SAFETY ASSESSMENT FOR DECOMMISSIONING

Annex I, Part C

Safety Assessment for Decommissioning of a Nuclear Laboratory

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CONTENTS

FOREWORD

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

The test cases selected for evaluation were:

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

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

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

Part C of Annex I is the safety assessment for the decommissioning of a nuclear laboratory. The laboratory is part of a laboratory complex in which some rooms in the building will remain operational for some time after completion of the decommissioning of the laboratory.

The safety assessment was developed for the Laboratory Test Case of the DeSa project to:

- Demonstrate compliance with safety criteria for protection of workers and public; and
- Define safety controls to be implemented in the decommissioning project.

The Laboratory Test Case illustrates that a small scale facility can lead to a complex decommissioning project and activities. As the facility is located at a multi-facility site, the estimated consequences for workers and public have to be carefully considered, taking into account exposures from other activities ongoing at the site.

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1. INTRODUCTION

1.1. BACKGROUND

Evaluation and demonstration of safety is an essential component of the successful planning, performance and completion of decommissioning of facilities using radioactive material. This has been highlighted by the recently published international safety requirements on decommissioning [1] and the supporting safety guides on decommissioning of nuclear power plants, research reactors [2], medial and research facilities [3], fuel cycle facilities [4], etc. Recognizing the need for exchange of information and experience and consolidation and harmonization of the best experience and lessons learned in the development and review of safety assessment for decommissioning, the International Atomic Energy Agency (IAEA) launched, in 2004, an international project on Evaluation and Demonstration of Safety for Decommissioning of Facilities Using Radioactive Material (DeSa).

The DeSa project aims to develop a harmonized methodology for evaluation and demonstration of safety during decommissioning and to illustrate the application of the methodology through the development of safety assessments for selected real facilities – a nuclear power plant (see Part A of Annex I of this report), a research reactor (see Part B of Annex I) and a nuclear laboratory (Part C of Annex I).

These test cases aim to provide practical illustration of the application of the safety assessment methodology (see main report), to illustrate the need and application of a graded approach due to the complexity and hazards of the facilities, and also to test the regulatory review procedure developed in the DeSa project. The test cases present safety assessments for a selected number of real facilities, volunteered by Member States, with different complexities and hazards following the individual steps of the methodology. By developing these test cases, practical issues related to the use of the methodology was identified, such as the criteria for selection and justification of hazards, scenarios and models, safety related systems, structures and components (SSCs), identification of different types of uncertainties and approaches for their treatment. Decisions on the important input data required, the use of generic vs. site specific data, as well as the depth for safety assessment necessary for demonstration of safety for decommissioning of various facilities with different hazards and risks is addressed.

The safety assessment for the nuclear laboratory presented in this Part C of Annex I should be considered as an illustration of the safety assessment methodology. It is important to note that it is not intended to be representative of all similar facilities. For example, the laboratory test case selected a Pu-laboratory with residual material; however, many laboratories have significantly less material or different isotopes of concern which may allow different level of evaluation of or consideration of additional hazards or SSCs in the safety assessment. The methodology applied in this report remains applicable; however, it needs to be applied to other types of facilities in a manner consistent with the level of hazard and risks. The level of information included in the report, such as facility description, was limited to that information directly relevant or necessary to prepare this test case (e.g. distance to site boundary to support modeling calculations). However, in the development of specific safety assessments for facilities, it is important that these assessments be based on specific requirements of the Regulatory Body and other interested parties, which may require more detailed information.

The safety assessment for the nuclear laboratory decommissioning is presented in this report (Part C of Annex I) and complements:

(a) The safety assessment methodology developed, and documented in a report “Safety Assessment Methodology for Decommissioning of Facilities Using Radioactive Material” (main report);

(b) The recommendations on the application of the graded approach presented in Annex II “Graded Approach to Safety Assessment for Decommissioning of Facilities Using Radioactive Material”; and

(c) The regulatory review procedure presented in Annex III “Regulatory Review of Safety Assessment for Decommissioning of Facilities Using Radioactive Material”.

1.2. SCOPE

Part C of Annex I documents the safety assessment for decommissioning of a nuclear laboratory, comprising five individual rooms within a larger Laboratory Complex, with other rooms which will remain operational. It covers immediate dismantling of the laboratory with the aim of releasing it as part of the release of the building for unrestricted use. Decommissioning of the Laboratory Complex, including demolishing of the building, will be deferred until laboratory operations and their decommissioning are completed at adjacent laboratories within the same building.

The Laboratory Test Case presents a safety assessment that is developed to support the application for authorization of decommissioning of the nuclear laboratory comprised of five rooms. In most cases, this information will be related and based on the main decommissioning activities and assumptions outlined in the decommissioning plan.

1.3. OBJECTIVES

The aim of Part C of Annex I is to illustrate the application of the safety assessment methodology developed as part of the DeSa project to a small size facility, a nuclear laboratory, by applying the graded approach. This report is aimed at presenting an assessment of radiological impact on members of critical groups from proposed decommissioning activities in order to demonstrate to the Regulatory Body that the activities can be conducted safely.

1.4. STRUCTURE

Part C of Annex I is structured as follows:

— Section 1 provides a background, scope and objectives of the report;

— Section 2 provides the safety assessment framework;

— Section 3 describes the nuclear laboratory and the planned decommissioning;

— Section 4 presents the identification and screening of hazards;

— Section 5 presents the evaluation of the scenarios during normal and accidental decommissioning conditions, and the associated modeling;

— Section 6 describes the engineering analysis;

— Section 7 presents the analysis of results and identification of safety measures;

- Section 8 presents the discussion on the approach to and decisions made for application of the graded approach in the development of the safety assessment of the nuclear laboratory;
- Section 9 outlines the confidence building measures applied to the safety assessment development; and
- Section 10 provides a summary of the lessons learnt and conclusions from the decommissioning of the laboratory.

2. SAFETY ASSESSMENT FRAMEWORK

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

2.1. CONTEXT OF SAFETY ASSESSMENT

The nuclear laboratory has been assessed herein for the risks to members of the critical group that could be incurred during decommissioning activities. The laboratory will be decommissioned at the initial phase of decommissioning of Laboratory Complex (see Fig. 2.1). This laboratory covers five rooms (see Fig. 1) and is one of a suite of laboratories, housed within a single facility – Laboratory Complex. Other laboratories within the facility will remain in operation for approximately three years after the completion of the decommissioning of the selected nuclear laboratory rooms (1-5) to provide radiochemical analysis to support other facilities located on the site. Decommissioning of the remaining laboratories is scheduled for a later phase.

On this basis, the goal for the decommissioning of the nuclear laboratory, presented in this test case, is:

(a) All equipment within the five rooms to be removed;

(b) The emptied rooms to be decontaminated and prepared for unrestricted release; and

(c) The laboratory to be maintained in the interim end point condition until the operation and subsequent decommissioning of the remaining laboratories has been completed.

Any residual contamination in the laboratory room that is a part of the building structure will be removed down to an approved acceptance level so that all remaining building material can ultimately cleared for disposal or reuse after demolition. The utilization of the loading dock will continue after decommissioning of the five rooms is completed. The safety analysis for the loading dock remains in a separate safety assessment for the operational facility.

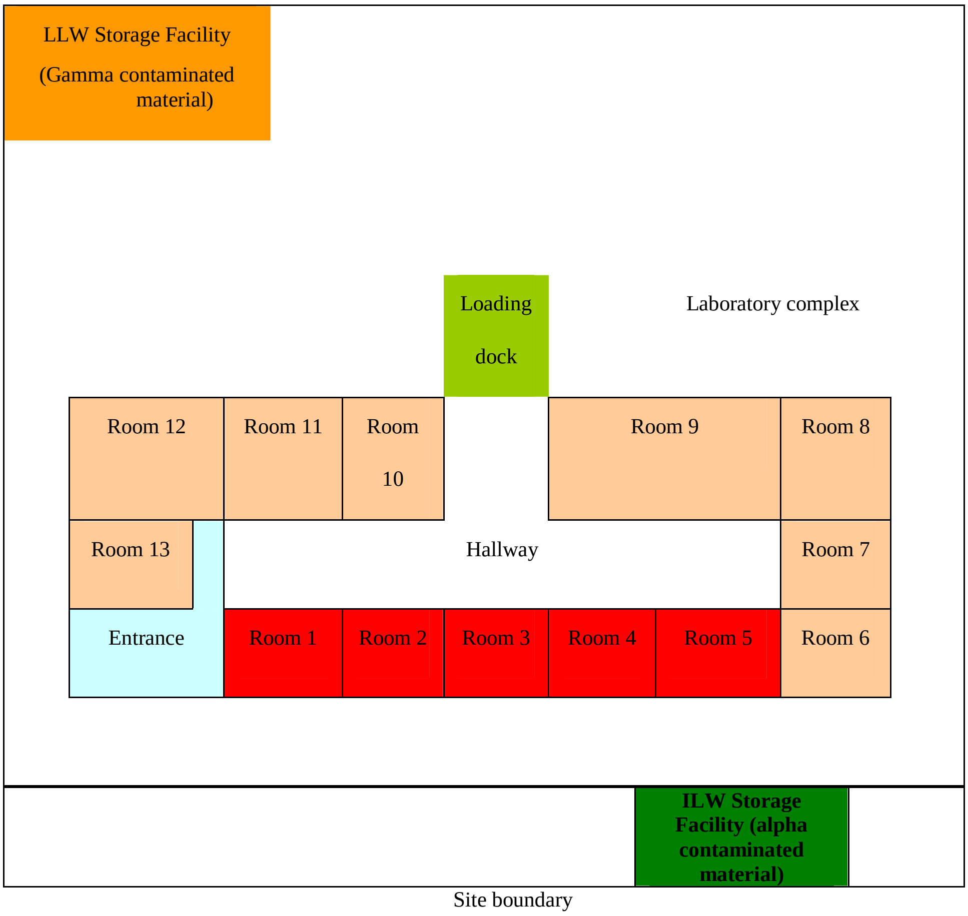


FIG. 1. Layout of the nuclear laboratory in the laboratory complex.

2.2. SCOPE OF THE ASSESSMENT

Only the nuclear laboratory comprising of five rooms (shown in red in 2-1) is in the scope of the safety assessment. The other ten laboratory rooms (rooms 6-13, hallway and entrance) that comprise the balance of the suite of laboratories within the Laboratory Complex remain operational.

This safety assessment evaluated the radiological consequences that may result in exposure to decommissioning workers, other workers on the site and members of the public. Potential exposures resulting from the planned decommissioning activities and any reasonably foreseeable accident conditions during decommissioning are considered. Non-radiological hazard evaluation, i.e. beryllium, is limited to treatment within the Safety Management Programmes or those hazards that could initiate radiological releases, prevent operator response to prevent radiological release, or present an unusual, significant risk in excess of those routinely governed by Safety Management Programmes.

The scope of the assessment includes activities beginning from termination of routine laboratory operations through to the defined end point of this stage of site decommissioning which is five empty, laboratory rooms decontaminated to an operationally clean (readily accessible material removed, penetrations or openings capped, remaining hold-up or contamination cannot be removed without structural modification) state.

The expected radioactive waste generated during decommissioning is low level waste (LLW) and some intermediate level waste (ILW) in accordance with –Ref. [5]. The radioactive waste management activities addressed in this safety assessment are collection, sorting, handling and packaging to meet waste acceptance criteria for storage of radioactive waste on the site. Once properly packaged the waste will be moved to a staging area in an operating portion of the laboratory facility and then transferred to the on-site storage facility for alpha contaminated material.

2.3. OBJECTIVES OF THE ASSESSMENT

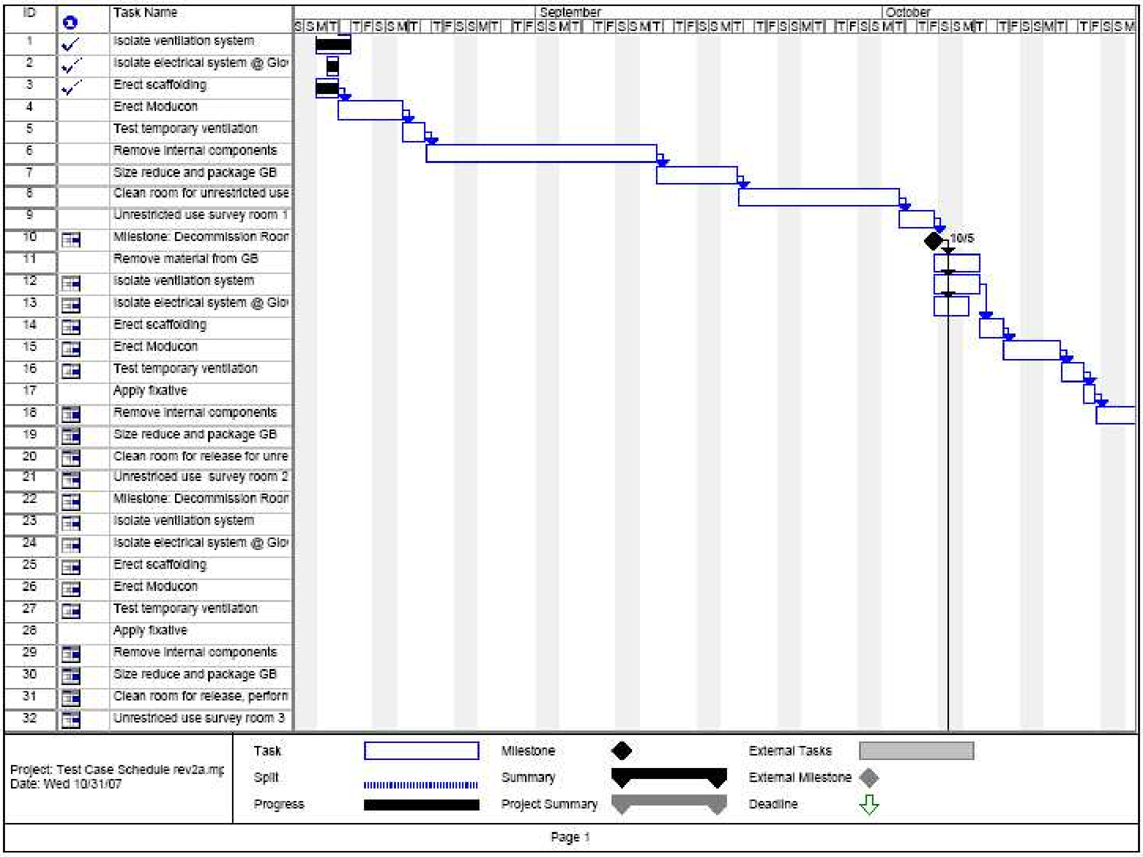
This safety assessment is performed with the aim to demonstrate that the decommissioning activities presented in the associated decommissioning plan comply with safety criteria. This assessment also aims to establish engineered and procedural safety control measures under which decommissioning activities will take place. This safety assessment has been independently reviewed by the Regulatory Review Working Group and the Graded Approach Working Group of the DeSa project (see also Section 7 and Annexes II and III of this report) prior to its finalization.

2.4. TIMEFRAMES

The decommissioning of the nuclear laboratory is envisaged to be performed within 3 years and 6 months. The active decommissioning of the five selected rooms is scheduled to be completed over a period of 6 months, followed by a 3 year period of monitoring until completion of operations in the other laboratory rooms and demolition of the building. The decommissioning of the laboratory is planned to be conducted sequentially (i.e. room by room) as described in Section 3. The decommissioning activities for each room are separated into the following four steps:

- Preparatory activities (e.g. partial isolation of ventilation system) – Step 1;
- Dismantling of contaminated systems and demolition of the non-active parts and clearance of material – Step 2;
- Final activities, including final survey and documentation – Step 3; and
- Care and maintenance of nuclear laboratory until remainder of laboratory building completes mission work and is also decommissioned (approximately 3 years) – Step 4.

The time schedule for the decommissioning of the nuclear laboratory is presented in Fig. 2.



ID		Task Name
1	✓	Isolate ventilation system
2	✓	Isolate electrical system @ Glo
3	✓	Erect scaffolding
4		Erect Moducon
5		Test temporary ventilation
6		Remove internal components
7		Size reduce and package GB
8		Clean room for unrestricted use
9		Unrestricted use survey room 1
10	▦	Milestone: Decommission Roor
11		Remove material from GB
12	▦	Isolate ventilation system
13	▦	Isolate electrical system @ Glo
14	▦	Erect scaffolding
15	▦	Erect Moducon
16	▦	Test temporary ventilation
17		Apply fixative
18	▦	Remove internal components
19	▦	Size reduce and package GB
20	▦	Clean room for release for unre
21	▦	Unrestricted use survey room 2
22	▦	Milestone: Decommission Roor
23	▦	Isolate ventilation system
24	▦	Isolate electrical system @ Glo
25	▦	Erect scaffolding
26	▦	Erect Moducon
27	▦	Test temporary ventilation
28		Apply fixative
29	▦	Remove internal components
30	▦	Size reduce and package GB
31	▦	Clean room for release, perforn
32	▦	Unrestricted use survey room 3

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FIG. 2. Schedule for decommissioning of nuclear laboratory.

2.5. END POINTS OF THE DECOMMISSIONING STAGES

The decommissioning of the nuclear laboratory will be performed in one stage and they can be described as follows:

(a) The working areas of the laboratory rooms will be decontaminated to a level of 0.04 Bq/cm^2 (α-emitters) and 0.4 Bq/cm^2 (for β-fixed emitters), such that demolition of the whole building can be performed at a later date (i.e. after 3 and a half years);

(b) All glove box material and radioactive waste, equipment, internals and services and above ground drains will be removed and packaged for shipment for storage and disposal as solid LLW or ILW, or cleared, as appropriate (see Section 2.6);

(c) Each laboratory room will remain connected to the facility ventilation at the entrance to the room only. Fire suppression systems will be removed from all laboratory rooms. Other facility support systems (process water, electrical systems, etc.) will be isolated and removed; and

(d) Radioactive waste generated during the decommissioning activities will be collected, packaged, handled for storage in the interim storage facility on the site (see Fig. 1) until the waste can be transferred to a disposal facility. The radioactive waste will be comprised of dismantled contaminated systems, equipment, and laboratory materials, as well as secondary waste generated during decommissioning (e.g. personnel protective clothing, waste packaging materials such as bags and tape) produced during decommissioning activities. Waste is not further addressed in this safety assessment once it is removed to the drum storage area.

2.6. REQUIREMENTS AND CRITERIA

This safety assessment is developed to meet the format and content requirements specified in the main report “Safety Assessment Methodology for Decommissioning of Facilities Using Radioactive Material” of this report and follows the IAEA safety standards and recommendations, e.g. [6], [7], [8]. Further criteria are derived from other international recommendations such as the technical requirements [9] to illustrate the application of specific regulatory requirements established in Member States. It is recognized in this report that other specific requirements and criteria may apply in Member States, defined by the appropriate Regulatory Body and other competent authorities.

2.6.1. Dose criteria for planned conditions

The radiological exposure predicted during the planned decommissioning needs to comply with the criteria specified in Basic Safety Standards [6] and IAEA Safety Requirements for Decommissioning [1]. The limit for potential effective doses to workers will not exceed [6]:

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

For the relevant critical groups of members of the public:

- The estimated average effective dose (from all sources) shall not exceed 1 mSv in a year;
- Dose constraint for the Laboratory Complex site is 0.3 mSv/y [10], [11];
- An equivalent dose to the lens of the eye of 15 mSv in a year; and
- An equivalent dose to the skin of 50 mSv in a year.

Not withstanding the dose limits work needs to be planned for optimization of safety so that predicted exposure can be demonstrated to be as low as reasonably achievable (ALARA).

The site is subject to a dose constraint of 300 μSv/y effective dose to the member of the public for the entire Laboratory Complex. The contribution of the nuclear laboratory decommissioning activities will be limited to 1/10th of this annual dose, prorated over the expected duration of decommissioning for the purposes of assessing the release of gaseous and liquid effluent to the most exposed member of the public. The balance of the public dose will be allocated to the remaining laboratory operations. In addition, the materials released from the site will be cleared to 0.1 mSv/y.

2.6.2. Risk criteria for accidental conditions

The safety assessment for the Laboratory Test Case used a classification system for accidental conditions during decommissioning, based on the assessed radiological consequences and assessed frequency without active safety control measures as shown in Table 1 (based on DOE STD 1120-2005, Appendix E) [12]) below.

TABLE 1 ACCIDENT CONSEQUENCE VS FREQUENCY RISK CLASSIFICATION SYSTEM

Probability → ↓ **Consequence**	***Beyond Extremely Unlikely***	***Extremely Unlikely***	***Unlikely***	***Anticipated***
	< 10^{-6}/y	**10^{-4} to 10^{-6}/y**	**10^{-2} to 10^{-4}/y**	**10^{-1} to 10^{-2}/y**
High Consequence **Off Site Public** **>100 - 1 000 mSv** **On site** **> 1 000 mSv**	III	II SAR, safety significant controls	I SAR, safety class controls for public, safety significant for workers	I SAR, safety class controls for public
Moderate Consequence **Off Site Public** **> 10 - 100 mSv** **On site** **> 100-1 000 mSv**	IV	III	II SAR, safety significant controls	I SAR, safety class controls for public, safety significant for workers
***Low Consequence** **Off Site Public** **< 1 - 10 mSv** **On site** **> 10 - 100 mSv**	IV	IV	III	III**

Note:

* SAR – Safety Assessment Report

** While screened from the safety assessment, low consequence/anticipated events are addressed in task analysis. See Section 4.3 for further discussion.

The classification system allows the extent of safety assessment, the independent and the regulatory review to be commensurate with the level of risk presented by each accident scenario selected for assessment. Such a classification system is therefore a fundamental aspect of a graded approach to the safety assessment process.

This risk ranking classification system provides a basis for the selection of engineered and administrative safety measures (further referred to as safety control measures) to reduce risk for each accident scenario to within the safety criteria. This is discussed further in Section 6. Administrative controls for the higher risk categories (i.e. I and II) would need to be included in the facility surveillance programme.

The risk ranking classification system presents results of evaluation of hazard and accident analysis without active safety control measures for the member of a critical group, on-site workers evaluated at 100 m from the release, and decommissioning workers at the nuclear laboratory. High, moderate and low consequence levels are quantitatively defined for the above groups (see Table 1):

(a) *Risk Class I* events are essentially those that could have a significant off site consequence, therefore the public must be protected with higher integrity engineered safety measures and administrative safety measures (with engineered measures being preferred). Events resulting in high off-site radiological consequences must be subject of detailed safety assessment irrespective of assessed frequency of occurrence.

