varying levels of detail. Examples can be found in Reference [10] where several approaches of differing complexity are provided for each step of the assessment (see Section 7.2.).

A graded approach could therefore be oriented to the overall hazard potential which is to be analyzed. If this potential is low, then it is recommended to use an enveloping, simple approach, e.g. a simple mixing in water or air. If the resulting doses are well below defined safety limits, this approach may suffice as it shows that compliance with limits can even be demonstrated with simple, enveloping assumptions. If, however, the resulting doses/risks are close to the prescribed safety limits or even exceed them, more refined techniques have to be used. For example, instead of using simple mixing assumptions for airborne releases, the dispersion can then be taken into account, or radioecological pathways (transfer or radionuclides to vegetation, to cattle, etc.) can be modelled using more sophisticated assumptions.

7.4. COUNTRY SPECIFIC EXAMPLES FOR GRADED APPROACH IN PERFORMING SAFETY ASSESSMENT

Country specific examples for the graded approach in performing safety assessment are provided in Appendix IV. These examples from Brazil, Cuba, the Czech Republic, Germany, the Netherlands and the USA cover various aspects of actual safety assessment, highlighting the idea of the graded approach. In these examples, grading consists of or is influenced by:

— Unique considerations required in decommissioning that result from the activities proposed for the specific facility, not specifically addressed (or envisioned) by its design and operational evaluations (e.g. tritium as a combustible gas and hydrogen generation during vessel grouting);
— Considerations on industrial safety, for which an assessment scheme is provided;
— The use of only one enveloping scenario for safety assessment of NPPs instead of several separate scenarios;
— The environmental and radiological pathways that need to be considered in the assessment, thus requiring the use of dedicated computer models; and
— The intended re-use of existing buildings of facilities.

These examples also address aspects such as time constraints for licensing procedures that are graded according to the type of facility and the scope of the license application, and the effect of the graded approach in safety assessment illustrated by the extent of the documentation provided for different types of facilities using radioactive material, all developed according to the same standards.

8. GRADED APPROACH IN THE IMPLEMENTATION OF SAFETY ASSESSMENT RESULTS (STEP 5)

8.1. OVERVIEW

The level of detail in which safety assessment is developed and implemented is mainly determined by the considerations given in Sections 4 to 7 above. However, there may also be other considerations that may have an impact on the way in which safety assessment is carried out, like the intended application of the safety assessment results (e.g. for decisions within the company to which the decommissioning project belongs) or other requirements and standards (e.g. the expectations of the authorities on the way in which safety assessment results are presented, the role of the safety assessment in the environmental impact assessment (EIA) and for interfacing with the public and other interested parties, etc.). Examples of these aspects influencing the grading are outlined in the following sections.

8.2. USE OF SAFETY ASSESSMENT RESULTS BY THE OPERATOR

The results of safety assessment can be used by the operator (or by other company/entity that is performing the decommissioning) to assess or decide upon certain strategies or approaches to take for the economical and safe execution of the decommissioning project. Such decisions may include for example:

- Whether upgrading certain safety related systems or the facility of (mobile or permanent) replacements, mainly for the ventilation system, the filter capacity, the fire detection system, the systems for permanent dose rate measurement, the surveillance of airborne radioactivity, etc. may significantly increase overall safety or may mitigate possible consequences from incident scenarios, or whether even downgrading of these systems may be possible;
- Whether specific training, e.g. to cope with incident situations that might arise during decommissioning work, will be required;
- Whether investments with respect to the improvement of the structural integrity of the facility are necessary or would mitigate effects of incidents or accidents, such as fires or explosions.

If the safety assessment results are intended to also provide answers for these safety related questions and to support these decisions, then the level of detail, the scenarios and parameters, as well as the presentation of the results, need to be chosen in such a way as to be able to draw the necessary conclusions. For example, in the case where the safety assessment is intended to provide input to decisions concerning upgrading of certain safety related systems, the analysis of appropriate accident scenarios as described in Section 7.3 needs to cover the appropriate situations, and it might not be enough to use just one enveloping incident/accident scenario. Likewise, a decision on the necessity to upgrade the ventilation system or the capacity or efficiency of the filters to cope with aerosol generation from dismantling or decontamination operations on the basis of the results of a safety assessment would only be possible if the appropriate pathways are analyzed.

8.3. DECOMMISSIONING AT COMPLEX SITES

The complexity of a decommissioning project or the interaction of the facility (under decommissioning or being prepared for decommissioning) with the other facilities on the site, can impose modifications or limitations on the safety assessment for decommissioning. In some case also new aspects (e.g. hazards, initiating events) need to be evaluated in comparison with isolated decommissioning cases and supporting facilities. The following considerations apply mainly to cases with a higher degree of complexity, like decommissioning of facilities after an incident or accident during the operational phase. An illustrative example on how the extent of the safety assessment is influenced by the type and complexity of facility is provided in Appendix IV, Section 4.6.

8.3.1. Complex decommissioning projects

The decommissioning projects for facilities may be complex, like for nuclear facilities shutdown after an accident during the operation. In these cases, the decommissioning plan can be developed for the whole decommissioning process or for a defined decommissioning phase (see main report), but due to the complexity of the decommissioning activities, some selected tasks of projects can be assigned for individual licensing. The graded approach in this case is represented by the safety assessment for the whole project under the condition that detailed safety assessments will be performed for these selected subtasks and the work in defined phases can continue only when the identified subtasks are licensed.

The complexity and risk imposed in some special decommissioning task may results in two-stage safety assessment for authorization or approval of equipment or a decommissioning procedure. In the first step the risks of the design of equipment or of a proposed procedure are evaluated and solutions for mitigation of the consequences of the evaluated risks are proposed.

As a following step, technical or organizational means are implemented into the design, like additional shielding, additional testing, modified procedure, training of personnel, etc., and sometimes also the additional requirements of the Regulatory Body. The upgraded design and when needed also the results of testing, are the subject of the final and updated safety assessment developed for authorization or approval of special equipment or decommissioning procedure.

8.3.2. Phased decommissioning

In some cases, the decommissioning projects are developed and licensed as a set of subsequent individual phases. The reasons for such a procedure could be the complexity of the project, lack of technical solutions due to non-standard conditions at the decommissioned facility or insufficient data for developing the decommissioning plan and for safety assessment in required quality and extent. The phased decommissioning facilitates to preparing the plan and licence for the nearest decommissioning phase and at the same time to prepare the conditions for the plan for the next decommissioning phase.

If the strategy of phased decommissioning is accepted, the graded approach in safety assessment results in two levels of details in the documentation for licensing of the next decommissioning phase. For the decommissioning activities planned to be performed within the next licensed phase, the detail safety assessment be developed in order to document the safety of performing the decommissioning activities.

For the next decommissioning phases, the safety assessment on conceptual level could be sufficient. The objectives of this level are the preliminary evaluation of risks, based on actual knowledge and data of those parts of the facility that will be the subject of decommissioning in next phases. The risks on which the detailed safety assessments for the next decommissioning phases need to concentrate are highlighted and additional facility characterization, data and information needed for the safety assessment of next phases are identified.

8.3.3. Site interactions

The situation in some nuclear sites may cover the operation of a facility or several facilities of various types and at the same time the decommissioning activities of another facilities on site. In some cases there are also common waste management facilities for processing of the waste from operation and from decommissioning. An example for this situation is the nuclear site Jaslovske Bohunice in Slovakia, where there are two operating NPPs, one NPP under decommissioning, one waste management complex and wet spent fuel store. Similar situations can be found at the Kozloduy NPP in Bulgaria.

In such cases the impact of decommissioning activities on the overall exposure from the site is significantly minor in comparison with the impact of the operating facilities. Example of the application of the graded approach could be the evaluation of the site impact on the environment, where the evaluation of decommissioning activities may be reduced to evaluation of effluents from decommissioning and comparing them with the effluent limits authorized for the site where the major contribution is from the NPPs in operation.

A similar approach can be applied in decommissioning safety assessments and plans with impact on the site level, where the risks are dominated by other on-site activities not related to decommissioning. On the contrary, operating NPPs or other facilities on the site may impose new types of hazards or limitations regarding the decommissioning activities, which would not occur in the case of isolated decommissioning activities.

8.4. APPLICATION OF SAFETY ASSESSMENT RESULTS IN LICENSING OF DECOMMISSIONING

As mentioned in the previous sections, there is common practice that a licence for decommissioning is required in many countries. Therefore, sound and mutual communication between the licensee and the respective Regulatory Body is a prerequisite for issuing a licence for decommissioning and successful performance of the whole project.

In order to facilitate smooth performance of each individual decommissioning task it could be a good approach to develop a licensing programme with agreed schedules. The graded approach could be implemented in the application of this programme and time constrains for issuing a licence required. Based on the complexity of the decommissioning project or individual decommissioning activity (task), these time constraints for granting a licence can vary from a few weeks to one year at the latest. Establishment of these time constraints is vital and leads to the fulfilment of decommissioning liabilities for both parties, the Regulatory Body and for the licensee. An example can be found in Appendix IV, Section 4.7. Time constraints in licensing procedures for providing a decision to the applicant are applied in a few countries, e.g. the Canada, the Slovak Republic and the US.

Good co-operation with the Regulatory Body from the early stages of developing the safety assessment (e.g. in definition of scope, approaches and methods of safety assessment, consulting the interim results) can result in keeping the proper timing and extent of the safety assessment, especially in non-standard cases. If the Regulatory Body is presented with the results of the safety assessment for the first time at the end of the assessment development, delays in approval of the safety assessment may occur due to requirements for additional substantiation of results or methods. In some cases, the Regulatory Body may demand additional evaluation or an independent review, which may prolong the overall duration of the safety assessment review and approval.

8.5. CONSIDERATIONS CONCERNING INFORMATION TO THE PUBLIC AND INTERACTION WITH OTHER INTERESTED PARTIES

The level of the safety assessment may also depend on the way its results are intended to be used for information purposes to the public and for interaction with other interested parties. If the results of the safety assessment needs to be used to answer certain questions or to respond to certain concerns that may be raised by the public or other interested groups, the safety assessment must be appropriately formulated. For example, if the public is interested in the potential dose to members of the critical group during the decommissioning, it may not be appropriate to use a conservative, bounding approach for calculation of doses from liquid or airborne releases, although that would suffice from a purely technical point of view. Such an approach would lead to a substantial overestimation of the real doses that a person of the critical group might receive, and would therefore lead to questions from interested parties, who might compare those conservatively high dose assessments more realistic results obtained for other decommissioning projects.

The graded approach in safety assessment from this perspective can be specific to a municipality, political viewpoint or geography due to possible additional requirements imposed by the public on the safety assessment.

9. APPLICATION OF THE GRADED APPROACH TO THE THREE DESA TEST CASES

9.1. OVERVIEW

This section deals with the application of the graded approach to the three DeSa Test Cases – an NPP, a research reactor and a nuclear laboratory (see Annex I of this report). In particular, it discusses the approaches and the results of the application of the graded approach to particular parts of the assessments were carried out for these three facilities. The approaches are compared with respect to the complexity of methods used for carrying out the assessments for all three types of facilities and are analyzed with respect to the question whether a link between the complexity of the analysis and the complexity of the decommissioning work for which the assessment is carried out can be observed.

In order to be able to apply common basis for such a comparison, guiding principles for comparing safety assessments have been developed, which were applied for the actual comparison presented here.

Using these principles, the three existing safety assessments for the DeSa Test Cases have been reviewed with special emphasis on grading between the approaches taken for the NPP, the research reactor and the laboratory for the various topics. The results are presented in Section 9.2.

9.2. COMPARISON OF THE APPROACHES OF THE THREE DESA TEST CASES

In the following subsections, the five steps at which a graded approach can be applied are discussed (see Sections 4 to 8). A brief description of the approaches used in all three DeSa Test Cases for each step is followed by the comparison, indicating similarities and differences in the approaches that can be linked to specificities of the decommissioning activities and/or the types of facilities.

Full details of the three DeSa Test Cases, i.e. the NPP, the research reactor and the nuclear laboratory, can be found in Annex I of this report.

9.2.1 Step 1: Identification of conditions for the safety assessment imposed by the regulatory framework

The comparison focuses on questions like: Did the safety assessment take account of grading allowed/demanded in the regulatory framework? Did the safety assessment use the adequate approach in accordance with the type of facility/activity inventory/hazard potential as required in the regulatory framework? This includes aspects like the use of the right dose criteria, scenarios, modelling approach, etc.

(a) NPP Test Case

— *Requirements for contents of safety assessment*

The NPP Test Case followed the recommendations for safety assessment approach and content presented in Volume I of this report and also in Ref [1]. In addition, the national regulations of Sweden[3] are used to develop the safety assessment for decommissioning of the NPP. These regulations require deterministic approaches in safety assessment. The approach to safety assessment is prescribed, while the use of a graded approach is not explicitly required.

— *Use of IAEA safety standards*

IAEA safety standards were used, mainly the Basic Safety Standards [12] and WS-G-5.2 [1], as outlined in the DeSa safety assessment methodology (see main report).

— *Origin of dose limits for workers and public*

The dose limits for workers and the public are taken from the IAEA Basic Safety Standards [1].

— *Release/clearance criteria*

The safety standards for radiation protection used in the NPP Test Case are harmonized with IAEA requirements for site release [2] and clearance of material [8].

(b) Research Reactor Test Case

— *Requirements for contents of safety assessment*

This Test Case also followed the recommendations for safety assessment approach and content presented in Volume I of this report and also in Ref [1]. It also made use of the checklist of hazards and explanations as provided in the main report.

— *Use of IAEA safety standards*

The following IAEA documents were used in the safety assessment; Basic Safety Standards [12]; WS-G-5.2 [1], Safety Reports Series No. 19 (modelling of dispersion in the environment) [10]; and of Section 49 of the German Radiation Protection Ordinance [13] (modelling of dispersion in the environment).

— *Origin of dose limits for workers and public*

The dose limits for workers and the public are taken from the IAEA Basic Safety Standards [12].

— *Release/clearance criteria*

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[3]) The nuclear power plant serving as the model for the DeSa NPP Test Case is situated in Sweden.

Clearance and waste management are taken into account only to the extent they are relevant for safety during decommissioning. As the clearance of radioactive material is performed outside the research reactor in a dedicated waste management facility, clearance is outside the scope of this test case. In addition, the evaluation of compliance of the end-state of the project with clearance criteria is not subject to the assessment of safety during decommissioning, but is subject to the assessment of the end-state during assessment of the decommissioning plan.

(c) Laboratory Test Case

— *Requirements for contents of safety assessment*

The Laboratory Test Case followed the recommendations for safety assessment approach and content presented in Volume I of this report and also in Ref [1]. Dose limits, risk levels, clearance levels, discharge limits and waste acceptance criteria are explicitly listed in the document. Chemical thresholds are included to establish screening criteria for hazards and initiating events. Criticality limits are also considered as not applicable.

— *Use of IAEA safety standards*

IAEA standards are applied as first resource. Additional examples or applications from US or UK experience are provided where such standards are available and applicable.

— *Origin of dose limits for workers and public*

The IAEA dose limits from the Basic Safety Standards [12] are applied. Dose constraints from the UK are used, but criteria in question are regarded as non-conservative in some countries.

— *Release/clearance criteria*

Clearance criteria are based on IAEA recommendations [8] and TECDOC-855 [14]. Site release criteria are also considered.

(d) Comparison of the three DeSa Test Cases

The comparison of the three safety assessments showed the following:

- There are no basic differences between the three Test Cases with respect to the assumed starting points, the regulatory framework to be applied, etc.
- All three Test Cases use IAEA standards as first resource, national regulations are taken into account if necessary. National legislation may impose requirements on or set standards for the safety assessment approach, like e.g. the use of deterministic approaches (used for the NPP Test Case) or probabilistic approaches (used here for the Laboratory Test Case).
- Dose limits are the same for all three Test Cases. Dose constraints differ according to the conditions being specific to the project, as a consequence of the ALARA principle. Dose constraints for off-site consequences differ according to the number of nuclear facilities on the site, as exemplified for the Laboratory Test Case.

— Clearance criteria are harmonized, as the criteria given in RS-G-1.7 [8] and TECDOC-855 [14] are in the same order of magnitude.

9.2.2. Step 2: Preliminary assessment of hazards

The comparison focused on questions like: What data have been collected for the characterization of the facility? What types of scenarios have been used for the preliminary hazard assessment, i.e. those scenarios that provide the right picture of the highest potential risks? Is the data that has been collected for this preliminary assessment sufficient to perform the necessary screening calculations? Are the simplifications made for this preliminary assessment justified? Are the calculation methods commensurate with the aim of a screening analysis?

(a) NPP Test Case

— *Description of the facility and site specific data/hazards*

A description of the whole facility is provided, as the NPP Test Case serves to illustrate the approach for a safety assessment of an entire NPP. For sake of practicability within the DeSa project, however, the NPP Test Case focuses only on two systems. Therefore, part of the comprehensive information for the whole facility is not used in detail in the Test Case study.

Exposure pathways were derived for site-specific conditions. No specific evaluation for hazards such as flooding, earthquake and storms were performed in the NPP Test Case as these hazards had been evaluated in a previous safety assessment.

— *Use of history of the facility and previous safety assessments*

History of the facility was used for categorization of systems and rooms in the NPP (e.g. system 321, 322, etc.). Previous safety assessment results were considered.

— *Description of decommissioning activities*

The decommissioning tasks are described in detail and systematically for the systems covered by the test case.

— *Screening evaluation of hazards to workers and public from normal operation and from accidents*

The HAZOP method and the approach described in Ref [1] were used for selecting potential accident scenarios. The results of HAZOP assessment were used for identification of the most critical scenarios for the NPP Test Case to limit the number of relevant scenarios. Selected scenarios were then used for detailed analysis (high external dose to workers, faults during cutting, dropped loads). The maximum dose to workers was calculated to 29 mSv without mitigating measures, making further consideration of mitigating measures necessary. The doses to the public via environmental pathways were calculated as negligible. Therefore, consequences to the public were not further evaluated. A dose criterion of 10 µSv/a was used for members of the public, which was possible as the safety assessment pertains only to the two systems 321 and 322 and not to the full decommissioning of the NPP.

— *Consideration of radiological and chemical/industrial hazards*

Industrial hazards are directly addressed, chemical hazards are considered only implicitly.

— *Database for preparation of the preliminary assessment*

No specific database was used for the development of the Test Case report. The database of calculation of collective doses to workers and waste streams from decommissioning was used (components, pipes, sizes, masses, inventory, dose rates).

(b) Research Reactor Test Case

— *Description of the facility and site specific data/hazards*

The information about the facility and on the site is mostly descriptive, referring to engineering details only in more general terms. The general site description is provided in sufficient detail to support the analysis.

— *Use of history of the facility and previous safety assessments*

A description of the history of the facility, as well as some general data are given (e.g. maximum neutron flux, total output, etc.).

— *Description of decommissioning activities*

The general steps for the decommissioning and dismantling process are provided in the report, while further details are given in a spreadsheet as an appendix.

— *Screening evaluation of hazards to workers and public from normal operation and from accidents*

A checklist is used to evaluate all types of hazard. Several hazards are then selected for further analysis, providing rough estimates of the upper bounds of doses that were estimated in the screening evaluation.

— *Consideration of radiological and chemical/industrial hazards*

The chemical/industrial hazards are identified and safety measures to prevent them are discussed. The comparatively small dimensions of the facility and the technically mostly uncomplicated work steps limit the chemical or industrial hazards.

— *Database for preparation of the preliminary assessment*

No specific database was used for the development of the Research Reactor Test Case report. A set of data that was available for the safety assessment is presented in Annex I of this report.

(c) Laboratory Test Case

— *Description of the facility and site specific data/hazards*

A general description of the site, facility, processes etc. is provided in sufficient detail to support the analysis (including e.g. distance to boundary, general meteorology). Limited discussion of the remaining equipment is given (i.e. no specific discussion of system location/configuration), but this is sufficient to support the development of the safety assessment. Additional details from previous safety analysis are available through reference.

— *Use of history of the facility and previous safety assessments*

The inventory is based solely on historical information produced at the time of the last operation, but applied using a representative isotope as Pu dominates analysis. This is an example of the use of bounding described in the graded approach document. Previous safety assessment was not used directly in the safety assessment with the exception of site description, site layout and natural phenomena hazards that remain applicable.

— *Description of decommissioning activities*

The Test Case report includes details of decommissioning activities and their sequence. This information is provided at a level of detail that allows adequate analysis. Specific assumptions with respect to the configuration of equipment, material present and factors to be considered at the time of a proposed operation is provided in the evaluation.

— *Screening evaluation of hazards to workers and public from normal operation and from accidents*

The Test Case report describes the HAZOP process used to evaluate hazards specifically related to decommissioning activities. A general description of the hazards included in the evaluation is provided in the text. All critical groups are addressed. A detailed example of the fault schedule is also included, but not presented in a complete form.

— *Consideration of radiological and chemical/industrial hazards*

A checklist (see main report) is used to evaluate all types of hazard. Several hazards are then selected for further analysis. Chemical/industrial hazards were evaluated, with the conclusion that all will remain below threshold quantities, thus not requiring further evaluation.

— *Database for preparation of the preliminary assessment*

The test case applied HAZOPs process, used existing safety analysis, operational history reports, information from existing safety assessment document.

(d) Comparison of the three DeSa Test Cases

The comparison of the three safety assessments showed the following:

— The descriptions of the facilities have been provided in a similar level of detail for all three Test Cases. The reason is that a certain level of detail is required for adequate identification of existing or potential hazards, definition of scenarios, calculations within the safety assessment, etc. This means that the overall amount of information to be provided if the NPP Test Case covered decommissioning of the whole facility and not only of the two systems would be substantially larger for the NPP Test Case than for the other two test cases.

- The description of the surroundings of the sites of the facilities has been graded according to the potential off-site consequences that had been identified in the preliminary hazard assessment. In the case of the Laboratory Test Case, reference was made to the existing safety assessments from operational phase as the preliminary hazard assessment has shown that no off-site consequences exist. In the case of the Research Reactor Test Case, a similar result was obtained from the preliminary hazard assessment so that only the information relevant for a simple conservative estimate for the atmospheric pathway was provided. The NPP Test Case provided a full set of information on the weather statistics, the hydrogeological situation, the population and land use in the surroundings etc. in order to allow a full-scale assessment of all environmental pathways. (This information was given with respect to a full safety assessment for the entire NPP decommissioning, while the actual safety assessment only pertained to the two systems 321 and 322 for which part of the information would not have been relevant). Some of the hazards (e.g. flooding) were screened out on the basis of the operational safety assessment results.

- Use of safety assessments from the operational phase and of other data from the history of the facility has been made in all three Test Cases, however, to a different degree. For example, the safety assessment for the NPP Test Case did not have to re-analyze the hazard from flooding again because this could be taken over from the operational safety assessment. Likewise, several scenarios could be taken over from the safety assessment carried out for the care and maintenance phase. Furthermore, interviews with key personnel from the operational phase were conducted to compile a base of information of incidents during operation. The Research Reactor Test Case used existing safety assessment as background information only; the scenarios analyzed in existing safety assessments, however, were found to be not relevant for the safety assessment for decommissioning. The Laboratory Test Case also used existing safety assessments for screening out certain initiating events.