(b) Risk Class II events are those that have lesser off-site consequences than Risk Class I but significant on-site effects. Both classes I and II must also be considered for protection with high level engineering and administrative safety measures. The consideration of control(s) needs to be based on the effectiveness and feasibility of the considered measures. Further controls for Class I and II accident sequences need to be considered over and above the requirements of the accident safety criteria if it is justified on ALARA grounds; sometimes described as defense in depth.

(c) *Risk Class III* events are those with localized consequences. They are generally considered to be adequately protected by the operator's Safety Management Programme. Class III accidents may be considered for defense in depth safety measures, if justified on ALARA grounds. A formal safety assessment would not normally be required, unless required by the Regulatory Body.

(d) *Risk Class IV* events are those with low consequences and do not require additional safety measures, but are considered to be adequately protected by the operator's Safety Management Programme, and consequently a documented safety assessment is not usually required.

It is common practice to classify a facility based on the highest Risk Class arising from the evaluation without active safety control measures in the accident safety assessment. This classification can then be used to define the level of independent and regulatory review of the safety assessment. For example, a Risk Class I facility safety assessment would be subject to full internal independent review as well as regulatory review. A Risk Class II facility may only be subject to internal review unless the Regulatory Body specifically chooses to carry out a review.

2.6.3. Selection of safety control measures from the safety assessment

The safety assessment identified those reasonably foreseeable accident conditions that could occur during planned decommissioning operations. The evaluation then grouped these accidents into categories to assess the maximum radiological exposure that could result without active safety control measure. The specific safety assessment of the five nuclear laboratory rooms can be found in Section 5. When any accident Risk Class I and II accident scenarios selected for detailed safety assessment have been evaluated the objective is to select (and verify the performance of) safety control measures that will reduce the Risk Class for each accident scenario evaluated with active safety control measures to at least Class III, but with a further requirement that defense in depth measures need to be adopted where justified on ALARA grounds.

Figure 3 below is an illustration that shows how active control measures reduce the radiological consequences. In this hypothetical example three control measures are identified as being required to reduce the consequences to an acceptable level. As discussed above further control measure(s) may be adopted if justified on ALARA grounds.

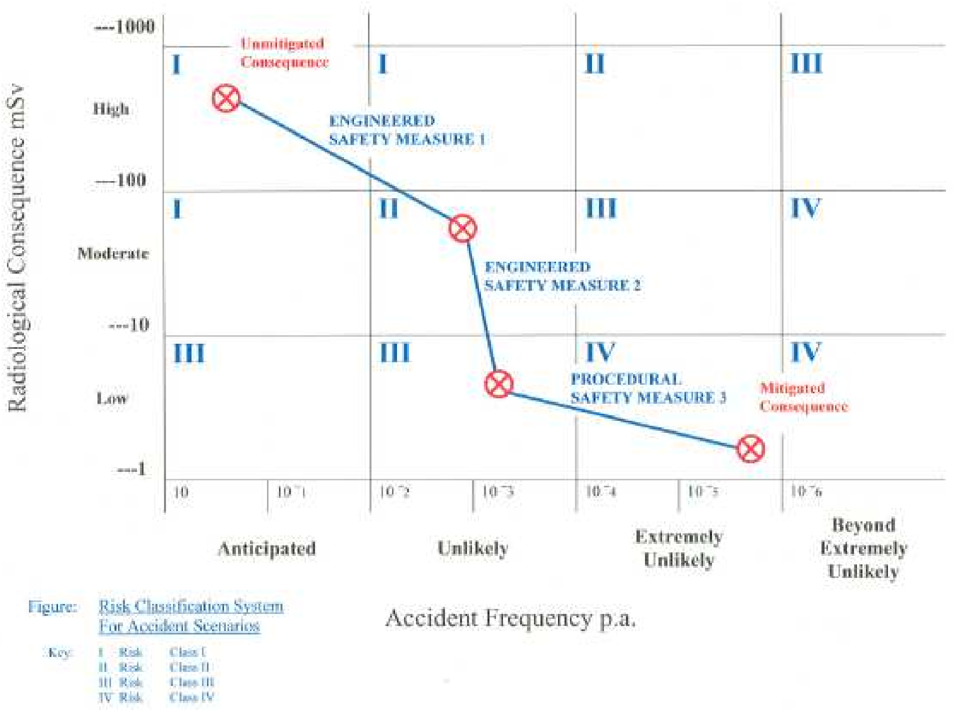

FIG. 3. Control measures and their combination to reduce exposure and or risk.

Physical barriers and other engineered controls are preferable with administrative controls to be used where they are the only practicable means.

The risk class determined for each analyzed event is then used to guide the decisions for additional control selections. Those control measures necessary to reduce to the dose to the public to below for Risk Class I accidents are designated as safety class controls. Additional controls needed to reach Risk Class III or IV for the public and the co-located worker are designated as safety significant controls. Controls deemed necessary to protect the immediate worker from high consequence events (Risk Class I and II) are also designated as safety significant controls. When controls are required to reduce the frequency or mitigate the consequences of an event, additional defense in depth controls are identified, when available. The selection of controls follows the hierarchy of prevention over mitigation, engineered over administrative and passive over active. All controls identified, including those identified as defense in depth controls, are then carried through to the Technical Safety Requirements (TSRs).

In addition to the engineered and administrative measures, identified as part of the safety assessment, there would also be task-specific controls (e.g., personal protective equipment – PPE) identified at a later stage, as needed, based on job hazard analyses as part of the work control process. The task specific controls are predominantly relied upon to provide general worker protection. The actual implementation of work control process is an integral part of the operator’s Safety Management Programmes. The safety assessment does not depend on the use of personnel protection equipment, although it is recognized to provide defense-in-depth and is required by the Safety Management Programmes.

Defence-in-depth

The principle of defense-in-depth needs to be applied to ensure that the facility is operated in a safe manner through the application of multiple layers of protection (safety measures) commensurate with the risk class of the events, and in particular the accidents with most hazardous consequences (i.e. highest level of risk, see Table 1). Defense-in-depth addresses the entire suite of safety controls, encompassing accident prevention measures, the establishment of limits and conditions for safe working, the establishment of safety measures to mitigate the consequence of accidents and other safety measures adopted on ALARA grounds. References, such as [10], [13] and [14] provide additional recommendations for consideration. Many important aspects of the defense-in-depth strategy are also implemented through the operator's Safety Management Programme.

2.6.4. Clearance criteria

The surface areas of the rooms are to be cleaned to a level of 0.04 Bq/cm^2 (for alpha emitters).

The surface areas of the room will be decontaminated to clearance levels defined in Ref. [15]. Clearance is applied on the basis of the concept of 10 μSv/y effective dose to a member of the critical group. Activity concentration values for unconditional clearance of material from control are the ones listed in Table 2 in the Ref. [8].

2.6.5. Classification of radioactive waste

The waste classification criteria used in this Test Case is based on the revised IAEA waste classification as presented in Ref. [5]:

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

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

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

(d) *Low level waste (LLW)* - Low level waste in the classification scheme set out in this publication is waste that is suitable for near surface disposal. This is a disposal option suitable for waste that contains such an amount of radioactive material that it requires containment and isolation for limited periods of time up to a few hundred years (i.e. up to around 300 years). A limit of long lived alpha emitting radionuclides of 4 000 Bq/g in individual waste packages and to an overall disposal facility average of 400 Bq/g has been adopted by an increasing number of countries for near surface disposal facilities.

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

(f) *High level waste (HLW)* - The high level waste class contains large concentrations ofboth short and long lived radionuclides, so that, as compared to ILW, a higher degree of containment and isolation from the biosphere, usually provided by the integrity and stability of deep geological disposal, with engineered barriers, is needed to ensure disposal safety. HLW generates significant quantities of heat from radioactive decay, and normally continues to generate heat for several centuries.

2.6.6. Waste management criteria

The waste will be put in a form that meets the waste store acceptance criteria. The waste acceptance criteria includes specific chemical characteristics, such as land disposal restrictions, which prevent material that could leach into the waste repository, precludes hazardous constituents such as flammable, reactive, oxidizers. Liquid is limited to less than 4l free per the waste acceptance criteria (see Refs. [16] and [17]). Waste from the nuclear laboratory decommissioning activities will be packaged in 200 ℓ drums or may be packaged in standard waste boxes.

In accordance with criteria established by the waste disposal facility, standard waste boxes are packaged at a maximum loading of 320 g Pu-Equivalent (Pu-E)[1]. Because containers in the Laboratory Complex have generally not been assayed to confirm this maximum loading, and to allow acceptance for repackaging of containers determined by assay to be over the 320 g limit, one container in each population of standard waste boxes is assumed to be overloaded by 25%, for a maximum loading of 400 g Pu-E for accident analysis.

For the purpose of this Test Case LLW is limited to a maximum of 3 700 Bq/g of waste matrix. The loading of the IP-2 metal LLW crate is limited to the A2 quantity [14], which is determined based on the isotopic mix of radionuclides present in the waste. To be consistent with other site analyses, each IP-2 box of low level waste is assumed to contain 3 g Pu-E [14]. This value includes sufficient margin to account for possible overloading.

To be consistent with other site analyses, each 200 ℓ drum of LLW is assumed to contain 1 g Pu-E. This value includes sufficient margin to account for possible overloading.

Further detail associated with specific acceptance criteria can be found in the source documents for LLW and ILW respectively [16].

It is important to note that this waste acceptance criteria are specific to the selected disposal site and they are provided as an example of site specific waste acceptance criteria.

Waste must be packaged and prepared to meet the waste acceptance criteria of the interim and final disposal site criteria. The LLW generated through decommissioning activities associated with the Laboratory will be prepared for interim storage at an onsite waste store. The LLW waste acceptance criteria applied in this analysis are taken from Ref. [16]. Intermediate waste will be packaged to meet the requirements of Ref. [17]. No high level waste will be generated through this activity. Further

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[1] Plutonium Equivalent (PuE) is a composite value used to represent all isotopes that may be present. The value specifically incorporates the Am ingrowth associated with the laboratory material based on its age.

treatment and storage of waste is outside the scope of this assessment. Waste is considered in this analysis only to the point that the package is closed and moved to the interim laboratory staging area.

2.6.7. Chemical and other industrial safety considerations

The applicable national occupational health and safety regulations will apply for the control of effects to workers from non-radiological hazards. However, these aspects are not addressed in the DeSa project and in the Laboratory Test Case they are discussed for illustrative purposes. The hazard analysis considers chemical hazards as initiators to nuclear release events. An example of relevant national regulation is used for illustration - threshold quantity in Ref. [18] or Ref. [19] or the threshold planning quantity listed in Ref. [20] used as screening thresholds for those chemicals listed in the facility inventory that require special consideration.

2.6.8. Criticality limits

Based on the form of residual material and its distribution, criticality is not a credible accident in the decommissioning of the nuclear laboratory. While the total mass of fissile isotopes that remain in inventory may exceed 3% of a minimum critical mass, these materials are dispersed throughout the facility. The nature of the planned decommissioning process will not separate or combine materials such that a minimum critical mass of 450 g Pu in solid form [21] would be collected. Material that is packaged to meet waste acceptance criteria properly considers the volume of free liquid, size of container and fissile material to ensure that packaged waste does not represent a criticality hazard [17].

2.7. ASSESSMENT OUTPUTS

The safety assessment the effective doses to workers and to members of the public, both during normal decommissioning operation and during accidental conditions. The results of the assessment are then compared with the relevant criteria set out in Sections 2.6.1 and 2.6.2.

2.8. SAFETY ASSESSMENT APPROACH

The safety assessment framework depicted in Fig. 4 (discussed in detail in the main report) was followed in a deterministic manner using the HAZOP[2] methodology [22].

The HAZOP analysis technique uses a systematic process to:

(a) Identify possible deviations from planned activities; and

(b) Ensure that appropriate controls are in place to prevent or mitigate potential accidents.

The HAZOP technique systematically considers all credible deviations from normal conditions. Characteristics of HAZOP include:

- A systematic, structured assessment conducted by a multidisciplinary team using a brainstorming session to generate a comprehensive list of upset conditions and potential control mechanism to prevent or mitigate events;
- Used most as a system or process level risk assessment technique; and
- Generates primarily qualitative results, although some basic quantification is possible.

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[2] Hazard Operability Study (HAZOP)

The HAZOP methodology identifies the hazardous material, energy sources and processing parameters associated with the planned decommissioning activities and it identifies potential events that could result in the release of material or expose receptors to unnecessary potential for harm. Standard hazard analysis techniques were applied which use a series of screening tools and hazard binning techniques to focus on a set of events that are representative and bounding, so that there is assurance that the resulting controls offer an appropriate level of protection. The evaluation is iterative and safety control measures are reviewed and consolidated as safety control measures are tailored to provide the minimum set of manageable controls to address the suite of hazards presented by a proposed decommissioning activity.

The approach to hazard analysis (further described in Section 4) adopted accident screening criteria used to eliminate any low consequence/frequency accident sequences that do not make a significant contribution to overall risk. Results of screening performed for the Laboratory Test Case are provided in Section 4.

Probabilistic methods were also used to establish appropriate controls and demonstrate that the overall risk of planned decommissioning activities meets the established criteria.

The code Radidose 1.4.3 [23] was used to perform dose calculations and to determine the effect of control measures applied.

The MACC code [24] was used to perform atmospheric dispersion calculations.

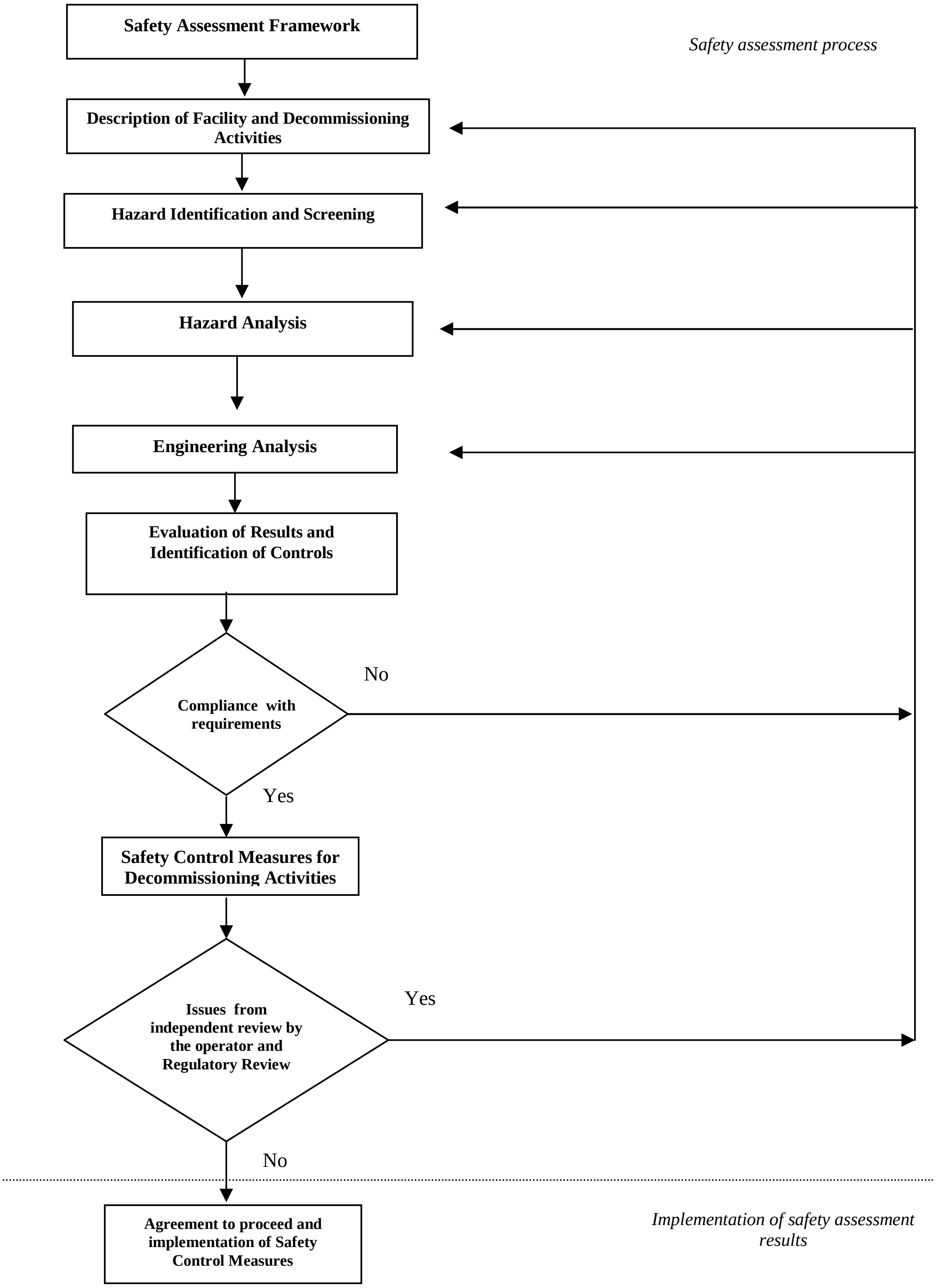

FIG. 4. The safety assessment process.

2.9. EXISTING SAFETY ASSESSMENT

The safety assessment for the nuclear laboratory in its operational phase has been taken into account in planning the decommissioning work and was applied where applicable to decommissioning activities. The site criteria associated with site location, geophysical and meteorological characteristics can be found in the operational safety assessment and are not repeated in detail in this test case. The remaining ten laboratories in the Laboratory Complex remains in operation, so that the conditions of the facility, land use, environmental pathways remain valid for the decommissioning safety assessment of the nuclear laboratory. No alteration of structural characteristics of the Laboratory complex that affect confinement or provide support to the nuclear laboratory is anticipated; therefore the operational safety analysis will be used in the development of the decommissioning safety assessment of the nuclear laboratory. Facility layout (Fig. 2.1) and description were directly incorporated as they were representative of the facility at the initiation of the decommissioning mission. Hazardous material inventory in the nuclear laboratory and processing information was used for reference only as specific inventory information was updated to reflect the bounding case for decommissioning. Natural phenomena and events from the operational safety assessment were used as bounding analysis and not repeated in the decommissioning safety assessment because of the limited remaining life of the nuclear laboratory and time that the hazard will be present. No further reference to it was considered necessary.

The decommissioning safety assessment of the nuclear laboratory also made use of the operational safety assessment that addresses the waste handling and storage at the Laboratory Complex.

2.10. SAFETY MANAGEMENT MEASURES

It is assumed that the operator of the Laboratory Complex has in place an effective safety management system that will ensure that all the planned decommissioning activities will be carried out in accordance with the operator’s policy and procedures and that staff and contractors will be suitably qualified and experienced for the work that they undertake. Consistent with approach illustrated in Fig. 2 of the main report, institutional controls will be implemented through the Laboratory Complex and the nuclear laboratory safety management programmes. The following programmes (consistent with those included in the decommissioning plan for the nuclear laboratory, see Appendix A of the main report) are included in the Laboratory Complex safety management system that is also applicable during execution of decommissioning activities described within this document:

- Radiation Protection;
- Nuclear Criticality Safety;
- Industrial Safety;
- Emergency Preparedness and Response;
- Fire Protection;
- Work Control (planning and authorization of planned work activities);
- Waste Management;
- Environmental Management;
- Quality Assurance and Record Keeping;
- Inspection and Maintenance;
- Conduct of operations (procedures and drawings);
- Configuration Management and Engineering; and
- Physical protection and Security.

The safety management system will ensure good site security; define interfaces of the operator with the independent reviewer and relevant regulators as work progresses. It will also require the adequate task-level safety assessment of individual work packages to ensure in particular, that all non-radiological hazards have necessary safety controls for the protection of workers identified, demonstrated to be in compliance with the risk criteria and further refined to demonstrate ALARA as appropriate. These programmes are not specifically credited in this safety assessment, but are assumed to be in place to provide fundamental bases for initiation of assessment.

3. DESCRIPTION OF THE FACILITY AND DECOMMISSIONING ACTIVITIES

3.1. SITE DESCRIPTION AND LOCAL INFRASTRUCTURE

(a) Nuclear laboratory as part of the laboratory complex

The Laboratory Complex provides radiochemical analytical services for the site, where other facilities are located. It is located in consists of a suite of interconnected laboratories (see Fig. 1).

Constructed in 1955, the Laboratory Complex is 140 m long (East-West) by 30 m wide (North-South). The Laboratory Complex is interconnected by three corridors; the central one giving access to all laboratories, the north service corridor giving access to the north laboratories and the south side corridor giving access to the south side laboratories. The south side corridor, added during the 1970's general refurbishment, is out of the controlled area and can be used as an emergency exit. The Laboratory Complex contains both operational and no longer required laboratories (see Table 2), including shielded cells, fume cupboards and gloveboxes. All laboratory facilities are connected to a common utility system that provides service to all individual laboratories.

TABLE 2 NUCLEAR LABORATORY PROFILE

Rooms	Contents	Comment
Laboratory room No. 1	Five gloveboxes	Empty and isolated from the facility ventilation system.
Laboratory room No. 2	Three gloveboxes	Connected to the building ventilation system, containing vials and residual Pu material.
Laboratory room No. 3	Six gloveboxes	One set of two interconnected one set of four interconnected, connected to the building ventilation system, containing with surface contamination.
Laboratory room No. 4	Two small gloveboxes	These boxes (not interconnected) were moved from their original locations and are not connected to the ventilation system. They are wrapped ready to be moved.
Laboratory room No. 5	One glovebox	Disconnected from the building ventilation system, empty of material.

Each of the gloveboxes has been used for operations with plutonium at some time during its operation.

FIG. 5 Example of three interconnecting gloveboxes in laboratory room 2.

(b) Local infrastructure and population

The Laboratory Complex is located 1580 m from the site boundary. About 800 persons are working on the site. Most of the immediate surroundings of the site are farmland used for sheep farming. About 2 km south of the site there is a settlement with about 200 single family houses. The population within a distance of 10 km from the site is about 5 000. The population between 10 and 15 km away from the site is about 25 000.

(c) Meteorology

The meteorological conditions at the nuclear laboratory have been evaluated and summarized in order to characterize the site climatology and to provide a basis for predicting the dispersion of gaseous effluents. Meteorological data from the national Meteorological Office indicate an annual mean wind speed of 4.9 m/s (17.64 km/h or 11.0 miles/h). The prevailing wind direction is wind from the south. The maximum five-second wind speed is 31.3 m/s. (112 km/h or 70 miles/h) from 200 degrees with respect to true north.

The climate is typical of a semi-arid region, with generally mild temperatures, low precipitation and humidity, and a high evaporation rate. Vegetation consists mainly of native grasses and some mesquite trees. During the winter, the weather is often dominated by a high-pressure system located in the east of the site and a low-pressure system located off the coast. During the summer, the region is affected by a low-pressure system normally located to the north.