- The description of the work packages/work steps in the three Test Cases is graded according to the complexity of the work sequences to be described. For example, in the Laboratory Test Case, the dismantling of a glovebox is described once in general, while only deviations for single gloveboxes (depending on contamination levels) are highlighted. The description for the NPP Test Case follows a tiered approach, describing the entire decommissioning work in general, while providing detailed descriptions only for systems 321 and 322 (which were analyzed in detail). In contrast to this, a less complex decommissioning project like the Research Reactor Test Case provided the entire detailed description of all work steps.

- The screening evaluation of hazards to workers and the public under normal decommissioning activities, as well as under accident conditions was performed using the general checklist for hazard identification (see main report). Certain scenarios were selected from this pre-analysis for further screening analysis for calculation of doses. Here, the selection criteria for these scenarios differed slightly between the three Test Cases: mainly expert judgment was applied in the NPP Test Case, expert judgment followed by conservative dose estimates for the Research Reactor Test Case, risk based screening for the Laboratory Test Case. This indicates that the most complex approach was used in the Laboratory Test Case with complex source term and dispersion pathways, as here the decisions on when it would be possible to remove controls (especially filtration, tents and respiratory protection) were not as straightforward as in the other two test cases. The calculations have been performed taking into account data that are specific for the facility and the site.

— Industrial and chemical hazards were treated only on a qualitative basis for all three Test Cases. The potential hazards were identified from the checklist, but were analyzed in more detail (outside the scope of the DeSa project).

9.2.3. Step 3: Radiological characterization and categorization of the facility and its systems, structures and components

The comparison of the three Test Cases focused on the following questions: Does the level of detail of the acquired data correspond to the assessment methods and models that have to be/are intended to be used (see Step 1)? Have enough data from all required categories been collected? If it was not possible to obtain real data from the facility itself, have sufficiently conservative estimates been used? Have enough samples and been taken and analyzed and have enough measurements been performed to obtain a sufficiently detailed picture of the situation of contamination/activation in the facility? Have the various scenarios and situations to be assessed in the Step 4 been assigned to the right categories of hazard potential, based on the data from the radiological characterization, with respect to the possible impact on the public, the radiological safety significance, the exclusion of any off-site consequences, etc.)?

(a) NPP Test Case

— *Sampling programme and measurements*

The sampling programme was graded according to potential hazards of the two systems 321 and 322 (around 50 sampling points at system 321, statistical sampling programme for system 322), additional dose rate and other measurements were performed appropriately.

— *Determination of radioactive inventory*

For the radioactive inventory, actual measurement results were used. No specific inventory determination has been made for the less contaminated system 322.

— *Application of measurement methods (direct/indirect methods)*

The usual spectrum of measurement methods is reported, i.e. wipe tests, dose rate measurements, material samples etc. The measurement methods applied for system 321 are, however, not described in detail.

— *Contaminated/activated areas and masses for which decommissioning is required*

Volumes and masses for the two systems under consideration were determined..

— *Distribution of activity in the facility (categorization of rooms and systems)*

A database of dose rate measurement at hot spots is available. It is reported that no contamination on floor and walls is present around system 321. A formalized categorization scheme for the systems is not used.

— *Use of knowledge of the history of the facility*

Knowledge of the history of the facility was taken into account in radiological characterization (e.g. grouping of rooms for system 321).

(b) Research Reactor Test Case

— *Sampling programme and measurements*

A limited sampling and measurement programme to support already available data from operational history and gamma-dose measurements was carried out. Samples were taken only to a limited extent.

— *Determination of radioactive inventory, including hard-to-measure radionuclides*

The radioactive inventory was derived from the sampling and measurement programme as well as from operational history and other measurements referred to above. Specific reference to hard-to-measure radionuclides is made only for Sr-90 and an upper estimate for α-emitters present in the facility.

— *Application of measurement methods (direct/indirect methods)*

Only direct measurement methods were applied.

— *Contaminated/activated areas and masses for which decommissioning is required*

A general description of the equipment that is to be dismantled and its masses is given. Data on masses were used that originate from design data.

— *Distribution of activity in the facility (categorization of rooms and systems)*

A general description of the facility and the categorization of rooms and systems is provided.

— *Use of knowledge of the history of the facility*

Historical data are used partly as background information for the safety assessment.

(c) Laboratory Test Case

— *Sampling programme and measurements*

No specific sampling was done or requested to support the Laboratory Test Case. The bounding analysis approach was determined to be sufficiently conservative for performance of analysis. Data was based on last measured condition and was modelled in worst-case configuration.

— *Determination of radioactive inventory*

Inventory was provided for each area of work. Material type was estimated as the bounding isotope rather than performing specific analysis to consider the actual radioactivity inventory.

— *Application of measurement methods (direct/indirect methods)*

Measurements of the inventory of the gloveboxes were performed. Further measurements with in situ gamma spectrometry of the building surfaces will be performed in the clearance process.

— *Contaminated/activated areas and masses for which decommissioning is required*

Quantities and form of material was estimated and considered in the worst case for all scenarios. This was done for simplicity of the analysis, recognized as a conservative representation of the hazards present.

— *Distribution of activity in the facility (categorization of rooms and systems)*

Information is provided which describes the configuration and distribution of material within the laboratory facility. Analysis considers the contribution from interconnected systems and areas using bounding assumptions.

— *Use of knowledge of the history of the facility ("plant memory")/historic documents*

A section on the operational history is included in the Test Case report with a brief description of the purpose or previous mission of equipment that will be removed. This information is general in nature and was carried forward as far as it may affect decommissioning activities (example: spills having caused residual contamination on floors/walls).

(d) Comparison of the three DeSa Test Cases

The comparison of the three safety assessments showed the following:

— The sampling programme in preparation for the safety assessments took into account the available information on the radioactive inventory and was tailored to the contamination level of the systems or areas (for example, quite different sets of measurements were used for systems 321 and 322 of the NPP Test Case). In choosing sampling locations, information on existing contamination (e.g. from known spills) were evaluated. In the NPP and the Research Reactor Test Cases, a sampling programme was part of the preparation of the safety assessment and has therefore been explicitly mentioned in the Test Case reports, while in the Laboratory Test Case, recourse was made to existing sampling and inventory information that had been obtained at the completion of operations.

— The sampling approaches and programmes were graded according to the available information on the radioactive inventory. The NPP and the Laboratory Test Cases mainly made use of direct measurements (material samples for system 321 in the NPP, wipe tests from the inside of gloveboxes in the Laboratory Test Case), while the Research Reactor Test Case mainly used dose rate measurements for inferring the contamination and used only a few material samples (especially graphite). The approach used in the Research Reactor Test Case is in line with the approach taken in the NPP Test Case for the low contaminated system 322 where only dose rate measurements were applied. The sampling programmes in the three Test Cases were further supported by dose rate measurements that were performed for the assessment of the decommissioning work steps.

— The NPP Test Case addressed the question which hard-to-measure radionuclides needed evaluation and which correlation methods were to be applied. The Laboratory and Research Reactor Test Cases selected the most relevant radionuclides and focused the evaluation on those radionuclides, which is an illustration of the application of the graded approach. The selection of the most relevant radionuclides was based on the activity percentage and the radiological significance of these radionuclides, as well as on experience from similar projects.

— The identification of contaminated areas and masses was performed in all three Test Cases to such a level that the decommissioning work could be adequately planned. However, the required effort differed between the Test Cases. Most effort (e.g. opening of closed systems for sampling) was invested in the NPP Test Case (system 321), as the knowledge of the physical and radiological inventory is of high relevance for the estimation of the resources required for decontamination and dismantling. In the Laboratory Test Case, the effort was lower because of the accessibility of the gloveboxes, the simpler sampling methods that could be used and the overall smaller physical inventory. The effort required in the Research Reactor Test Case was in between the other two Test Cases as the area to be characterized was larger than in the Laboratory Test Case, the activation of structures had to be determined and the operation records indicated spills during the operation history, and was much lower than in the NPP Test Case because of the much lower radioactive inventory. This comparison indicates that stocktaking of contaminated areas and masses can be graded according to factors like physical and radioactive inventory, type of contamination, accessibility of surfaces and the history of the facility.

— All three Test Cases made use of the knowledge of the history of the facility and historic documents as far as possible, including the personal experience of key personnel of the facilities. However, the impact of the availability of data from the operational history correlates with the complexity of the facility, as the use of historical can efficiently reduce the effort for characterization.

9.2.4. Step 4: Performance of safety assessment

The comparison focuses on questions like: Have all relevant exposure situations for workers and (if relevant) for the public been identified for normal decommissioning conditions? Have all relevant initiating events been identified for incident/accident situations? Does the model cover all relevant scenarios/pathways for workers and for the public both during normal decommissioning conditions and during incident/accident conditions? Have the calculation methods been chosen according to the legal requirements (Step 1) and according to the results of the screening assessment (Steps 2 and 3)? Are the required data available with the correct level of detail (Steps 2 and 3)? Are the results presented and analyzed with the correct level of detail in order to draw all relevant conclusions (e.g. are the workplaces grouped into a sufficiently large number of categories according to the hazard potential so that the most relevant workplaces with respect to safety can easily be identified)?

(a) NPP Test Case

The general procedure for hazard analysis that was followed in the safety assessment for the NPP Test Case is outlined in Fig. 4.

— *Link between decommissioning work steps and the radioactive inventory involved*

A link between dose rates/inferred radioactive inventory and the work steps has been established when modelling the doses to workers for normal decommissioning conditions using the OMEGA computer code and for accident conditions by using the HAZOP method.

— *Approach for the calculation of doses to workers from normal decommissioning conditions*

A list of work steps was compiled for cutting operations on system 321, which were related to six occupations. Furthermore, an additional breakdown of the work into a large number of plant items was carried when the OMEGA code was implemented (see Annex I of this report). As the maximum size of the material after dismantling was determined by the waste containers, the numbers of plant items are mainly determined by the physical complexity of the system. For the larger system 322, a number of 1 031 plant items were defined, for system 321 a number of 373. This approach also determined the single work steps for which doses were calculated using the OMEGA code.

— *Approach for the calculation of doses to the public from normal decommissioning conditions*

Doses to the public were calculated mainly by using the DecDose computer code (see Annex I of this report). This programme assesses the annual public dose from radioactive gas and airborne particles discharged into the environment through various pathways from the facility where dismantling activities such as cutting and decontamination are conducted. This computer code takes into account all relevant environmental pathways.

— *Approach for the calculation of doses to workers from accidents*

The dose calculation from accident scenarios was started from a list of scenarios covering high external dose to a worker, faults during cutting operations and dropped loads. First, unmitigated consequences were evaluated, then – if dose limits were exceeded – safety measures were featured into the calculations and ALARA considerations were applied. The highest dose from an accident scenario was calculated as 96 mSv to a worker, which exceeds the 20 mSv reference level but was nevertheless assessed to be ALARA because of the extremely low probability.

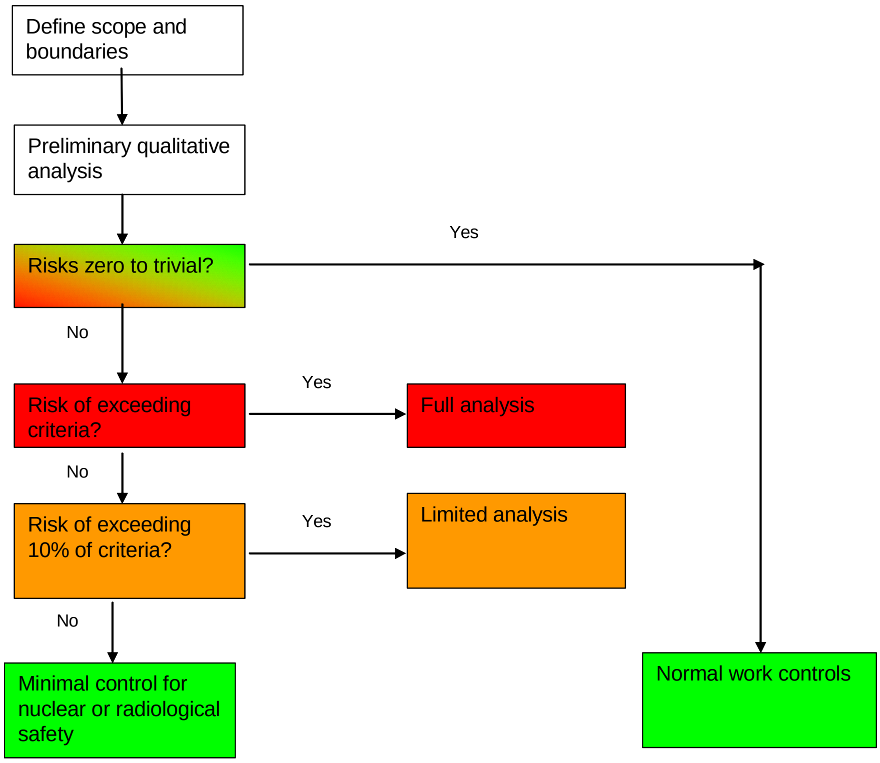

FIG. 4. The graded approach as applied in the NPP Test Case.

— *Approach for the calculation of doses to the public from accidents*

The DecDose code was used for calculation of public exposure for normal decommissioning and for accident conditions.

(b) Research Reactor Test Case

— *Link between decommissioning activities and the radioactive inventory involved*

A link between the decommissioning activities and the radioactive inventory was established for the analysis of worker scenarios (normal operation and accident), mainly via dose rates and contamination data, as well as the list of work steps referred to in Annex I of this report.

— *Approach for the calculation of doses to workers from normal decommissioning conditions*

The most relevant steps were analyzed on the basis of external exposure; using a model developed for the computer code VISIPLAN. This model was based on the dose rates determined at the start of decommissioning.

— *Approach for the calculation of doses to the public from normal decommissioning conditions*

Exposure for members of the public from inhalation and direct exposure from ground deposition, as well as from the secondary ingestion, via radioecological pathways was considered in this test case. The calculations were based on a Gaussian model for the routine releases incorporated in the Fortran 77 computer code, developed in Vinča Institute of Nuclear Sciences and using the RESRAD computer code [5], which provided a very detailed presentation of the assessment results.

— *Approach for the calculation of doses to workers from accidents*

No detailed analysis of doses to workers beyond the scenarios mentioned above were performed (only pre-analysis of internal and external exposure), and reference was made to the proposed mitigating measures.

— *Approach for the calculation of doses to the public from accidents*

One scenario was evaluated (fire) using German computer code for the calculations (see Annex I of this report).

(c) Laboratory Test Case

— *Link between decommissioning activities and the radioactive inventory involved*

The decommissioning to which the safety assessment pertains has been divided into four main steps; (i) vent removal, (ii) construction of ModuCon (the modular containment for the gloveboxes), (iii) size reduction and (iv) waste management (followed by clearance). Each step is further divided into 2 or 3 groups of scenarios, which are distinguished by the dose rate. The number of personnel, the dose rate and the estimated exposure time for the particular work step are given to calculate external exposure. No further breakdown of the work is done for estimation of doses from inhalation.

The analysis of accident scenarios is based on numerous initiating events and assumptions related to the inventory in the area of the facility that might be affected by the scenarios.

— *Approach for the calculation of doses to workers from normal decommissioning conditions*

The calculation of doses to workers from external irradiation during normal decommissioning conditions is carried out on the basis of the four main steps and various sub-steps referred to above. For each step/sub-step, the collective dose to workers is calculated as the product of number of personnel involved in this work step, the time required for execution of this work step and the dose rate. Individual doses are estimated on the basis of an individual taking part in all steps of the process.

An additional estimate of doses from internal doses (inhalation) is carried out on the basis of ambient activity concentration measured by air sampling times the number of working hours. The same assumptions are used for all staff assigned to one task.

— *Approach for the calculation of doses to the public from normal decommissioning conditions*

The doses to the public from normal decommissioning conditions are addressed with respect to the dose criterion and are then treated together with accident scenarios (see below) that are used as conservative upper bounds.

— *Approach for the calculation of doses to workers from accidents*

Numerous accident scenarios are introduced and described in detail in Annex I of this report, based on the previous screening of hazards and initiating events. The scenarios are grouped according to the type of initiating event and then on the basis of the area of the facility that might be affected. Detailed assumptions for each scenario are outlined and are used in the subsequent dose calculation. The scenarios are analyzed for dose consequences both for workers and – if offsite consequences are possible – to members of the public. In the interpretation of the results, the estimated frequency of occurrence of the scenarios is taken into account.

— *Approach for the calculation of doses to the public from accidents*

The calculation of doses to the public from accidents is performed based on the same initiating events and scenarios as addressed above for workers. No detailed information is given on how the dispersion of the activity in the atmosphere after release (i.e. the activity concentration at the receptor point) is modelled and what types of exposure from these releases are taken into account.

(d) Comparison of the three DeSa Test Cases

The comparison of the three safety assessments showed the following:

— Normal decommissioning conditions for workers: The basic approach (breakdown of the work into work packages, estimation of man-hours, number of personnel and dose rate, calculation of dose as product of these three factors) is similar for all three Test Cases. Dose rates were determined from measurements with the help of computer codes, considering the distances during the work steps. The main difference between the NPP Test Case on the one hand and the Research Reactor and Laboratory Test Case on the other hand was the number of major work packages into which the work was divided: the number of work steps for one system of the NPP Test Case was of the same order as for the entire Laboratory and Research Reactor Test Case.

— Normal decommissioning conditions for public: The estimation of the source term was done in a similar way for all three Test Cases by assuming a certain percentage of the activity inventory to become mobilized and dispersed into the atmosphere.

— While the HAZOP analysis for the NPP Test Case indicated that virtually no dose to the public would be possible from decommissioning of the two systems under consideration, this appraisal was underpinned by DecDose calculations leading to an estimation for the source term for releases from cutting operations and subsequent assumptions on retention in the building and in the filtration.

— The most complex dispersion model was used for the NPP Test Case, assuming release via the stack, while the dispersion models used for the Research Reactor and Laboratory Test Cases used a similar approach (straight-line Gaussian model) at ground level (Laboratory Test Case) and via the stack (Research Reactor Test Case). The full characteristics of the meteorological conditions (annual basis) were taken into account only for the NPP Test Case, while steady-state conditions were assumed for the Research Reactor and Laboratory Test Cases. The reason for using simple assumptions for the latter two facilities is the fact that the preliminary assessment showed that the expected doses are low and a more sophisticated approach would therefore not be necessary.

— The exposure pathways were modelled for all three test cases using dedicated computer codes covering all relevant exposure pathways.

— Accident conditions for workers: Detailed analyses of doses to workers from accident conditions were only performed for the NPP and the Laboratory Test Cases, while no further detailed analysis was considered necessary for the Research Reactor Test Case. The reasons are the activity inventory (amount of activity for the NPP Test Case, transuranium radionuclides for the Laboratory Test Case) and its physical form (dispersible for NPP and Laboratory Test Cases during cutting operations and opening of systems, mainly fixed for the Research Reactor Test Case).

— The approaches used in the NPP and Laboratory Test Cases were similar. Probabilistic approaches were used in the hazards screening of the Laboratory Test Case.

— Accident conditions for public: The estimation of the source term for the Research Reactor Test Case was done using one enveloping, conservative scenario (fire). For the Laboratory Test Case, the same scenarios were considered for determination of source terms as for the dose analysis for workers (see above).

— For dispersion and exposure pathways analysis, similar assumptions as for normal operation are applied for all three test cases.

9.2.5. Step 5: Implementation of Measures based on the Results of the Safety Assessment

(a) NPP Test Case

On the basis of the safety assessment results, the use of protective masks and the ventilation was prescribed for cutting operations.

(b) Research Reactor Test Case

On the basis of the safety assessment results, reference to mitigating measures for accident scenario for workers is made.

(c) Laboratory Test Case

On the basis of the safety assessment results, the use of protective masks and the ventilation was prescribed for cutting operations.

(d) Comparison of the three DeSa Test Cases

The comparison of the three safety assessments showed the following:

- The results of the safety assessments for the NPP and the Laboratory Test Cases show that strong protective measures against inhalation need to be applied during normal decommissioning conditions, while this is of less significance for the Research Reactor Test Case.
- Apart from the statement above, the three Test Case reports (currently) do not contain enough information on implementation measures following from the safety assessments to draw further conclusions on application of the graded approach. This aspect is planned to be addressed in the DeSa follow-up project.

9.3. SUMMARY OF OBSERVATIONS

The comparison of the three DeSa Test Cases reveals that the assessment approaches have been graded, i.e. adjusted according to the complexity of the analysis and the hazard potential, in a number of steps. This applies mainly to the preliminary hazard evaluation and the characterization of the facility, as well as to the actual implementation of the safety assessment. Other areas of a more general character, like the description of the facility and its surroundings, exhibit grading only to a limited extent. The comparison illustrates that the three DeSa Test Cases, although carried out by independent Working Groups have applied grading quite naturally.

10. CONCLUSIONS AND RECOMMENDATIONS

The application of a graded approach in performing safety assessment for decommissioning of facilities using radioactive material is a general requirement [2]. It can save substantial efforts in development and review of safety assessment and can help to direct these analyses to safety relevant areas and hazards and to avoid the use of resources on irrelevant aspects that are covered by other parts of the assessment. There are examples of the application of the graded approach from a large number of decommissioning projects, as well as from various countries. The DeSa project, through the Graded Approach Working Group worked on the consolidation of the lessons learned from the applications and the outcomes are presented in this volume of the report.

Application of the graded approach has also become apparent when analysing safety assessments of differing levels of complexity for the three Test Cases within the DeSa project, as working groups are currently performing safety assessments for an NPP, for a research reactor and for a laboratory to which the methodology developed within the DeSa framework was applied (see main report). The characteristics of these assessments have been analyzed by the Graded Approach Working Group with respect to grading.

(a) The lesson learned concerning Step 1 of the safety assessment process lies in the fact that the regulatory framework of a particular country may address the graded approach for various aspects of safety assessments, such as the prescription of the use of certain calculation models for larger facilities. Furthermore, the use of a graded approach required by the Regulatory Bodies', i.e. Regulatory Bodies to be applied in safety assessments commensurate with the hazard of the facility or the decommissioning work to be analyzed.

(b) The lesson learned concerning Step 2 of the safety assessment process is that upon completion of this step, the operator and the safety assessor need to have a clear idea of the necessary level of detail in which the safety assessment for decommissioning is to be carried out. This needs to be based on a qualitative and – if possible – a bounding quantitative preliminary assessment of the facility and decommissioning activities.

(c) A lesson learned from accomplishing the tasks of Step 3 is that adequate characterization and categorization of areas within a facility under decommissioning and of the decommissioning work to be performed can focus and direct the identification of the most relevant and critical hazards that need to be evaluated during decommissioning.

(c) The lessons learned from Step 4 is that the safety assessment can be carried out:

- By using enveloping approaches and methods by which compliance with limits is demonstrated using simple approaches that overestimate the real situation; or
- By focussing the effort in the assessment on those hazards, which are most relevant to safety and on which the assessment results will depend to the largest extent.

(d) Although being different in complexity and scope, all three DeSa Test Cases have identified areas where the safety assessment indicated certain additional safety measures to be beneficial. Therefore, a lesson learned from Step 5 is that even simple safety assessments of a low degree of complexity allow to draw conclusions on measures how to increase worker safety and how to reduce potential exposures.

(e) The lesson learnt from the comparison of the three DeSa Test Cases with respect to grading in the various steps of the safety assessments reveals that all Test Cases have been based on similar approaches, using the IAEA safety framework as the first resource, and that the level of detail has been tailored according to the complexity of the work to be analyzed and to the hazard potential of the facility and the work to be performed.

In summary, experience from numerous decommissioning projects shows that the use of a graded approach in the various steps of safety assessment may save effort and time and will help to focus on the areas of highest importance. The use of a graded approach can thus contribute to the overall safety of decommissioning projects, and it has therefore been incorporated in decommissioning planning from the very beginning, analysing how to deal with the various issues of safety assessment in the most efficient way.

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Appendix I

EXAMPLES FOR A GRADED APPROACH FOR SAFETY ASSESSMENT IN THE REGULATORY FRAMEWORK

I-1. GRADED APPROACH IN THE REGULATORY FRAMEWORK FOR PERFORMING ASSESSMENTS OF DOSES DUE TO RADIOACTIVE RELEASES FROM NUCLEAR FACILITIES (GERMANY)

An example for a graded approach in assessment of doses due to radioactive releases from nuclear facilities can be found in the German Radiation Protection Ordinance (*Strahlen schutzverordnung*) [13] (Sect. 47 in combination with Appendix VII, Part D). The main aspects of the regulatory framework are interpreted below:

"§ 47: Limitation of the Discharge of Radioactive Substances

(1) For the planning, construction, operation, **decommissioning**, safe enclosure and dismantling of facilities or installations, the following limits of the radiation exposure of individual members of the public related to discharges of radioactive substances from these facilities or installations by means of air or water per calendar year apply:

1.	Effective dose	0.3 mSv
2.	Organ absorbed dose for gonads, uterus, bone marrow (red)	0.3 mSv
3.	Organ absorbed dose for colon, lungs, stomach, bladder, breast, liver, gullet, thyroid gland, other organs or tissues as specified in Appendix VI, Part C, and subpara. (2), Footnote 1, unless specified in subpara. 2	0.9 mSv
4.	Organ absorbed dose for bone surface, skin	1.8 mSv

Steps shall be taken to ensure that radioactive substance is not discharged into the environment unmonitored.

(2) In the planning of facilities or installations, the radiation exposure as specified in para. (1) shall be applied for a reference person at the most unfavourable receiving points, considering the exposure pathways specified in Appendix VII, Parts A to C, the living habits of the reference person and the other assumptions; the average consumption rates specified in Appendix VII, Part B, Table 1 multiplied by the factors specified in Column 8 shall be used. With the consent of the Federal Council, the Federal Government will issue administrative provisions relating to further assumptions to be made. The competent authority may consider the limits specified in para. (1) To have been complied with if this is demonstrated on the basis of said general administrative provisions.

(3) For the operation, decommissioning, safe enclosure and dismantling of facilities or installations, the competent authority shall determine the permitted discharges of radioactive substances from these facilities or installations by means of air or water by restricting the concentrations or

quantities of activity. Proof of compliance with the limits specified in para. (1) is deemed furnished when these restrictions are not exceeded.

(4) For facilities or installations not requiring a license granted under §§ 6, 7 or 9 of the Atomic Energy Act or a plan approval granted under § 9b of the Atomic Energy Act, the competent authority may refrain from determining quantities and concentrations of activities and consider the proof as specified in para. (2) regarding compliance with the limits referred to in para. (1) to have been furnished, insofar as the permitted activity concentrations for discharges of radioactive substances by means of air or water from radiation protection areas as specified in Appendix VII, Part D are not exceeded on average per year. Unless the competent authority determines otherwise, the permitted activity concentrations shall be complied with at the boundary of a radiation protection area. The first sentence shall not apply if the competent authority issues criteria according to which the limits referred to in para. (1) may be exceeded at a site through discharges from facilities or installations or previous practices."

Appendix VII of the Radiation Protection Ordinance has the following structure:

- Exposure paths to be taken into account in the assessments (Part A);
- Lifestyles (including details on the diet, breathing rates, exposure times etc.) (Part B);
- Remaining assumptions (Part C); and
- Maximum permissible activity concentration from radiation protection areas (Part D).

Appendix VII Part D of the Radiation Protection Ordinance also defines the following:

"For several radionuclides, the sum of the ratios from the mean annual concentration of the radionuclides in air or in water in Bq/m^3 ($\overline{C}_{i,a}$) and the relevant calculated mean annual concentration value of the given radionuclide (C_i) in Table 4 or 5 shall be determined (sum formula), where i is the given radionuclide. This sum shall not exceed the value 1.

$$\sum_i \frac{\overline{C}_{i,a}}{C_i} \leq 1 \,. \quad \text{(A1)}$$

Daughter radionuclides shall be taken into account."

The maximum permissible activity concentration in the air from supervised areas is defined for:

(a) Inhalation

The activity of the radionuclide i in the annual average per cubic metre of air.

— For exhaust air streams $Q \leq 10^4$ m^3 h^{-1} may not be higher than ten times the values given in Table 1, Column 2 or Table 2, Column 2; or

— For exhaust air streams 10^4 m^3 h^{-1} < $Q \leq 10^5$ m^3 h^{-1} may not be higher than the values given in Column 2 of Table 1. or Table 2.

(b) Submersion

The activity of the radionuclide i in the annual average per cubic metre of air is defined as follows:

— For exhaust air streams $Q \leq 10^4$ m^3 h^{-1} may not be higher than ten times the values specified in Table 3, Column 2; or

— For exhaust air streams 10^4 m^3 h^{-1} < $Q \leq 10^5$ m^3 h^{-1} may not be higher than the values of Table 3, Column 2.

Maximum permissible activity concentration in water that is released from supervised areas into sewers.

(a) Ingestion

The activity of the radionuclide i in the annual average per cubic metre of air.

— For waste water quantities $\leq 10^5$ m^3 a^{-1} may not be higher than ten times the values given in Table 1, Column 3 or Table 2, Column 4; or

— For waste water quantities > 10^5 m^3 a^{-1} may not be higher than the values given in Table 1, Column 3 or Table 2, Column 4.

TABLE 1 REPRODUCTION OF THE BEGINNING OF APPENDIX VII (TABLE 4 OF THE GERMAN RADIATION PROTECTION ORDINANCE)

Radionuclide		**C_i**	
A = aerosole (air) **E = elemental (air)** **O = organic**		**in air [Bq/m³]**	**in water [Bq/m³]**
1		**2**	**3**
H-3 H-3	A O	1×10^{2}	1×10^{7} 7×10^{6}
Be-7 Be-10	A A	6×10^{2} 1	5×10^{6} 6×10^{4}
C-11 C-14	A A	6×10^{2} 6	3×10^{6} 6×10^{5}
F-18	A	5×10^{2}	2×10^{6}
Na-22 Na-24	A A	1 90	4×10^{4} 3×10^{5}
Mg-28	A	20	7×10^{4}
Al-26	A	0.5	1×10^{4}
Si-31 Si-32	A A	3×10^{2} 0.3	5×10^{5} 1×10^{5}
P-32 P-33	A A	1 20	3×10^{4} 3×10^{5}
S-35 S-35	A E	20	7×10^{5} 1×10^{5}
Cl-36 Cl-38 Cl-39	A A A	0.1 5×10^{2} 6×10^{2}	1×10^{4} 6×10^{5} 9×10^{5}
K-42 K-43 K-44 K-45	A A A A	2×10^{2} 2×10^{2} $1 \cdot 10^{3}$ 2×10^{3}	2×10^{5} 4×10^{5} 9×10^{5} 1×10^{6}
…	…	…	…

TABLE 2 REPRODUCTION OF APPENDIX VII (TABLE 6 OF THE GERMAN RADIATION PROTECTION ORDINANCE)

Radionuclide mixture	**C_i in air [Bq/m³]**	**Radionuclide mixture**	**C_i in water [Bq/m³]**
1	**2**	**3**	**4**
Any mixture	1×10^{-5}	Any mixture	10
Any mixture if Ac-227 and Cm-250 can be ignored	1×10^{-4}	Any mixture if Po-210, Ac-227, Ra-228 and Cm-250 can be ignored	50
Any mixture if Ac-227, Am-241, Am-242m, Am-243, Cm-245, Cm-246, Cm-247, Cm-248, Cm-250, Pa-231, Pu-238, Pu-239, Pu-240, Pu-242, Pu-244, Th-229, Th-230 and Th-232 can be ignored	5×10^{-4}	Any mixture if Po-210, Ac-227, Ra-228, Th-229, Pa-231, Bk-247, Cm-248, Cf-249, Cm-250, Cf-251 and Cf-254 can be ignored	1×10^{2}
Any mixture if Ac-227, Am-241, Am-242m, Am-243, Bk-247, Cf-249, Cf-251, Cf-254, Cm-243, Cm-244, Cm-245, Cm-246, Cm-247, Cm-248, Cm-250, Np-237, Pa-231, Pu-236, Pu-238, Pu-239, Pu-240, Pu-242, Pu-244, Th-228, Th-229, Th-230, Th-232 and U-232 can be ignored	1×10^{-3}	Any mixture if Sm-146, Sm-147, Gd-148, Gd-152, Po-210, Pb-210, Ra-223, Ra-224, Ra-225, Ra-226, Ra-228, Th-228, Ac-227, Th-229, Th-230, Pa-231, Th-232, U-232, Pu-236, Pu-238, Pu-239, Pu-240, Pu-244, Cm-245, Cm-246, Bk-247, Cm-247, Np-247, Cf-248, Cm-248, Cf-249, Cf-250, Cm-250, Cf-251, Cf-252, Cf-254, Es-254 and Fm-257 can be ignored	1×10^{3}

TABLE 3 REPRODUCTION OF APPENDIX VII (TABLE 5 OF THE GERMAN RADIATION PROTECTION ORDINANCE)

Radionuclide	C_i in air [Bq/m³]
1	2
C-11	3 x 10^3
N-13	2 x 10^3
O-15	1 x 10^3
Ar-37	2 x 10^8
Ar-39	6 x 10^3
Ar-41	2 x 10^2
Kr-74	2 x 10^2
Kr-76	5 x 10^2
Kr-77	2 x 10^2
Kr-79	9 x 10^2
Kr-81m	5 x 10^6
Kr-81	4 x 10^4
Kr-83m	4 x 10^6
Kr-85	4 x 10^3
Kr-85m	1 x 10^3
Kr-87	2 x 10^2
Kr-88	1 x 10^2
Xe-120	6 x 10^2
Xe-121	1 x 10^2
Xe-122	3 x 10^3
Xe-123	3 x 10^2
Xe-125	9 x 10^2
Xe-127	9 x10^2
Xe-129m	1 x 10^4
Xe-131m	2 x 10^4
Xe-133	7 x 10^3
Xe-133m	7 x 10^3
Xe-135m	5 x 10^2
Xe-135	9 x 10^2
Xe-138	2 x 10^2

Conclusions

The German Radiation Protection Ordinance states the general requirements pertaining to dose assessments for gaseous or liquid releases from nuclear facilities and facilities to be or under decommissioning. There are two ways in which these assessments can be performed:

(a) For large facilities that require a licence according to the Atomic Energy Act, the safety analyses need to be performed in a detailed way. Some details of such an assessment are even laid down in the Radiation Protection Ordinance itself (Appendix VII parts A, B and C as outlined above) while the complete details are given in a General Administrative Regulation (*Allgemeine Verwaltungsvorschrift*), including all formulae and default parameter values. This is the normal case for nuclear facilities in Germany, such as NPPs, large research reactors, fuel cycle facilities etc.

(b) For smaller facilities (i.e. those not requiring a licence according the Atomic Energy Act or a plan approval granted under § 9b of the Atomic Energy Act) default values for concentrations in air and water are given (see reproduction of Appendix VII Part D as well as Tables 1, 2 and 3 above). If those values are complied with in the way described in Section 47 para. 4 and in Appendix VII Part D, it is assured that the dose constrictions listed in Section 47 para. 1 Radiation Protection Ordinance are complied with.

In conclusion, the type of facility and thus the hazard potential determines whether a detailed and complex analysis is necessary or whether simply default concentration values may be used. This is associated with a vast difference in effort required for performing the analysis vs. implementing the default concentration values.

I-2. DERIVATION OF RELEASE LEVELS WITH THE RESRAD CODE (USA)

The RESRAD code is a computer model designed to estimate doses and risks from RESidual RADioactive materials [5] and is issued by the Environmental Assessment Division of Argonne National Laboratory (USA). There are model versions for materials, buildings and sites. The computer codes for these models are used in dose/risk assessments for nuclear facilities in the USA as a kind of reference.

A graded approach with these computer codes is possible as follows: The RESRAD models incorporate all necessary (default) parameter values and exposure pathways that would cover any generic exposure condition. The user can apply these data and assumptions for deriving suitable exposure assessments and release criteria. It is, however, also possible to use site specific parameters which are derived from evaluations of the conditions prevailing at that particular facility and/or to exclude certain exposure pathways on the basis of site specific evaluations.

(a) The first alternative to use generic data is the simplest approach and will lead to manageable results which may, however, in some cases lead to overly conservative results. As long as the release criteria that are derived from the generic approach can be met without unjustified effort, this simpler alternative can be applied which would usually be the case for smaller facilities.

(b) The other alternative of establishing a site-specific set of input data for the calculations requires more effort and time, which may later be compensated by saving effort and money, which would otherwise have been spent uselessly by meeting overly conservative criteria. This approach is more relevant for larger nuclear facilities.

A graded approach is also used in USA to establish the level of documentation and approval procedures required, determined by the facility categorization. For each category, there is a specific requirement for performance of safety assessment (see the Appendices to main report). The processes are illustrated in Fig. 5 and Fig. 6.

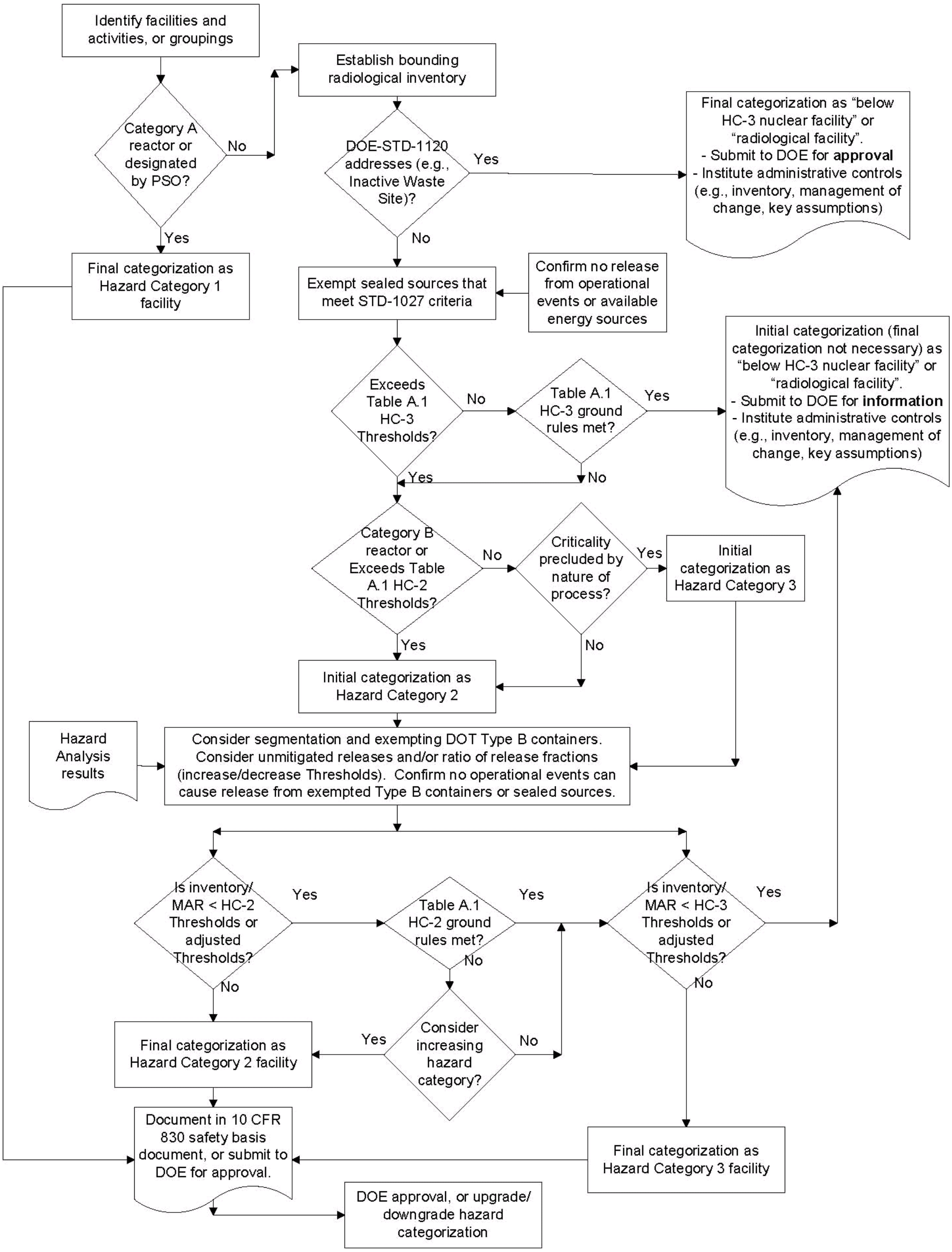

FIG. 5. Hazard categorization process flow [15].

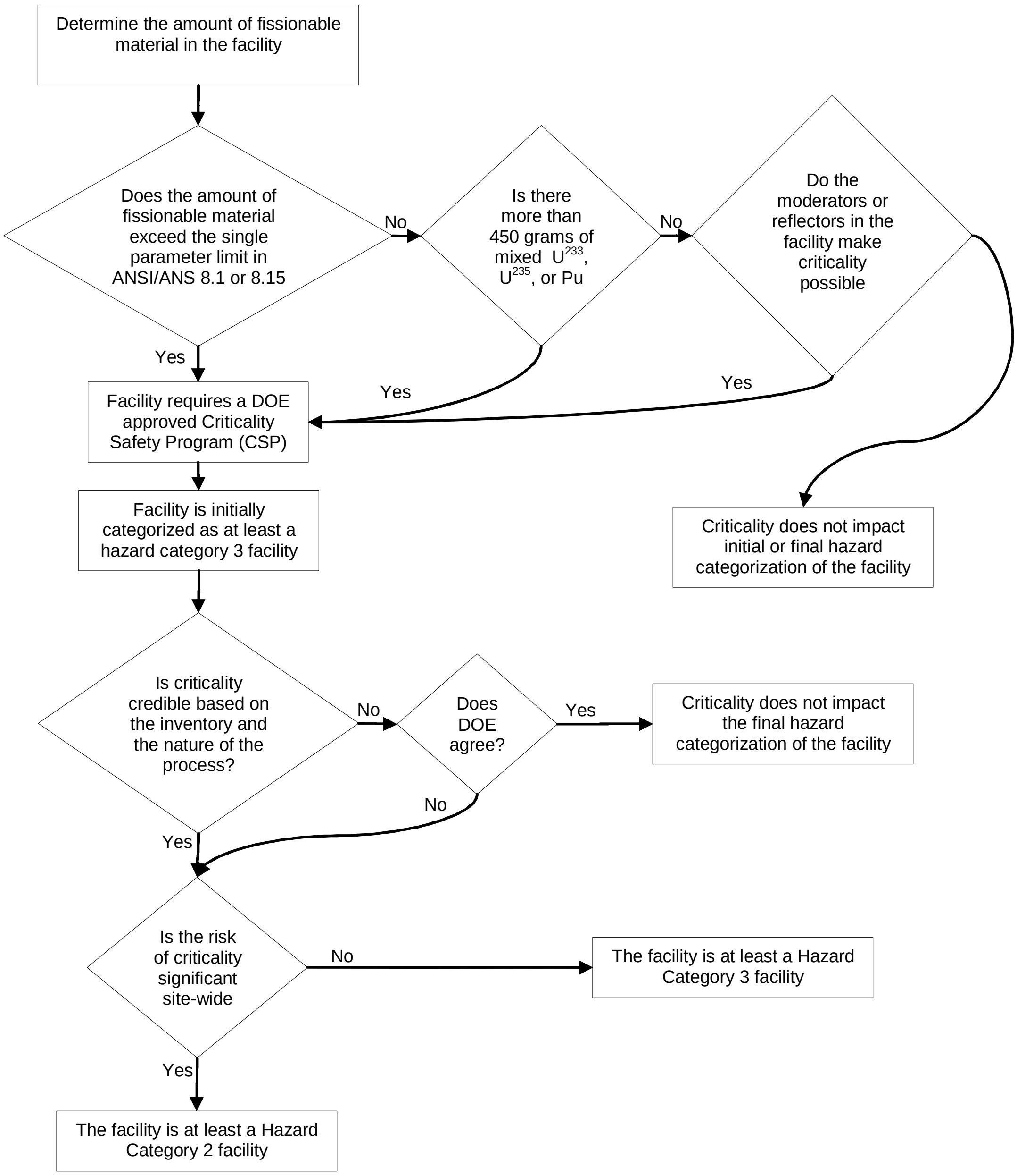

FIG. 6. Criticality hazard categorization process flow [15].

I-3. EXAMPLE FOR THE GRADED APPROACH IN REGULATIONS CONCERNING SAFETY ASSESSMENT FOR NORM FACILITIES (THE NETHERLANDS)

The Dutch Nuclear Energy Act specifies that it is prohibited to store, apply, transport or dispose of materials specified by the Dutch authorities as radioactive materials without authorization. The Dutch Radiation Protection Decree ("Besluit Stralingsbescherming") [6] specifies when materials have to be treated as radioactive materials by the definition of exemption and clearance levels. In addition to regulations with regard to practices, the *Besluit Stralingsbescherming* defines when work activities, activities dealing with naturally occurring radioactive material (NORM), fall under reporting or authorization requirements. More detailed regulation regarding work activities and standard forms for reporting or license applications are given in the ministerial guideline mr-NABIS [7]. The legislation is applicable to all operations dealing with NORM including decommissioning.

Studies performed on behalf of the Dutch authorities have identified which industries in the Netherlands use materials or processes that may cause enhanced exposure of workers or members of the public due to the presence of NORM. These studies showed that, except for exposure of aircraft personnel to cosmic radiation, most exposure situations in the Netherlands are related to the processing of mineral sands and large amounts of ores, and to the oil and gas production process, which, due to the production or extraction process applied, may lead to waste with enhanced concentrations of natural radionuclides. An inventory of relevant industries is given in mr-NABIS [7].

In the studies, doses were calculated for a large number of activities based upon scenarios for normal and unfavourable but possible conditions. The results of the various studies have led to a system of exemption/clearance, reporting and authorization of activities, as specified in the Dutch Radiation Protection decree, based on a single set of exemption and clearance levels for each radionuclide so there is no numerical difference between these two concepts. Industries or work activities identified in mr-NABIS are obliged to investigate if they process or produce materials on their premises that exceed the exemption levels specified in *Besluit Stralingsbescherming [6]*.