The normal annual total rainfall as measured is 46.1 cm. Precipitation amounts range from an average of 1.22 cm in March to 7.95 cm in September. Record maximum and minimum monthly totals are 35.13 cm and zero respectively.

For the purpose of the safety assessment presented in the Laboratory Test Case, conservative assumptions of a standard weather situation at the site with westerly winds are used. Standard Meteorology was applied: wind speed 6.4 km/h from 270° Stability D, no precipitation. This is noted to be generally representative of the mean weather conditions described above.

3.2. SAFETY RELATED STRUCTURES, SYSTEMS AND COMPONENTS

The laboratory gloveboxes are constructed of stainless steel and Perspex (acrylic sheeting), with the largest sized at some 2 m long x 1 m wide x 1 m high. As noted above, post operational cleanup and fixing of internal contamination has been done. For those gloveboxes that have been disconnected from the building ventilation system the ventilation extracts are isolated at the glove box outlet valves, and the gloveboxes now are ventilated via HEPA filters to the laboratory environment, in readiness for size reduction within a Modular containment facility. The ventilation systems associated with removed enclosures have been removed to approximately about 3 m above the floor. The overhead consists of ductwork, fire protection system piping and various other support systems (water, electric, air, etc.). The primary sources of residual fissile material is that remaining in gloveboxes, minimal if any in b-boxes and hoods, ductwork, recirculation tunnel and filter plenum, and residual liquids in associated process piping.

The Modular containment system consists of sets of glass reinforced plastic panels that can be assembled to form a self-supporting enclosure around an object of almost any size. Panels are handled and installed manually and are self-supporting. Originally designed for operations involving plutonium, the Modular containment system has a 15 year track record of preventing the spread of toxic/radioactive materials. Modular containments typically operate at a depression of 10 Pa but are still effective at 100 Pa and above. Designed to be easily decontaminated and flat-packed for re-use, the internal faces of the panels are smooth and unblemished with the option of a strippable coating to aid decontamination after operations with a higher contamination factor. The designs may also incorporate entry and exit areas with optional showers, ports for material 'postings' or services, windows for viewing and lighting, mobile ventilation and filtration facilities and lifting equipment, as the job requires.

For the glovebox decommissioning, the Modular containment will consist of a central enclosure built around each glovebox or connected suite of gloveboxes. It will incorporate personnel access and a standard bagging drum posting port. Typical indicative arrangements of Modular containments for the planned tasks are shown below in Fig. 6. Access for personnel will be via a three stage arrangement as detailed in the Ref. [25]. The stages are separated by barriers and provision is included for air sampling at the final entry point.

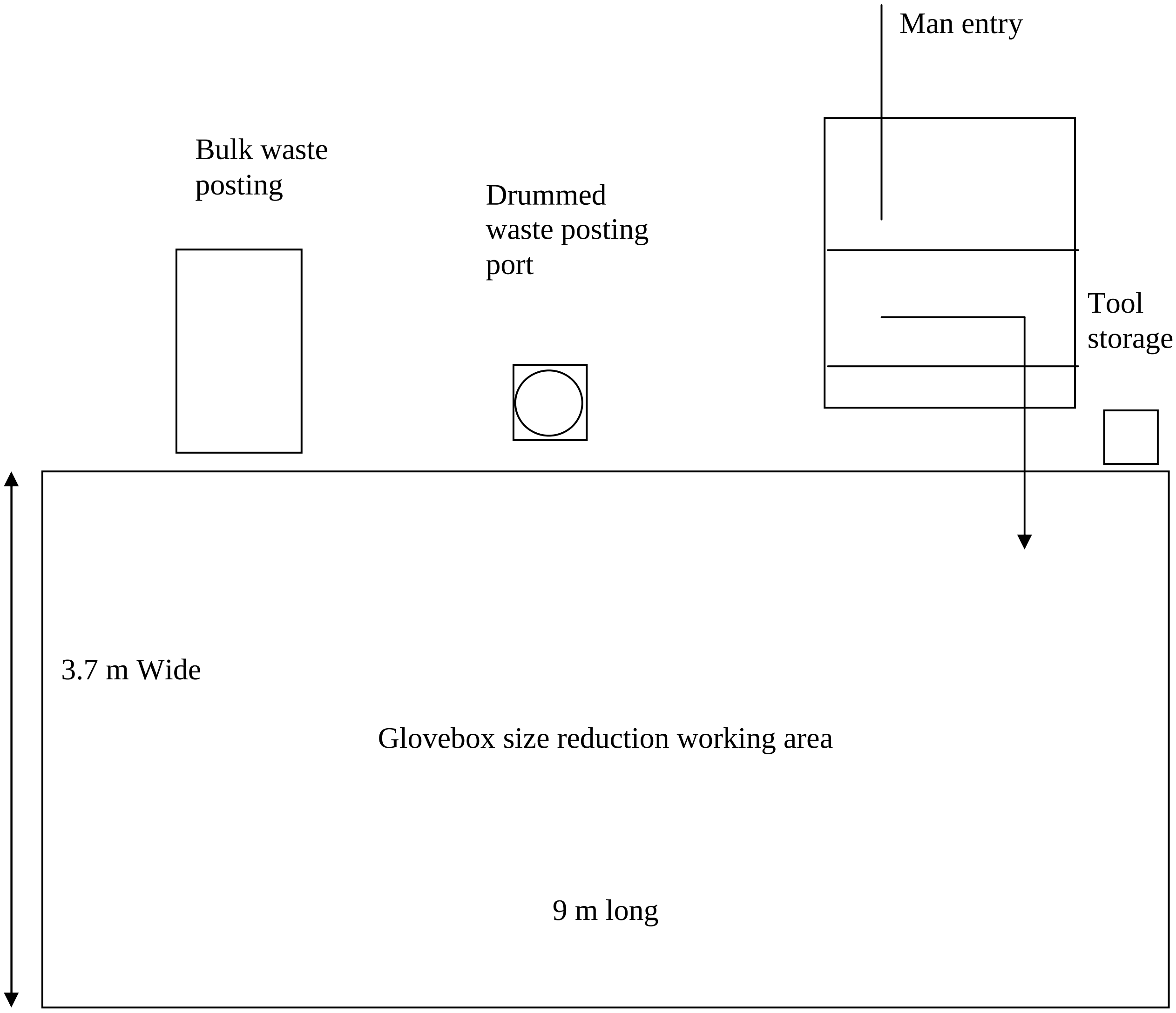

FIG. 6. Layout of Modular containment facility.

Ventilation of the Modular containment will be via a dedicated portable ventilation unit, fitted with a HEPA filter and discharging into the laboratory.

Following assembly, taping of seams between panels and initial inspection of the Modular containment, the panels will be coated internally with a tie down spray coating to act as a sacrificial layer protecting the panel surfaces. Fire retardant sheeting will be used to cover the floor and will be taped into position against the walls of the Modular containment. The ventilation unit will be commissioned and tested with particular reference to air flow velocities through the entry aperture and any leakage points.

After each glovebox or glovebox suite has been size reduced, the cleared internal area of the Modular containment will be cleaned by means of a dedicated vacuum cleaner to remove loose contamination, swarf or dust. The vacuum filter is designed to minimize fine material carryover into the postfilter. The inside cylindrical part of the vessel holds three filter bundles, each with 19 pleated elements. The filters are constructed of 316 ℓ stainless steel sintered fiber, and designed to remove 99.97% of all solids greater than 0.3 micron. The surfaces will then be sprayed with a further tie down coat to fix any remaining contamination. The double layer will then be stripped off and consigned as waste, following which the Modular containment structure must be able to be monitored, declassified and dismantled.

The laboratory high-efficiency particulate air (HEPA) filter unit consists of four filter tiers, stacked vertically. Each filter tier is one filter high by two filters wide. Filter elements measure 61 cm. wide × 61 cm high × 30 cm and are arranged in each filter section with bubble-tight dampers in the inlet and outlet from each tier. The off-gas flow is split equally between the tiers. The off-gas piping and the process HEPA filters housing are insulated to maintain temperature in the off-gas system above 120°C. The HEPA filter housings will be manufactured of AL6XN alloy or equivalent, meeting relevant standards Ref. [26] and Ref. [27] in accordance with Section VIII, Division 1 of the ASME code, although the filter housing will not operate at sufficient pressure to mandate compliance with ASME boiler codes. All piping meets the requirements established in Ref. [28]. A fire screen is located upstream of the HEPA filters to prevent hot embers from reaching the HEPA filter system.

Each filter tier is comprised of a prefilter, a test inlet section, a first-stage HEPA, a test combination section, a second-stage HEPA, and an outlet test section. The design flow air flow in cubic meters per minute for each filter within the filter tier is 35 m^3/min, which gives a total flow through each tier of 70 m^3/min, providing a treatment capacity of 280 m^3/min for all four filter tiers. The design process off-gas filter is 192 m^3/min, thus the process can operate with only three tiers online.

The HEPA filters are designed for a maximum temperature of 225°C and a pressure range of −25 to + 34 kPa. The normal operating temperature is 150°C and a normal pressure is −18.96 kPa. The filters are constructed of boron glass microfiber media in a stainless-steel frame for high temperatures.

Each fire area in the facility is separated from adjacent fire areas by fire barriers as required by the International Building Code. All fire barriers are required to have a 2 h fire-resistance rating, with 1 h fire-resistance rated opening protective features (such as doors, dampers). Per established criteria for fire area boundaries Ref. [29] these barriers will be increased to a 2 h fire resistance rating, with 1.5 h fire-resistance rated opening protective features.

The laboratory is sprinklered and also expected to meet the requirements of Ref. [30]. All sprinkler systems within the facility are automatic wet pipe systems with the exception of the manual deluge (HEPA filters). The sprinkler system design complies with Ref. [31], [32] or other applicable NFPA documents.. Fire suppression system design includes the use of an existing dedicated fire water supply and mains. The existing supply system is capable of providing a firewater flow of at least 19.87 m^3/min [33] (at a residual pressure of 137 kPa) for at least 4 h.

A fire detection system will detect the presence of a fire and activate the alarm system to allow for prompt personnel evacuation and initiation of containment and suppression activities. The alarm system will automatically notify the Fire Alarm Centre of all alarm conditions. The alarm system will be on an uninterruptible power supply (UPS) or have battery backup in accordance with the requirements of Ref. [34], Automatic detection is provided in the HEPA filter system (heat detection).

The building design includes the provision to contain at least 20 min. of flow from the fire water system in order to contain possible contamination. Firewater collection is provided for all areas of the process building to prevent contamination spread. Collection will include the use of berms and sumps to control the spread of firewater during and after a discharge.

In accordance with Ref. [29], fire screens will be located ahead of all HEPA filter banks and plenums for the purpose of reducing flame propagation and to prevent glowing embers from reaching the final

HEPA filters. Prefilters will be located ahead of HEPA filters to reduce the percentage of large particles that would potentially plug the HEPA filters.

The building ventilation system HEPA filters have both manual and automatic water spray systems. The HEPA filters are also equipped with metal demisters between the water spray heads and the first series of HEPA filters. The water spray heads are located in front of (upstream) of the prefilter. Water spray is designed in accordance with Refs. [32] and [29].

3.3. RADIOACTIVE INVENTORY

Calculated plutonium inventories for the gloveboxes in all five rooms of the nuclear laboratory are given in Table 3 and Table 4., based on non-destructive assay measurements taken after completion of the laboratory operation, following post operational cleanup. The results are further broken down into individual glovexbox suites, to demonstrate their characterization. The residual contamination values are used as the measured or nominal value based on non-destructive assay plus approximately two (sometimes 1.96) times the standard deviation (N + 2σ) or as the Lower Limit of Detection (LLD), which is the upper bound of the 95% confidence interval, where available.

The “best case” results assume that all the plutonium is present on the windows and ports, whereas the “worst case” results assume it is present in the middle of the glovebox floor.

With time, the ratio of plutonium isotopes will change subtly due to their different half-lives, but more importantly, the in growth of ^{241}Am will alter the ratio of alpha emitters significantly. Dose conversion factors based on the International Commission on Radiological Protection (ICRP) [35] are used for all scenarios. All forms of plutonium are modeled with a moderate lung clearance rate (similar to the ICRP-30 solubility class W [36]), due to the presence of significant amounts of fluorides in the nuclear laboratory operations. In accordance with ICRP-68 recommendations [35], a particle size of 5 μm is assumed for all without active safety control measure scenarios

TABLE 3. FINGERPRINT FOR GLOVEBOX PRIOR TO DECOMMISSIONING

Percentage of Total Alpha Activity [%]					**Percentage of Total Beta Activity ^{241}Pu [%]**
^{238}Pu	**^{239}Pu**	**^{240}Pu**	**^{242}Pu**	**^{241}Am**	
30.7	21.7	29.8	0	17.8	100

TABLE 4 GLOVEBOX RESULTS FROM NON-DESTRUCTIVE ANALYSIS

	Before Post Operational Cleanup		**After Post Operational Cleanup**	
	Best case [g]	**Worst case [g]**	**Best case [g]**	**Worst case [g]**
Laboratory Room 1				
Glovebox 1.1	0.5	1.4	0.25	0.65
Glovebox 1.2	0.6	1.7		
Glovebox 1.3	0.5	1.4	0.25	0.65
Glovebox 1.4	0.3	0.8		
Glovebox 1.5	0.2	0.7		
Laboratory Room 2				
Glovebox 2.1	441.6	1 413.2	80	412
Glovebox 2.2	404.9	1 417.0	120	413
Glovebox 2.3	151.4	1 444.5	100	442
Laboratory Room 3				
Glovebox 3.1	0.09	0.28	Trace	Trace
Glovebox 3.2	0.08	0.25	Trace	Trace
Glovebox 3.3	0.13	0.37	Trace	Trace
Glovebox 3.4	0.08	0.25	Trace	Trace
Glovebox 3.5	0.11	0.35	Trace	Trace
Glovebox 3.6	< 0.03	< 0.08	Trace	Trace
Tunnel	< 0.03	< 0.08	Trace	Trace
Laboratory Room 4				
Glovebox 4.1	0.08	0.25	Trace	Trace
Glovebox 4.2	0.11	0.35	Trace	Trace
Laboratory Room 5				
Glovebox 5.1	< 0.03	< 0.09	Trace	

The Laboratory gloveboxes in the five rooms were cleaned up after the end of operation and were coated internally with ‘Detex’ tie-down coating.

3.4. OPERATIONAL HISTORY

The operational history of the nuclear laboratory can be summarized as follows:

(a) *Laboratory Room 1* gloveboxes were used for preparation of samples for analysis including dilution, evaporation, ion exchange, solvent extraction and centrifugation. Wet operations included fuel dissolution, flowsheet development and general plutonium/uranium chemistry. Dry operations involved re-packing and weighing fissile material and fuel ball-milling operations.

(b) *Laboratory Room 2* gloveboxes were used mainly for plutonium nitrate concentrate analysis and preparation of plutonium solutions for use as controls and standards for analytical techniques elsewhere.

(c) *Laboratory Room 3* gloveboxes were used for preparation and examination of various metallic and ceramic samples of non-irradiated uranium/plutonium oxide/silicide or carbide fuels, gas chromatography and small scale experiments on wet oxidation of organic materials. General chemical work on the storage of plutonium and uranium solutions was also undertaken.

(d) *Laboratory Room 4* gloveboxes were moved from the X –Ray Diffraction Laboratory where they were used for sample preparation.

(e) *Laboratory Room 5* glovebox was used for determination of the density of radioactive material.

The review of the operational history of the nuclear laboratory showed that the facility has been consistently used as a laboratory facility as described above and that spill events, including nitric acid, resulted in fixed contamination in portions of the concrete floor. These areas have been covered and identified using magenta paint. There are no other significant events reported during the operational history of the nuclear laboratory.

3.5. DECOMMISSIONING ACTIVITIES AND TECHNIQUES

The approach for size reduction is therefore to construct a Modular containment around each individual glovebox or glovebox suite (where they are interconnected) in their current location. For all gloveboxes, any installed glovebox extract ventilation pipework will first be disconnected, allowing it to be size reduced along with the gloveboxes. This initial stage of work (service disconnection with no invasive work) will be undertaken within a tent constructed above each set of gloveboxes.

The decommissioning plan for the nuclear laboratory defines the progression of the decommissioning process as specific steps of hazard reduction, which correspond to the removal of specific types of systems and/or equipment. In general, these stages are:

(a) Glovebox/component removal;
(b) Duct removal; and
(c) Cold strip-out (removal of embedded components requiring structural demolition).

Each step is not applicable to every defined confinement area. For example, decommissioning of the five rooms will involve glovebox/component removal, and duct removal. Final survey and surveillance and maintenance will be performed simultaneously at the end of the decommissioning process.

The overall sequence of activities will be, for each room in turn, to separate the gloveboxes from the ventilation extract system, isolate the live system by fitting an end cap and dismantle the ventilation pipework for later size reduction within the Modular containment. This first stage of work will be done by tenting.

The Modular containment will then be assembled around the appropriate glovebox units and prepared for active decommissioning operation. Size reduction of glovebox and the dismantled ventilation pipework will take place via a series of airline suit entries, following which the Modular containment will be decontaminated and dismantled in situ, ready for re-use if appropriate.

The portions of the ventilation systems that remain will be rebalanced and the configuration management program will be updated to reflect the new configuration. Finally, the maintenance system will be updated to delete all scrapped items from the maintenance schedule.

Figures 7 and 8 present a pictorial representation of the size reduction techniques.

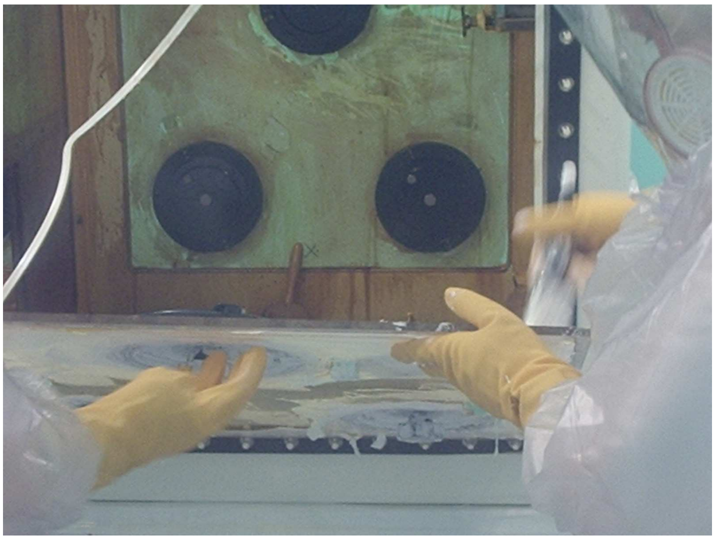

FIG. 7. Removal of the first glovebox window, with freshly exposed surfaces being coated in tie-down coating.

FIG. 8. Progressing the application of fresh tie down coating, to supplement coating already in place, prior to size reduction commencing.

The major decommissioning work activities that will be performed include:

(a) Administrative operations;

(b) General operations; non-radioactive, hazardous material handling;

(c) Radioactive waste and contaminated equipment handling; and

(d) Decommissioning – decontamination, dismantlement, and demolition of system structures and components.

The following detailed decommissioning tasks will be undertaken:

Step 1 - Preparations

- Remove as much as possible of any remaining glovebox external services, particularly electrical;
- Apply tie down coating; and
- Reduce the height of the current wooden cladding around the glovebox for protection during scaffold erection. Fitting of new wood covers over the individual windows is an alternative. Combustible material in the area have to be reduced before using cutting tools.

Step 2 includes the following set of activities (see also Fig. 9 to Fig. 12):

- **Glovebox size reduction**

- Erect Modular containment;
- Remove first window, apply tie down coating to assessable internal areas if not previously coated;
- Repeat for all windows;
- Remove remaining services;
- Stitch drill panels to allow access for reciprocating saw;
- Using the reciprocating saw and band saw to size reduce the glove box panels;
- Size reduce glovebox base using nibbler; and
- Size reduce glovebox stand using band saw.

- **Ventilation system dismantling**

- Erect scaffold platforms and three-stage tents to permit removal of the ventilation pipework between the glovebox flange (under the local isolation damper) and as close as practical to the laboratory ceiling.
- Remove the duct sections, securely sealing end caps as near to roof level as practicable.
- Dismantle tent and scaffold.

- **Radioactive waste management**

- Radioactive waste generated during the decommissioning process will be packaged and stored in an interim package facility until the balance of facilities is decommissioned and material can be transferred to a designated disposal facility. The waste material from the nuclear laboratory is packaged in place and transferred immediately (by the end of the working shift) to an active staging area governed by the operational safety case for the remaining laboratory facility.

— The handling of radioactive waste generated by decommissioning activities is included in this safety assessment as it is governed by this document until it is transferred to the storage facility.

Step 3 - Survey

Iterative process to survey surface and remove areas that do not meet acceptance criteria.

Step 4 - Monitoring and surveillance

Access will be controlled to cleaned areas. No activities are allowed. Personnel enter to perform surveillance and inspection activities on an annual basis. No active systems are left in place; therefore no maintenance activities are expected.

FIG.9. Temporary low voltage transformer to provide tooling supplies.

FIG. 10. Temporary air handling unit to provide extract from containment (via pre-filter and HEPA filters) and maintain set airflow rate across all barriers.

FIG. 11. Temporary air particulate monitors installed to check for activity in the containment change areas.

FIG.12. Interior of containment prior to commencing size reduction. Temporary fluorescent lighting (with internal battery backup in case of power loss) installed over roof lights. Floor covered by several layers of non-slip sheeting. Strippable coating (tinted) applied. All 'low active' glovebox peripherals removed.

3.6. WASTE MANAGEMENT

3.6.1. Management of radioactive waste

Decommissioning and size reduction of the gloveboxes and ventilation pipework will result in the generation of LLW and ILW to be handled in 200 ℓ drums.

No high level waste will be generated through this activity. Further treatment and storage of waste is outside the scope of this assessment. Waste is considered in this analysis only to the point that the package is closed and moved to the interim laboratory staging area. There is no liquid waste expected to be generated during glovebox decommissioning.

ILW drums having a nominal 200 ℓ capacity are assumed to be packaged at a maximum loading of 200 g Pu-E (see Section 2 and Ref. [37]. Because containers in the Laboratory Complex have generally not been assayed to confirm this maximum loading, and to allow acceptance for repackaging of containers determined by assay to be over the 200 g limit, one container in each population of 200 ℓ ILW drums is assumed to be overloaded by 25%, for a maximum loading of 250g Pu-E.

ILW is considered to comprise the basis of each glovebox, pus port and window seals and decontaminating materials (secondary waste), approximately 4.8 m^3 of waste generated per glovebox. The balance assumed to e LLW, is estimated to be approximately 4 times the volume of ILW (see Table 5). It is the intention to stream ILW and LLW within the containment (see Fig. 13).