All work activities must be reported or authorized when both the total activity and the activity concentration exceed the radionuclide specific exemption/clearance levels specified in the *Besluit Stralingsbescherming*. This regulation contains a single set of total activity and activity concentration levels. These values apply to exemption as well as clearance. For discharges in water or air exemption levels have been calculated based on generic scenarios and a dose criterion for members of the public of 10 µSv/a.

The system of exemption and clearance, reporting and authorization of activities, as described in the Dutch Radiation Protection Decree, can be summarized as shown in the following Table 4:

TABLE 4 SYSTEM OF EXEMPTION/CLEARANCE, REPORTING AND AUTHORIZATION OF WORK ACTIVITIES IN THE DUTCH RADIATION PROTECTION DECREE

Case	Resulting action
Total activity < EL/CL value	Exemption/Clearance
Total activity ≥ EL/CL value	
• Concentration < EL/CL value	Exemption/Clearance
• Concentration ≥ EL/CL value	Reporting
• Concentration ≥ 10 times EL/CL value	Authorization
Total discharged activity	
• < EL/CL value	Exemption/Clearance
• ≥ EL/CL value	Authorization

EL = exemption level; CL = clearance level (as specified in the Dutch regulations)

The requirement for reporting or authorization has a number of implications. Reporting requirements lead to a less stringent regulatory process framework then authorization (a graded approach). Once a work activity has been reported only general rules apply, while in the case of authorization a licence is granted in which specific requirements are given with regard to e.g. administration, dose registration.

A summary of these data showing compliance with the rules specified in the legislation and license has to be sent to the authorities at the end of each year.

Guidelines and methods for risk assessments are given in another ministerial guideline, the mr-AGIS. The mr-AGIS also refers to supporting documents for more complicated situations: DOVIS-A and DOVIS-B. The risk assessment for a work activity falling under reporting requirements usually can be performed using conservative rough estimates based on e.g. extrapolation of dose rate measurements or gamma dose constants. Only if specific dose limits are exceeded, more detailed calculations are required. Once an activity has been reported, yearly registration of the doses for workers is not required.

For authorized or licensed work activities, more specific data are usually necessary which also show a graded approach. The calculation of the dose to the public by external radiation can be done by conservative but realistic estimation in the first instance. If the result of this rough estimate is less than 10 µSv/a, a more precise assessment is not required (e.g. using a computer code such as Microshield). Also, no stringent requirements are prescribed with regard to the implementation of ALARA. If the total activity discharged is lower than the exemption values for discharges specified in *Besluit Stralingsbescherming*, the discharges are exempted and no site-specific risk estimate is required. Discharges above the exemption level fall under authorization. In this case, a site-specific risk estimate is required.

NORM waste falling under reporting requirement can be sent to special landfill sites and is stored under conditions equal for non-radioactive hazardous waste, while radioactive waste falling under authorization needs to be sent to the central organization for storage of radioactive waste, COVRA. The difference in storage methods and costs of both options is considerable. Reuse is allowed for material falling under reporting as well as authorization requirements. Due to the administrative consequences of the stringent regulations and the public perception of radioactivity, the reuse options for this relatively small quantity of material are however limited in practice.

I-4. IDENTIFICATION OF REQUIREMENTS AND CRITERIA FOR THE SAFETY ASSESSMENT (CUBA)

No national regulation specifically addressed to decommissioning activities exists in the country. The main requirements for decommissioning of facilities and for conduct of safety assessment are outlined in the National Basic Safety Standards and other regulations such as the “Regulation for the Authorization of Practices associated with the use of Nuclear Energy” (Resolution 25/98).

- *Safety assessment aspects in the National Basic Safety Standards:*

According to the National Basic Safety Standards (from January 2002, which is based on the IAEA Basic Safety Standards [12]), it is stated:

– (Chapter III, Section IV, Paragraph 20) “Safety assessments related to protection and safety measures for sources within practices shall be made at different stages, including sitting, design, manufacture, construction, assembly, commissioning, operation, maintenance and decommissioning, as appropriate, in order:

 (a) To identify the ways in which normal exposures and potential exposures could be incurred, account being taken of the effect of events external to the sources as well as events directly involving the sources and their associated equipment;

(b) To determine the expected magnitudes of normal exposures and, to the extent reasonable and practicable, to estimate the probabilities and the magnitudes of potential exposures; and

(c) To assess the quality and extent of the protection and safety provisions.”

– (Chapter V, Section I, Paragraph 157 “The safety assessment shall include, as appropriate, a systematic critical review of:

(a) The nature and magnitude of potential exposures and the likelihood of their occurrence;

(b) The limits and technical conditions for operation of the source;

(c) The ways in which structures, systems, components and procedures related to protection or safety might fail, singly or in combination, or otherwise lead to potential exposures, and the consequences of such failures;

(d) The ways in which changes in the environment could affect protection or safety;

(e) The ways in which operating procedures related to protection or safety might be erroneous, and the consequences of such errors; and

(f) The protection and safety implications of any proposed modifications. ”

- *Safety assessment according to the Resolution 25/98:*

The general requirements for safety assessment are outlined in the “Regulation for the Authorization of Practices associated with the use of Nuclear Energy” (Resolution 25/98). The operators of facilities of 1st and 2nd categories, when applying for a decommissioning licence need to prepare the following documentation: (i) plan for the termination of practice; (ii) decommissioning plan; and (iii) radiation safety manual and submit it to the Regulatory Body for approval. The assessment of the initial radiological situation (type of contamination, the exposure and contamination levels, etc.), the proposed schedule for termination of practice, the assessment of the decommissioning options (strategies) and the assessment of the final radiological situation in the facility need to be included in the decommissioning plan for the termination of practices. The selected strategies for decommissioning, including the decontamination processes and dismantling techniques, estimation of types and volumes of radioactive waste to be generated, as well as the measures for reduction of occupational doses must be described in the decommissioning plan.

Additional to these specific requirements for the decommissioning activities, for any operational license the operator needs to submit a Safety Report to the Regulatory Body for approval. The content of the Safety Report is described in the Annex No. 2 of the Resolution 25/98. Regarding safety assessment, the Safety Report must contain a description and assessment of the response to postulated initiating events such as: malfunctioning or equipment failures, common cause failures, human mistakes, external events which could entail accidental events. This analysis could be extended to a combination of these failures, mistakes or events. The results of this analysis needs to be expressed when possible, in terms of occurrence likelihood of the sequential accident, the magnitude of the damage of the barriers between the radiation source and the workers and/or the public and the magnitude of the doses they could receive.

- *Radiological criteria for a particular decommissioning project:*

The National Institute of Oncology and Radiobiology (INOR) was one of the pioneers in the use of radioactive material in medicine in Cuba. The brachytherapy services have been provided since the forties. The first brachytherapy facility was located in section A. Later on, the brachytherapy service was moved to another section within the hospital and the former facility was then used as temporary storage facility for disused sealed sources. One or more Cs-137 sources stored there were leaking, causing a radioactive contamination in the area. Different dismantling and decontamination activities were carried out in the facility between 1988 and 1999. But for different reasons, the requirements established by the Regulatory Body for decommissioning could not be achieved. Then the selection of appropriate radiological and clearance criteria played an important role in the definition of decommissioning strategy and the final release of the facility from regulatory control.

For this particular case, the radiological criteria proposed for clearance in the decommissioning plan considered that the annual dose received by members of the public (after the facility is release from regulatory control for non-nuclear use) must not exceed 0.3 mSv above the natural background, in the worst case scenario. Following this criterion, operational reference levels in term of dose rate and specific activity were derived and used during decommissioning:

— Dose rate: Dose rate at 10 cm from any surface (walls, floors and roofs) must not exceed 0.1 μSv/h above the natural background. By considering this exposure condition, an occupancy factor 2/3 and a two meter radius plane source, the annual effective dose estimated was 0.22 mSv. This result was obtained as an example for a 2 meter radius room. For smaller surfaces the dose rates would be lower.

— Activity concentration: Specific activity in the soil (in the garden and floor filling materials) must not exceed 1 Bq/g. The annual dose was estimated considering a two meter radius area filled with typical soil (density 1.6 g/cm^3), activity concentration 1 Bq/g and an occupancy factor 2/3. The maximum annual effective dose was 0.24 mSv. It was considered that the Cs-137 concentration was 1 Bq/g in all the profile of the soil. It was an overestimation because the distribution coefficient (Kd) is high and consequently the activity concentration needs to decrease in depth. The occupancy factor, the dimensions of the contaminated area, as well as the depth of contaminated soil were estimated, taking into account conservative assumptions. It is expected that the annual doses would be lower than the calculated.

These radiological criteria were approved by the Regulatory Body in the decommissioning licence.

Appendix II

EXAMPLES FOR THE GRADED APPROACH IN THE RADIOLOGICAL CHARACTERIZATION OF A FACILITY

II-1. ASSESSMENT OF THE CONTAMINATION OF THE REACTOR AND AUXILIARY SYSTEMS AT THE CAORSO NUCLEAR NPP (ITALY)

In the framework of the activities carried out to plan the decommissioning of Caorso NPP (an AMN-GETSCO direct cycle Boiling Water Reactor with an output of 860 MWe, shut down in 1986 after an operating period of about 8 years), a campaign for a preliminary estimation of the radiological contamination deposited on the equipment of reactor drywell, auxiliary and turbine buildings was performed.

While contamination of vessel, pools and turbine systems was estimated using either direct measurement performed inside the relevant components (turbine piping and equipments), or historical data (vessel and pools), as regards the other reactor and auxiliary systems, the assessment of radionuclide contamination, deposited on the inner surfaces of piping and equipments, involved:

- The direct measurement of dose rate in contact with the components;
- The application of suitable *conversion factors* between dose rate and surface β/γ contamination, properly derived by experimental procedures; and
- The application of *scaling factors*, in order to estimate the total (α, β, β-γ, X) activity of deposited contamination.

By such an “indirect” approach, a reliable estimate of the total activity distributed in the reactor and auxiliary buildings was obtained with significant savings in time, money and occupational exposure of workers. The operative criteria and the adopted methodologies are detailed below.

General methodology

A number of dose rate measurements were directly performed on piping by using GM-detectors, with a spatial frequency distribution reflecting dose variability of the systems examined (the role of radiation background variability was taken into account as well). Particular attention was paid to assess the “circumferential” average exposure intensity, related to any measurement point located on horizontal pipes. Pipes with diameter smaller than 2 inches were not considered in the survey, and valves were considered as pipe segments. Concerning other equipments (e.g. heat exchangers, filters), measurements were done on the component itself and on the entering pipes.

On the basis of such collected data, the radiometrically homogeneous portions of the systems considered in the survey (pipes and components) were accurately defined, assuming that analogue homogeneous conditions could be referred to the deposited contamination as well. Then, the inner surfaces related to each homogeneous portion were calculated.

For each homogeneous portion of pipes, the specific surface β-γ-contamination was determined multiplying the related mean exposure intensity (derived from averaging the relevant survey measurement results) by a specific conversion factor calculated taking into account the pipe design features and the measurement distance. Regarding the measurement distance, it was determined by considering the detector head dimensions and the average pipe insulator thickness.

For a generic j^{th} system, including the i = 1, …, n homogenous piping portions, the following expression, for total β-γ-activity, was used:

$$A_j(kBq) = \sum_{i=1}^{n} S_i(m^2) \cdot E_{mi}(\mu Sv \cdot h^{-1}) \cdot F_{ci}(kBq \cdot m^{-2}/\mu Sv \cdot h^{-1}) \qquad \text{(II.1.)}$$

where:

A_j	indicates the total β-γ activity deposited on the j^{th} system;
n	indicates the number of homogenous pipe portions included in the j^{th} system;
S_i	indicates the inner surface of the i^{th} homogenous pipe portion;
E_{mi}	indicates the mean exposure intensity related to the i^{th} homogenous pipe portion;
F_{ci}	indicates the conversion factor of the i^{th} homogenous pipe portion, as a function of pipe design and measurement distance.

For equipment, the specific β-γ-contamination related to the entering pipes, and determined by the same approach indicated above, was assigned to the entire surface of the component. Then, the total β-γ activity deposited on the component was inferred by calculating its effective surface in contact with the contaminant fluids.

The conversion factors between the exposure intensity measured on piping and the specific surface β-γ-activity, were calculated according to the experimental methodology outlined below.

Conversion factors calculation

In order to determine the conversion factors, four pipes with a length of 20 cm and a diameter of, respectively, 2, 4, 8 and 12 inches were "traced" by a known quantity of radioactivity (such diameter values are representative of piping of the considered systems of the facility). More specifically, in order to model a homogenous distribution of crud on the inner surface of each pipe, an annular cavity, with a width of about 2 ÷ 3 mm, was obtained and filled with a solution containing a known activity of Co-60 (Co-60 was the main γ-emitter in the contaminated crud of piping and components of primary systems). Performing several dose rate measurements, at distances from the source terms of the same order of magnitude of typical piping insulator thicknesses[4], it was possible to collect a set of distance-depending conversion factors for each pipe typology. Also, experimental evidence confirmed that, for calculation purposes, the presence of water filling the pipes had no significant relevance.

The results of the experimental evaluations are summarized in Fig. 7.

Total activity assessment

Once the total deposited β-γ contamination was determined, the total α, β, β-γ, X activity was assessed by using proper scaling factors relative to Co-60, derived from analyses of suitable crud samples taken from the examined systems.

4) Insulator density has no relevance for measurement results.

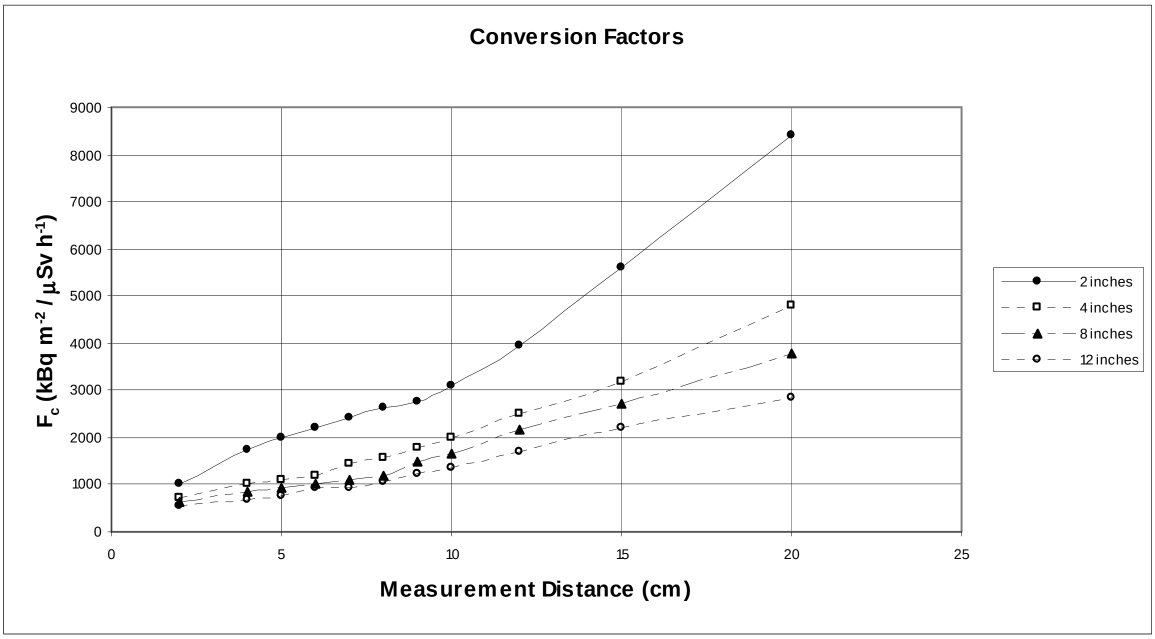

FIG. 7. Comparison of conversion factors calculated for different pipe typologies and measurement distances.

Radiochemical analyses, performed on representative samples of the examined systems, have subsequently confirmed a good correlation between the predicted and actual values of radiological contamination.

Among the Reactor Building systems (where most of the activity in the facility has been deposited due to radioactive contamination) the most contaminated one turned out to be the “Reactor Water Clean-Up System” (RWCS) responsible of the 52% of the total amount of contamination activity deposited on the Reactor Building systems (except for vessel and pools). The estimated total contamination activity of RWCS can be summarized as follows (activity data refer to Feb. 1992):

Piping:	1.42×10^{8} kBq
Equipments:	4.11×10^{8} kBq
Total:	5.53×10^{8} kBq

The reactor vessel provides the most significant contribution to the whole contamination, and was estimated to be equal to about 2.20×10^{10} kBq (as of Feb. 1992).

Conclusions

The most time consuming and exposure-demanding activity of the radiological characterization campaign was the Caorso NPP was devoted to the plant systems dose rate mapping. The total collective dose incurred by workers, during a period of 7 months (1 400 man-hours), was of about 4 100 µSv·man, as a substantial optimization of radiation protection was obtained by using detectors with telescopic probes and by an appropriate work planning.

The preliminary assessment of the contamination present on the systems of Caorso NPP has represented a fundamental step for the accurate planning of successive decommissioning activities.

From this point of view, the imparted collective dose due to the radiometric mapping of the various systems of the plant can be considered adequately justified by the advantages deriving from improving the knowledge of the radiological status of systems and components.

II-2. COMPUTATIONAL MODEL FOR THE ASSESSMENT OF CONTAMINATION ON THE PRIMARY CIRCUIT OF THE TRINO NPP (ITALY)

The radiological characterization prior decommissioning of pipes and equipment in shutdown nuclear reactors represents a major effort in the general framework of the activities preceding removal and dismantling operations.

An example of "graded approach" in radiological characterization of NPPs can be the use of computer codes for the assessment of contamination on components and piping. This method needs a limited number of sample analyses for validation of the output results, and can be usefully applied to normally operated pressurized water reactors (PWRs).

Radiological characterization activities at Trino NPP (a Westinghouse four loops Italian Pressurized Water Reactor with an output of 270 MWe, shut down in 1986 after an operating period of 23 years) have recently involved a calculation model in order to assess the radiological contamination present in the primary circuit of the plant.

The model, the LLWAA-DECOM code (Low Level Waste Activity Assessment - Decommissioning) developed by Tractebel, is designed to enable the assessment of activities of critical radionuclides deposited on the equipment of primary and auxiliary systems.

Originally implemented to support the decommissioning of Belgian Doel and Tihange PWRs, the LLWAA-DECOM code has been applied to derive a radiological profile of the most representative equipment of Trino primary loops, i.e. the hot and cold legs, the re-circulation pumps, the four steam generators, and the pressurizer. These results are aimed to provide a useful reference to successive experimental investigations, and to contribute to the optimization of the whole radiological characterization process.

The LLWAA-DECOM code is site-specific as it takes into account the design characteristics and operating conditions of the reactor of the site; in particular, the calculation requires a proper preliminary modelling of the Reactor Coolant Systems and Purification Systems. Then, using suitable mathematical algorithms, on the basis of a number of specific input data and general parameters, the code is able:

— To model the formation, activation, migration and disposal of activated corrosion products (Table 5) on primary loops, by considering the most important physical and chemical phenomena occurring in the reactor coolant and in relevant operating systems;

— To model, in normal operation conditions, the release, migration and disposal of fission products and transuranic radionuclides (Table II.2.1), deriving from fuel failures and cladding contamination, along the primary circuit; and

— To estimate the radionuclide activity deposited, in steady-state conditions, on the primary piping and components in contact with the contaminated coolant fluids.

Input data and parameters, to be considered in plant modelling and calculation implementation, include:

— Characteristics of the equipment and components of the primary circuit (materials, constructive and geometrical data, pressure vessel characteristics, coolant data, etc.);

— Characteristics of the core assembly and fuel elements (number of fuel elements, fuel materials, fuel cladding characteristics, fuel radionuclide inventory, etc.);

— Operating conditions (average burn-up, temperature, average fluid velocity, coolant pH and chlorine concentration, number of operating cycles, CVCS flow, etc.);

— The corrosion product characteristics (particle density, particle diameter distribution, etc.);

— Physical and chemical characteristics of the isotopes (decay rate λ_i, solubility product constant, etc.);

— Characteristics of decontamination processes possibly performed (decontamination factors, etc.); and

— Time elapsed between the reactor final shutdown and the decontamination or dismantling.

One significant benefit of the code results from it enabling to easily perform sensitivity analyses to assess the effect, on the radionuclide inventories, due e.g. to the change of operating procedures or operating conditions and to uncertainties related to some model parameters.

TABLE 5. LIST OF RADIONUCLIDES TAKEN INTO ACCOUNT BY LLWAA-DECOM CODE

Activation Products	C-14, Cl-36, Co-58, Co-60, Fe-55, Ni-59, Ni-63, Nb-94, H-3
Fission Products	Sr-90, Tc-99, I-129, Cs-134, Cs-137, U-234, U-235, Np-237, U-238, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, Am-241, Am-234, Cm-244

Among the radionuclides considered in contamination assessment by LLWAA-DECOM, notably, some “hard-to-measure” long-life isotopes, such as C-14, Cl-36, Tc-99, I-129 and Pu-239, can be automatically addressed by the code (suitable scaling factors for reactor coolant and primary wastes can be determined as well). This is of particular interest from the perspective of radioactive waste disposal, as the above mentioned radionuclides represent a major concern in final repository assessment.

Validation of output results of the model will require a comparison between calculated activities of deposited radionuclide and direct measurements performed on plant components, in order to refine code “calibration”. These measurements can involve either sample collection and analyses, and/or dose rate determinations. Code estimates validation needs generally a number of sample analyses remarkably smaller than a complete experimental characterization of the plant primary circuit usually requires. So, such a semi-empirical approach may result in significant savings in time, money and occupational exposures of workers.

II-3. GRADED APPROACH CONCERNING THE NUMBER OF SAMPLES FROM THE RESEARCH REACTOR IN SOFIA (BULGARIA)

- *Short history of IRT–2000*

The research reactor IRT, a pool-type, light-water cooled and moderated reactor, is located 8 km east of the center of Sofia, the capital of Bulgaria. It was constructed in the period 1959 - 1961 by the Kurchatov Institute in Moscow. First criticality was reached 1961 with nominal power 1 MW, followed by upgrades to 1.5 MW in 1965 and to 2 MW in 1970. The reactor was permanently shut down in 1989. Up to 48 fuel and graphite assemblies were in the core with 14, 15, or 16 fuel rods in the assembly. Fuel rods were of EK-10 type (10% enrichment) or C-36 (36% enrichment). The reflector consisted of 13 graphite blocks. There were 11 horizontal and 12 vertical experimental channels, with a maximum neutron flux of 2×10^{13} $n/cm^2 \cdot s$ at 2 MW thermal power.

The decommissioning strategy for IRT-2000 is a partial dismantling prior to its reconstruction into a low power reactor, with an intention to re-use the concrete biological shield for the new low-power research reactor.

Removal of the reactor core and replacement of old equipment will not pose any significant problems for dismantling. Many of the activities are within the scope of what would be termed maintenance, which are usual during power upgrading of pool type reactors.

The old data from the radiological characterization through measurements and calculations made in 1985 – 1986 are updated in the “Plan for partial dismantling” according requirements in the “Technical Assignment for Partial dismantling” developed by the Institute of Nuclear Research and Nuclear Energy, INRNE (the operator) specialists according the new regulatory requirements.

The radiological characterization programme includes the following steps:

— Review of historical information;
— Calculation methods implementation;
— Sampling and analyses plan preparation;
— Measurements sampling and analyses performance; and
— Review, evaluation and comparison of data obtained.

- *Performance of sampling, measurements and analyses*

The implementation of the sampling and analyses plan includes measurements before draining of the water of the first cooling circle and measurements, taking wipe tests and samples after removal of the water (see Fig. 8.).

It has been considered by the operator to use a statistical approach with a goal to reduce the number of the wipe tests and sampling and to reach a maximum effectiveness in performed measurements and analysis. The purpose was to determine the type of measurements, the necessity of taking smears and sampling and their position. In this case it has been considered not necessary to perform a characterization of the whole facility, because it will not be final decommissioning of the research reactor. It has been decided to do characterization of the equipment and systems, which will be replaced with new one, according the Technical Project for maintenance of the IRT-Sofia into a low power reactor. A complete list of parts and equipment of the IRT-2000 reactor, which are to be dismantled, was prepared according to a time schedule. The list includes:

- Reactor core according to each component part;
- Water protective shield;
- Fuel assembly holder;
- Ejector;
- Console for carrying fuel assemblies;
- Thermal column;
- Lead plate;
- CMD fixture and cables;
- Pipelines of first cooling loop (force and suction pipeline);
- Vertical experimental channels (11 pieces);
- Control rods channels (7 pieces);
- Channels of the fixed ionization chambers (5 pieces);
- Fixed ionizing chambers (5 pieces);
- Drives of movable ionization chambers with ionization chambers (4 pieces);
- Electric motors of control rods and EP (5 pieces);
- Upper plate of platform (above the reactor core);
- Horizontal channels (11 pieces);
- First cooling loop circulation pumps (3 pieces);
- Circulation pumps motors of first cooling loop (3 pieces);
- Heat exchangers (2 pieces); and
- Pipelines and fixture in the first loop room.

The fact that the new reactor vessel will be built into the old vessel has been taken into account.

FIG. 8. Sampling during radiological characterization at the Sofia research reactor.

II-4. A SEMI-EMPIRICAL MODEL FOR ASSESSMENT OF HARD-TO-DETECT RADIONUCLIDE LEVELS AND THEIR SIGNIFICANCE IN DECOMMISSIONING WASTE FROM ACCIDENTALLY SHUT DOWN NPP BOHUNICE A1 (SLOVAK REPUBLIC)

A-1 NPP at Bohunice was a prototype NPP of HWGCR type (heavy water moderated, gas cooled) with channel type reactor KS 150 and installed power 143 MWe. This NPP was shut down in 1977 (after 5 years of operation) after an accident related to a primary circuit integrity failure. Significant fuel cladding damage during the accident and later cladding corrosion during spent fuel storage at the A-1 NPP resulted in contamination of the NPP construction surfaces by radionuclides of composition specific for this facility, e.g. elevated levels of actinides. It is known that the direct determination of hard-to-detect radionuclide (HD-RN, RN emitting only alpha or beta particles or gamma radiation in the presence of RN with higher energy of radiation) in the waste to be stored or material to be cleared is too expensive, and therefore unsuitable for routine control. Moreover, considering the A-1 NPP shutdown after primary circuit integrity failure, the HD-RN in radioactive waste might exhibit specific features characteristic only for this type of reactor [16]. This is why to characterize HD-RN it is necessary to use as much empirical data for the facility studied as possible.

In such situation, one of the possible solutions for characterization and evaluation of the significance of the HD-RN content at the A-1 NPP is to apply the combined theoretical-empirical approach utilizing a calculated radionuclide inventory in spent fuel and a developed model with effective empirical release coefficients (ERC) relative to relative Cs-137, describing the released fraction of HD-RN from the spent fuel [17].

- *Effective release coefficients*

The ERC are describing the release into a particular form of radioactive waste (e.g. the liquid phase) and contrary to total release coefficients (RC) are depending also on physico-chemical conditions and on particular distribution processes between liquid and solid phase in the system. On the other hand the RC are influenced mainly by the release mechanism from the spent fuel and its velocity, e.g. for the HD-RN released from the intergranular space this factor is determined by the dissolution of base material velocity which contains HD-RN generated in this area.

The model assumes that the ERC for the HD-RN belonging to the same release group are equal. Three to four types of release mechanisms are known and according to the literature [18] on modelling of radiological consequences of highly radioactive waste geological disposal, the relevant long-lived radioisotopes can be divided into the following release groups: Release group from:

(a) The gap between the matrix and the fuel cladding (inter-fuel gap release) - isotopes of Cs, I, C, Tc and Se;

(b) The grain boundary space - Sr, Nb, Pd and Sn (noble metals);

(c) Spent fuel element (SFE) constructional materials - Ni, Co, Mo and Ca (with the similar dissolutionvelocity as at the previous group); and

(d) The fuel matrix – transuranium elements (Pu, Am, Cm), Sm and Zr.

As release markers and in the same time the representatives of these release groups were according to the available empirical data selected Cs-137, Sr-90 and Am-241 (or Pu-239 and Pu-240) respectively, while Sr-90 was shown to be suitable as release marker from the fuel assembly also for constructional material, as well (mainly Ni-59 and Ni-63 that arise in NPP A1 from SFE and not from primary circuit material (carbon steel) corrosion/activation like in the case of Co-60).

- *Evaluation of historical and newly acquired empirical data*

In this respect all available empirical data from radiochemical analyses for A-1 NPP operational waste system has been evaluated and used to determine the RC for the above-mentioned release markers and the operational radioactive waste system (concentrates and sludges from the operational system in object 41 and 44/10, which as a whole can be considered as a one- compartment system). In two compartments model, sorption properties of HD-RN and their redistribution between solid and liquid phase can be accounted for by the available values of effective distribution coefficients K_d [18].

The main idea of the current evaluation was to show that for the most of the HD-RN, which have to be declared for the purpose of waste disposal or radioactive material clearance, their contribution to the total dose limit is negligibly small compared with relatively easily measured dominant radionuclide Cs-137. Moreover, this evaluation enables to identify those HD-RN, on which the main attention needs to be focused to determine their content by the expensive radiochemical analyses so as to develop a sufficiently large database of relevant release and scaling coefficients.

A disadvantage of the proposed method of effective release coefficients is that the conclusions of assessment are limited to particular partial systems of radioactive waste (waste streams) and the time interval for which the database of specified empirical data is available (needed for determination of corresponding release and consecutively scaling coefficients for individual HD-RNs).

Consequently part of the work in this field has been devoted to the acquisition of new empirical data on relative HD-RN content (normalized to the Cs-137 content) for the important streams of waste and materials designated to disposal or clearance into the environment: gravel bed from the waste water tank of object 44/10, decontaminated inner surfaces of steel pipes, concrete debris (a floor) from dismantling machinery at A-1 NPP machine hall, different types of concrete stored in object 44/20, glass wool from pipe insulation, dismantled cables insulation. Table 6 presents selected results on relative content of the most significant HD-RN of which some are also damaged spent fuel release markers (VUJE research report [19]). It can be seen from Table 6 that the abundance of ^{90}Sr and transuranium elements (TRU) in concrete samples from object 44/20 most resembles that of balanced liquid waste (B-KRAO) or the sludge from the tank of object 41/10. On the other hand, the elevated relative contents of transuranics and Sr-90 in concrete debris from the turbine hall is probably influenced by the rust from the steel pipes stored and segmented before decontamination in this hall.

TABLE 6 AVERAGE RELATIVE CONTENT OF SELECTED HD-RN IN STUDIED RADIOACTIVE WASTE AND MATERIALS FROM THE DECOMMISSIONING OF NPP A1

Type of sample	No. of samples	Sr-90/ Cs-137	Pu-239, Pu240/Cs-137	Pu-238/Cs-137	Am-241/ Cs-137	Tc-99/ Cs-137
Concrete debris obj. 44/20, 2003	7/33*	0.084 (0.03-0.18)	0.0033 (0.0012-0.006)	3.0×10^{-4} (5×10^{-5} – 8×10^{-4})	0.0040 0.002-0.007)	-
Concrete debris turbine hall, 2002	10/9*	0.23 (0.13 – 0.32)	0.025 (0.01 – 0.062)	0.0053 (0.0013 – 0.014)	0.024 (0.01-0.054)	0.07 (0.05 – 0.09)
Rust, inner side of steel pipes, 2001	13	0.29	0.049	-	0.040	0.11**
Sludge, obj 41 a 44, 1990	8	0.015	0.0016	0.00015	0.0013	-
B-KRAO***	16	0.015	0.0016	0.00015	0.0013	3×10^{-5}

* The second number (/x) means number of samples analyzed for Am-241 by gamma spectrometry.
** Only 3 samples were radiochemically analyzed and seemed to be overestimated.
***B-KRAO - balanced reconstructed contaminated liquid radioactive waste in the radioactive waste system.

- *Radiological importance of particular HD-RN in operational radioactive waste system of NPP A1*

On the basis of the proposed semi-empirical approach and the database of available data on HD-RN, the radiological importance (relative dose contribution according to concentration limits of the national radioactive waste repository Mochovce) of 18 prescribed HD-RN in the A1NPP operational radioactive waste system has been evaluated. All historically available data from A1NPP acquired in 1992-93 have been used [17]. The results of evaluation are summarized in Table 7. It shows that only a few HD-RN are important from the point of view of their contribution to the dose limit. Critical HD-RN for A1NPP concentrate with the same order of importance are I-129 , Tc-99 and Cs-137. Such critical HD-RN for the sludges are Pu-239, Pu-240 and Am-241 and their contribution to dose exceeds that of Cs-137 by 2 orders of magnitude. The contributions of the other HD-RN in both cases are even less important, i.e. by 2 or more orders of magnitude.

In case of reconstructed operational liquid radioactive waste (KRAO or LRW) the significance of HD-RN is similar to that of sludges. For the comparison the relative dose contribution for RLW of the spent fuel elements storage system (*chrompik* and *dowtherm*), where the only significant contributor to the dose limit was identified Cs-137, has been evaluated according to available historic data. All the rest HD-RN in this system are less significant by about one order of magnitude.

- *The inner surfaces of contaminated steel pipes*

According to measured data it was found that relative content of Sr-90 and Pu-239 and Pu-240 relative contents (related to Cs-137) exceeds the reconstructed ones of source liquid waste. This increase (the concentration) probably occurred gradually by natural deposition processes in a CO_2 cooling system during the system operation. On the basis of estimated release coefficients and conservatively estimated concentration coefficients, the dose contribution of the other HD-RN was found to be negligible.

- *Release of concrete debris*

The content of HD-RN in this material stream is considered to be similar to concentrate or to reconstructed KRAO. On the basis of the respective release limits it was possible to show, that the only significant contributor to the corresponding dose limits is Cs-137. The content of the rest of HD-RN from the point of view of dose contribution plays by an order less significant role. The HD-RN contributions for the purpose of declaration were estimated according to the proposed method of effective release coefficients.

Conclusions

The main conclusion of the study is that the proposed semi-empirical method enabled an effective evaluation of radiological significance of HD-RN as well as their content declaration in released waste streams. Moreover, such an evaluation enables to identify those HD-RN on which the main emphasis needs to be focused in determination of their contents by radiochemical analysis so as to develop statistically sufficiently large database of the relevant SFE-release and scaling coefficients. The proposed method so far has been applied on declaration of HD-RN content at the disposal of the processed radioactive waste from the pebble bed, sludge from the underground tanks in object 44/20 as well as the release into the environment the decontaminated steel pipes and construction debris from the turbine hall of A1NPP.

TABLE 7 RELATIVE DOSE CONTRIBUTION VALUES (LIM $FR_{I,CS}$ IN UNITS OF REFERENCE Cs-137 DOSE) FOR SELECTED HD-RN WITH AVAILABLE EMPIRICAL DATA FOR THE MAIN TYPES OF WASTE AT A1NPP

HD-RN	Lim_{Cs}/Lim_i	KC-44 average $Lim\_fr_{i,Cs}$	Sludge liquor, 41 $Lim\_fr_{i,Cs}$	Sludge obj.44a41 $Lim\_fr_{i,Cs}$	Rek-WWa* 44a41 $Lim\_fr_{i,Cs}$	Chrompik** KS-2 $Lim\_fr_{i,Cs}$	DS** water $Lim\_fr_{i,Cs}$
Sr-90	2.0	2.0×10^{-2}	1.2×10^{-1}	2.9×10^{-2}	1.5×10^{-2}	3.0×10^{-4}	3.9×10^{-4}
Pu-239, Pu-240	8.8×10^{4}	1.7×10^{-1}	1.3×10^{1}	1.4×10^{2}	1.4×10^{2}	1.0×10^{-3}	2.1×10^{-2}
Am-241	3.7×10^{4}	1.2×10^{-2}	3.3	5.0×10^{1}	4.8×10^{1}	2.0×10^{-3}	-
Tc-99	4.1×10^{3}	1.4	<	7.1×10^{-2}	1.2×10^{-1}	3.6×10^{-2}	1.5×10^{-3}
Co-60	1.0	9.3×10^{-2}	<	3.6×10^{-3}	7.2×10^{-3}	-	-
I-129	1.7×10^{5}	1.5	1.2×10^{2}	1.4×10^{-2}	8.0×10^{-2}	7.2×10^{-2}	6.2×10^{-1}
C-14	2.1×10^{3}	5.1×10^{-2}	8.2×10^{1}	4.6×10^{-2}	4.7×10^{-2}	1.0×10^{-1}	4.9×10^{-1}
Ni-63	3.6×10^{1}	2.4×10^{-1}	2.0	2.6×10^{-3}	1.2×10^{-2}	-	-
Cs-137	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Cs-137 [Bq/kg]	7.2×10^{9}	2.6×10^{7}	1.5×10^{5}	4.7×10^{9}	3.4×10^{8}	1.3×10^{9}	2.0×10^{8}
Lim_fr Cs-137	1	1.8×10^{-2}	1.0×10^{-2}	3.2×10^{-1}	2.4×10^{1}	1.8×10^{1}	1.4×10^{1}
C/Uncond***	1	5	500	0.5	500	100	500

* Rek WWa - radioactive waste water data reconstructed according to data for concentrate and sludge.
** KS – short term, DS long term SFE store, chrompik – chrom dyoxide solute.
*** C/Uncond - factor between conditioned and unconditioned radioactive waste.

TABLE 8 RELATIVE DOSE CONTRIBUTION VALUES (LIM FRI,CS IN UNITS OF REFERENCE Cs-137 DOSE) FOR THOSE HD-RN FOR WHICH MEASURING DATA ARE UNAVAILABLE

HD-RN	LimCs/Limi RWR**	Lim_fri,Cs KC44	Lim_fri,Cs KC44-wc	Lim_fri,Cs Sludge41&44	Lim_fri,Cs Liq-RW	Lim_fri,Cs ChP KS-2	K_{d-eff}, [l/kg]
Se-79	2.7×10^{4}	2.5×10^{-1}	1.6	1.8×10^{-1}	1.9×10^{-1}	1.9×10^{-1}	248
Nb-94	2.6×10^{5}	1.6×10^{-5}	1.1×10^{-4}	1.4×10^{-5}	1.4×10^{-5}	1.5×10^{-7}	302
Pd-107	1.8×10^{1}	2.2×10^{-6}	9.6×10^{-6}	5.9×10^{-7}	6.6×10^{-7}	6.9×10^{-9}	90
Sn-126	2.5×10^{4}	8.6×10^{-3}	5.5×10^{-2}	5.8×10^{-3}	5.9×10^{-3}	6.2×10^{-5}	224
Ni-63	3.6×10^{1}	3.3×10^{-1}	3.6×10^{-1}	1.2×10^{-3}	1.5×10^{-2}	1.6×10^{-4}	1,2
Mo-93	2.2×10^{2}	2.9×10^{-5}	7.3×10^{-5}	2.6×10^{-6}	3.7×10^{-6}	3.9×10^{-8}	30
Ca-41	2.3×10^{3}	4.6×10^{-5}	8.8×10^{-5}	2.3×10^{-6}	4.1×10^{-6}	<	16
Sm-151	1.9×10^{-1}	2.0×10^{-5}	1.5×10^{-4}	2.6×10^{-5}	2.6×10^{-5}	1.2×10^{-9}	436
Zr-93	2.3×10^{2}	9.9×10^{-4}	4.9×10^{-3}	3.3×10^{-4}	3.6×10^{-4}	1.7×10^{-8}	111
Cs-137	1.0	1.0	1.0	1.0	1.0	1.0	335

* dose fractions Ni-59/ Ni-63= 0.077, Cs-135/Cs-137=0.013
** RWR- republic waste repository, Mochovce

Appendix III

EXAMPLES FOR A GRADED APPROACH IN INITIAL HAZARD CATEGORIZATION

III-1. GRADED APPROACH TOWARDS SAFETY ASSESSMENT IN THE UK

The application of the graded approach is inherent in the regulatory framework found in the UK. The basis of UK law for all aspects of safety is that risks must be made as low as reasonably practicable (ALARP). Within the ALARP concept are two levels of risk that will be of regulatory significance.

- There will be a relatively high level of risk that regulators will declare to be intolerable. If a facility using radioactive material finds itself with a risk that is greater than the level of intolerance, then it must make every reasonably practicable step to reduce this risk.

- There will be a lower level of risk, below which risks are acceptable. Although the legal duty of as low as reasonably practical (ALARP) is still present, regulators will devote less effort to inspection.

- In between these two levels is an area where the operator needs to show that he has reduced the risks from an operation or a facility to be ALARP.

The licence that is granted to a nuclear site to operate is a brief enabling document that lasts until the site is taken out of regulatory control at the end of its lifetime. The licence simply requires the operator to have in place a number of arrangements, e.g. for safety cases, control of operations, maintenance, etc., to the satisfaction of the regulator.

In practice, arrangements are a little more complicated than this. Additional regulatory processes are in force, but these are subordinate to the site licence and are under the control of the Regulatory Body. Thus, the Regulatory Body will publish detailed guidance on how to comply with a site licence and design principles needed to achieve the level of safety that regulators expect. The operator will also be expected as part of their safety management system to set out how they will comply with the site licence requirements. The regulator may formally approve these, which means that the operator does not have the discretion to change them in isolation. Finally, when it comes to the performance of tasks, each one will be subject to a risk assessment. The degree of rigour in the risk assessment and the effort put into safety management will be proportional to the risk.

Thus, one arrives at a situation where in principle every task is being regulated on the basis of the risks it poses. Of course, to evaluate the risk that a task poses so that it can be graded according to its risk needs a safety assessment and the safety assessment must be appropriate to the risk. The way out of this circular argument is that when the operator is proposing to do new tasks, experience and precedent will show the likely level of significance.

The categorization forms used by one UK operator are shown in the following Figure 9. Other UK operators have similar categorization processes and similar categories to grade their safety assessments.

Plant Modification Proposal

PMP number ____________________ ***Revision:*** __________

Section 6 Initial Safety categorisation

The purpose of this section is to categorise the safety significance of the modification with respect to people, plant and environmental performance.

Section 6.1 Radiological Safety Categorisation (to be completed by originator).

Radiological categorisation is based on a balanced subjective judgement of the potential nuclear or radiological consequences of the modification and includes consideration of any impact on radioactive waste and its disposal.

The probability of the faults occurring cannot be taken into account when determining the category.

The judgement of the category should be based on the most significant potential consequences which may arise from realistic fault scenarios, during all stages of the modification.

The potential consequences arising from the proposal being inadequately implemented or misconceived must be considered and should be addressed by asking 'What might go wrong?' and 'What might the consequences be?'

Category D has no (trivial) radiological safety implications
Category C has no more than a minor radiological safety significance
Category B has more than a minor radiological safety significance
Category A may have a major impact on a member of the public

Safety significance **Circle 'Y' for yes or 'N' for no in response to each question**	**Likely categorisation** D	C	B	A
1 Are there any radiological or nuclear safety implications (other than working in the active area etc)? If *yes*, continue below	N	Y	Y	Y
2 Are there planned individual operator exposures (internal plus external) > 15mSv per year?		N	Y	Y
3 Will planned aerial discharges: • Exceed or lead to an increased (or new) plant or stack aerial discharge trigger level (alpha or beta), or		N	Y	Y
• Exceed 0.1% of the current authorised limit for approved places?		N	Y	Y
4 Will planned liquid discharge to sea lead to an increased (or new) plant discharge ceiling (alpha or beta)?		N	Y	Y
5 Is there a realistic fault scenario which could lead to a building (widespread) evacuation?		N	Y	Y
6 Is there a realistic fault scenario in which an Operator could exceed an annual dose limit?		N	Y	Y
7 Is there a realistic fault scenario which could lead to an aerial release exceeding 50 microSv to critical group?		N	Y	Y
8 Is there a realistic fault scenario which could lead to a breach of the site liquid discharge authorisation?		N	Y	Y
9 Are there any significant changes, including additions, necessary to any Operating Rules?		N	Y	Y
10 Are there departures from the original design safety principles or major changes of scope?		N	Y	Y
11 Is there a single fault sequence likely to give a consequence of more than 1milliSv to a member of the public?		N	N	Y

NOTE:
It is possible for a PMP to achieve a lower categorisation despite 'yes' answers to the questions above if adequate justification can be made.

Provisional radiological categorisation ___________

12 Could the modification impact upon the generation or minimisation of radioactive waste (aerial, liquid, solid) or its disposal? (consideration should be given to any impact on plants/processes upstream and downstream). **Yes/No**

If the answer is 'yes' or if the PMP category is A/B then review the Best Practicable Means (BPM) requirements in section 6.1.1.

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FIG. 9. Examples of a categorization forms for hazards screening and proposals for plant modifications

Plant Modification Proposal

PMP number ______________________ *Revision:* ___________

Section 6.1.1 BPM review (to be completed by the Environmental Performance Manager)

Indicate if there is a requirement for a new BPM case as a consequence of the modification.	**Yes/No**
Alternatively	
Review and update existing BPM case in line with SSP 2.01.03 'Management of radioactive waste using best practicable means'.	**Yes/No**
Or where one exists	
Review and update section D10 of the Safety Case (Integrated Environmental Assessment) in line with SSP 1.25.01 'Preparation of Safety Cases'.	**Yes/No**
Identify any document production/update that has been indicated by the BPM review in section 10 of this PMP.	**Yes/No**

BPM review complete (Name) ________________ Signature: ____________________ Date: __________

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FIG. 9 Cont.

Plant Modification Proposal

PMP number ______________________ *Revision:* ___________

Section 6.2 Non Radiological Safety Categorisation (to be completed by the originator)

NOTE:
All questions below must be answered: Where clarification and/or justification is necessary for a decision then use the right hand column. If *all* answers are no then category is E3. If *any* answers are yes, then an environmental Risk Assessment must be provided. The resulting categorisation will be determined on the basis of the environmental assessment carried out.