TABLE 5 WASTE BREAKDOWN

Laboratory Room	**Number of ILW drums***	**Volume of ILW [m^3]**	**Number of LLW drums***	**Volume of LLW [m^3]**
1	5	1.00	20	4.00
2	9	1.80	36	7.20
3	2	0.40	8	1.60
4	6	1.20	32	6.40
5	1	0.20	4	0.80
Total	**23**	**4.8**	**100**	**20.00**

* Note: waste was estimated by drum quantity, note that standard waste boxes may also be used to package waste materials; however the volume produced will be consistent with these values

FIG. 13. Waste posting port in containment.

The municipal and county authorities have established a number of requirements that impose a number of duties on the waste producer with regard to separation, storage, handling, transport and disposal of building - and construction waste. A large proportion of the non-radioactive waste generated from decommissioning of the nuclear facilities will be building and construction waste, which is defined as all waste generated in:

(a) The construction of new buildings;

(b) Renovation, including sand blasting, etc.;

(c) Demolition;

(d) Construction and repair of roads;

(e) Construction and repair of utility systems; and

(f) Other construction work.

The quantity of waste does not exceed 10 m^3, therefore it is not required that the waste be separated in situ into the following recyclable materials: paper; cardboard; iron and metals; concrete and brick; and gravel and sand; however, this requirement will be applicable to further decommissioning of the laboratory facility. If hazardous waste substances such as lead, cadmium and asbestos are found, they will be handled in accordance with the rules governing those substances.

3.6.2. Clearance of the rooms

Prior decommissioning the rooms and structures in the building were classified in three groups, Class A, Class B and non-impacted, according to the likelihood of finding any contamination or activation (see Fig. 14):

— The surfaces in the group with the highest likelihood of being contaminated have been classified as Class A and will be measured for clearance to a coverage of 100%; and

— Those areas in the middle group, Class B, will generally be measured to a coverage of 10 to 50%, while a few random measurements will be carried out on the non-classified surfaces that have had no or very little contact with radioactive materials.

The laboratory rooms 1-3 and 5 were classified as Class A. Part of the Laboratory room 4 was classified as Class A and part as Class B. There were no non-classified surfaces within the nuclear laboratory area. Survey of outside areas will be deferred to subsequent stages of decommissioning of the Laboratory Complex.

Clearance measurements for the rooms will be carried out after completion of decommissioning as a combination of contamination measurements with hand-held instruments and spectrometric measurements with Ge-detectors or NaI-detectors. Ge-detectors will be used in larger rooms as one or two measurements can measure the surface-contamination in the whole room. Ge-detectors can also measure the γ- emitting radionuclides that have penetrated into the floor or walls. Furthermore, gamma spectrometric measurements can determine the radionuclide composition of γ-emitters. The measurement results with the Ge-detectors are analyzed by means of the commercially available software such as ISOCS [38]. Glovebox and equipment surfaces will be measured with measured with contamination monitors.

FIG. 14. Monitoring of glovebox interior to check for ‘high’ areas to target for early removal. NDA analysis had already been used to indicate inventory of the glovebox. Swab samples taken at this point would be sent for analysis to confirm activity ‘fingerprint’.

The walls will then be measured with Ge-detectors. It will be assumed conservatively that all the activity seen by the detector is located in one square meter that is positioned farthest away from the detector. Further characterization will be done of hot spots to determine how material is actually distributed on the wall. Contamination of the ceiling will be measured with Ge-detectors.

The floor will be measured with both Ge-detectors and contamination monitors.

3.7. UPPORTING FACILITIES

The only mobile supporting facility that will be used in the decommissioning of the nuclear laboratory is the Modular containment (see Fig. 15). No additional new facilities are considered necessary. Radioactive waste will be moved to an existing waste storage facility on-site.

3.8. END-STATE

The end point for the completion of the decommissioning activities at the nuclear laboratory can be described as follows:

(a) The laboratory working areas will be decontaminated so that surface areas are cleaned to a level of 0.04 Bq/cm^2 that can allow the building of the Laboratory Complex to be demolished at a later date with the main laboratory’s building structure;

(b) All radioactive waste, equipment, internals, and services and above ground drains from the five rooms will be removed for on-site storage and subsequent disposal as ILW, LLW or cleared as appropriate; and

(c) Each laboratory room will remain connected to the facility ventilation at the entrance to the room only. Fire suppression systems will be removed from all five laboratory rooms. Other facility support systems (process water, electrical systems, etc.) will be isolated and removed.

FIG. 15. Gloveboxes removed, stands decontaminated and covered for disposal as low level waste, containment floor sheeting layers removed to expose an uncontaminated layer, and Modular containment interior marked out and confirmed free of contamination. After this point the tie-down coating and joint tape will be removed, and the containment dismantled for re-use elsewhere.

4. HAZARD ANALYSIS: IDENTIFICATION AND SCREENING

A combination of the checklist from Volume I of this report and Ref. [10], in conjunction with the HAZOP process was used to identify those reasonably foreseeable hazards associated with the planned decommissioning activities. The hazard identification and hazard assessment process was conducted by suitably qualified and experienced persons from a wide range of disciplines including facility operators and engineers; radiological protection adviser; safety engineers; human factors and criticality specialists assisted by decommissioning workers.

4.1. APPROACHES TO HAZARD IDENTIFICATION

To aim for completeness in the identification of all reasonably foreseeable accident scenarios, a well structured hazards analysis method needs to be followed and documented. The following formal processes were employed for the identification and analysis of all reasonably foreseeable hazards and initiating events such that the overall risk estimate for the facility is conservative:

- Assessment of normal operations are contained within the operational history of the facility;
- Review of significant events during the facility lifetime, including interviews with operations personnel with specific facility history;
- Human factors/ergonomic walk down;
- A checklist was used to initially identify hazards, initiating events and eliminate standard industrial hazards that do not warrant further consideration in this analysis (Table 6);

- Identification of accident sequences was carried out primarily by HAZOP studies (both desktop and facility walk down) The final approach used to ensure all reasonably foreseeable hazards were identified was to perform a “what-if” evaluation of potential failure modes, including process upsets like power loss, fire, operator error;
- Construction of a hazard/event schedule for internal and external scenarios was based on the HAZOP. The identified hazards and initiators were grouped logically into event groups based on similar hazard/initiators types to enable the number of scenarios to be analysed to be kept to a minimum. Any bounding cases that were used are identified in the particular significant event scenario (Tables 7);
- Analysis of individual hazards including calculation of risk and, where appropriate, design basis analysis and identification of the engineered and procedural limits and conditions necessary in the interests of safe decommissioning operations;
- Input from the engineering assessment i.e. in relation to any new initiating events and status of the facility safety functions identified during the operator/engineering analysis of the facility and engineering walkdown; and
- Collating and summarizing the results of the individual hazard assessments into an overall assessment of the facility.

4.2. HAZARD IDENTIFICATION

The main groups of radiological and non-radiological hazards to workers, the public and the environment were identified in the table below.

TABLE 6 CHECKLIST FOR IDENTIFYING HAZARDS AND INITIATING EVENTS

HAZARD ID No.	HAZARDS	RELEVANT FOR PLANNED WORK	RELEVANT FOR ACCIDENTS
1	**INTERNAL INITIATING EVENTS**		
1.A	**Radiological Initiating Events**		
1.A.1	***Criticality***		
1A.1-1	**Residue of fissile materials in equipment and process lines**	**Yes, residue of fissile material is present, dispersed over surfaces and in small quantities in several locations**	**No, material is in solid form and cannot be collected in quantities of 450 g in a manner that would result in a favorable geometry**
1A.1-2	**Residue of fissile radioactive liquids in tanks**	**No, there are no residual fissile liquids**	**No**
1A.1-3	**Presence of moderators (e.g., water, PVC etc) in the vicinity of fissile material**	**Yes, Moducon materials may have moderating effect**	**No, material is in solid form and cannot be collected in quantities of 450 g in a manner that would result in a favorable geometry No**
1A.2	***Spread of contamination***		
1A.2-1	**Loss of containment/ barriers**	**Yes, barriers could be inadvertently damaged due to planned activities**	**Yes**
1A.2-2	**Dismantling of**	**Yes, glove boxes and**	**Yes**

HAZARD ID No.	HAZARDS	RELEVANT FOR PLANNED WORK	RELEVANT FOR ACCIDENTS
	containment/barriers	**ventilation systems will be deliberately opened**	
1A.2-3	**Drop of radioactive materials, packages and waste**	**Yes**	**Yes**
1A.2-4	**Cleanup of buildings (activated or contaminated)**	**Yes, use of acid washes, hydrolazing (steam or power wash), scabbeling could result in puncture of a primary barrier**	**No, Planned activity is cleanup, therefore, upsets must not result in accidental release of any significant material quantity**
1A.3	***External exposure***		
1A.3-1	**Activated materials and equipments**	**No**	**No**
	Direct radiation sources	**Yes, Am, and other isotopes typical of fuels are present**	**Yes**
1A.3-2	***Internal exposure***		
1A.3-3	**Physical and chemical state of the radioactive materials**	**Yes, Pu is present in oxide form that may be easily dispersed**	**Yes**
1A.4	***Contamination, corrosion, etc.***		
1A.4-1	**Pathways (inhalation, ingestion)**	**Yes, breach of airline suit or Modular containment or puncture wound**	**Yes**
1A.4-2	**Spectrum, activity, emitters (presence of alpha emitters)**	**Yes**	**Yes**
1A.4-3	**Contaminated materials**	**Yes, both fixed and loose contamination is expected**	**Yes, items with loose surface contamination could**

HAZARD ID No.	HAZARDS	RELEVANT FOR PLANNED WORK	RELEVANT FOR ACCIDENTS
			result in release
1A.4-4	**Gaseous Effluent**	**Yes, gaseous effluents may be produced by welding/cutting activities**	**No, release would be very localized**
1A.4-5	**Liquid Effluent**	**Yes, some liquids may be produced by steam cleaning or hydrolazing activities**	**No, release would be very localized**
B	**NON RADIOLOGICAL INITIATING EVENTS**		
1B.1	***Fire***		
1B.1-1	**Thermal cutting techniques (zircalloy, etc.)**	**Yes**	**Yes**
1B.1-2	**Decontamination process (chemical, mechanical, electrical methods or mixed methods to remove contamination from metals, concrete or others surface)**	**Yes, fixatives and structural foams, may be flammable in liquid form, decontamination solutions may enhance ignitability characteristics of cellulose materials**	**Yes**
1B.1-3	**Accumulation of combustible materials and radioactive waste**	**Yes, fire retardant wood may be used to protect windows, additional plastics and other materials to support waste collection may be increased**	**Yes**
1B.1-4	**Flammable gases and liquids**	**Yes, MAPP and acetylene may be used to support thermal cutting**	**Yes**
1B.2	***Explosion***		
1B.2-1	**Decontamination process**	**No**	**No**

HAZARD ID No.	HAZARDS	RELEVANT FOR PLANNED WORK	RELEVANT FOR ACCIDENTS
1B.2-2	**Dust (graphite, zircalloy, etc.)**	**Yes, dust may be produced by cutting operations**	**No**
1B.2-3	**Radiolysis phenomena (radioactive waste storage, transport)**	**Yes, packaged waste may generate H2 gas**	**Yes**
1B.2-4	**Compressed gases**	**Yes, welding/cutting**	**No, insufficient quantity to be of concern, standard industrial hazard**
1B.2-5	**Explosive substances**	**No, not present**	**No**
1B.3	***Flooding***	***No***	***No***
1B.3-1	**Leak of liquid storage**	**No, no liquid storage**	**No**
1B.3-2	**Leak of pipes**	**Yes**	**No, standard industrial hazard**
1B.4	***Toxic and hazardous materials***		
1B.4-1	**Asbestos/glasswoolin thermal insulation system**	**Yes**	**No, standard industrial hazard**
1B.4-2	**Lead in paint, shielding**	**Yes**	**No, standard industrial hazard**
1B.4-3	**Beryllium and other hazardous materials**	**Yes, other hazardous materials include cyanide gas caused by burning of structural foam components**	**No, standard industrial hazards**
1B.4-4	**Polychlorinated biphenyls (PCBs)**	**No**	**No**
1B.4-4	**Oils**	**No**	**No**
1B.4-5	**Pesticide use**	**No**	**No**

HAZARD ID No.	**HAZARDS**	**RELEVANT FOR PLANNED WORK**	**RELEVANT FOR ACCIDENTS**
1B.4-6	**Biohazards**	**No**	**No**
1B.5	***Electrical hazards***		
1B.5-1	**Loss of power supply**	**Yes**	**Yes**
1B.5-2	**High voltage**	**No, not present**	**No,**
1B.5-3	**Non-ionizing radiation sources (lasers, …)**	**No, not present**	**No**
1B.6	***Physical hazards***		
1B.6-1	**Falling of heavy loads**	**Yes**	**Yes**
1B.6-2	**falling loads on SSCs important for safety**	**No**	**No**
1B.6-3	**falling loads on radioactive materials (packages)**	**Yes**	**Yes**
1B.6-4	**Collapse of structure (due to ageing)**	**No**	**No**
1B.6-5	**Demolition activities**	**No, demolition is not included in this stage of decommissioning**	**No**
1B.6-6	**Working at heights**	**Yes**	**No, standard industrial hazard**
1B.6-7	**High noise area**	**Yes**	**No, standard industrial hazard**
1B.7	***Human and organizational initiating events***		
1B.7-1	**Operator error/violation**	**Yes**	**Yes**
1B.7-2	**Inadvertent entry into high-radiation areas**	**No, high radiation areas expected**	**No**

HAZARD ID No.	HAZARDS	RELEVANT FOR PLANNED WORK	RELEVANT FOR ACCIDENTS
1B.7-3	**Misidentifications**	**Yes**	**Yes**
1B.7-4	**Contractor and sub-contractor**	**Yes**	**No, standard practice**
1B.7-5	**Performing incompatible activities**	**No, laboratory operations are sufficiently separated from decommissioning activities**	**No**
1B.7-6	**Disabling services to other facilities**	**Yes, the utilities are common to the balance of the laboratory and site.**	**No, resources sufficiently isolated, other facilities/operations include proper analysis, standard industrial hazard**
1B.7-7	**Poor ergonomic conditions**	**No**	**No**
2.C	**EXTERNAL INITIATING EVENTS**		
2.C.1	***Earthquake***	**No, duration of activity is so short that probability of earthquake during decommissioning is BEU**	**No, while release may occur as a result of earthquake the consequences would be reduced from the operational evaluation preceding decommissioning as radiological material has been removed.**
2.C.2	***External flooding***		
2.C.2-1	**River**	**No**	**No**
2.C.2-2	**Sea**	**No**	**No**

HAZARD ID No.	**HAZARDS**	**RELEVANT FOR PLANNED WORK**	**RELEVANT FOR ACCIDENTS**
2.C.2-3	**Infiltration of groundwater**	**No**	**No**
2.C.3	***External fire (oil storage, etc.)***	**No**	**No**
2.C.4	***Extreme weather conditions (temperature, wind, snow, etc.)***	**No**	**No**
2.C.5	***Hazards due to industrial environment (explosion, etc.)***	**No**	**No**
2.C.6	***Airplane crash***	**No**	**No**
3.D	**OTHERS INITIATING EVENTS**		
3.D.1	***High temperature and pressure***	**No**	**No**
3.D.2	***Corroded barriers***	**No**	**No**
3.D.3	***Unknown or unmarked materials***	**Yes**	**Yes**

Those items that were identified as “Relevant for Planned Work” were further considered as to whether they could initiate an accident that would result in release of radioactive material, i.e. initiate an event or result in significant exposure consequences. Those that could be indicated by yes in the “Relevant for Accident” column. These hazards are carried forward to the HAZOPs review for further consideration. Those hazards identified as relevant for planned work, but not considered to be initiators will be carried forward in work planning processes as standard industrial hazards and protective measures must be included in work packages, but are not carried further in this assessment.

4.3. PRELIMINARY HAZARD ASSESSMENT AND SCREENING

The risk assessment itself needs to be as rigorous as is reasonably practicable. It is important that the greatest analytical resources are focused on those hazards and initiating events which have the greatest safety significance. The safety significance of a hazard was judged purely in terms of its potential radiological consequences. Hazard screening is necessary to ensure that resources are directed at the analysis of hazards and initiating events which are potentially serious and that resources are not expended on those which contribute little to the overall risk or are incapable of producing events which add significantly to the radiological consequences.

Small doses (low consequences, see Table 1) ariser from very minor accidents can be regarded as being within the range of normal operations doses. It should be noted that due to the conservative assumptions employed in hazard analysis calculations for without active safety control measure dose, an estimate of without active safety control measure dose quite possibly exceeds the dose which actually be delivered in an accident by at least one order of magnitude dose from normal operation of the facility.

The process for hazard screening begins with the checklist from Appendix I of Volume I of this report. It is not the intention of the hazard analysis to cover safety as it relates to the common industrial hazards that make up a large portion of basic regulatory compliance of Ref. [39] or to expend resources on those hazards for which institutional Safety Management Programmes already define and regulate appropriate practices without the need for special analysis. These types of standard industrial hazards are to be screened from further consideration in the hazard analysis unless the industrial hazard could affect radiological or large chemical inventories or cause facility wide effects, for example, are included in the facility safety basis, where they may be controlled through application of safety measures.

As part of the hazard identification process, the hazards which may be screened will be differentiated using the logic provided by the flowchart in Fig. 16. Ref. [40] defines standard industrial hazards for this purpose as: “Hazards that are routinely encountered in general industry and construction, and for which national consensus codes and/or standards (e.g., OSHA, transportation safety) exist to guide safe design and operation without the need for special analysis to design safe and/or operational parameters.” The questions in the flowchart guide this determination as part of the screening process. For these determinations, the hazards analysis process interfaces with other programmes such as specific topics of OSHA compliance, industrial safety, or fire protection. The analyst is expected to verify coverage of hazards within Safety Management Programmes during screening of standard industrial hazards. The following guidelines apply for hazard screening determinations:

- The first question determines whether the identified hazard is a standard industrial hazard (i.e., one routinely encountered in general industry or construction applications outside the nuclear industry) that can be screened. See Table 6 for an example of the identification and screening process. Unique hazards cannot be screened and must be carried forward for further evaluation. In determining whether a hazard is unique, consider any variations from standard practice, the magnitude of the hazard, etc. The intent is to carry forward hazards that warrant case specific hazard evaluation even though they may be concluded to be a standard industrial hazard (SIH) following specific evaluation (e.g., unusually large quantities of hazardous materials), but not anything that is evidently an SIH (e.g., high-voltage electrical equipment).

- If the hazard is an SIH, the second question determines whether the hazard is adequately covered by Safety Management Programmes based on national consensus standards. The intent is to verify that the program scope includes appropriate requirements, not that the field implementation of the Safety Management Programme be verified as well (this is reserved for the implementation review). Where exceptions are taken to national consensus standards, the hazard cannot be screened and requires hazard evaluation.

- The last question determines whether the hazard entails potential for significant interactions with nuclear hazards. Such interactions may not be addressed by consensus standards and requires more thorough evaluation than screening would afford. A “yes” answer is appropriate if the hazard requires evaluation to verify or determine appropriate controls. Some hazards are adequately controlled, but may still serve as an initiator for a nuclear accident. Electrical power is an example. In such an instance, Question 3 would be answered “no”, but the initiator potential would be noted to ensure its inclusion indicating further consideration is required Table 6 in the column “Relevant for Accidents”.

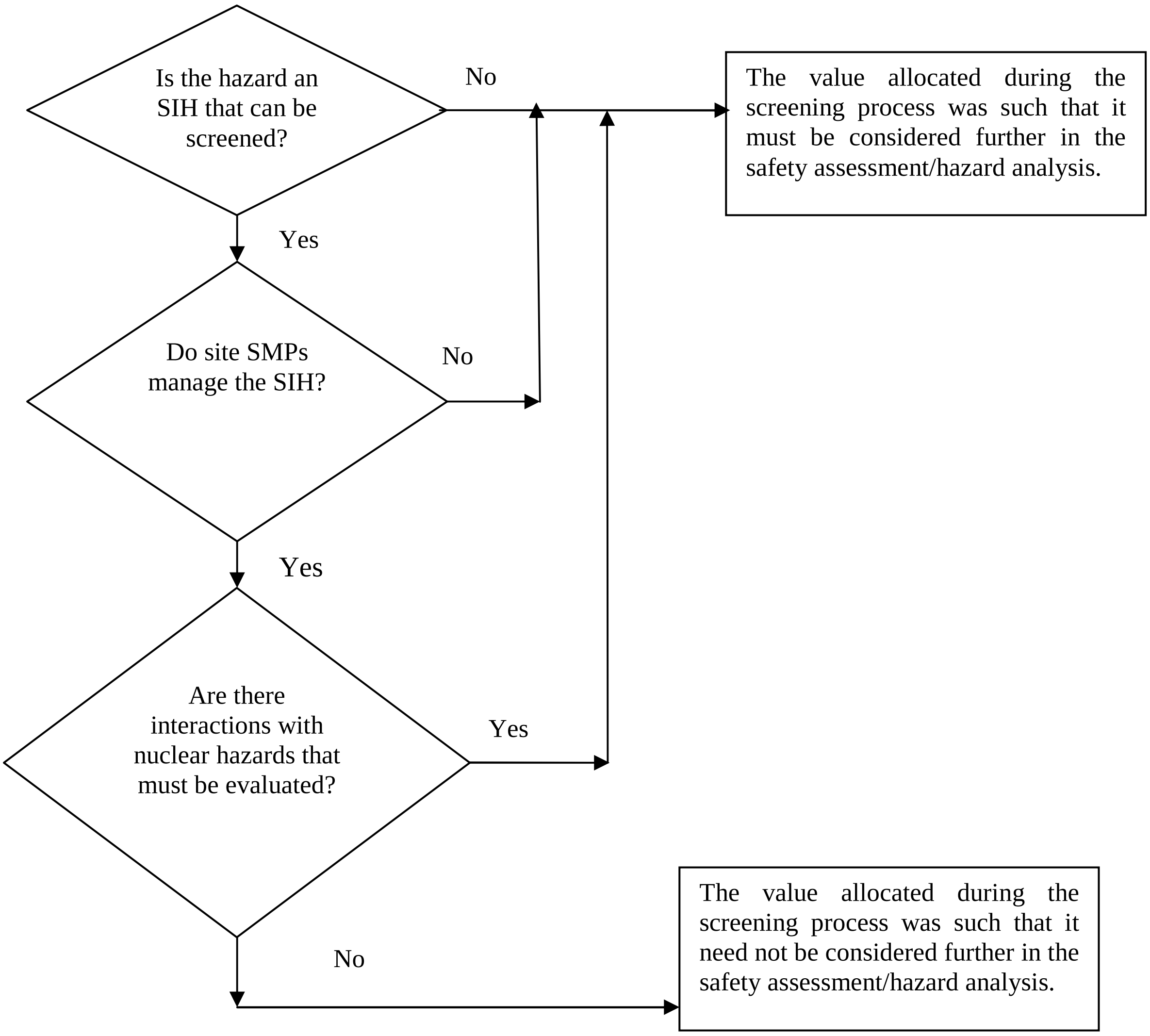

FIG. 16. Hazard screening process[3].