	Factor	Yes	No	Details (whether yes or no)
1	Could the proposal lead to new or increased discharges of prescribed substances, or a discharge requiring a new authorisation or consent?			
2	Could the proposal lead to an increase in any current authorised/consented discharge or result in 70% of any consent/authorisation or local limit being exceeded?			
3	Could the proposal result in any detrimental effect, or breach of any limit on physiochemical properties such as pH, volume or temperature?			
4	Does the proposal involve modification to or have any effect on plant, equipment or procedures for the control (for example, sentencing, monitoring/alarms or sampling) of discharges?			
5	Could any deviation from normal operations (for example, leaks, increased discharges, Operator error and instrument failure) *fail* to be detected by new or existing control systems in time to prevent significant environmental harm?			
6	Will the chemical process change? (new chemicals used, stored, produced or a change in intermediates).			
7	Is there potential for any incompatible effluents or materials to interact?			
8	Could the modification increase non-radiological waste disposals, or generate wastes for which there are no identified disposal routes?			
9	Could the modification increase the use of energy and/or raw materials (for example, water, fuel, chemicals)?			
10	Could the modification result in increases of noise, odour, detrimental visual impact or other nuisance in the external environment?			

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FIG. 9 Cont.

Plant Modification Proposal

PMP number ____________________ *Revision:* __________

Section 6.2 Non Radiological Safety Categorisation (continued)

	Factor	Yes	No	Details (whether yes or no)
11	Could the modification cause harm to the environment in any other way, considering *all* discharge or release routes (for example, greenhouse gases, ozone depleting substances, suspended solids)?			
12	Could the modification have any of the above effects (1 to 11) on any preceding or subsequent process, plant or sentencing facility, including off site (for example, lack of capacity, control capability)?			
13	Could the physical act of implementing the modification lead to any of the above effects (1 to 12) (for example, accidental damage leading to breach of containment)?			

Yes → Environmental Risk Assessment is required - *See section 6.2.2.*

No → Category E3 (no formal assessment required)

NOTE:

1 Any uncertainties regarding an E3 categorisation should be clarified by conducting an environmental Risk Assessment.

2 If the outcome is clearly E3, then categorisation is complete and local procedures should be followed prior to implementation.

3 If in doubt at any point during the categorisation, guidance is available in EPGN 07 'Guidance on environmental categorisation and Risk Assessment for plant modifications'.

Section 6.2.2 Environmental Risk Assessment (to be completed by the Environmental Assessor)

Based upon the environmental assessment, reference number ______________________, *this* PMP is categorised E1/E2/E3 *(delete as necessary)*.

Mitigating steps have been identified which, if implemented correctly, will reduce the consequences to a level equivalent to E1/E2/E3 *(delete as necessary)*. (This does not change the category)

Where the unmitigated risk category remains E1/E2 following environmental Risk Assessment, then where it exists, review and update Section D10 of the Safety Case (IEA) in line with SSP 1.25.01 'Preparation of Safety Cases'.

Risk Assessment complete (Name) ______________ Signature: ______________ Date: ________

The PMP has been initially categorised as A or B or C or D and E1 or E2 or E3

Name: ______________________ Signature: ______________ Date: ________

Plant Manufacturing Manager

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FIG. 9. Cont.

III-2. GRADING IN THE INITIAL HAZARD CATEGORIZATION (USA)

For initial hazard categorization, the facility radioactive material inventory shall be compared against the Threshold Quantities (TQs) identified in Table A.1 of Attachment 1 of DOE Standard 1027 [9]. Initial hazard categorization is a simple screening step that does not involve detailed computations. The consideration of material form, location, dispersibility and interaction with available energy sources called for in final hazard categorization is not applicable to initial hazard categorization. The overall hazard classification decision process is shown in Fig. 10.

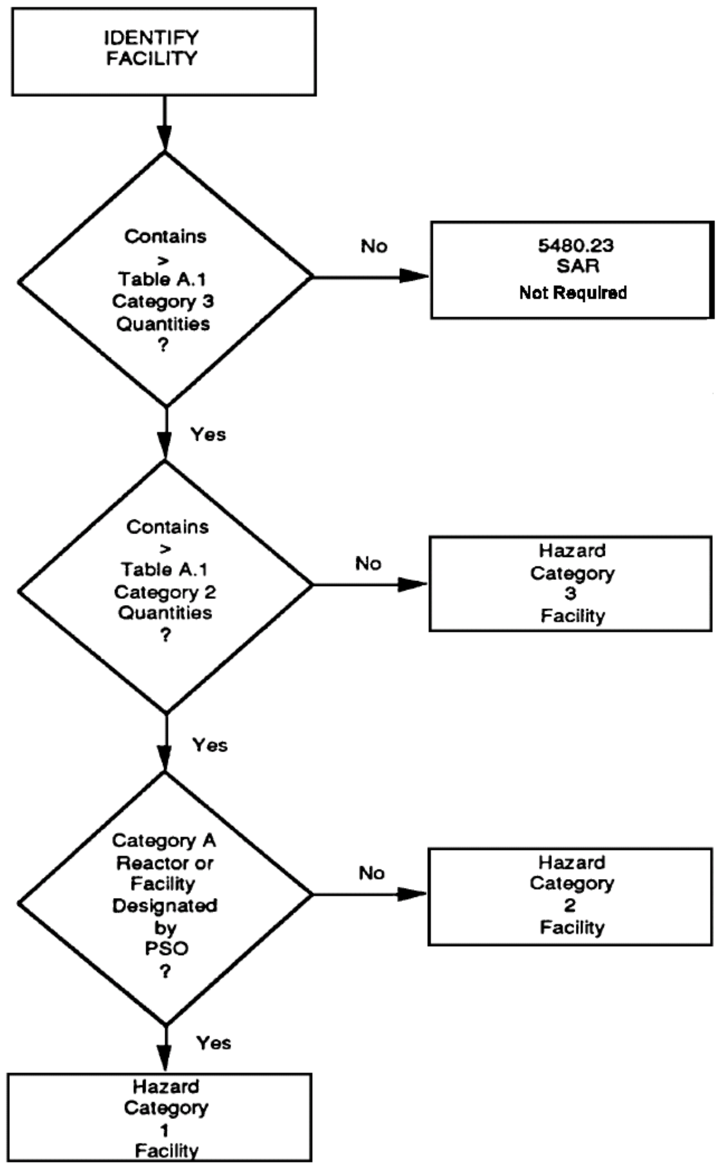


FIG. 10. Hazard classification decision process as described in DOE-STD-1027-92 [9].

The purpose of the final hazard categorization is to ensure that facility and accident specific factors that could:

(a) Either change the fraction of material released in an accident, or

(b) Change the amount of the total inventory of material subject to an accident are addressed to ensure the facility is properly categorized.

The first case (change in fraction of material released) is addressed by considering whether the release fractions that were utilized in derivation of the TQs used in the initial hazard categorization has to be adjusted. Note, conditions that may increase or decrease the hazard category must be considered.

In the second case (change in material subject to an accident), two conditions need to be considered in determining the final hazard categorization:

(a) Whether the facility inventory can be reduced (for the purpose of hazard categorization) due to segmentation (e.g., where facility features preclude bringing material together or causing harmful interaction from a common severe phenomenon); and

(b) Whether the facility inventory can be reduced (for the purpose of hazard categorization) due to excluding inventory contained in Type B containers.

DOE Standard 1027 [9] states that “for final Categorization, for facilities initially classified as Hazard Category 2, if the credible release fractions can be shown to be significantly different than these values based on physical and chemical form and available dispersive energy sources, the threshold inventory values for Category 2 in Table A.1 may be divided by the ratio of the maximum potential release fraction to that found on Page A-9.”

The release fraction assigned in DOE Standard 1027 for non-volatile solids/powders/liquids (1×10^{-3}) is based on release fractions used by the NRC in NUREG-1140, as modified by DOE as described in DOE Standard 1027 Attachment 1. Alternate release fractions, other than specified in DOE Standard 1027 must not be used unless there is some obvious inconsistency between a facility’s material forms or circumstances that warrant adjustment. Examples might include exceptions such as contaminated soil, activated metals in a de-inventoried facility, and vitrified glass.

If alternate release fractions are used, they must be appropriate for worst-case conditions, considering all materials in the facility and all accident stresses to which those materials might be subjected. DOE-HDBK-3010-94 [20] provides a useful source of information on ARF/RFs.

For the purpose of specifying alternate release fractions, applicable bounding airborne release fraction values need to be assigned. Where DOE-HDBK-3010-94 [20] identifies alternate release fractions significantly different than 1×10^{-3}, the applicability of that value needs to be verified for the form and stress under consideration. Where DOE-HDBK-3010-94 does not provide information directly applicable to a given situation, analysts may either:

(a) Derive conservative analogies to information in DOE-HDBK-3010-94; or

(b) Present new data and relevant calculations. In either case, the proposed application of alternate release fractions must be conservative, clearly explained, justified and approved by DOE.

If an alternate release fraction is accepted by DOE, new TQs can be calculated by multiplying the DOE Standard 1027 TQs by the ratio of the maximum potential release fractions and the release fractions on Page A-8 and A-9 of DOE Standard 1027 [9]. The final hazard categorization can be reduced if the sum of the fractions (i.e., fraction of the actual radionuclide inventory to the new Hazard Category 2 TQs for each radionuclide) is less than 1.

The conditions, parameters, and assumptions that form the basis for the hazard category of the facility must be protected. For facilities that are adjusting the facility's category based on form, dispersibility, segmentation, etc., Technical Safety Requirements administrative controls (or other functionally equivalent contractor controls for less than Hazard Category 3 facilities) need to be established to maintain the conditions, parameters, and assumptions that form the basis of the hazard categorization. Several examples of these inventory control process elements and assumptions (and how they may be changed) are presented below:

- Radionuclide inventory (increase in material to be stored or processed, change in the process, new sample data or analysis, discovery of new or different materials, for example during decommissioning of a facility);
- Form of material (change in how materials are contained, processed, or treated, or a newly discovered material characteristic);
- Dispersibility (change in container, process, or treatment, discovery of new or different materials, change in type or intensity of energy sources, change in project environment – drier or wetter than assumed);
- Interaction with available energy sources (change in adjacent facility or process, change in process, change in location, change in conditions surrounding area);
- Segmentation (change in facility physical features, change in process, change in energy sources, change in operations); and
- Changes in the nature of processes that may affect criticality safety assumptions.

If a configuration change is made or new information discovered that affects a condition, parameter, or assumption that helps form the basis for a hazard category downgrade, the approved hazard categorization must be re-evaluated. This hazard characterization basis must then be reviewed by DOE prior to making a change to ensure that the basis for the approval of the hazard category has not changed. The revised final hazard categorization must provide justification that demonstrates that the change or new information does not adversely affect the hazard category or establishes a new hazard category.

DOE has made provisions in some decommissioning safety basis documents that pre-authorizes change. It is anticipated that the hazard categorization and relevant hazard controls may be reduced as the hazard is reduced (i.e. source term removed). DOE has allowed reduction of hazard categorization and removal of controls upon demonstration of predetermined criteria. Similarly, a facility that was categorized as hazard Category 3, or low hazard facility may implement or "reinstate" controls if a sample returns unexpected results, or if unexpected hold-up is found once an inaccessible area is opened.

Procedure MCP-2451 illustrates the process/questions used by the US DOE site in Idaho to review proposed work to ensure that hazards introduced do not alter the established category. If the proposed work has to increase the hazard or energy such that hazardous materials could be released, additional safety analysis may be required.

III-3. HAZARD ASSESSMENT FOR THE ETR FACILITY HAZARD CATEGORIZATION, IDAHO (USA)

This section describes an application of the hazard categorization presented in the previous Section 3.2. The Engineering Test Reactor (ETR) at Idaho is part of the Idaho Cleanup Project. This water-cooled reactor was started in 1957, after taking only 2 years to build. At that time, it was the largest and most advanced materials test reactor available. The 175 MW reactor provided larger test spaces and a more intense neutron flux than the older Materials Test Reactor. The ETR evaluated fuel, coolant, and moderator materials under environments similar to those of power reactors. In 1972, the ETR was modified by the addition of a Sodium Loop Safety Facility into the reactor core, thus playing a new role supporting DOE's breeder reactor safety program. The deactivation of ETR was initiated in December 1981, including defueling the reactor, draining all the liquid systems, and preparing all major equipment for long-term storage. The deactivated ETR complex consists of the reactor building and a number of attached supporting buildings or structures.

The ETR Facility Hazard Categorization prepared in 2006 evaluates the radiological and hazardous materials in the Engineering Test Reactor (ETR) and determines the facility hazard characterization based on U.S. Department of Energy (DOE) requirements during planned facility decommissioning that includes vessel removal and facility demolition. The radiological and hazardous material source terms were evaluated to determine the categorization of the ETR facility. The evaluation of the facility hazards results in a categorization of less than Hazard Category 3 (LTHC3), based on the criteria in DOE-STD-1027-92 [9]. The assignment of Hazard Category 3 has been based on the considerations described in the following Figure 11.

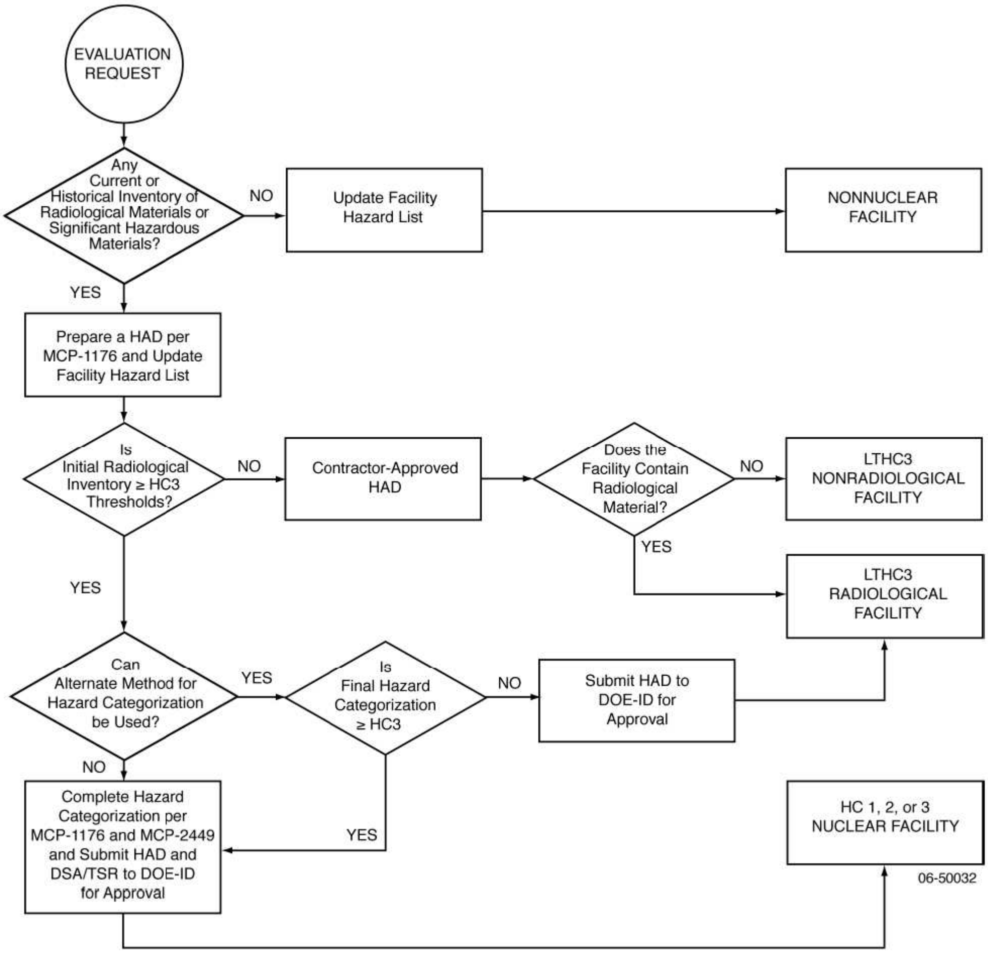
A current bounding inventory suitable for hazard categorization of the ETR complex has been performed. The sum of ratios has been calculated, based on the Hazard Category 3 thresholds by isotope, per DOE-STD-1027-92 [9]. For the ETR complex, the sum of ratios for the Hazard Category 2 thresholds is well below 1.0 (approximately 0.12), while the sum of ratios for the Hazard Category 3 thresholds is 14.6 (see Table 9). Based on these results, the initial Hazard Category for the ETR complex is Hazard Category 3.

This result is derived from the six isotopes within the reactor vessel that dominate the calculated sum of ratios, as shown in Table 9.

TABLE 9 REACTOR VESSEL RADIOLOGICAL INVENTORY DATA FOR DOMINANT ISOTOPES

Radionuclide	Activity [Ci]	Hazard Category 3 TQV [Ci]	Hazard Category 3 Ratio
H-3	3.11×10^{4}	1.60×10^{4}	1.94
Co-60	1.97×10^{3}	2.80×10^{2}	7.04
Ni-63	2.42×10^{4}	5.40×10^{3}	4.48
Pu-238	8.38×10^{-2}	0.62	0.135
Am-241	1.82×10^{-1}	0.52	0.35
Cm-244	3.20×10^{-1}	1.04	0.308
Sum of ratios (all isotopes)			14.58

The assignment of Hazard Category 3 then had consequences to the depth and complexity of the overall hazard analysis, consisting of release mechanism analysis, sequence selection, engineering analysis and consequence analysis. It was possible to choose simple enveloping approaches, minimizing the efforts required.



DSA Documented Safety Analysis
HAD Hazard Assessment Document
HC hazard category
LTHC3 less than Hazard Category 3
MCP management control procedure
TSR technical safety requirement

FIG. 11. Safety basis flow chart for less than hazard category 3 facilities or activities.

III-4. CATEGORIZATION AND ZONING FOR THE RESEARCH REACTOR HD-I AT THE DKFZ HEIDELBERG (GERMANY)

In the German Cancer Research Center (*Deutsches Krebsforschungszentrum*, DKFZ), two TRIGA reactors were built for medical and biological studies and investigations. The TRIGA HD I was operating from August 1966 to March 1977, and the following TRIGA HD II was in operation from February 1978 to December 1999. The decommissioning of the TRIGA HD I was deferred into the year 2005, and the decommissioning of the TRIGA HD II was done by immediate dismantling at the same time as TRIGA HD I in the years 2004/2005. The time schedule of decommissioning of the TRIGAs Heidelberg is shown below in Fig. 12 and Fig. 13.

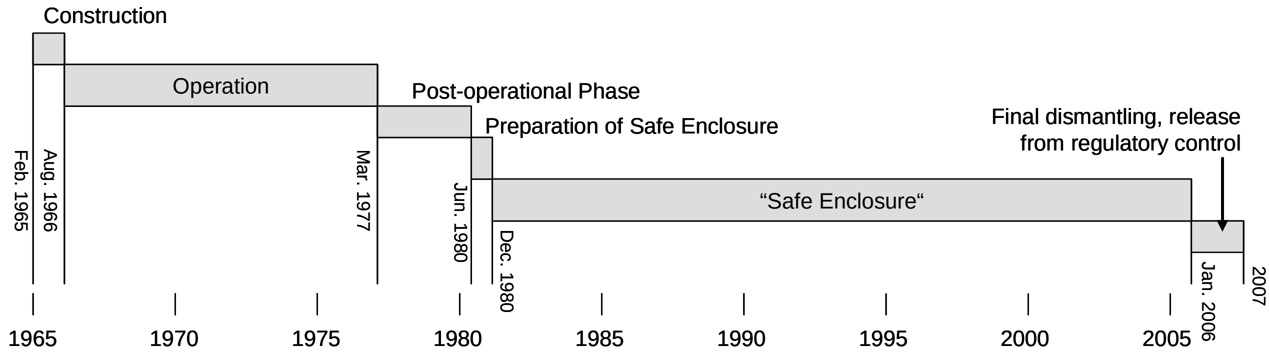


FIG. 12. Time schedule for operation, safe enclosure and decommissioning of the TRIGA HD I research reactor.

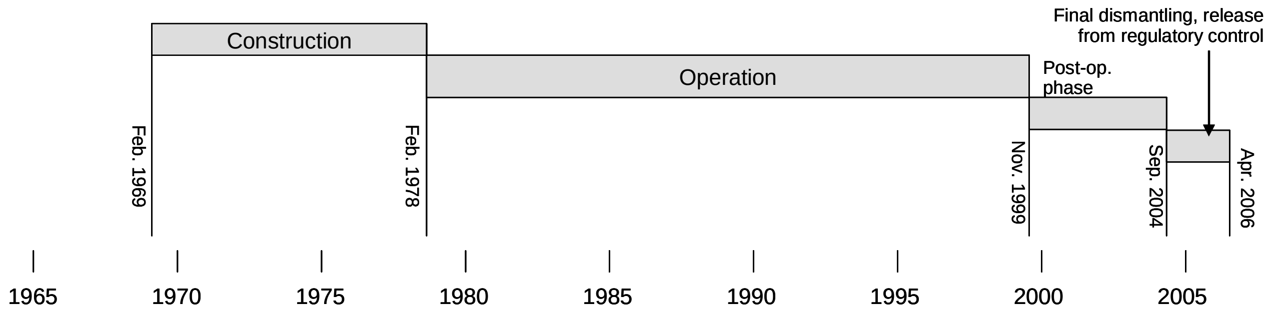


FIG. 13. Time schedule for operation and decommissioning of the TRIGA HD II research reactor.

The building of the TRIGA HD I was handed over to the Radiochemical Institute of the Heidelberg University immediately after moving the reactor vessel and the fuel to the TRIGA HD II, while the infrastructure (air conditioning, electricity, sanitary) of the building was still in function and in good condition. During the early years, the laboratories were used also as radioactive controlled area. In the meantime, the reactor core was taken out of the tank and the reactor equipment was dismantled. The tank was cleaned and decontaminated and afterwards it was closed by a concrete lid. Then the reactor hall was also used by the university (Radiochemistry Department), but the DKFZ had to fulfill some requirements and periodical measurements according to the requirements of the German Atomic Law.

In the year 2004, the DKFZ took over the forty-year-old building, which had not been well maintained. Some parts were even destroyed and different groups had used the laboratories, rebuilding some parts, and nobody could definitely identify which radionuclides had been treated and used in the controlled area. It was not even clear where exactly the boundary of the controlled area was placed. In addition, the license of the reactor operation was not very precise in this regard, so the licensee was fully responsible for all measurements of the contamination of the building.

This situation led to extended discussions with the authority concerning the number of measurements that were necessary for performing an initial characterization and for providing a reasonable basis for screening evaluations of hazard. It was this point where a graded approach was used, taking into account the overall low hazard potential of the facility: DKFZ was finally allowed to provide plausibility arguments for restricting the characterization programme to the lower part of the reactor building, avoiding measurements in the upper range of the building, because there all the offices, the library and the meeting rooms were situated.

This approach was to a large extent built on the trust that the authorities had developed during the good progress of the work. All tasks during the decommissioning work had been carried out without mistakes, and any problems had been openly discussed between the operator and the authority, seeking for a way of correction and improvement.