Table 1 defines the point in the fault analysis process at which this low consequence filter needs to be applied. The first step in the procedure is to assess the operator and/or public consequences on a conservative basis with failed protection and all mitigating systems removed from consideration i.e. without active safety control measure dose. If the doses are below these threshold values shown in Table 1, then they may be considered as candidates for the low consequence methodology, resulting in a “No” response to “Relevant for Accidents”. If without active safety control measure doses are above the respective screening criteria then the hazard is carried forward for full hazard assessment.

The following is a summary of the more complete list of hazards identified in the checklist presented in Section 4.2 after application of the HAZOPs process. This summary provides a brief description of those events selected to be carried forward for consideration in further accident analysis or the basis for dismissal of those categories not considered in the accident analysis. The more complete results of the hazard event schedule are included as Table 7.

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[3] SMP = Safety Management Programme, and SIH = Standard Industrial Hazard.

(a) Radiological hazards

— Criticality

The maximum quantity of fissile material in a single glovebox is estimated to be 442 g. The total amount of fissile material inventory in the 5 rooms is estimated to be approximately 1 270 g. While there is some amount of Pu remaining in the equipment, the fissile material is distributed as residual, fixed in gloveboxes and ventilation ducting, or surface contamination. The proposed decommissioning activities will not preferentially separate or collect fissile material in a fashion that could result in quantities of 450 g or more in solid form that could be placed in a favourable configuration which may result in a criticality. There are no fissile liquids present in the facility. Excess fissile material will be removed from gloveboxes before they are opened to the room such that water from fire suppression could interact with material. Waste handling and packaging criteria impose limits that are designed to preclude criticality (mass limits, geometry, spacing, etc.). Criticality safety is established using American Nuclear Standards Institute series ANS 8 [21]. No further evaluation is provided in this document.

— Spread of contamination

Dismantlement and size reduction activities directly compromise primary confinement barriers. Holes, punctures, cuts are planned activities and will typically be performed within a Modular confinement system; however there is potential for an inadvertent release if the wrong line is cut or the line was not properly identified as containing material. Scabbling or power washing could also result in an unplanned cut or puncture of equipment or barrier; however these activities are typically conducted on structural surfaces after all loose fissile material has been removed.

Drops and spills of packaged material are also a means to spread contamination. Individual waste packages may be moved by fork trucks or hand dollies, prepared glove boxes, piping components and other equipment may be moved by cranes and hoists.

Some liquid effluent may be generated from hydrolazing (pressurized wash of contaminated surfaces). Liquid generated from decontamination processes will be collected and processed through waste evaporators. On this basis no gaseous effluents are expected to result from routine decommissioning activities.

Drops and spills are evaluated further in Scenarios G1, H1, H2, K1, K2 and L1 (see Section 5.2., fire and electrical hazards for additional loss of confinement events).

— Direct exposure

The nature of the work conducted in the laboratory used Pu which is primarily an alpha emitter. One of the daughter products of Pu is ^{241}Am, ^{242}Am, and ^{243}Am which does present a direct exposure hazard. In addition, some of the laboratory operations included radioactive fuels. Therefore activated metals and some residual materials that produce a direct exposure hazard such as ^{60}Co, ^{134}Cs, ^{137}Cs, ^{129}I, ^{90}Sr and ^{99}Tc. Exposure from these direct radiation hazards are evaluated as normal activities in Section 5.1.

— Internal exposure

The most prominent internal exposure hazard associated with the proposed decommissioning activities is Pu. While most of the loose material has been removed or fixed in place, there is risk of dispersion when currently closed systems are opened and contamination may be dislodged by mechanical agitation associated with size reduction activities. In addition, material could be dispersed by fire which could expose personnel outside of the facility. The potential for inhalation is of particular concern. Potential dose from internal exposure resulting from injection wounds is evaluated as normal activities in Section 5.1. Other mechanisms for release that could result in inhalation or ingestion from accidental release are discussed by individual events below:

— Fire

Dismantlement and size reduction activities introduce the use of thermal cutting techniques. Cutting with MAPP (a combination of liquefied petroleum and Methylacetylene-Propadine), acetylene, or oxyacetylene gases introduce both ignition sources and flammable gases.

Fixatives, structural foams, and paints may be flammable in liquid form (e.g. before they are dried). Efforts are made to use materials with limited flame spread characteristics, but the presence of these materials serves as a fuel source can result in an increased potential for fire. Some decontamination solutions (e.g. cerium nitrate) can produce materials that may have increased ignitability characteristics when used with cellulose products and not properly rinsed.

The nature of decommissioning requires the use of combustible materials (plastics and tapes) in packaging of waste. Fire retardant protective plywood may be used to protect glovebox windows. Filters and collection devices have to be uniquely evaluated for potential as ignition sources.

Fires are evaluated in Scenarios A1, A2, B1, C1, E1 and F1 presented in Section 5.2.

— Explosion

Particular attention should be paid to dust generating activities such as metal cutting and collection of fines in filters or vacuums.

Although radiolysis may result in generation of H_2 at lower flammability limits (LFL) the physical dimension of drums and standard waste boxes does not physically support the development of a detonation event. The drum deflagration event is included as fire. Newly generated waste is packaged in vented containers designed to preclude collection of H_2 and other volatile organic compounds that could result in such an event.

Compressed gasses are present to support welding/cutting operations. These gases are provided in approved containers (Department of Transportation) designed to with stand pressure/drops, etc. Gas is not supplied in sufficient quantity for any of the laboratory rooms to reach lower flammability limit.

Further Explosions are found in Section 5.2, Scenario J1.

— Flooding

No potential exists for flooding and therefore no further evaluation is provided in this safety assessment.

(b) Non-radiological hazards

— Toxic and hazardous materials

Construction records indicate that asbestos was used as an insulating material in the Laboratory complex. Routine laboratory chemicals must have been removed; however some residual amounts may be present in areas that may be exposed once equipment is moved. Decommissioning may use structural foam to secure glovebox interiors and meet packaging requirements. These foams may produce toxic gases (e.g. cyanide) if involved in a fire. Lead paint was used to cover contamination from a spill in Laboratory room 2. Beryllium was suspected to have been handled in glovebox A, Laboratory room 2. The assessment shows that any accidental exposure to either Pb or Be will be at a level less than the dangerous dose and that any long term exposure would be at a level less than the appropriate occupational exposure limit. In conclusion, the level of risk is sufficiently low that no specific constraints or restrictions are necessary beyond those provided by compliance with the site's safety management programme (see Table 8).

— Electrical hazards

Electrical sources will be removed from gloveboxes and other equipment. Normal lighting and power systems will remain in place until the balance of the Laboratory Complex completes mission activities and undergoes structural demolition. Temporary power may be required if electrical conduit must be removed to support surface decontamination. Electrical power may be the cause of a spark that could initiate a fire (see Fire above). Additionally, loss of power could result in loss of ventilation or flow reversal. This case is covered in Section 5.2, Case I1.

— Physical hazards

No unique hazards have been identified with the decommissioning of the 5 laboratory rooms. Heavy loads will be lifted and handled by crane and fork truck. Events associated with drops are covered above in drops and spills. Equipment will not be lifted over credited SSCs, so there is no danger of damaging systems. Demolition is not authorized in the first stage of decommissioning as the end point is an empty room that will support demolition in the final stage of decommissioning, no structural degradation will result from authorized activities. Standard industrial hazards associated with physical hazards including working at heights, high noise, etc. will be addressed through work planning processes.

— Human and organizational initiating events

There is high probability for human errors in the decommissioning process. Those most common include failure to perform work in accordance with planning documents and procedures which may encompass cutting the wrong line, opening a line that may have been mischaracterized, or erroneous free-release of materials. The continuing laboratory operations are sufficiently separated from the decommissioning activities by hallways and fire walls so as not to introduce hazards from the laboratory operations or vice versa. The consequences of accidents initiated by human error are included above. Planning of work packages have to include review for error likely situations and precursors that may be avoided with proper consideration.

— External initiating events

No specific risk was considered resulting from natural phenomena hazards (earthquake, external flooding, extreme weather), during the limited duration of decommissioning of the 5 laboratory rooms. The probability of occurrence of a design basis event during this limited window of operation is beyond extremely unlikely. The decommissioning activities reduce the material at risk. No structural changes are planned; therefore, the operational evaluation of natural hazards remains bounding for planned decommissioning activities. The decommissioning activities planned for the five laboratory rooms do not introduce any new hazards to the industrial environment (e.g. no cranes or hoists external to the facility, no additional gas/propane to support operation). No additional consideration is included for external initiating events evaluation.

— Other initiating events

There are no other initiating events identified. Unknown material is discussed above.

TABLE 7 HAZARD/EVENT SCHEDULE - RADIOLOGICAL HAZARDS

Hazard ID/ Ref. No.	**Event Description/Type**	**Cause**	**Material at Risk**	**Without Active Safety Control Measure Evaluation**		**Potential Controls (Prevention/Mitigation)**		**Accident Scenario Ref. No.**
				Frequency	**Receptor/ Dose [Sv]**	**Engineered Controls**	**Administrative Controls**	
1A.2-1, 1B.7-3, 3.D.3	Loss of containment/ barrier Misidentifications *Unknown or unmarked materials*	Size reduction and decontamination tools such as saws, grinders, shears, hydrolazers etc cut or puncture primary barrier (glove box, waste container) leading to direct exposure or unconfined material	1 350 g	Anticipated	Worker: 4.67 x 10^{-2} Public: 6.34 x 10^{-4}	Confinement Building ventilation Moducon	Drums over 200 g Pu-E not be staged/stored outside confinement, Drums/SWBs must not be stacked	G1, H1, H2, K1, K2, L1
1A.2-2, 1A.3-2, 1B.7-3	Dismantling of containment/Barriers Dose received from direct radiation sources Misidentifications	Inadequate survey or relocation/improper placement of shielding or removal of containment barriers before removal of source term result in hot spots	5-25 μSv/h			Shielding CAM	Radiological Protection Program, Alarming dosimetry	Section 5.2
1B.1-1 1B.1-3	Fire initiated by thermal cutting techniques (zircalloy, etc.) Accumulation of combustible	Cutting activities occur too close to surrounding material, or cutting operation started before removing excess materials.	1 276 g	Anticipated	Worker: 1.1 x 10^{-1} Public: 1.5 x 10^{-3}	Confinement (1 stage HEPA) fire suppression	Combustible material controls, hot work controls, Drums/standard waste boxes must not be stacked	A1, A2, B1, E1, F1

Hazard ID/ Ref. No.	Event Description/Type	Cause	Material at Risk	Without Active Safety Control Measure Evaluation		Potential Controls (Prevention/Mitigation)		Accident Scenario Ref. No.
				Frequency	Receptor/ Dose [Sv]	Engineered Controls	Administrative Controls	
	materials and radioactive waste							
1B.1-2	Fire resulting from decontamination process (chemical, mechanical, electrical methods or mixed methods to remove contamination from metals, concrete or others surface)	Surfaces cleaned using cerium nitrate, wiped with cellulose cloths ignited by sparks produced by mechanical cutting operations.	1 595 g	Anticipated	Worker: 1.3×10^{-1} Public: 1.8×10^{-3}	Confinement (1 stage HEPA)	Combustible material controls, hot work controls	D1
1B.1-4	Explosion caused by flammable gases and liquids	Ignition of flammable gas produces deflagration/overpressure, jet release caused by sheared regulator or vapor cloud explosion from accumulated leak	468 g	Anticipated	Worker: 7.8×10^{-3} Public: 1.1×10^{-4}	Confinement	Drums over 200 g Pu-E shall not be staged/stored outside confinement, use of non sparking tools,	J1
1B.2-3	Radiolysis phenomena (radioactive waste storage, transport) results in drum	Sealed waste container with hydrocarbon materials produces hydrogen gas without ventilation path,	2 000 g	Anticipated	Worker: 9.7×10^{-1} Public:	Waste container, drum vent, HEPA filtration	Use of lid restraints, remote handling, non-sparking tools	J2

Hazard ID/ Ref. No.	**Event Description/Type**	**Cause**	**Material at Risk**	**Without Active Safety Control Measure Evaluation**		**Potential Controls (Prevention/Mitigation)**		**Accident Scenario Ref. No.**
				Frequency	**Receptor/ Dose [Sv]**	**Engineered Controls**	**Administrative Controls**	
	deflagration	venting, or movement introduces oxygen and sufficient energy to ignite			1.3 x 10^{-2}			
1B.5-1	Loss of power supply causes ventilation system flow reversal	Loss of flow with a momentary backward pressure gradient disperses airborne material	562 g	Anticipated	Worker: 5.7 x 10^{-4} Public: 7.7 x 10^{-6}			I1

TABLE 8 NON-RADIOLOGICAL HAZARDS

Hazard ID/ Ref. No.	**Event Description/ Type**	**Material at Risk**	**Without Active Safety Control Measure Evaluation**		**Potential Controls (Prevention/Mitigation)**		**Accident Scenario Ref. No.**
			Frequency	**Receptor/ Dose**	**Engineered Controls**	**Administrative Controls**	
1B.4-1	Asbestos/glass woolin thermal insulation system	< TQ	Anticipated	N/A	N/A	N/A	The amount of asbestos in the facility is below threshold quantities; therefore, the level of risk is sufficiently low that no specific constraints or restrictions are necessary beyond those provided by compliance with the site's safety management programme.
1B.4-2	Lead in paint, shielding	< TQ	Anticipated	N/A	N/A	N/A	The assessment shows that any accidental exposure to Pb will be at a level less than the dangerous dose and that any long term exposure would be at a level less than the appropriate occupational exposure limit. In conclusion, the level of risk is sufficiently low that no specific constraints or restrictions are necessary beyond those provided by compliance with the site's safety management programme.

Hazard ID/ Ref. No.	**Event Description/ Type**	**Material at Risk**	**Without Active Safety Control Measure Evaluation**		**Potential Controls (Prevention/Mitigation)**		**Accident Scenario Ref. No.**
			Frequency	**Receptor/ Dose**	**Engineered Controls**	**Administrative Controls**	
1B.4-3	Beryllium and other hazardous materials (cyanide gas caused by ignition of structural foam components)	< TQ	Anticipated	N/A	N/A	N/A	The assessment shows that any accidental exposure to Be will be at a level less than the dangerous dose and that any long term exposure would be at a level less than the appropriate occupational exposure limit. Structural foam will be used to meet waste acceptance criteria in some cases. The amount of material required is minimal and therefore the amount of cyanide off gas produced would not reach toxic levels. In conclusion, the level of risk is sufficiently low that no specific constraints or restrictions are necessary beyond those provided by compliance with the site’s safety management programme.

5. HAZARD ANALYSIS: EVALUATION

5.1. ANALYSIS OF NORMAL ACTIVITIES

Hazards during normal operations of the redundant glovebox decommissioning project may arise from external and internal radiation doses and from industrial hazards.

5.1.1. External radiation dose

Routine external radiation doses are expected to be low in consequence of the low background dose rates in the laboratory areas, each of which has been subject to post operational cleanup and routine radiation monitoring. The calculated inventories are based on non-destructive analysis measurements taken following post operational cleanup in 1994. The limits of the non-destructive assay required the application of a best case, worst case model. The “best case” assumes that all the plutonium is present on the windows and ports, distributed evenly, consistent with swipe samples and radiological meter dose readings, whereas the “worst case” results assume the material is collected at the middle of the glovebox floor. The main potential cause of external dose arises from the gamma radiation sources within the gloveboxes that give doses of up to 100 μSv/h at the glovebox surface. These dose rates reduce to 25 μSv/h at the protective boarding (shielding) and would be expected to be < 1 μSv/h outside a radius of 1 m.

Table 9. gives a conservative assessment of the external dose from decommissioning operations. All work adjacent to glovebox surfaces is assumed to be at a minimum dose rate of 5 μSv/h, with a small amount of high dose work at 25 μSv/h. Other preparatory activities are calculated at a nominal 1 μSv/h.

TABLE 9 DOSE ASSESSMENT FOR DECOMMISSIONING ACTIVITIES

Stage	**Activity**	**Number of People**	**Dose Rate [μSv/h]**	**Exposure Time [h/person]**	**Total Dose [μSv]**
Vent removal	Assemble scaffold and tent	3	1	12	36
	Remove vent pipework	4	1	16	64
	Strike scaffold and tent	2	1	10	20
Build ModuCon	Work adjacent to radiation sources	2	25	2	100
	Work away from radiation sources	2	5	16	160
Size reduction	Work adjacent to radiation sources	2	25	16	800
	Work away from radiation sources	2	5	80	800
Waste operations	Handling high dose waste drums	2	5	2	20
	General waste drum ops	2	1	8	16
Clearance	Survey equipment and room surfaces	2	1	80	160
Total dose [person μSv]					**2 176**

Hence the provisional dose budget for the decommissioning operations is 2 176 man mSv for three gloveboxes in Laboratory room 2.

Assuming the total dose budgets can be calculated pro rata from the maximum dose rates and the number of gloveboxes to be processed suggests the following total operator doses from normal operations (see Table 10):

TABLE 10 OPERATOR DOSE FROM NORMAL OPERATION

Location	**Number of boxes**	**Maximum Dose Rate [μSv/h]**	**Conversion Factor [Man h/box]**	**Dose [Man μSv]**
Laboratory room 1	5	10	6.72	336
Laboratory room 2	3	100	6.72	2016
Laboratory room 3	2	6*	6.72	81
Laboratory room 4	6	6	6.72	242
Laboratory room 5	1	18	6.72	121
Total	**17**			**2 796**

* Taken to be the same as Laboratory room 4 dose rate. These boxes have been wrapped and transported and have no hot spots.

Hence the total dose budget is calculated as some 2.8 man·mSv, of which 72% is accounted for by Laboratory room 2. Assuming an individual took part in each of the stages in all of the labs doing the tasks leading to greatest exposure they would receive less than 1 mSv.

5.1.2. Dose constraint objective

Consideration of the dose budget given in the previous paragraphs, together with the fact that detailed planning of operations will focus on dose reduction, suggest that a dose constraint objective of 0.75 mSv/y, resulting in 0.375 mSv for the 6 month period of decommissioning activities, for each individual worker is reasonable. One half of the dose will be assumed for conduct of this activity as it is projected to take 6 months. This will be reviewed monthly to ensure that it remains a challenging target to assist in keeping doses ALARA.

5.1.3. Internal radiation dose

The redundant laboratories in which the work will be undertaken have all been subject to post-operational cleanup, involving the removal or isolation of all sources of contamination with the exception of that fixed within the gloveboxes. Installed alarming air monitors are placed within each laboratory to warn of any instance of air activity. Indicative calculations based on general air sampler levels of 0.01 to 0.03 derived air concentration (DAC)[4] hours per shift was measured during the care and maintenance of the laboratories over the period 1993 – 1997 suggest that internal doses for the whole programme of work will be very low. For example, assuming an individual may accumulate 1000 hours exposure at 0.03 DAC hours per 8 h shift (and 2000 DAC hours equivalent to 50 mSv for old data), the estimated internal dose is 0.03/8 x 1 000/2 000 x 50 = 0.094 mSv. This dose indicates a very small contribution of routine internal dose by comparison either with external radiation or with possible release events, which are considered in the next Section.

The following scenarios provide upper and conservative estimates of potential doses that may arise from normal and from accidental conditions both for workers and for members of the public. This preliminary analysis serves to put the potential hazards from the laboratory into perspective with other facilities and to justify the choice of the level of detail in which the detailed accident analysis is carried out in Section 5.2.

5.2. ANALYSIS OF ACCIDENT SCENARIOS

The criteria presented in Section 4.3 above were used to determine what hazards/potential scenario types to carry forward for additional quantitative analysis. The hazards/potential scenario types (fires, spills, explosions, etc.) that meet these criteria are annotated in Table 7. The following summarizes the bases for those scenarios presented for additional analysis:

- Hazards/initiating events that lead to scenarios that could result in on-site or off-site consequences. At this stage, each hazard was considered independent of other hazards.
- Hazards/initiating events that lead to scenarios that are of sufficient complexity to require more detailed analysis to understand consequences and the impact of controls.
- Hazards/potential events that provide technical justification for control set reduction (i.e. removal of specified controls once hazard has been reduced).
- Qualitative analysis was performed for each hazard to determine what equipment (structures, systems or components) or administrative controls are required to protect immediate workers. The following criteria, based on definitions in Ref. [40], were used to identify safety measures, either as engineered systems, SSCs, or as administrative controls.

Those hazards and initiating events identified in Table 7. that require further evaluation were first grouped by event categories to support more detailed analysis. The major event categories are:

- Operational accidents (caused by facility conditions or operations);
- External events (caused by activities outside the facility that may or may not be related to decommissioning operations); and
- Natural phenomena hazard events (acts of nature).

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[4] The DAC is a value calculated from a fixed airhead sample station. The samples represent the airborne concentration of a room of known volume for a specific period of time. The derived value provides the estimate of internal dose an individual in the room would receive by remaining in the room for a complete shift.

Operational accidents are further subdivided into fires, spills, explosions, and criticalities. Additional sorting is based on the area that the accident may occur. These include outside of the laboratory building (however, no activities occur outside the laboratory building structure and will not be included), within the laboratory rooms but outside of confinement, and within the laboratory rooms within confinement.

The gloveboxes in Laboratory room 2 are used to represent the worst potential accident conditions as these boxes contain the highest amount of potential source material. The material is modeled as loose surface contamination to ensure the most bounding case. The material at risk (MAR = the amount of material that may be available to be involved in any scenario) in the glovebox fire and room fire involve these materials. The second case presents evaluation of materials in Laboratory room 1 to illustrate that controls required to support the cleanup of Laboratory room 2 will not be required once these boxes have been removed and may be discontinued at this point. No external events were identified to be carried forward. NPH events presented in the operational analysis are considered bounding and will not be reevaluated here.

5.2.1. Modelling assumptions

MELCOR Accident Consequence analysis Code (MACCS2) [24] dispersion model was applied to obtain dose estimates at 100 m for the onsite worker population (noted as WORKER in associated analysis) and at the site boundary for potential does to the critical group, also known as the public receptor who is presumed to be located at the nearest site boundary to the facility. The distance from closest point of laboratory decommissioning activities to the nearest site boundary is 1 580 m.