Appendix IV

EXAMPLES FOR THE GRADED APPROACH IN EVALUATION OF ACCIDENTS DURING DECOMMISSIONING

IV-1. EXAMPLE FOR AN ENVELOPING SCENARIO FOR THE ASSESSMENT OF DOSES TO CRITICAL GROUPS DURING DECOMMISSIONING OF AN NPP WITH LIGHT WATER REACTOR (GERMANY)

The study [11] has been performed with the aim of creating an enveloping scenario for the assessment of doses to critical groups during decommissioning of a light water reactor that would be compatible to the regulatory requirements in Germany. The term "enveloping scenario" means that this scenario would lead to higher releases than similar individual scenarios being usually analyzed. It does not mean that in each step of the enveloping scenario all parameters are chosen at their maximum possible value.

In [11] it is assumed that the radionuclide vector in the plant consists of Co-60, Cs-134, Cs-137, Mn-54, Eu-152, Eu-154 as well as Sr-90.

The scenario takes into account a fire in the plant by which a certain amount of the activity on outer surfaces of buildings and components, of the activity on inner surfaces of components which are assumed to have been opened for dismantling work, and of the activation is released into the atmosphere of the plant. The contaminated air will then be distributed in the plant because of the ventilation and the heat. Part of the aerosols will be deposited on the colder walls. Another part will be kept back by the filters. It is, however, assumed that a larger percentage of the activity will pass the filters assumed to have become clogged, and will leave the plant without filtration. The assumptions for each step are outlined in Table 10.

TABLE 10. ASSUMPTIONS FOR CALCULATION OF THE SOURCE TERM IN THE ENVELOPING ACCIDENT SCENARIO

	Outer surfaces	**Inner surfaces**	**Activated material**
Total activity [Bq]	1×10^{10} Bq	1×10^{12} Bq	1×10^{17} Bq
Percentage of high-energy γ- and β-emitters	100 %	100 %	10 %
Activity of γ-emitters and Sr-90	1×10^{10} Bq	1×10^{12} Bq	1×10^{16} Bq
Activity affected by the incident (fire)	100 %	10 %	10 %
Affected activity of γ-emitters and Sr-90	1×10^{10} Bq	1×10^{11} Bq	1×10^{15} Bq
Percentage of activity being resuspended into the atmosphere of the facility	10 %	10 %	0.1 %
Resuspended activity of γ-emitters and Sr-90	1×10^{9} Bq	1×10^{10} Bq	1×10^{12} Bq
Retention in the facility on surfaces and by filters	50 %	50 %	50 %
Release into the environment	5×10^{8} Bq	5×10^{9} Bq	5×10^{11} Bq

It is in particular a conservative approach to assume a percentage of 0.01 % of the activated material to be affected and released by the fire, as there fill be almost no inflammable material near the reactor pressure vessel and the biological shield. However, studies on fires in NPPs (HDR Großwelzheim) [21] show that damage to components is likely and that activation can thus be released.

In addition, the activity in the contamination of the controlled area of the plant is chosen at a rather high level. It is also a conservative approach to assume that the fire will affect nearly the entire controlled area. Instead of the assumption of 100 % presented in Table 10, a value of 10 % would be more pertinent to a real case.

The percentage of contamination released as the consequence of a fire is typically on the order of 0.1 % to 1 % [22]. The value of 10 %, which is assumed in Table 10, is therefore a conservative value that corresponds to releases observed during the application of thermal cutting techniques.

As experiments with real fires in nuclear facilities demonstrate [21], the effectiveness of filters is significantly reduced by heat and smoke. The release of aerosols into the environment will occur mainly via the stack (here assumed as 90 %), but some doors (escape routes) in the controlled area might be opened so that release at lower heights may occur (here assumed as 10 %).

On the basis of these assumptions, doses to members of the public have been calculated using the requirements on dose calculation for incidents and accidents [23]. The resulting dose remains below the individual dose limit of 50 mSv.

This enveloping scenario had been intended to be used instead of facility specific assumptions. The approach of using one or a few enveloping scenarios has been applied many times in German safety assessments, thereby greatly simplifying the number of scenarios to be analyzed, but those enveloping scenarios have employed different sets of parameters and assumptions.

IV-2. EXAMPLES FOR SPECIAL HAZARD ASSESSMENTS (USA)

The following evaluations were included in Hazard Assessment Document (HAD-200) [24] for the Engineering Test Reactor (ETR) Facility Hazard Categorization. These represent unique consideration required in decommissioning that result from the proposed activities, not specifically addressed (or envisioned) by the design and operational evaluations of the facility. Two unique hazards were posed by decommissioning of the ETR. The first, release of tritium was not a concern during operation of the reactor or in the post operational surveillance and maintenance because the material was contained within the reactor vessel. Decommissioning required opening the reactor, exposing the Be plates contaminated with tritium. Fire events associated with equipment removal presented a new means of release during decommissioning. Similarly, filling the reactor with grout, thereby producing potentially explosive levels of hydrogen gas was an event unique to decommissioning that was not applicable to other phases of the facility's life cycle.

(a) Tritium as a combustible gas – Tritium is a combustible gas, chemically similar to hydrogen. Ignition is not a concern while the gas is confined throughout the solid beryllium material structure. Possible release of significant quantities by heating of the beryllium in a fire is addressed as a potential radiological hazard below and concluded not to be credible. The reaction of hot beryllium with water to produce hydrogen is also precluded by the conclusion of that evaluation, that fire affecting the reactor internals is not credible.

The potential for hydrogen diffusion from the beryllium at room temperature to pose a combustible gas hazard was given further consideration as follows:

- The total quantity of tritium calculated in the beryllium is approximately 3.2 g (31 110 Ci/9669 Ci/g). The quantity 9669 Ci/g is the specific activity of H-3 given in Table 1 of LA-12 846-MS, the Hazard Category 2 support document for DOE-STD-1027-92 [9]. This is approximately 1.0-g mole of tritium. Upon dilution in the mass of air contained in the reactor vessel, approximately 3 kg of air (100 g moles) would ensure a concentration below 1 % (specifically, below 25 % of the 4 % lower flammability limit [LFL] for hydrogen). This corresponds to less than 3 000 l, or approximately 100 ft^3.

- If diffusion is occurring at all, it is not a new phenomenon and the releasable tritium is likely to be gone after 25 years. Assuming conservatively that diffusion has been linear with time and is still taking place, it would be limited to approximately 4% per year and would be safely diluted in a volume as small as 4 ft^3 (even stratification under the head could not pose a concern).

- Beryllium tends to retain tritium generated by irradiation until heated to sufficient temperature to anneal it. One study focused on small beryllium particles in fusion applications identifies two mechanisms for retention: (i) it can be bound chemically in the form of beryllium hydroxide (where BeO was present); or (ii) it can migrate to helium bubbles trapped in the lattice. Another study concludes that there was no significant open porosity in beryllium, even as it swelled significantly during irradiation. The third study concludes that for irradiation below approximately 400°C, tritium is completely captured in irradiated beryllium.

Based on these considerations and the fact that samples of the reactor atmosphere taken in support of decommissioning detected no tritium above the detection threshold, potential diffusion of tritium to a hazardous concentration (specifically, >25% LFL), either within the reactor or outside, is concluded not to be credible.

(b) Hydrogen Generation During Vessel Grouting – Hydrogen gas may be generated during reactor grouting as a result of chemical reaction between the grout and the aluminium. Other metals, including beryllium, do not have a similar potential to interact with grout and produce hydrogen. The bounding potential rate of hydrogen generation during ETR grouting has been determined to ensure that adequate ventilation is provided. The analysis in EDF-7228 bounds the quantity of aluminium that could be present and reacting with grout (1,211 ft^2), bounds the grout temperature during pouring and subsequent curing based on the allowed composition (≤ 60 C), and determines the bounding hydrogen generation rate that could occur for these conditions based on Ref [25] (< 1.5 ft^3/min). Based on these results, the following additional considerations have been addressed to ensure that the hydrogen generation hazard can be safely managed:

- Given the 1 761 ft^3 volume of the upper portion of the ETR vessel, the peak hydrogen generation rate of 1.46 ft^3/min implies a hydrogen concentration of 0.083% per minute requiring 12 minutes to reach 25% of the hydrogen LFL, and 48 minutes to reach a flammable concentration (4% hydrogen is the LFL). An airflow rate of 150 cfm is sufficient to prevent the actual hydrogen concentration from exceeding 25% of the LFL. A minimum ventilation flow rate of 600 ft^3/min is required and a nominal flow rate of 2 000 ft^3/min is planned.

- Based on PNNL-15156 [25], hydrogen generation can last for up to 152 min (Test No. 5, Attachment 3). This would imply a total hydrogen concentration without ventilation of no more than 12.6 % (152 · 0.083 %). Recognizing that the peak generation is not maintained for the entire 152 minutes, a more accurate estimate of the total possible hydrogen concentration based on Test No. 5 is 67.8 % of this value, or 8.6 %. This value is sufficiently low to preclude detonation. Deflagration, which is not precluded if the planned ventilation is interrupted, would occur rapidly, largely at elevations above the activated structures, would not exceed a peak deflagration pressure of 2.6 atmospheres, and would not cause enough heat transfer to heavy activated metal structures to result in release of radiological material. The water inlet pipe will be configured to provide a vent path to the unoccupied pipe trench in the event that a deflagration does occur.

— Per EDF-7228, the project requirements for ventilation during grouting ensure excess capacity above the minimum 600 ft^3/min and require redundant available equipment for reliability, including a backup power supply. The exhaust will be connected to the top of the vessel to effectively remove hydrogen, while the makeup air will be introduced at as low an elevation as practical (water inlet nozzle). Ventilation will be maintained for at least 24 h. Provisions will be made to ventilate the top of the inner tank, both to preclude the accumulation of hydrogen and to ensure that the grout placement meets waste disposal requirements. Airflow will be verified at an interval chosen to ensure that a flammable concentration of hydrogen does not accumulate (that is, less than 48 min between flow checks).

Thus, without considering the planned ventilation (a control that cannot be credited in establishing hazard categorization), detonation is precluded and a deflagration could not release the radiological material from the activated structures. While a control to prevent deflagration is not required or relied upon to support hazard categorization, the facility safety management programs (SMPs) will ensure sufficient ventilation to make an actual deflagration unlikely (i.e., planned ventilation will be sufficient to prevent deflagration with significant margin).

Several events that are applicable during operation or extended surveillance and maintenance of a facility may require discussion in a decommissioning safety analysis, but may not necessarily require detailed evaluation.

The analysis that was presented in the last approved safety analysis can be relied upon with discussion that addresses those elements that are pertinent to decommissioning. Relative results may be scaled up or down if assumptions (material form, ARF/RF) remain valid. Some of those aspects that need to be addressed:

— Reduced material at risk. Typically, the hazard is reduced during decommissioning. Events such as earthquake, flood or wind may use the initial value as bounding as decommissioning activities serve to remove the material.

— Exposure of new material at risk. Activities such as removal of ventilation ducting with holdup may expose materials that were not previously subject to accident events. Removal of safety support systems, introduction of construction activities/materials may increase potential of specific events.

— Time dependent frequency. The probability of a design basis earthquake striking during the decommissioning process in conjunction with the removal of materials may provide sufficient basis for not considering the event in the decommissioning safety analysis. It is not appropriate to eliminate such events from consideration if the facility will remain in a surveillance and maintenance mode for an extended period of time.

— The sequencing of structural demolition or activities that could reduce the structural capacity of the facility must be factored into planning such that the majority of material is removed before such changes could affect the frequency of collapse or similar events (challenging previous analysis).

IV-3. EXAMPLE FOR ASSESSING INDUSTRIAL SAFETY (THE NETHERLANDS)

While extending to areas outside the DeSa project, industrial safety also has an implication on nuclear safety and is therefore taken into account in performing safety assessment to the appropriate extent. The performance of safety assessment for industrial hazards can be graded according to the risk level, just as safety assessment for nuclear and radiological hazards can be graded according to the hazard potential or the resulting doses. In the following, an example is described, which provides a practical scheme to evaluate the grade of safety assessment for industrial hazards.

In the Netherlands, industrial safety is regulated in the Arbeidsomstandighedenwet [26] and its supporting regulations. This is the national implementation of the European Council Directive of 12 June 1989 on the introduction of measures to encourage improvements in the safety and health of workers at work [27].

Companies are required to develop a (industrial) health and safety strategy. In order to know the industrial risks associated to the work it is obligatory to perform a risk inventory and evaluation (RI&E). This is a phased process to reduce industrial risks to an acceptable level.

(a) Risk inventory; checklists have been developed, to facilitate the risk inventory process.

(b) Risk evaluation; the risks observed in the inventory are given a priority rating or scoring. One of the common approaches is to rank the risks according to the probability of the initial event, the probability that the initial event will lead to harm, and potential effect (harm). In some cases correction factors are used if mitigating measures are still possible once the initial event took place. An example of practical application is presented below.

(c) Planning of actions; a plan must be made to reduce the risks until the resulting risks are at an acceptable level. It is allowed to spread actions over several years for financial reasons.

(d) Updating the risk inventory and the evaluation if both regularly basis and circumstances change; e.g. working methods or conditions change, available knowledge or techniques (state of the art) improves, or operational experience shows that improvement is required (e.g. after an accident).

One of the commonly used systems for risk scoring is based on the method by Fine and Kinney:

$$R = E \times P \times C \qquad \text{(D–1)}$$

where:

R = risk score;
E = exposure or probability of the initial event;
P = probability that the initial event will lead to harm; and
C = potential consequences of an accident.

Scoring tables like the following are used to quantify E, P and C.

TABLE IV.11 PROBABILITY OF THE INITIAL EVENT AND VALUES FOR E

Probability of the initial event	E
Hardly thinkable	0.1
Almost impossible	0.2
Imaginable but improbable	0.5
Unlikely but possible in borderline case	1
Unusual	3
Possible	6
To be expected	10

TABLE 12 PROBABILITY THAT THE INITIAL EVENT WILL LEAD TO HARM AND VALUES FOR P

Probability that the initial event will lead to harm	P
< 1 × per year	0.5
Annually	1
Monthly	2
Weekly	3
Daily	6
Continuously	10

TABLE 13 POTENTIAL EFFECTS AND VALUE FOR C

Potential effect	C
Little harm, without absence or nuisance	1
Absence and nuisance	3
Irreversible effect (disablement)	7
Death of one person; either immediately or later	15
Death of several persons; either immediately or later	40

The resulting risk score gives in indication of the action required. An example thereof is presented in the following table.

TABLE 14: RISK SCORES AND ASSOCIATED ACTIONS

Risk ranking	R	Action
Little harm, without absence or nuisance	< 20	No action
Absence and nuisance	20 –70	Attention required
Irreversible effect (disablement)	70 - 200	Measures required
Death of one person; either immediately or later	200 - 400	Immediate improvement required
Death of several persons; either immediately or later	> 400	Stop work immediately

In practice, for a risk ranking above 70 additional analyses are required to be able to find the primary causes and potential ways to reduce the risk. When searching for ways to reduce the industrial hazard the obligatory order to investigate options is:

(a) Reduction of the source term;

(b) Mitigation of the exposure pathway;

(c) Change of the organization of the work; and

(d) Use of personal protection devices.

The validity of the priorities or the decisions based on this method is a function of the validity of the estimates of the parameters P, E and C. The individual scores are determined from a combination of statistical failure data and operational experience and therefore require the collection of information, the visit of the workplaces and discussion with the workers about the exact nature of the activities. The above tables are only examples of scoring tables. Several ranking systems have been developed to fit to the standards applied in different companies.

Special attention is needed for combinations of radiological and industrial risks. For example, a small cut in the hand is usually not very serious in an industrial environment, but when this happens while cleaning a plutonium containing glove box, a serious risk of internal exposure exists. Another example is a small fire, which would be easily extinguished in a non-nuclear facility, but would expose fire fighters to radiation if the fire were in a controlled area with high dose rates.

This example provides illustration for an approach to assess industrial hazards, forming the basis for the decision which level of complexity would be appropriate for a more thorough investigation and which industrial hazards might need to be taken into account as part of the (radiological) safety assessment, e.g. as an initiating event for incidents or accidents.

IV-4. APPROACH TO DECOMMISSIONING OF A NORM FACILITY (BRAZIL)

The mineral industry uses many different types of ores that contain naturally occurring radioactive materials (U-238 and Th-232 series), called NORM. These materials are typically produced in very large volumes with relatively low specific activities. The concentration of radionuclides during processing often results in relatively high dose rates, also later during dismantling of the facilities involved, including vessels, storage facilities, baghouses, electrostatic precipitators, metallurgical furnaces and other ancillary equipment.

One large NORM facility of concern is placed in the north part of Brazil, and produces concentrates of tantalum, niobium, tin, zircon using physical procedures, and utilizes a pyrometallurgical process to generate the metal alloys. The average grade of natural radionuclides in the fresh rock is approximately 0.02 % U_3O_8 and 0.18 % ThO_2.

Part of the milling site has been recently decommissioned, and a safety assessment was made using the DeSa safety assessment methodology (see main report).

The mining complex is composed of specific units:

— The open pit mine (hard rock);

— The physical concentration (crushing and grinding site, pre-concentration, gravimetric and electromagnetic separation);

— The pyrometallurgy furnaces; and

— The waste trench.

In order to assemble all of the physical processes on the same site, part of the complex is being transferred. The schematic representation of Fig14 shows this part of the facility (MU 1, 2 and 3). Some equipment will be transferred to the new site, and the rest disposed of as radioactive waste (contaminated pavement, metallic pieces, soil, and bricks). The Brazilian Nuclear Energy

Commission (CNEN) requires detailed reports of the planned dismantling activities, to be analyzed and approved before any action can be taken. These reports must include the radiological procedures that are to be followed by the workers involved, an estimate of individual doses, and information regarding waste management procedures that will be adopted during this critical phase.

The decommissioning plan must define:

(a) The radiological criteria to be used for clearance;

(b) Information of the radioactive waste disposal site to be used, including an appropriate description of the landfill;

(c) A description of the scenarios that will be used for long-term safety assessment of the waste disposal site;

(d) Radiological procedures for personnel involved in the cleanup;

(e) Scenarios for incidents/accidents; and

(f) Pprocedures for controlling and guaranteeing that the doses for the critical group will not exceed 0.3 mSv y^{-1} (dose constrain).

For the workers, a maximum value of 20 mSv y^{-1} during operations will be adopted; they are also obliged to use special garments, masks, gloves and individual dosimeters.

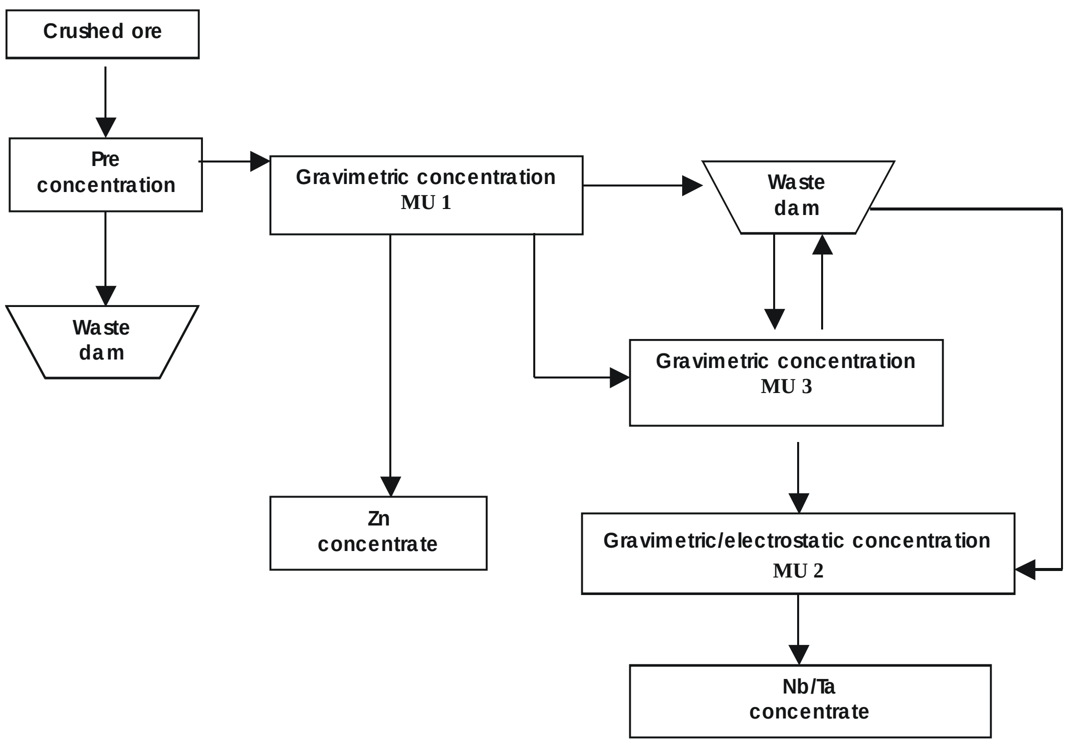


FIG. 14. Part of the facility to be decommissioned.

Surface contamination of floors, walls and metallic pieces is to be estimated doing measurements at every 2 m x 2 m area by means of wipe tests (to determine removable contamination). Clearance limits established for floors, walls, materials and equipment are the same as those applied to the

USAM decommissioning in 1997 (Table 15). Clearance limits for soil, to be determined by CNEN, will account for the high natural radioactivity in the region. For liquid effluents, the permitted maximum concentration will be defined using risk assessment, once Brazil does not adopt a certain prescribed level.

TABLE 15 CLEARANCE LEVELS FOR SURFACE CONTAMINATION

Type of radiation	Surface contamination	
	Regulatory limit [Bq cm^{-2}]	Operational limit, [Bq cm^{-2}]
alpha	0.3	0.25
beta-gamma	3	2.5

A preliminary analysis of the decommissioning work was carried out (screening phase), based on risk assessment and the operational work at the facility. The maximum exposure rate was 5 mR/h, while the workforce doses varied between 0.8 and 1.8 mSv month^{-1} (unit MU 1), and between 1.2 and 2.7 mSv/month (MU 2 and 3). The estimated average internal dose was taken to be 5 mSv y^{-1}, but up to 10 times higher in case of accidents (e.g., spills, inhalation of dust). A preliminary survey was done using wipe testes in order to determine removable contamination; values varied from 0.02 to 0.5 Bq m^{-2}. The average concentration measured at the milling units (see Table 16) was used for the safety assessment of the future waste repository.

TABLE 16 CONCENTRATION OF RADIONUCLIDES

Material	Concentration			
	^{238}U - ^{234}U [Bq g^{-1}]	^{230}Th [Bq g^{-1}]	^{226}Ra [Bq g^{-1}]	^{210}Pb [Bq g^{-1}]
Ore concentrate	30	30	25	18

Contaminated material resulting from the decommissioning work (MU 1, 2 and 3) will be disposed of in an industrial landfill inside the facility. The landfill will be an engineered earthen structure (open excavation above the water table, with placement and compaction of the waste), in order to ensure integrity of the deposit over relatively large geologic times (thousand of years). According to the Brazilian law [28], the initial deposit of wastes containing natural radioactive material can become the final repository and can be placed at the mining facility site. For this reason, the Regulatory Body must analyze if the safety assessment provide enough information to ensure the protection of the future generations and the environment.