Plume dispersion is based on 95th percentile meteorology in all scenarios except for those that are postulated to be caused by high winds or tornadoes. This represents conservative weather conditions corresponding to low wind and neutral stability, minimization of plume dispersion, and a conservatively high estimate of dose to the receptor. Plume dispersion will be higher and doses to receptors will be lower than predicted by this value 95% of the time. Plume dispersion will be lower and doses to receptors will be higher than predicted by this value 5% of the time [33]. The public is represented by a hypothetical critical group, who is presumed to be located directly downwind of the facility and at the nearest Site boundary. Standard Meteorology assumes wind speed 6 km/h from 270° Stability D, no precipitation.

For scenarios that involve a lofted plume, the public is presumed to be located downwind at the point of maximum dose.

The breathing rate corresponding to heavy activity (3.6×10^{-4} m^3/s), which would provide maximum internal exposure to airborne releases, is used for all cases.

All dose consequences are reported in units of Sv, 50 y CEDE (Committed Effective Dose Equivalent).

Dose consequences are primarily reported in units of Sv 50 y CEDE (Committed Effective Dose Equivalent).

There are no credible criticality accidents; therefore doses from criticality are not modeled.

Dose conversion factors based on the most recent International Commission on Radiation Protection standards (ICRP-68) [35] are used for all scenarios. In accordance with ICRP-68 recommendations, a particle size of 5 μm is assumed for all without active safety control measure scenarios except where the postulated exposure is to a non-volatile (aqueous) liquid release. For these cases, a particle size of

1 µm is assumed and corresponding dose consequences are slightly higher. Releases through HEPA filters also use the 1 µm particle size as a credited active safety control measure.

Release events are modeled as ground level release for unfiltered events. Discharge from the ventilation stack is released at 25 m.

The basic assumptions used to establish the source term for the Laboratory Test Case were the following:

(a) During dismantling of the gloveboxes, inner surfaces are exposed and building surfaces are decontaminated, leading to a mobilization of radionuclides into the facilities' atmosphere. The without active safety control measure case assumes that all material is available for release unless protected by a confinement barrier (such as a glove box or sealed waste container). Damage ratios for confinement barriers are described by scenario to represent the physical conditions under evaluation.

(b) Intermediated level waste (ILW) drums having a nominal 200 ℓ capacity that are ready to ship are packaged at a maximum loading of 200 g Pu-E. Because containers in the Laboratory have generally not been assayed to confirm this maximum loading, and to allow acceptance for repackaging of containers determined by assay to be over the 200 g limit, one container in each population of 200 ℓ ILW waste drums is assumed to be overloaded by 25%, for a maximum loading of 250 g Pu-E. In general practice to prevent reworking drums, many are packaged to less than 200 g. While the dose consequences of an event would be slightly higher if the event involved only these drums, a reduction in frequency of that specific event would be justified since there are so few. Thus assuming that all drums are packaged to 200 g with one overbatched at 250 g continues to provide an adequate conservatism for analysis purposes.

(c) Standard Waste Boxes are packaged to a maximum loading of 320 g Pu-E. The waste acceptance criteria for the waste disposal facility[5] and criticality safety limits are 325 g Pu (Fissile gram equivalent) and 342 g Pu + U. Procedurally, the boxes are packaged to much lower levels to prevent approaching theses limits. However, for analysis, one container in each population of Standard Waste Boxes is assumed to be overloaded by approximately 25%, for a maximum loading of 400 g Pu-E.

(d) Low Level Waste (LLW) is limited to a maximum of 3.7 kBq activity per gram of waste matrix. The loading of the IP-2 metal LLW crate is limited to the A2 quantity, which is based on the isotopic mix of radionuclides present in the waste. To be consistent with other Site analyses, each IP-2 box of low level waste is assumed to contain 3 g Pu-E. This value includes sufficient margin to account for possible overloading.

(e) LLW is limited to a maximum of 3.7 kBq activity per gram of waste matrix. Each 200 ℓ drum of low level waste is assumed to contain 0.5 g Pu-E. For analysis purposes, each drum is assumed to contain 1 g Pu-E to provide sufficient margin to account for possible overloading.

(f) Residual values are used as the measured or nominal value based on non-destructive assay plus approximately two (sometimes 1.96) times the standard deviation ($N + 2\sigma$) or as the Lower Limit of Detection (LLD).

(g) Container integrity provides a primary boundary between the material at risk and the surrounding environment.

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[5] For the purpose of this safety assessment the isolation pilot plant (WIPP) was used.

Additional assumptions that are relevant to multiple scenarios include the following:

- Radioactive waste will be packaged only in metal containers approved for on-site shipping. Radioactive waste will not be packaged in wooden containers. Plastic containers used for LLW liquid wastes (e.g., from eyewash station testing) are also permitted.
- Moducon size reduction structures will be constructed of fire retardant material.
- Fixative coatings used on process components to prevent the spread of contamination will not continue to burn without an external heat source when dry.
- This analysis credits one tested stage of HEPA filtration. Filter stage particle removal efficiency is at least 99.9% for the tested stage (equivalent to a Building leakpath factor of 0.001). Each of the accident scenarios begins with the without active safety control measure event.

Although the standard assumption is that containerized waste is modeled as confined material, some of the analyses model portions of the waste stream as unconfined noncombustible material. This is done to more realistically portray the hazard of decommissioning waste, much of which will be surface contaminated metal and other contaminated noncombustible material. The rationale for the various modeling methods is provided on a case-by-case basis.

5.2.2. Modelling and calculation of consequences

The hazard identification process identified numerous fires involving radioactive materials that could result in uncontrolled releases of radioactive materials. Fires are evaluated as *anticipated* events. A fire can be initiated by a spark or heat from various activities that ignites accumulated combustible material (e.g., hot work, residual pyrophoric material).

While the scenarios are evaluated at different frequencies, including *anticipated*, every effort is taken to prevent the occurrence of any fires. The frequencies selected are used for evaluation purposes to demonstrate the adequacy of proposed controls.

A small fire is defined as the largest fire that can cause radiological material release without activating the suppression system. The small fire is modeled as a 1 MW fire with release duration of 10 min.

A medium fire is modeled as a 1 to 5 MW fire with a release duration of 15 min. that occurs based on lapses in combustible material controls and hot work controls. While a small fire is modeled as too small to actuate the fire suppression sprinkler system, a medium fire will actuate the system. Thus, the fire suppression sprinklers cannot be credited to reduce the frequency of a medium fire (since it reaches medium size prior to actuation). However, once actuated, the fire suppression sprinklers will either extinguish the fire, thereby reducing the consequences.

A large fire is defined as the largest fire that warrants consideration based on lapses in combustible material controls and hot work controls and a failure of the fire suppression sprinklers where available. Thus, large fires would result from a failure of the combustible material controls such that an accumulation of material to support a 30 min. fire occurs. An ignition source is then assumed to start the fire and the fire suppression sprinklers are then assumed to fail such that the fire grows to a large (5 to 10 MW) size. The fire suppression sprinklers not only prevent smaller fires from growing but also limit radiological releases by limiting pressurization of drums and other containers. The heat release rate for the postulated large fire is 9.26 MW from combustion of 210 kg of polyethylene drum liners.

The accumulation of the incomplete combustion of gases and/or particulate could occur in ventilation ductwork due to large or major fires and subsequently be ignited causing a pressure transient (similar to, but less severe than, a deflagration) in the ductwork. This condition has the potential to exist given the occurrence of various factors, which include but are not limited to:

- Incomplete combustion products resulting from the initial fire;
- Mixing of the combustible products with fresh air in the ventilation system;
- Sufficient residual heat or other ignition source necessary to ignite the combustible products;
- Configuration of the ventilation system;
- Ventilation airflow; and
- Type and quantity of combustible materials involved in the fire.

While this phenomenon could potentially occur in the exhaust ventilation ductwork, it is not considered likely due to the configuration of the system. The ensuing pressure transient, should it occur, is not expected to cause any damage to the downstream HEPA filters due to their location. The filters are more vulnerable to clogging or hot embers.

A postulated outcome of a fire is loss of filtration due to HEPA filter plugging. The potential for smoke to plug the HEPA filters in the exhaust plenum is considered because it provides a mechanism for room exhaust to escape the building unfiltered and also presents a potential for damage to HEPA filters. Plugging of the HEPA filters could result in sufficient differential pressure across the filter banks to cause collapse or blow through. Approximately 5 kg of plastic or 27 kg of ordinary combustibles such as wood is required to produce sufficient solid combustion products to clog a single filter.

Since all material for the above containers is modeled as confined material and assuming seal failure for all containers in fires, the source term evaluated for ILW drums bounds that source term for standard waste boxes. Since ILW drums experience lid loss with material ejection and the Air borne release fraction/resprirable fraction values are higher, the ILW drums bound both Standard Waste Boxes for pools fires also.

(a) Waste container staging fires

The staging, or aggregating, of full waste containers is considered as a non-location-specific activity. Waste containers used in staging areas are metal containers approved for use in on-site shipping. Containers are assumed to have lids (permanent or temporary metal) in place. This assumption does not require that the lid be crimped or torqued (for metal crates or Standard Waste Boxes), or the ring-bolt tightened (for open-head drums), as required for transportation. For medium and large fires, no pressure relief credit is taken for drums that have lids that are not secured, even though this effect would help to prevent lid loss (does not apply to small fires since drums do not experience lid loss in a small fire).

Small waste container staging fires (Scenario A)

The following scenarios consider a small fire (less than 1 MW) involving staged containers of nuclear debris. Such material could involve ILW in 200 ℓ drums or Standard Waste Boxes, LLW in metal (IP-2) crates or 200 ℓ drums. A fire could be initiated by a spark or heat from operations, maintenance, or closure activities (e.g., hot work such as thermal cutting or sparks from mechanical size reduction) that ignites accumulated combustible material. Other possible initiators could include exothermic chemical reactions from incompatible container contents, improper hot work, equipment malfunction (e.g., electrical short, overheat) or improperly operated, degraded electrical equipment, power supplies, or damaged electrical power cords.

- **Scenario A1**: Small Drum Fire inside Confinement

This scenario postulates a small fire involving ILW in twelve drums inside Laboratory room 2 with active confinement. As postulated, the fire consumes the ILW and internal plastic packaging, exposing the burning contents to the building atmosphere.

Nine drums containing a total of 1 267 g, based on Table 4, are modeled as confined material and a damage ratio of 0.2 is used for the material in drums.

Control set and risk class

The dose consequences without active safety control measure for this scenario are 44 mSv and 0.6 mSv for the public. When evaluated at an *anticipated* frequency, the scenario is Risk Class III for the worker and the public (see Table 11).

The computer code Radidose 1.4.3 [23] was used to model dose consequences and the effects of implemented controls. The results are displayed in tables after the description of the scenario. In those tables, the following abbreviations were used:

χ/Q	is notation for the dispersion coefficient;
SCO	Surface contaminated objects;
DCF	Dose conversion factor;
ICRP	International Commission on Radiation Protection;
DR	Damage ratio;
LPF	Leak path factor;
ARF	Air borne release fraction; and
RF	Respirable Fraction.

TABLE 11 SMALL DRUM FIRE INSIDE CONFINEMENT (SCENARIO A1)

Small Drum Fire Inside Confinement (Non-Lofted Fire)										
Radidose Parameters										
Contributor	**Material**	**χ/Q[6]**	**Breathing Rate**	**Form of Material**	**DCF ICRP-68**	**DR**	**LPF**	**Release Duration**	**ARF**	**RF**
a. drum fraction	Pu-E	95th	Heavy	Confined Materials	Moderate	0.2	1.0	10	5 x 10^{-4}	1.00

Radidose Results							
Contributor	**Without active safety control measure**			**With active safety control measures**			Notes:
	Pu-E	**Dose Consequence Sv**		**Pu-E**	**Dose Consequences Sv**		
		WORKER	**PUBLIC**		**WORKER**	**PUBLIC**	
a. drum fraction	1 267	4.4 x 10^{-2}	6.0 x 10^{-4}	1 267	4.4 x 10^{-2}	6.0 x 10^{-4}	
	Total:	4.4 x 10^{-2}	6.0 x 10^{-4}	**Total:**	4.4 x 10^{-2}	6.0 x 10^{-4}	

Risk Class Evaluation		**Dose Consequences with Active Safety Control Measures [Sv]**							
		0 HEPA		**1 HEPA**		**2 HEPA**		**0.1 LPF[7]**	
		WORKER	**PUBLIC**	**WORKER**	**PUBLIC**	**WORKER**	**PUBLIC**	**WORKER**	**PUBLIC**
Dose Consequences:		4.4 x 10^{-2}	6.0 x 10^{-4}	6.4 x 10^{-5}	8.7 x 10^{-7}	1.3 x 10^{-7}	1.7 x 10^{-9}	4.4 x 10^{-3}	6.0 x 10^{-3}
Consequence Level:		Low	Low	Low	Low	Low	Low	Low	Low
Risk Class	**Anticipated**	III	III	III	III	III	III	III	III
	Unlikely	III	III	III	III	III	III	III	III
	Extremely Unlikely	IV	IV	IV	IV	IV	IV	IV	IV

Controls	**Classification of Control**
Combustible material controls	specific administrative control
Confinement (1 stage HEPA)	SSC Category 4
Hot work controls	administrative control

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[6] χ/Q is notation for the dispersion coefficient. This notes the meteorology used in the analysis. 95th indicates that standard meteorology, or weather representative of the 95th percentile was applied.

[7] The LFP can be reduced by a factor of 0.1 for a passive confinement (no unfiltered pathways to the environment).

— **Scenario A2:** Surface Contaminated Objects, LLW, and ILW Fire without Confinement

This scenario postulates a fire that involves LLW in metal crates and ILW waste in 200 ℓ drums. The case could occur in any operationally clean area that:

- Has had major source term removed;
- Is used to package remaining debris from removal of piping, lighting and other debris from the laboratory rooms); and
- Where remaining waste is being loaded into waste containers.

Work in operationally clean areas will produce two major waste streams:

- Combustible waste, which is modeled as confined material; and
- Contaminated noncombustible waste, such as metal and concrete, which is modeled as unconfined noncombustible.

It is assumed that these activities will not generate ILW other than contaminated non-combustibles (e.g., metal piping, duct stubs). ILW is, therefore, modeled as contaminated noncombustible material. Surface contaminated objects are also modeled as unconfined noncombustible material. LLW is modeled as confined material since combustible wastes produced during cold strip-out activities (contaminated protective clothing, plastic used for contamination control, fiberboard ceiling tiles, etc.) will be packaged as LLW. Significant amounts of noncombustible LLW will also be produced (sprinkler piping, light fixtures, scabbled concrete). The choice to model all LLW as combustible is therefore conservative.

The without active safety control measure case is assumed to involve:

- Two large surface contaminated objects (total of 12 g Pu-E),
- Eight metal LLW crates (total of 24 g Pu-E),
- Two ILW drums (total 450 g Pu-E, assuming one drum at 200 g; and
- One overbatched at 250 g), and an overbatched Standard Waster Box (400 g Pu-E).

A damage ratio of 0.2 is used to represent seal failure for metal LLW crates and a damage ratio of 1.0 is used for ILW drums and the Standard Waste Box, since the containers are assumed to be open and in use. Although this quantity of ILW is not anticipated to remain in areas declared Operationally Clean, this allows the introduction and use of additional containers prior to removal of filled containers, as necessary. In addition to the above material at risk (MAR), embers from the fire are assumed to propagate through the plenum and ignite the HEPA filters. Based on the residual measurements of residual material the ducting to the filter with the highest amount of residual material residual contains less than 100 g Pu-E. For conservatism, the HEPA filter is assumed to contain 150 g Pu-E. This conservatism accounts for some amount of residual that might be present in the HEPA filters of air movers that are in use in the area.

The amount of material selected conservatively bounds what might be present in an operationally clean area. Although a small fire could not involve this amount of material, the fire is modeled as a 10 min. release (i.e., small fire). Since the fire involves all material in the area and a medium and large fire have longer release duration, which result in lower dose consequences, this fire bounds a medium and large fire involving the same amount of material.

Control set and risk class

The without active safety control measure dose consequences for this scenario are relatively *low* to the worker (24.0 mSv) and *low* to the public (0.325 mSv). When evaluated at an *anticipated* frequency, the scenario is Risk Class III for the worker and Risk Class III for the public.

Since the dose consequences for both the worker and public are relatively *low* and Risk Class III for the without active safety control measure case, no additional controls to mitigate the consequences or reduce the frequency of the event are warranted. For the immediate worker, the without active safety control measure dose consequences are qualitatively assessed as *low*. When evaluated at an *anticipated* frequency, the event represents Risk Class III. No additional controls to the ones that the immediate workers will wear in accordance to the safety programmes are envisaged to specifically protect the immediate workers (see Table 12).

TABLE 12 SURFACE CONTAMINATED OBJECTS, LLW, AND ILW WASTE FIRE IN OPERATIONALLY CLEAN AREA (SCENARIO A2)

Surface Contaminated Objects, LLW, and ILW Waste Fire in OPS Clean Area (Non-Lofted Fire)

Radidose Parameters

Contributor	Material	χ/Q	Breathing Rate	Form of Material	DCF ICRP-68	DR	LPF	Release Duration	ARF	RF
a. SCO fraction	Pu-E	95th	Heavy	Unconfined Noncombust.	Moderate	1.0	1.0	10	6 x 10^{-3}	1.00
b. LLW fraction	Pu-E	95th	Heavy	Confined Materials	Moderate	0.2	1.0	10	5 x 10^{-4}	1.00
c. TRU drum fraction	Pu-E	95th	Heavy	Unconfined Noncombust.	Moderate	1.0	1.0	10	6 x 10^{-3}	1.00
d. SWB fraction	Pu-E	95th	Heavy	Unconfined Noncombust.	Moderate	1.0	1.0	10	6 x 10^{-3}	1.00
e. HEPA filters	Pu E	95th	Heavy	HEPA Filters	Moderate	1.0	1.0	10	1 x 10^{-4}	1.00

Radidose Results

Contributor	Without active safety control measure			With active safety control measures			Notes: Ventilation system is assumed to be unavailable in Operationally Clean Area
	Pu-E	Dose Consequences [Sv]		Pu-E	Dose Consequences [Sv]		
		WORKER	PUBLIC		WORKER	PUBLIC	
a. SCO fraction	12	2.50 x 10^{-4}	3.39 x 10^{-6}	12	2.50 x 10^{-4}	3.39 x 10^{-6}	
b. LLW fraction	24	8.32 x 10^{-4}	1.13 x 10^{-5}	24	8.32 x 10^{-4}	1.13 x 10^{-5}	
c. TRU drum fraction	450	9.36 x 10^{-3}	1.27 x 10^{-4}	450	9.36 x 10^{-3}	1.27 x 10^{-4}	
d. SWB fraction	400	8.32 x 10^{-3}	1.13 x 10^{-4}	400	8.32 x 10^{-3}	1.13 x 10^{-4}	
e. HEPA filters	150	5.20 x 10^{-3}	7.06 x 10^{-5}	150	5.20 x 10^{-3}	7.06 x 10^{-5}	
	Total:	2.40 x 10^{-2}	3.25 x 10^{-4}	**Total:**	2.40 x 10^{-2}	3.25 x 10^{-4}	

Risk Class Evaluation		Dose Consequences with Active Safety Control Measures Sv							
		0 HEPA		1 HEPA		2 HEPA		0.1 LPF	
		WORKER	PUBLIC	WORKER	PUBLIC	WORKER	PUBLIC	WORKER	PUBLIC
	Dose Consequences:	2.40 x 10^{-2}	3.25 x 10^{-4}	3.51 x 10^{-5}	4.76 x 10^{-7}	7.02 x 10^{-8}	9.53 x 10^{-10}	2.40 x 10^{-3}	3.25 x 10^{-5}
	Consequence Level:	Low	Low	Low	Low	Low	Low	Low	Low
Risk Class	**Anticipated**	III	III	III	III	III	III	III	III
	Unlikely	III	III	III	III	III	III	III	III
	Extremely Unlikely	IV	IV	IV	IV	IV	IV	IV	IV

Controls	Classification of Controls
No specifically credited controls are warranted	N/A

Medium fire, waste container staging (Scenarios B)

Scenario description

The following scenarios consider a medium fire (1 to 5 MW) involving staged containers of material such as ILW in 200 ℓ drums or standard waste boxes, low level waste in metal (IP-2) crates or 200 ℓ drums. This fire could be initiated by a spark or heat from operations, maintenance, or closure activities (e.g., hot work or electrical equipment) that ignites accumulated combustible material.

Fires initiated inside the building are typically not considered to be fast growing fires due to the combustible material types with one exception. Waste oil in a drum may be found staged in various areas. The consequences and the footprint of the medium fire are different for a fast-growing (pallet, drum liner, or flammable liquid) fire than for an ordinary combustible fire. The medium, fast-growing fire develops a heat flux sufficient to cause lid loss in 200 ℓ drums. Lid loss is not assumed for metal waste crates or standard waste boxes. The lack of a gasket seal in metal waste crates allows venting throughout the heating of the container and contents, and prevents buildup of sufficient pressure in the container to result in lid loss. For the standard waste boxes, the weight of the lid (approximately 90 kg) and the method of fastening (bolted to the container) are considered to effectively prevent lid loss in the fast-growing fire as analyzed.

Scenario B1: Medium drum fire inside confinement

This scenario postulates a medium fire involving ILW within a non-standard wooden crate that also involved drums inside the laboratory rooms with confinement (e.g. ventilation). As postulated, the fire consumes the ILW and internal plastic packaging, exposing the burning contents to the building atmosphere. The dominant cause or initiator for this scenario is size reduction activities or other ignition sources such as transportation equipment, maintenance, or closure activities. Other possible initiators could include exothermic chemical reactions from incompatible container contents, improper hot work, equipment malfunction (e.g., electrical short, overheat) or improperly operated, degraded electrical equipment, power supplies, or damaged electrical power cords.

For the case without active safety control measures, a fire involving 21 stacked drums (826 g Pu-E) that experience seal failure plus two ILW drums (total 450 g Pu-E, assuming one drum at 200 g and one overbatched at 250 g) that experience lid loss. All waste is modeled Pu-E and as confined material. All of the drums that experience seal failure are evaluated with a damage ratio of 0.2 and the drums that experience lid loss are evaluated with a damage ratio of 1. For the case without active safety control measures, all drums are assumed to experience seal failure.

Control set and risk class

The dose consequences without active safety control measures for this scenario are *moderate* to the worker (107 mSv) and *over the limit* to the public (1.49 mSv). When evaluated at an *anticipated* frequency, the scenario is Risk Class I for the worker and Risk Class III for the public (see Table 13).

Fires are considered *anticipated*. Medium fires are considered *unlikely* when crediting combustible material controls and hot work controls. Crediting one stage of HEPA filtration reduces the consequences to *low* for the worker.

The assumption that fire suppression sprinklers will remain operational provides a reasonably practicable control which must be maintained as a measure of the fire protection program.

For the scenario with active safety control measures, the *low* dose consequences to the worker (0.15 mSv) evaluated at an *unlikely* frequency results in a Risk Class III category and the *low* dose consequences to the public (0.002 mSv) evaluated at an *unlikely* frequency also results in a Risk Class III category.

For the immediate worker, the dose consequences without active safety control measure are qualitatively assessed as *moderate*. When evaluated at an *anticipated* frequency, the event represents Risk Class I. The potential for serious injury or significant radiological exposure can be further reduced by evacuating the immediate area of the fire. Various aspects of the Safety Management Programmes such as training and fire protection ensure that workers in the immediate vicinity of the fire evacuate and that other workers in the facility are notified via the building fire alarms. The building alarms can be activated automatically by smoke detectors or manually by pull stations or fire phones. As these are all governed by Safety Management Programmes, no additional controls to protect the immediate workers require elevation as safety measures. With the immediate worker protection afforded by the Safety Management Programmes, the consequences with active safety control measures are qualitatively assessed as *low*. When evaluated at an *unlikely* frequency, the event represents Risk Class III (see Table 13).

TABLE 13 MEDIUM DRUM FIRE INSIDE CONFINEMENT (SCENARIO B1)

Medium Drum Fire inside Confinement (Non-Lofted Fire)										
Radidose Parameters										
Contributor	**Material**	**χ/Q**	**Breathing Rate**	**Form of Material**	**DCF ICRP-68**	**DR**	**LPF**	**Release Duration**	**ARF**	**RF**
a. drums - lid loss fraction	Pu-E	95th	Heavy	Confined Materials	Moderate	1.0	1.0	15	5 x 10^{-4}	1.00
b. drums - seal failure fraction	Pu-E	95th	Heavy	Confined Materials	Moderate	0.2	1.0	15	5 x 10^{-4}	1.00

Radidose Results							
Contributor	**Without active safety control measure**			**With active safety control measures**			Notes: The totals to the left are different from the sum of the values listed due to round-off error and significant digits.
	Pu-E	**Dose Consequences [Sv]**		**Pu-E**	**Dose Consequences [Sv]**		
		WORKER	**PUBLIC**		**WORKER**	**PUBLIC**	
a. drums - lid loss fraction	450	7.8 x 10^{-2}	1.1 x 10^{-3}	450	1.1 x 10^{-4}	1.6 x 10^{-6}	
b. drums - seal failure fraction	826	2.9 x 10^{-2}	3.9 x 10^{-4}	826	4.20 x 10^{-5}	5.71 x 10^{-7}	
	Total:	1.07 x 10^{-1}	1.49 x 10^{-3}	**Total:**	1.5 x 10^{-4}	2.17 x 10^{-6}	

Risk Class Evaluation		**Dose Consequences with Active Safety Control Measures [Sv]**							
		0 HEPA		**1 HEPA**		**2 HEPA**		**0.1 LPF**	
		WORKER	**PUBLIC**	**WORKER**	**PUBLIC**	**WORKER**	**PUBLIC**	**WORKER**	**PUBLIC**
Dose Consequences:		1.07 x 10^{-1}	1.49 x 10^{-3}	1.5 x 10^{-4}	2.17 x 10^{-6}	3.14 x 10^{-7}	4.20 x 10^{-9}	1.7 x 10^{-1}	1.49 x 10^{-4}
Consequence Level:		Moderate	Low	Low	Low	Low	Low	Low	Low
Risk Class	**Anticipated**	I	III	III	III	III	III	III	III
	Unlikely	III	III	III	III	III	III	III	III
	Extremely Unlikely	III	III	IV	IV	IV	IV	IV	IV

Controls	**Classification of Controls**
Combustible material controls	specific administrative control
Confinement (1 stage HEPA)	SSC Category 2
Hot work controls	Administrative Control
Fire suppression (sprinklers)	SSC Category 3

Scenario B2: Medium fire that involves ventilation ducting

This scenario postulates a large drum fire that occurs in the Laboratory room 2. A total failure of the ductwork resulting in unfiltered release to the environment is not evaluated in this case as the ducting travels a significant distance before reaching the filter-housing and finally the stack.

Total filter plenum bypass leakage factors, estimated with maximum undetected door leakage are 3.7 x 10^{-5} and 5.1 x 10^{-5} for filter room 1 and filter room 2 respectively. In the case of a fan seal failure for filter room 1 or filter room 2, the bypass leakage factor is the sum of the previous bypass leakage factor for filtered and unfiltered releases (5.1 x 10^{-5}), and an additional 5.7 cmm bypass leakage factor (5.7 cmm unfiltered bypass flow divided by the total volumetric flow of the room). An unfiltered leakpath of 5.7 m^3/min. was evaluated for its safety impact since this value bounds the worst observed leakage in any laboratory facility. The volumetric flow in Room 2 is 2137 m^3/min. Thus, the additional bypass leakage factor is 5.7/2 137 = 2.7 x 10^{-3}. The total bypass is thus 5.1 x 10^{-5} plus 2.7 x 10^{-3} = 2.7 x 10^{-3}.

The case with the largest source term involves drums exposed to a fire that is fueled by polyethylene drum liners or a flammable/combustible liquid. For this case, 21 drums (826 g Pu-E) experience seal failure and an addition two drums (450 g Pu-E) experience lid loss. A damage ratio of 0.2 is used for drums experiencing seal failure and a damage ratio of 1.0 is used for drums that experience lid loss. All drum waste is evaluated as Pu-E and is modeled as confined material. Based on the values measured during planning and characterization the most contaminated section of ducting and filters contain less than 100 g Pu-E. For conservatism, the ducting is assumed to reach the filters contain 150 g Pu-E of residual material burning as HEPA filters.

Control set and risk class

The dose consequences without active safety control measure for this scenario are *moderate* to the worker (110 mSv) and *over the dose limit* to the public (1.55 mSv). When evaluated at an *anticipated* frequency, the scenario is Risk Class I for the worker and Risk Class III for the public (see Table 5.6).

Fires are considered *anticipated*. Medium fires are considered *unlikely* when crediting combustible material controls and hot work controls in conjunction with fire suppression sprinklers.

As described above, the bypass leakage for the rooms was 2.7 x 10^{-3}, was used in previous versions of the safety case. Because multiple instances of small patches have been identified on the ductwork and no fire rating information has been identified for the material used, a building leakpath factor of 0.1 is conservatively assumed for the return plenum rooms. When using the 0.1 building leakpath factor, the dose consequences to both the worker and public are *low*.

For the immediate worker, the dose consequences without active safety control measures are qualitatively assessed as *moderate*. When evaluated at an *anticipated* frequency, the event represents Risk Class I. The potential for serious injury or significant radiological exposure can be further reduced by evacuating the immediate area of the fire. Various aspects of the Safety Management Programmes such as training and fire protection ensure that workers in the immediate vicinity of the fire evacuate and that other workers in the facility are notified via the building fire alarms. The building alarms can be activated automatically by smoke detectors or manually by pull stations or fire phones. As these are all governed by Safety Management Programmes, no additional controls to protect the immediate workers require elevation to Safety Measure. With the immediate worker protection afforded by the Safety Management Programmes, the consequences with active safety control measures are qualitatively assessed as *low*. When evaluated at an *extremely unlikely* frequency, the event represents Risk Class IV (see Table 14).

TABLE 14 MEDIUM FIRE IN RETURN PLENUM ROOMS (SCENARIO B2)

Medium Fire that Involves Ventilation Ducting (Non-Lofted Fire)										
Radidose Parameters										
Contributor	**Material**	**χ/Q**	**Breathing Rate**	**Form of Material**	**DCF ICRP-68**	**DR**	**LPF**	**Release Duration**	**ARF**	**RF**
a. drums - lid loss	Pu-E	95th	Heavy	Confined Materials	Moderate	1.0	1.0	30	5×10^{-4}	1.00
b. drums - seal failure	Pu-E	95th	Heavy	Confined Materials	Moderate	0.2	1.0	30	5×10^{-4}	1.00
c. residual	Pu-E	95th	Heavy	HEPA Filters	Moderate	1.0	1.0	30	1×10^{-4}	1.00

Radidose Results							
Contributor	**Without active safety control measure**			**With active safety control measures**			Notes: The totals to the left are different from the sum of the values listed due to roundoff error and significant digits. Evaluated using a 0.1 building leakpath factor Ventilation system is involved in the event, compromised and not available
	Pu-E	**Dose Consequences [Sv]**		**Pu-E**	**Dose Consequences [Sv]**		
		WORKER	**PUBLIC**		**WORKER**	**PUBLIC**	
a. drums - lid loss	450	7.8×10^{-2}	1.1×10^{-3}	450	7.8×10^{-3}	1.1×10^{-4}	
b. drums - seal failure	826	2.9×10^{-2}	3.9×10^{-4}	826	2.9×10^{-3}	3.9×10^{-5}	
c. residual	150	4.18×10^{-3}	5.67×10^{-5}	150	4.18×10^{-4}	5.67×10^{-6}	
	Total:	1.1×10^{-1}	1.55×10^{-3}	**Total:**	1.1×10^{-2}	1.55×10^{-4}	

Risk Class Evaluation		**Dose Consequences with Active Safety Control Measures [Sv]**							
		0 HEPA		**1 HEPA**		**2 HEPA**		**0.1 LPF**	
		WORKER	**PUBLIC**	**WORKER**	**PUBLIC**	**WORKER**	**PUBLIC**	**WORKER**	**PUBLIC**
Dose Consequences:		1.1×10^{-1}	1.55×10^{-3}	4.33×10^{-3}	5.82×10^{-5}	4.18×10^{-3}	5.67×10^{-5}	1.1×10^{-2}	1.55×10^{-4}
Consequence Level:		Moderate	Low	Low	Low	Low	Low	Moderate	Low
Risk Class	**Anticipated**	III	III	III	III	III	III	III	III
	Unlikely	III	III	III	III	III	III	III	III
	Extremely Unlikely	III	III	IV	IV	IV	IV	IV	IV

Controls	**Classification of Controls**
Combustible material controls	Specific administrative control
Fire suppression (sprinklers)	SSC Category 3
Hot work controls	Administrative Control

Large fires: residual glovebox/component fire (Scenarios C)

Scenario description

The medium fire scenarios involved all material therefore is not necessary to evaluate a large fire (5 to 10 MW) for containerized waste.

The glovebox/component fire is postulated to occur during decommissioning, in which gloveboxes, and other process components are removed from the process rooms. Components examined for these scenarios are the worst-case gloveboxes in which size reduction, processing, or storage activities will be performed, and the worst-case gloveboxes based on residual, in which only activities related to the decommissioning of the glovebox will be performed (unused gloveboxes).

Gloveboxes may be used for size reduction of process piping, or for storage and/or batch solidification of aqueous or organic liquids. Liquids may also be stored "out-of-line," in drums, flammable liquid cabinets, and in plastic bottles. Unused gloveboxes are assumed to contain nuclear material only in the form of residual. For the small and medium fires, events involving gloveboxes bound those involving other components. For the large fire, additional types of components are examined to provide a representative component fire for areas without gloveboxes.

Gloveboxes are also used for air-drying of materials used in clean-up of radioactive liquid spills. Wet spill clean-up materials are hung in gloveboxes to dry to prevent the introduction of free liquids into waste containers. The glovebox provides confinement for radioactive material that may be released in the drying process. Once these materials are dry, they fall under the facility Combustible Material Control Program, and are promptly removed from the glovebox and placed into a waste container. Due to the low frequency and short duration of this activity, it is not specifically analyzed, but is bounded by other analyzed glovebox fires.

The potential initiators for a glovebox fire could be:

- Heat/sparks from electrical equipment or hot work,
- Physical activities in or involving the glovebox/component, or
- An exothermic/pyrophoric reaction within the glovebox/component.

The analyzed cases for this scenario bound small fires induced by exothermic/pyrophoric reactions in ducts, tanks, pumps, or other equipment that may result from decommissioning activities.

The largest residual in a glovebox in the decommissioning area was 1 445 grams. That glovebox has been decontaminated. All the remaining gloveboxes in the facility have less residual. However, since 1 445 grams was measured as the largest value prior to post operational cleanup, it will be maintained as a conservative, bounding value.

Small fire: glovebox/component fire (Scenarios D)

Scenario description

These scenarios evaluate a small fire (less than 1 MW) that occurs within a glovebox still in place in a process area and a fire that occurs near an open tank. The maximum residual in a glovebox in the decommissioning area was 1 445 g Pu-E. Fires could occur due to electrical shorts, or by general facility operations including waste handling activities or decommissioning activities.

Assumptions

It is assumed that any fixative coating used on a glovebox/component is not combustible when dry. A non-combustible fixative coating may be in place, but is not credited for any mitigating effect. The contents of unused gloveboxes are assumed to be limited to glovebox residual.

Scenario D1: Small glovebox fire during size reduction of process piping

This scenario postulates a small fire that occurs during size reduction of process piping in a glovebox. Although available free liquids have been removed, some residual liquids are assumed to remain within the piping components being size reduced. For evaluation purposes, 1 ℓ of solution at a concentration of 150 g/ℓ is assumed to be contained within the piping component. For a small fire in a glovebox, 100% of material not in containers is involved in the fire. This material is evaluated as non-volatile liquid with a damage ratio of 1. The glovebox with the large amount of residual material is estimated to contain 1 445 g. This material is modeled as unconfined noncombustible material. A damage ratio of 1 is used for all material. This scenario also bounds a small fire in an unused glovebox since that case would be similar but involve less source term.

Control set and risk class

The dose consequences for this scenario without active safety control measure are *moderate* to the worker (130 mSv) and *low* to the public (0.1 mSv). When evaluated at an *anticipated* frequency, the scenario is Risk Class I for the worker and Risk Class III for the public (see Table 15).

Small fires are *anticipated* during the remaining life of the facility. Combustible material controls inside gloveboxes and hot work controls effectively reduce the frequency of this scenario from *anticipated* to *unlikely*. However, the small fire is always considered to be *anticipated*, no controls are credited to reduce the frequency of the event.

Crediting one stage of HEPA filtration reduces the consequences to *low* for both the public and the worker.

For the scenario with active safety control measures, the *low* dose consequences to the worker (0.19 mSv) evaluated at an *anticipated* frequency results in Risk Class III. The *low* dose consequences to the public (2.7 x 10^{-3} mSv) evaluated at an *anticipated* frequency also results in Risk Class III.

For the immediate worker, the dose consequences without active safety control measures are qualitatively assessed as *moderate*. When evaluated at an *anticipated* frequency, the event represents Risk Class I. The potential for serious injury or significant radiological exposure can be reduced by evacuating the immediate area of the fire. Various aspects of the Safety Management Programmes such as training and fire protection ensure that workers in the immediate vicinity of the fire evacuate and that other workers in the facility are notified via the building fire alarms. The building alarms can be activated automatically by smoke detectors or manually by pull stations or fire phones. As these are all governed by Safety Management Programmes, no additional controls to protect the immediate workers require elevation to Safety Measures. With the immediate worker protection afforded by the Safety Management Programmes, the consequences with active safety control measures are qualitatively assessed as *low*. When evaluated at an *anticipated* frequency, the event represents Risk Class III (see Table 5.7).

Since this event is assumed to involve all of the material, the only difference between the small and medium would be the release duration. Since the small fire has more conservative release duration, this event bounds a medium fire involving the same material.

TABLE 15 SMALL GLOVEBOX FIRE DURING SIZE REDUCTION OF PROCESS PIPING (SCENARIO D1)

Small Glovebox Fire during Size Reduction of Process Piping (Non-Lofted Fire)

Radidose Parameters

Contributor	**Material**	**χ/Q**	**Breathing Rate**	**Form of Material**	**DCF ICRP-68**	**DR**	**LPF**	**Release Duration**	**ARF**	**RF**
a. residual fraction	Pu-E	95th	Heavy	Unconfined Noncombust.	Moderate	1.0	1.0	10	6×10^{-3}	1×10^{-2}
b. liquid fraction	Pu-E	95th	Heavy	Nonvolatile Liquid	Moderate	1.0	1.0	10	2×10^{-3}	1.00

Radidose Results

Contributor	**Without active safety control measure**			**With active safety control measures**			Notes: The totals to the left are different from the sum of the values listed due to roundoff error and significant digits. Note that residual material in duct will typically be 1 micron as it has passed the glovebox prefilter.
	Pu-E	**Dose Consequences [Sv]**		**Pu-E**	**Dose Consequences [Sv]**		
		WORKER	**PUBLIC**		**WORKER**	**PUBLIC**	
a. residual fraction	1 445	3.0×10^{-2}	4.10×10^{-4}	1 445	4.1×10^{-4}	4.4×10^{-5}	
b. liquid fraction	150	1.0×10^{-1}	1.4×10^{-3}	150	1.4×10^{-3}	1.5×10^{-4}	
	Total:	01.3×10^{-1}	1.81×10^{-3}	**Total:**	1.94×10^{-4}	2.7×10^{-6}	

Risk Class Evaluation		**Dose Consequences with Active Safety Control Measures [Sv]**							
		0 HEPA		**1 HEPA**		**2 HEPA**		**0.1 LPF**	
		WORKER	**PUBLIC**	**WORKER**	**PUBLIC**	**WORKER**	**PUBLIC**	**WORKER**	**PUBLIC**
Dose Consequences:		1.3×10^{-1}	1.81×10^{-3}	1.94×10^{-4}	2.70×10^{-6}	3.88×10^{-7}	5.3×10^{-9}	1.3×10^{-2}	1.81×10^{-4}
Consequence Level:		Moderate	Low	Low	Low	Low	Low	Low	Low
Risk Class	**Anticipated**	I	III	III	III	III	III	III	III
	Unlikely	II	III	III	III	III	III	III	III
	Extremely Unlikely	III	IV	IV	IV	IV	IV	IV	IV

Controls	**Classification of Controls**
Combustible material controls inside gloveboxes	Specific Administrative Control
Confinement (1 stage HEPA)	SSC Category 2
Hot work controls	Administrative Control

Size reduction enclosure fires

Fires involving size reduction operations are postulated to occur due to ignition of transient combustible materials by sparks from size reduction operations, or heat/sparks from electrical equipment. An additional potential initiator could be an exothermic/pyrophoric reaction in a component in size reduction, or in an open waste container. The fire is most likely to occur during size reduction activities, when it can be detected and extinguished by personnel at an incipient or smoldering stage. Components undergoing size reduction include, but are not limited to, gloveboxes, equipment, ducts, and HEPA filters.

Size reduction work will produce three major waste streams:

- Contaminated metal;
- HEPA filters; and
- Combustible waste.

The primary waste stream will be contaminated metal, since the objective of size reduction is to cut large contaminated components into smaller pieces that will fit into drums or standard waste boxes. It is assumed that approximately 70% of any size reduction operation will be surface contaminated metals.

Size reduction tents or Moducon units that are constructed of fire retardant material will be used to size reduce large components. To facilitate size reduction activities, open containers such as drums or standard waste boxes will be sleeved to the enclosure. The contents of these containers would not normally contribute to the material at risk since the contents of these container represents smaller pieces of the component being size reduced. However, these containers are assumed to contain 50% of their maximum inventory to allow partially filled containers to be sleeved to the enclosure so that they may be filled to capacity. When a sleeved container is filled, it is closed and staged next to the enclosure until the material handlers can move it. Thus, several filled containers may be staged next to the enclosure at any given time. Containers that are sleeved to the enclosure are modeled as open containers. Containers that are staged next to the enclosure awaiting movement are modeled as closed containers.

For evaluation purposes, it is assumed that two ILW drums and one standard waste box are sleeved to the enclosure. Using the standard assumption that one ILW drum is overbatched at 250 g and one is at the maximum of 200 g, 50% of the total maximum capacity for the sleeved transuranic drums is 225 g. Also using the standard assumption that one standard waste box is overbatched at 400 g, 50% of the maximum capacity for the sleeved standard waste box is 200 g. It is also assumed that four ILW drums (each at 200 g) and two standard waste boxes (each at 320 g) are staged next to the enclosure awaiting movement.

Small fire, size reduction enclosure (Case E Scenarios)

Scenario description

These scenarios evaluate small fires (less than 1 MW) as originating within a size reduction enclosure or in the immediate vicinity of size reduction operations. The fire is postulated to involve a maximum of ten linear feet of any component (e.g. pipe or ventilation ducting) in size reduction. Due to the size of this fire, and the inability of the enclosure itself to propagate fire, the fire is postulated to involve only materials within the enclosure (e.g., a portion of the component in size reduction). The enclosure is not sufficiently breached to involve materials outside the enclosure or containers sleeved to it, or to activate room sprinklers. This fire could be initiated by general facility operations, waste handling activities, or decommissioning (e.g., size reduction) activities.

Assumptions

The size reduction enclosure is constructed of fire retardant material in accordance with procedures. The enclosure material is therefore considered to be incapable of propagating the fire.

A noncombustible fixative coating may be present on metal components; however, no credit is taken for the mitigating effect of such a coating.

Scenario E1: Small fire involving a glovebox in a size reduction enclosure

This scenario evaluates a small fire that is postulated to occur in Laboratory room 2 and involves size reduction of gloveboxes from Laboratory room 2 in a fixed-location size reduction enclosure. The highest residual in a glovebox is 442 g Pu-E.

The material-at-risk is the residual material in the glovebox and is modeled as unconfined noncombustible material, with a damage ratio of 1.0, since residual in contact with the fire is assumed to be entirely available for involvement.

Control set and risk class

The dose consequences for this scenario without active safety control measures are *low* to the worker (9.2 mSv) and *low* to the public (0.124 mSv). When evaluated at an *anticipated* frequency, the scenario is Risk Class III for the worker and Risk Class III for the public (see Table 16).

Since the case without active safety control measures scenario is Risk Class III for both receptors, no specifically credited controls are warranted. However, since the dose consequences to the worker are approaching the level that would result in a change to Risk Class II, one stage of HEPA filtration is credited to reduce the dose consequences to both receptors.

For the scenario with active safety control measures, the *low* dose consequences to the worker (0.013 mSv) evaluated at an *anticipated* frequency results in a Risk Class III category. The *low* dose consequences to the public (1.8×10^{-4} mSv) evaluated at an *anticipated* frequency also results in a Risk Class III category. While no defense in depth controls are specified, hot work controls and combustible material controls are identified as specifically required since the prevention of fires is preferable.