The aim of a risk assessment for a NORM waste disposal facility is to demonstrate compliance with the safety requirements, related to the human being and the environment. These results are used to judge the design ability to meet the radiological standards for long-term protection of the public, established by the governmental authorities. The pathways analysis and scenarios give a systematic way to evaluate the potential routes by which people could be exposed to radiation.

The performance assessment of the disposal facility is carried out using a leaching and off-site scenario. The leaching or small farm scenario was modelled by assuming that rainfall percolated vertically downward through the disposal landfill, the liner and the unsaturated zone and then, finally, moved rapidly into the aquifer. Radionuclides transported from the waste repository via subsurface groundwater is intercepted by a well and also discharged directly into the stream. The final assumption for this example was that the water from that well was the only source of water available to the resident farmer, and all the fish consumed comes from a nearby stream. This scenario assumes that the receptor is located 100 m from the landfill, along the gradient, and that contaminated water will be used in the biosphere compartment.With respect to the biosphere, the following processes have been considered in the model:

(a) Ingestion of well water;

(b) Irrigation;

(c) Re-suspension and inhalation;

(d) External radiation exposure;

(e) Consumption of home-grown produce;

(f) Consumption of contaminated meat;

(g) Ingestion of contaminated milk;

(h) Accidental ingestion of contaminated soil;

(i) Inhalation of radon and decay products from soil; and

(j) Contact with surface water, transfer to fish and to humans.

The model for vertical transport of radionuclides through the landfill assumes that all leachate from the landfill is homogeneous, and local data was used in order to have a more realistic risk assessment. The water flux was simulated using the computer code Hydrus-1D [29] and the results were used for the radionuclides transport to the aquifer. The infiltration obtained was 0.65 m y^{-1}, and the waste is placed in the landfill in a layer of 6 m height (clay soil), and overlain by a thin layer of clean soil. A compacted clay liner is underlining the waste layer and the unsaturated layer (5.0 m, clay loam soil) is coupled with the aquifer. In order to obtain the groundwater concentration, a numerical simulation was done using the symbolic computation software Mathematica 5.1 [30] for the radionuclides transport in a decay chain mode. The final risk assessment was performed coupling these results with the biosphere model. The time scale chosen is 10 000 y (next glacial era), once for longer periods of time the results become more and more uncertain.

Different simulations were performed, taking into consideration all the pathways and the radionuclides of the U-238 decay chain. The first option modelled the landfill placed directly above the aquifer, considering a compacted clay soil for the waste region, and gave a final dose much higher than 0.3 mSv y^{-1} (Fig. 15). The simulation of U-238 and U-234 were made separately, but the final results for both were added in the plots.

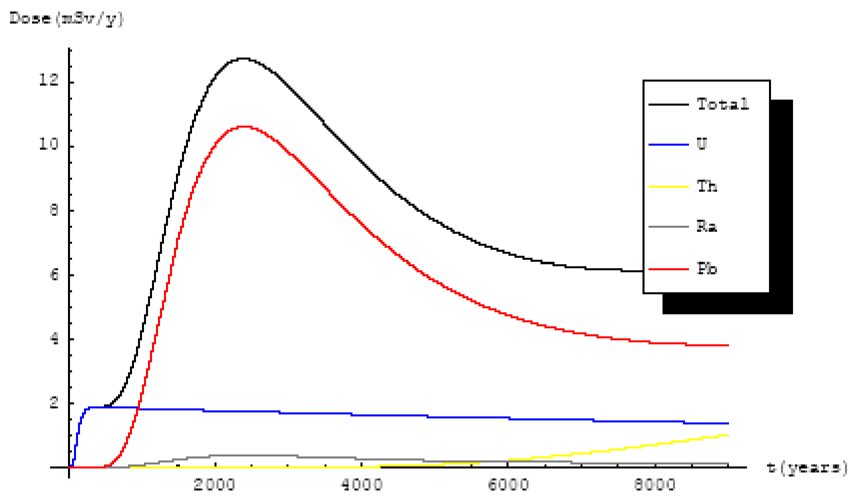


FIG. 15. Dose assessment for NORM disposal (option 1).

The results demonstrated that a graded approach could be applied for the dose risk assessment, and in this case, the unsaturated zone was taken into consideration, between the waste layer and the aquifer. The local data obtained shows that the vadose was 5.0 m deep, with characteristics of a clay loam soil. The doses, for the period of 10 000 years, were below 0.3 mSv y^{-1} (Fig. 16), showing compliance with the safety requirements defined by the Brazilian standards for this kind of facility [31].

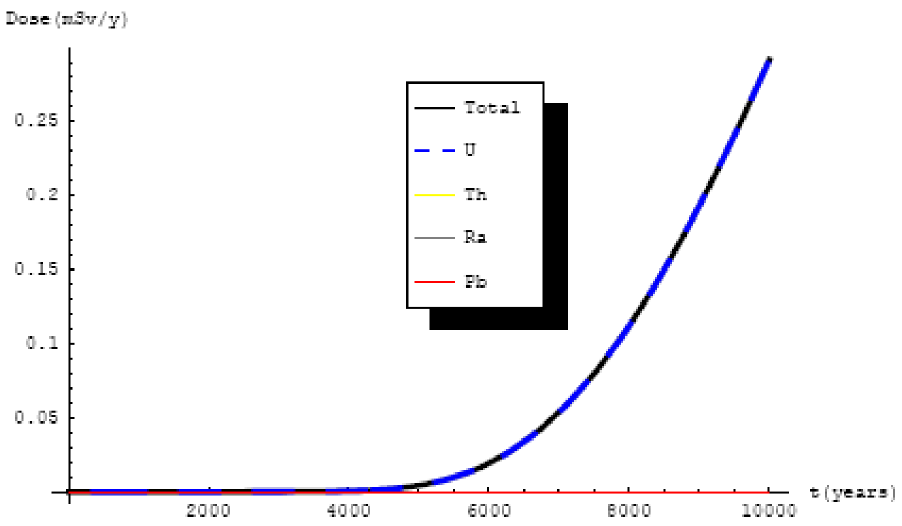


FIG. 16. Dose assessment for NORM disposal (model with vadose).

In order to verify the maximum dose for the simulation with vadose, the risk assessment were performed also for the period of 100 000 years (Fig. 17). It can be noticed that the maximum dose will be around 80 000 y, because of the growing of Pb-210 and Th-230 in the aquifer. In this case, a new graded approach can be used in order to improve the models and local data, and a new modelling must be performed.

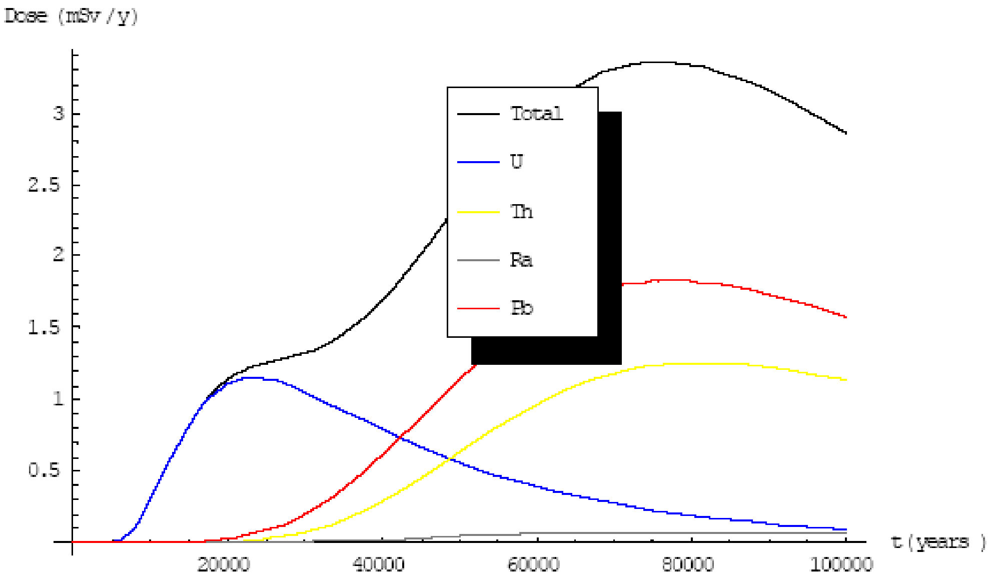


FIG. 17. Determination of maximum dose.

IV-5. SAFETY ASSESSMENT FOR DOSE ASSESSMENTS FROM CS-137 SOURCES (CUBA)

The National Institute of Oncology and Radiobiology (INOR) was one of the first institutions in Cuba that applied ionizing radiations in medicine. There was a facility used for brachytherapy service. Some years later, the brachytherapy service was moved to another section in the hospital.

At the beginning of the 1980s, no centralized storage facility for radioactive waste was in operation in the country. A room belonging to the former brachytherapy service at INOR was then used as storage facility for disused sealed sources arising from nuclear applications in medicine and industry. One or more Cs-137 sources stored in this area were leaking, causing a radioactive contamination in the facility (eight rooms and the garden).

No regulations to address decommissioning were in place in the country at that time. This resulted in a lack of early decommissioning considerations and planning at the INOR. When the contamination was detected some attempts were carried out, but for different reasons, the requirements established by the Regulatory Body for decommissioning could not be achieved, and therefore the facility could not be released from regulatory control. The facility has remained closed for many years because of the remained contamination.

The decommissioning activities were considered justified taking into consideration: (1) the unacceptable radiological risk. It was estimated that one person, 1/3 year in this area could receive an effective dose of 100 mSv (due to external exposure), and (2) the need for reusing these rooms (as part of a hospital) and therefore the need to release them from Regulatory Control

Safety requirements and criteria to be met for the final release of the facility from regulatory control are described in the decommissioning plan. After decommissioning the facility would be use for a non-nuclear purpose, therefore the annual dose for any member of the public has to be lower than 0.3 mSv above the natural background.

This report documents the safety analysis carried out for the decommissioning activities at the INOR facilities. Particular emphasis is given to the application of the graded approach in carrying out safety assessment for decommissioning as planned. The dose assessments during and after decommissioning are described below. The graded approach consists in not only performing a site-specific evaluation of the facility, but in deriving site-specific criteria for dealing with the situation. Because of the relative simplicity of this nuclear facility, the approach is very transparent.

(a) Safety assessment for personnel during decommissioning

It was identified that the hazards at workplaces during decommissioning activities may arise from:

— The external irradiation caused by existing contamination; and

— The inhalation of aerosols originating during the demolition and decontamination activities.

A map of dose rates and contamination conditions was elaborated as important tool for hazard identification. Unexpected contamination was found at doorframes in almost all the rooms.

Adopted considerations for the estimation of the effective dose constrains for operators due to external radiation are summarized in Table 17. The expected individual doses for decommissioning activities were estimated from the required time the operators would spend at particular locations (room) and the maximum dose rates at these locations. According to the decommissioning plan, the effective dose for workers due to decommissioning activities has to be less than 5 mSv (4 mSv for external radiation, 1 mSv for inhalation).

TABLE 17 ESTIMATION OF EFFECTIVE DOSE FOR OPERATORS, DUE TO EXTERNAL RADIATION DURING DECOMMISSIONING

Room	**Dose rates on the floor surface [μSv/h]**		**Estimated working time [h] (1)**	**Effective doses in [mSv] (2)**
	at 50 cm	**at 100 cm**		
1	7.0	4.0	8	0.05
2	30.0	20.0	32	0.85
3	50.0	30.0	64	2.77
4	2.0	1.5	32	0.06
5	2.5	2.0	24	0.06
6	0.6	0.6	16	0.01
7	3.0	2.0	32	0.09
8	0.6	0.6	20	0.01
Garden	2.0	1.5	32	0.06
Totals			260	**3.95**

(1) Conservative time. The dose rates will diminish during decommissioning

(2) (2/3·D(50 cm) + 1/3·D(100 cm) · Working time

The doses to be received by the operators were estimated prior to each operation. Individual radiological surveillance was maintained as well as the monitoring of working areas, in order to verify the compliance with the dose constraints established in the decommissioning plan. The monitoring activities included: dose rate levels, surface contaminations, activity concentration in aerosols and individual doses from personal dosimetry.

Actual values of external doses received by the operators during the 5 months decommissioning project are shown in Table 18.

TABLE 18 INDIVIDUAL EFFECTIVE DOSES FOR OPERATORS

Operator	**Doses [mSv]**					
	Feb.	**March**	**Apr.**	**May**	**June**	**TOTAL**
Operator-01-S	0.00	0.00	0.00	0.00	0.00	0.00
Operator-02-C	0.00	0.00	0.53	0.00	0.00	0.53
Operator-03-M	0.00	0.26	1.33	0.24	0.00	1.83
Operator-04-R	0.00	0.11	0.41	0.12	0.00	0.64

Samples of aerosols for verification of radioactive contamination levels in air and for the estimation of the commitment effective doses by inhalation were taken daily in work areas (Fig. 18). Air samples were taken during 6 hours with a caudal of 7.5 m^3/h. Therefore, 45 m^3 of air were monitored daily and all values were reported. No significant contamination was detected on the filters.

FIG. 18. Taking of aerosol samples for verification of radioactive contamination levels in air and for the estimation of the commitment effective doses by inhalation in work areas

The individual radiological surveillance for internal contamination of operators involved in the decommissioning project was carried out. Table 19 contains the reports of the commitment effective doses by inhalation. Doses from internal exposures were kept near zero by using of respiratory protection in any circumstances.

TABLE 19 COMMITTED EFFECTIVE DOSES BY INHALATION

Operator	Committed effective doses by inhalation [mSv]
Operator-01-S	0.001
Operator-02-C	0.010
Operator-03-M	0.000
Operator-04-R	0.005

(b) Safety assessment for post-decommissioning activities

As some amounts of radioactive material were left at the site, and the end-point of the decommissioning project was the release of the facility from regulatory control; a safety assessment for the post-decommissioning state was performed. The assessment included dose estimation during reconstruction operations and annual dose for members of the public for free reutilization of the site.

(c) Dose assessment for reconstruction operations

Once the dismantling and decontamination activities were concluded, the reconstruction phase of the facility took place. For that reason, some constructers were exposed to the low-remaining radiation exposures; which was assessed.

For this assessment, the following assumptions were considered:

— Any work is considered under public conditions, therefore the participating constructers were considered as member of the public;

— As the remaining contamination was fixed, the exposure pathway considered was the external irradiation;

— One person takes part in all operations, during each reconstruction phase, during the whole 30 days (conservative assumption);

— The dose rate at 50 cm from the surface of the floor in the centre of each room was use for calculation; and

— The gradual reduction of the dose rates during the operations were not considered (conservative assumption).

Table 20 summarizes the estimated effective doses for constructers during the facility reconstruction.

TABLE 20 ESTIMATED EFFECTIVE DOSES FOR CONSTRUCTERS DURING THE FACILITY RECONSTRUCTION.

Room	Dose rate at 50 cm [μSv.h^{-1}]	Working time		Doses
		Days	Hours	[μSv]
1	0.28	2	16	4.48
2	0.20	5	40	8.00
3	0.34	5	40	13.60
4	0.03	3	24	0.72
5	0.02	2	16	0.32
6	0.05	3	24	1.20
7	0.15	5	40	6.00
8	0.02	2	16	0.32
Garden	0.15	3	24	3.60
Totals		30	240	38.24

The expected effective doses, even under conservative assumptions, do not exceed 0.04 mSv. It is around 25 times below the annual doses due to the natural background and the annual dose limit for member of the public. Therefore, no additional radiological measures were necessary to take into consideration during the reconstruction of the facility.

(d) Assessment of the annual doses for free reutilization of the facility

Once decommissioning activities were concluded in all the areas, a final radiological survey was carried out. It included dose rate measurements at the surface of floors, walls and roofs. The reference level in terms of dose rate was achieved in almost all the areas, except around the doorframes. An assessment of the radiological situation in each room was carried out. As the dose rate levels were not significant (7 μSv/h is the maximum dose rate at the surface of the holes), and continuing removing contaminated soil would generate a considerable amount of very low level radioactive waste, the strategy for decommissioning was then change to “entombment”. That strategy was based on the assumption that some construction works were necessary in any case. The “holes” must be filled with soil or other materials, which at the same time would serve as shielding. The deep of the holes was calculated in order to guaranty that after filling them with new material the reference level in terms of dose rate will be achieved.

Taking into account that the half value layers and ten value layers of ordinary concrete (2.35 g.cm^{-3}) are 4.8 cm and 15.7 cm respectively for the energy of Cs-137, it is possible to assume that if the holes are filled with concrete (more than 20 cm in all the zones), the dose rate levels would be reduced more than 10 times. Consequently, the dose rates at the surface of the floor would be less than 0.1 μSv/h. Considering the dimensions of the rooms, the dose rates (above the natural background) at 50 cm and 100 cm from the surface were estimated. The dose rates will additionally be reduced after the floor of the room will be laid with tiles.

Two situations were considered for evaluation of annual dose:

— Residential condition, the exposed person lives in the room and therefore an occupancy factor of 2/3 was considered; and

— Working condition, the exposed person is inside the room 8 hours per day, during 5 days a week and 50 weeks a year.

The results of annual effective dose estimations are summarized in Table 21.

TABLE 21 ANNUAL EFFECTIVE DOSES

Room	Area, [m^2]	Annual effective dose for residential condition [mSv]	Annual effective dose for working condition [mSv]
1	4.55	0.076	0.026
2	29.18	0.149	0.051
3	5.94	0.117	0.040
4	15.05	0.073	0.025
5	7.44	0.044	0.015
6	3.48	0.263	0.090
7	19.44	0.158	0.054
8	19.08	0.044	0.015
garden	20.00	0.038	0.013

As expected, the estimated annual effective dose in all the rooms and in the garden are below the radiological criteria approved for decommissioning: 0.3 mSv/year.

The collective effective dose was also estimated, as a global indicator of the radiological risk for the population. As the total area of the facility is 104 m^2, it was assumed that around 10 persons would stay systematically in the facility, for residential as well as for working conditions.

So the estimated collective dose for residential condition would be 1.11×10^{-3} manSv and for working conditions $0.38 \cdot 10^{-3}$ manSv.

IV-6. GRADING ILLUSTRATED BY THE EXTENT OF DOCUMENTATION (DECOMMISSIONING PLANS) FOR DIFFERENT TYPES OF NUCLEAR FACILITIES USING RADIOACTIVE MATERIAL (CZECH REPUBLIC)

While other examples in this report illustrate grading by analysing differences in the actual approaches or methods used for different types of nuclear facilities, a more phenomenological approach is to compare the length or extent of the decommissioning plans generated for different types of facilities. This approach, however, only makes sense if these decommissioning plans have been prepared under comparable conditions.

Examples for which such a comparison is possible have been found in the Czech Republic. The comparison has been carried out by relating the extent of various sections of the decommissioning plans for an NPP, a research reactor and a laboratory and expressing the results in fractions of the length the sections have in the NPP decommissioning plan (100 %). The information provided in Table IV.4.13 is based on Czech experience and is relevant to the first update of the initial decommissioning plan. This plan is prepared during the life of the facility and is required by the Regulatory Body for operating licence renewal.

The comparison in Table 22 clearly shows that the documentation required for the NPP has in all cases the largest extent, followed by the research reactor and the laboratory. It is clear that such a comparison has to be interpreted with great caution; for example, it would not be possible to infer the required number of pages a decommissioning plan needs to have for a specific decommissioning project from an existing plan for a different facility. However, this comparison shows that facilities of different level of complexity also require safety assessment of different extent.

TABLE 22 EXAMPLE OF THE EXTENT OF VARIOUS SECTIONS OF DECOMMISSIONING PLANS FOR AN NPP, A RESEARCH REACTOR AND A RADIOCHEMICAL LABORATORY, NORMALIZED TO THE DOCUMENTATION FOR THE NPP

		NPP Temelín (2 units WWER 1000/320)	**Research reactor (LVR-15 NRI ŘEŽ)**	**Radiochemical laboratory**
1.	Introduction	-	-	-
2.	Facility Description	1	0.35	0.10
3.	Decommissioning Strategy	1	0.37	0.06
4.	Regulatory Requirements	Regulatory requirements are incorporated into these plans by reference, only short summary is provided		
5.	Decommissioning Activities	1	0.23	0.08
6.	availability of services, engineering and decommissioning techniques	1	0.05	0.05
7.	Waste Management	1	0.08	0.05
8.	Cost Estimate and Funding Mechanisms	1	0.28	0.13
9.	Safety Assessment	1	0.06	0.06
10.	Project Management	1	0.43	0.28
11.	Surveillance and Maintenance	1	0.50	0.17
12.	Environmental Assessment	Environmental assessment is considered to be outside the scope of these plans		
13.	Compliance and environmental monitoring	1	0.20	0.20
14.	Health and Safety	1	0.06	0.06
15.	Quality Assurance	QA is considered to be outside the scope of these plans		
16.	Emergency Planning	1	0.18	0.09
17.	Physical Security and Safeguards	1	0.12	-
18.	Final radiological Survey	1	0.50	0.50
19.	Stakeholder Involvement	Stakeholder involvement is considered to be outside the scope of these plans		
20.	Plan Extent	1	0.22	0.09

IV-7. TIME CONSTRAINTS IN LICENSING PROCEDURES RELEVANT TO NUCLEAR FACILITIES USING RADIOACTIVE MATERIAL AND FACILITIES USING IONISING RADIATION (CZECH REPUBLIC)

In the Czech legislation, namely in the Act No. 18/1997 Coll. (Atomic Act, 90) time limits are established within which SÚJB (State Office for Nuclear Safety) has to decide on the issue of a licence for particular practice.

These time limits are the following:

- Four months, for issuing a licence for siting of a nuclear facility or very significant ionising radiation source (i.e. nuclear reactor);
- One year, for issuing a licence for construction of a nuclear facility or very significant ionising radiation source;
- Six months, for issuing a licence for the first fuel loading into a reactor and 10 days in the case of other stages of commissioning;
- One day, for issuing a licence for restart of a reactor to the criticality following a fuel reloading; and
- 60 days, for issuing other licences (e.g. particular stages of decommissioning of a nuclear facility or category III or IV workplaces, discharge of radionuclides into environment, radioactive waste management, ionising radioactive waste management, etc.).

Note: Nuclear facilities using radioactive material mean in the context of the Czech Republic:

(a) Structural and operational units containing a nuclear reactor;

(b) Facilities for production, processing, storage and disposal of nuclear materials, except uranium ore treatment plant and storages of uranium concentrate;

(c) Repositories of radioactive waste, with the exception of repositories containing only natural radionuclides; and

(d) Facilities for the storage of radioactive waste with activity exceeding values given by implementing regulation.

Examples of workplace Category III:

- Workplace with a stationary industrial irradiator intended for irradiation of foodstuffs and other materials;
- Workplace with a facility containing a sealed radionuclide source intended for radiotherapy, including brachytherapy, classified as a significant source; and
- Workplace for mining and treatment of uranium ore (including mining, treatment, and uranium concentrate handling, decontamination units operation, etc.).

Examples of workplace category IV:

- Construction and operational units containing a nuclear reactor; and
- Spent fuel storage facility.

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