For the immediate worker, the dose consequences without active safety control measures are qualitatively assessed as *low*. When evaluated at an *anticipated* frequency, the event represents Risk Class III. While not required, the risk to the immediate worker can be reduced by evacuating the immediate area of the fire. Various aspects of the Safety Management Programmes such as training and fire protection ensure that workers in the immediate vicinity of the fire evacuate and that other workers in the facility are notified via the building fire alarms. The building alarms can be activated automatically by smoke detectors or manually by pull stations or fire phones. As these are all governed by Safety Management Programmes, no additional controls to protect the immediate workers require elevation to the safety measure level. With the immediate worker protection afforded by the Safety Management Programmes, the consequences with active safety control measures are qualitatively assessed as *low*. When evaluated at an *anticipated* frequency, the event represents Risk Class III (see Table 16).

TABLE 16 SMALL FIRE INVOLVING A GLOVEBOX IN A SIZE REDUCTION ENCLOSURE (SCENARIO E1)

Small Fire Involving a Glovebox in a Size Reduction Enclosure (Non-Lofted Fire)										
Radidose Parameters										
Contributor	**Material**	**χ/Q**	**Breathing Rate**	**Form of Material**	**DCF ICRP-68**	**DR**	**LPF**	**Release Duration**	**ARF**	**RF**
a. glovebox residual	Pu-E	95th	Heavy	Unconfined Noncombust.	Moderate	1.0	1.0	10	6 x 10^{-3}	1 x 10^{-2}

Radidose Results							
Contributor	**Without active safety control measure**			**With active safety control measures**			Notes:
	Pu-E	**Dose Consequences [Sv]**		**Pu-E**	**Dose Consequences [Sv]**		
		WORKER	**PUBLIC**		**WORKER**	**PUBLIC**	
a. glovebox residual	442	9.2 x 10^{-3}	1.2 x 10^{-4}	442	1.3 x 10^{-5}	1.8 x 10^{-7}	
Total:		9.2 x 10^{-3}	1.2 x 10^{-4}	**Total:**	1.3 x 10^{-5}	1.8 x 10^{-7}	

Risk Class Evaluation		**Dose Consequences with Active Safety Control Measures [Sv]**							
		0 HEPA		**1 HEPA**		**2 HEPA**		**0.1 LPF**	
		WORKER	**PUBLIC**	**WORKER**	**PUBLIC**	**WORKER**	**PUBLIC**	**WORKER**	**PUBLIC**
Dose Consequences:		9.2 x 10^{-3}	1.2 x 10^{-4}	1.3 x 10^{-5}	1.8 x 10^{-7}	2.7 x 10^{-8}	3.7 x 10^{-10}	9.2 x 10^{-4}	1.2 x 10^{-5}
Consequence Level:		Low	Low	Low	Low	Low	Low	Low	Low
Risk Class	**Anticipated**	III	III	III	III	III	III	III	III
	Unlikely	III	III	III	III	III	III	III	III
	Extremely Unlikely	IV	IV	IV	IV	IV	IV	IV	IV

Controls	**Classification of Controls**
Combustible material controls	Specific Administrative Control
Confinement (1 stage HEPA)	SSC Category 3
Hot work controls	Administrative Control

Large fire, size reduction enclosure (Case F Scenarios)

Scenario description

This scenario evaluates a postulated large fire (5 to 10 MW) that is assumed to originate within or immediately adjacent to a size reduction enclosure. This fire is modeled as a fast growing fire that results in lid loss for some containers. The fire is postulated to involve the entire component enclosed by the enclosure, any immediately adjacent component (for in-situ size reduction enclosures), contents of open waste containers sleeved to the enclosure, and any waste containers located adjacent to the enclosure, with temporary or permanent lids, awaiting transfer to an approved container staging location. Containers sleeved to the enclosure are modeled as open and containers located adjacent to the enclosure are modeled as closed and experiencing seal failure with the exception of two drums adjacent to the enclosure, which are assumed to experience lid loss.

In the case of an in-situ size reduction enclosure, it is expected that waste containers in the immediate vicinity of the enclosure will contain waste produced in the size reduction of the component(s) contained within the enclosure. The movement of nuclear material from this component and its placement in a waste container does not alter the total amount of nuclear material available for involvement in the scenario. For conservatism, containers sleeved to in-situ size reduction enclosures are assumed to contain material not generated in size reduction at that location. This also allows for the possibility of the movement of partially filled containers to the size reduction enclosure for addition of waste generated within the enclosure. Fires could be initiated by general facility operations, waste handling activities, or decommissioning (e.g., size reduction) activities.

Assumptions

The size reduction enclosure is constructed of fire retardant material in accordance with procedures. The enclosure material is therefore considered to be incapable of propagating the fire.

A noncombustible fixative coating may be present on metal components; no credit is taken for the mitigating effect of such a coating.

Scenario F1: Large fire involving a glovebox in a size reduction enclosure

This scenario postulates that a large fire occurs and involves a size reduction enclosure being used to size reduce a glovebox. The highest residual in a glovebox is 442 g Pu-E. The glovebox residual is modeled as unconfined noncombustible material, with a damage ratio of 1.0, since residual in contact with the fire is assumed to be entirely available for involvement. A damage ratio of 1 is used for the open containers sleeved to the enclosure. The ILW drums sleeved to the enclosure are assumed to be 50% full (based on one normal and one overbatched drum) and 70% of the components in the enclosure are assumed to be surface contaminated metals. Thus, the source term contribution for contaminated metal from the ILW drums is 155 g, which is evaluated as unconfined noncombustible material. The remaining 70 g contribution from the sleeved ILW drums is evaluated as confined material. The standard waste box sleeved to the enclosure is also assumed to be 50% full (based on an overbatched standard waste box) and 70% of the content is assumed to be surface contaminated metals. Thus, the source term contribution for contaminated metal from the standard waste box is 140 g, which is evaluated as unconfined noncombustible material. The remaining 60 g contribution from the sleeved standard waste box is evaluated as confined material.

For the four ILW drums staged outside the enclosure, two are assumed to experience lid loss and the other two are assumed to experience seal failure. Since the drums staged outside the enclosure are assumed to contain size reduced materials (i.e., 70% surface contaminated metals), 560 g is evaluated as unconfined noncombustible material and 240 g is evaluated as confined material. A damage ratio of 1.0 is used for the drums that experience lid loss and a damage ratio of 0.2 is used to represent seal failure. In addition, the two standard waste boxes staged outside the enclosure are also assumed to experience seal failure. The source term contribution from the staged standard waste boxes (based on 70% of the contents being surface contaminated metals) is 440 g evaluated as unconfined noncombustible material and 200 g evaluated as confined material. A damage ratio of 0.2 is also used for the standard waste boxes.

Control set and risk class

The dose consequences for this scenario without active safety control measures are 42.8 mSv for the worker and 0.38 mSv for the public. When evaluated at an *anticipated* frequency, the scenario is Risk Class I for the worker and Risk Class III for the public.

Fires are considered *anticipated*. Large fires are considered *extremely unlikely* when crediting combustible material controls and hot work controls in conjunction with fire suppression sprinklers.

One stage of HEPA filtration provides defense in depth.

For the scenario with active safety control measures, the *low* dose consequences to the worker (42.8 mSv) evaluated at an *extremely unlikely* frequency results in a Risk Class III category. The *low* dose consequences to the public (0.38 mSv) evaluated at an *extremely unlikely* frequency results in a Risk Class IV category.

For the immediate worker, the dose consequences without active safety control measures are qualitatively assessed as *low*. When evaluated at an *anticipated* frequency, the event represents Risk Class III. While not required, the risk to the immediate worker can be reduced by evacuating the immediate area of the fire. Various aspects of the Safety Management Programmes such as training and fire protection ensure that workers in the immediate vicinity of the fire evacuate and that other workers in the facility are notified via the building fire alarms. The building alarms can be activated automatically by smoke detectors or manually by pull stations or fire phones. As these are all governed by Safety Management Programmes, no additional controls to protect the immediate workers require elevation to the Safety Measure level. With the immediate worker protection afforded by the Safety Management Programmes, the consequences with are qualitatively assessed as *low*. When evaluated at an *extremely unlikely* frequency, the event represents Risk Class IV (see Table 17).

This scenario also bounds a similar large fire involving a size reduction enclosure when a piece of ductwork is being size reduced, since the source term contribution from the ductwork is less than the contribution from the glovebox.

TABLE 17 LARGE FIRE INVOLVING A GLOVEBOX IN A SIZE REDUCTION ENCLOSURE (SCENARIO F1)

Large Fire Involving a Glovebox in a Size Reduction Enclosure (Non-Lofted Fire)

Radidose Parameters

Contributor	Material	χ/Q	Breathing Rate	Form of Material	DCF ICRP-68	DR	LPF	Release Duration	ARF	RF
a. glovebox	Pu-E	95th	Heavy	Unconfined Noncombust.	Moderate	1.0	1.0	30	6 x 10^{-3}	1 x 10^{-2}
b. sleeved drums - contaminated metal	Pu-E	95th	Heavy	Unconfined Noncombust.	Moderate	1.0	1.0	30	6 x 10^{-3}	1 x 10^{-2}
c. sleeved drums - remaining waste	Pu-E	95th	Heavy	Confined Materials	Moderate	1.0	1.0	30	5 x 10^{-4}	1
d. sleeved SWB - contaminated metal	Pu-E	95th	Heavy	Unconfined Noncombust.	Moderate	1.0	1.0	30	6 x 10^{-3}	1 x 10^{-2}
e. sleeved SWB - remaining waste	Pu-E	95th	Heavy	Confined Materials	Moderate	1.0	1.0	30	5 x 10^{-4}	1
f. staged drums - lid loss - metal	Pu-E	95th	Heavy	Unconfined Noncombust.	Moderate	1.0	1.0	30	6 x 10^{-3}	1 x 10^{-2}
g. staged drums - lid loss - waste	Pu-E	95th	Heavy	Confined Materials	Moderate	1.0	1.0	30	5 x 10^{-4}	1
h. staged drums - seal failure - metal	Pu-E	95th	Heavy	Unconfined Noncombust.	Moderate	0.2	1.0	30	6 x 10^{-3}	1 x 10^{-2}
i. staged drums - seal failure - waste	Pu-E	95th	Heavy	Confined Materials	Moderate	0.2	1.0	30	5 x 10^{-4}	1
j. staged SWBs - contaminated metal	Pu-E	95th	Heavy	Unconfined Noncombust.	Moderate	0.2	1.0	30	6 x 10^{-3}	1 x 10^{-2}
k. staged SWBs - remaining waste	Pu-E	95th	Heavy	Confined Materials	Moderate	0.2	1.0	30	5 x 10^{-4}	1

Radidose Results

Contributor	Without active safety control measure			With active safety control measures			Notes: The totals to the left are different from the sum of the values listed due to roundoff error and significant digits.
	Pu-E	Dose Consequences [Sv]		Pu-E	Dose Consequences [Sv]		
		WORKER	PUBLIC		WORKER	PUBLIC	
a. glovebox	442	9.2 x 10^{-3}	1.2 x 10^{-4}	442	1.3 x 10^{-5}	1.8 x 10^{-7}	
b. sleeved drums - contaminated metal	155	2.59 x 10^{-3}	3.51 x 10^{-5}	155	2.59 x 10^{-3}	3.51 x 10^{-5}	
c. sleeved drums - remaining waste	70	9.74 x 10^{-3}	1.32 x 10^{-4}	70	9.74 x 10^{-3}	1.32 x 10^{-4}	
d. sleeved SWB - contaminated metal	140	2.34 x 10^{-3}	3.17 x 10^{-5}	140	2.34 x 10^{-3}	3.17 x 10^{-5}	
e. sleeved SWB - remaining waste	60	8.35 x 10^{-3}	1.13 x 10^{-4}	60	8.35 x 10^{-3}	1.13 x 10^{-4}	
f. staged drums - lid loss - metal	280	4.68 x 10^{-3}	6.34 x 10^{-5}	280	4.68 x 10^{-3}	6.34 x 10^{-5}	
g. staged drums - lid loss - waste	120	1.67 x 10^{-2}	2.27 x 10^{-4}	120	1.67 x 10^{-2}	2.27 x 10^{-4}	
h. staged drums - seal failure - metal	280	9.35 x 10^{-4}	1.27 x 10^{-5}	280	9.35 x 10^{-4}	1.27 x 10^{-5}	
i. staged drums - seal failure - waste	120	3.34 x 10^{-3}	4.53 x 10^{-5}	120	3.34 x 10^{-3}	4.53 x 10^{-5}	
j. staged SWBs - contaminated metal	440	1.47 x 10^{-3}	1.99 x 10^{-5}	440	1.47 x 10^{-3}	1.99 x 10^{-5}	
k. staged SWBs - remaining waste	200	5.57 x 10^{-3}	7.55 x 10^{-5}	200	5.57 x 10^{-3}	7.55 x 10^{-5}	
	Total:	4.28 x 10^{-2}	3.83 x 10^{-4}	**Total:**	3.35 x 10^{-2}	2.63 x 10^{-4}	

Risk Class Evaluation		Dose Consequences with Active Safety Control Measures [Sv]							
		0 HEPA		1 HEPA		2 HEPA		0.1 LPF	
		WORKER	PUBLIC	WORKER	PUBLIC	WORKER	PUBLIC	WORKER	PUBLIC
	Dose Consequences:	4.28 x 10^{-2}	3.83 x 10^{-4}	3.35 x 10^{-2}	2.63 x 10^{-4}			4.28 x 10^{-3}	3.8 x 10^{-5}
	Consequence Level:	Low	Low	Low	Low			Low	Low
Risk Class	**Anticipated**	III	III	III	III			III	III
	Unlikely	III	III	III	III			III	III
	Extremely Unlikely	III	IV	IV	IV			IV	IV

Controls	Controls
Combustible material controls	Specific Administrative Control
Fire suppression (sprinklers)	SSC Category 3
Hot work controls	Administrative Control
Confinement (1 stage HEPA)	SSC Category 3

(b) Spills

The hazards assessment identified spills that could result in uncontrolled releases of radioactive materials. This section presents analyses of three different scenarios that address spills of varying types affecting different activities within the 5 laboratory rooms:

- Container spills;
- Component drop/impact; and
- Ventilation reversal.

These scenarios are intended to bound the spectrum of potential spills in any of the 5 laboratory rooms.

- **Container spill (Scenarios G)**

These spills result from container movement activities, or other activities performed in the vicinity of staged containers. A container or containers are punctured, impacted, or dropped, resulting in the release of radioactive material.

Potential initiators for this event include kinetic energy sources such as operational, maintenance, or closure activity equipment, waste container or component handling equipment, and internally generated missiles; potential energy sources (e.g., falls or drops from high storage locations or during movement accompanied by lifting, elevator shafts, or stairwells); external events (e.g., surface vehicle impact); and natural phenomena events (e.g., earthquake, wind-generated missiles).

Scenario description

The following scenarios consider a spill involving various containers of nuclear material such as a 200 ℓ ILW or LLW drum, an standard waste box of ILW, or metal LLW crate. This spill could be caused by puncturing the container with a fork truck, dropping the container, or otherwise damaging the container (i.e., impact to the container by a compressed gas cylinder projectile).

Assumptions

For the purposes of this spill, it is assumed that waste container lids are held in place (e.g., chained) but not fastened (e.g., crimped or torqued) except for drum lids, which are assumed to have rings in place with bolts at least finger-tight. Waste containers used in staging areas are metal containers approved for use in on-site handling.

Scenario G1: ILW spill outside confinement

This scenario evaluates a spill that involves four ILW drums. The material at risk for the largest 4 drums based at maximum loading 800 g. Waste in the container is modeled as confined material, with a damage ratio of 1.0.

Control set and risk class

The dose consequences without active safety control measure for this scenario are over the exposure limit for the worker (28 mSv) and *low* to the public 0.38 mSv. When evaluated at an *anticipated* frequency, the scenario is Risk Class III for the worker and Risk Class III for the public (see Table 18).

For the immediate worker, the dose consequences without active safety control measures are qualitatively assessed as *low*. When evaluated at an *anticipated* frequency, the event represents Risk Class III. No controls to specifically protect the immediate workers require elevation to safety measure.

TABLE 18 TRANSURANIC WASTE SPILL OUTSIDE CONFINEMENT (SCENARIO G1)

Transuranic Waste Spill outside Confinement (Spill)											
Radidose Parameters											
Contributor	**Material**	**χ/Q**	**Breathing Rate**	**Form of Material**	**DCF ICRP-68**	**DR**	**LPF**	**Release Duration**	**ARF**	**RF**
a. drums	Pu-E	95th	Heavy	Confined Materials	Moderate	1.0	1.0	10	1×10^{-3}	1×10^{-1}

Radidose Results							
Contributor	**Without active safety control measure**			**With active safety control measures**			Notes: Outside of confinement, no mitigation available Source Term limited to 800g
	Pu-E	**Dose Consequences [Sv]**		**Pu-E**	**Dose Consequences [Sv]**		
		WORKER	**PUBLIC**		**WORKER**	**PUBLIC**	
a. drums	800	2.8×10^{-2}	3.8×10^{-4}	800	2.8×10^{-2}	3.8×10^{-4}	
Total:		2.8×10^{-2}	3.8×10^{-4}		2.8×10^{-2}	3.8×10^{-4}	

Risk Class Evaluation		**Dose Consequences with Active Safety Control Measures [Sv]**							
		0 HEPA		**1 HEPA**		**2 HEPA**		**0.001 LPF**	
		WORKER	**PUBLIC**	**WORKER**	**PUBLIC**	**WORKER**	**PUBLIC**	**WORKER**	**PUBLIC**
Dose Consequences:		2.8×10^{-2}	3.8×10^{-4}						
Consequence Level:		Low	Low						
Risk Class	**Anticipated**	III	III						
	Unlikely	III	III						
	Extremely Unlikely	III	III						

Controls	**Classification of**
Drums over 200 g Pu-E shall not be staged/stored outside confinement	Specific Administrative Control
Drums/SWBs shall not be stacked	Administrative Control

(c) Component drop/impact (Scenarios H)

These scenarios evaluate spills that could result from component movement activities or other activities performed near components. In these scenarios, a component is punctured, impacted, or dropped, which results in a release of radioactive material.

Potential initiators for this event include kinetic energy sources such as operational, maintenance, or closure activity, waste container or component handling equipment, and internally generated missiles, potential energy sources (e.g., falls or drops from high locations or during movement accompanied by lifting, elevator shafts, or stairwells), external events (e.g., surface vehicle impact), and natural phenomena events (e.g., earthquake, wind-generated missiles).

These scenarios also evaluate spills that could occur either inside or outside of confinement. Centralized size reduction areas may be established. Thus, contaminated equipment that requires size reduction may require movement from one room of the laboratory to another. The least hazardous method to accomplish this is by using the elevator.

Scenario description

The most likely scenario for the component drop/impact occurs during movement of a component from its previous fixed location to staging, size reduction, or packaging in a cargo or other container. Although components are generally provided with some radiological containment prior to movement, no credit is taken for the containment provided by measures such as shrink wrapping, end-capping, or fixative coating.

Assumptions

Component are assumed to be cut in 6 m lengths, as this is conservatively longer than the maximum 4.7 m length that can currently be placed in a size reduction enclosure, or the 5.5 m length that can be placed in a cargo container. The majority of contamination is assumed to be relatively non-dispersible.

Scenario H1: duct drop/impact

This scenario assumes a spill occurs that involves the drop/impact of a 6 m worst-case segment of contaminated Zone I duct from Laboratory room 2. The worst-case contaminated duct is assumed to have an average concentration of 6 g Pu-E per 0.3 linear meter of duct. A 6 m segment of this worst-case duct would contain 120 g Pu-E. A damage ratio of 0.1 is used, and contamination is modeled as unconfined noncombustible material.

Control set and risk class

The dose consequences without active safety control measures for this scenario are *low* to the worker (4.16 mSV) and *low* to the public (0.056 mSv). When evaluated at an *anticipated* frequency, the scenario is Risk Class III for the worker and Risk Class III for the public (see Table 19).

Since the scenario is Risk Class III to both receptors and the dose consequences are below the evaluation guidelines, no controls are specifically required to reduce the frequency or consequences of the event. This event also bounds the spill of a low level waste box since these contain less source term. Since this event does not require HEPA filtration, the scenario bound the event inside or outside of the facility. For the immediate worker, the dose consequences without active safety control measures are qualitatively assessed as *low*. When evaluated at an *anticipated* frequency, the event represents Risk Class III (see Table 19). No controls to specifically protect the immediate workers require elevation to the safety measure.

TABLE 19 DUCT DROP/IMPACT (SCENARIO H1)

Duct Drop/Impact (Spill)

Radidose Parameters

Contributor	Material	χ/Q	Breathing Rate	Form of Material	DCF ICRP-68	DR	LPF	Release Duration	ARF	RF
a. duct	Pu-E	95th	Heavy	Unconfined Noncombust.	Moderate	0.1	1.0	10	1×10^{-3}	1.00

Radidose Results

Contributor	Without active safety control measure			With active safety control measures			Notes:
	Pu-E	Dose Consequences [Sv]		Pu-E	Dose Consequences [Sv]		
		WORKER	PUBLIC		WORKER	PUBLIC	
a. duct	120	4.16×10^{-3}	5.65×10^{-5}	120	4.16×10^{-3}	5.6×10^{-5}	
Total:		4.16×10^{-3}	5.65×10^{-5}	Total:	4.16×10^{-3}	5.65×10^{-5}	

Risk Class Evaluation		Dose Consequences with Active Safety Control Measures [Sv]							
		0 HEPA		1 HEPA		2 HEPA		0.001 LPF	
		WORKER	PUBLIC	WORKER	PUBLIC	WORKER	PUBLIC	WORKER	PUBLIC
Dose Consequences:		4.16×10^{-3}	5.65×10^{-5}	6.10×10^{-6}	8.27×10^{-8}	1.22×10^{-8}	1.65×10^{-10}	4.16×10^{-4}	5.65×10^{-6}
Consequence Level:		Low	Low	Low	Low	Low	Low	Low	Low
Risk Class	Anticipated	III	III	III	III	III	III	III	III
	Unlikely	III	III	III	III	III	III	III	III
	Extremely Unlikely	IV	IV	IV	IV	IV	IV	IV	IV

Controls	Classification of Controls
No specifically credited controls are warranted	N/A

Scenario H2: glovebox drop/impact

This scenario assumes that a spill occurs that involves the drop of a worst-case segment of contaminated glovebox from Laboratory room 2. The worst-case glovebox segment is estimated at 442 g Pu-E. A damage ratio of 0.1 is used, and contamination is modeled as unconfined noncombustible material. This spill could occur anywhere inside confinement.

Control set and risk class

The dose consequences without active safety control measures for this scenario are *low* to the worker (1.5 mSv) and *low* to the public (0.21 mSv). When evaluated at an *anticipated* frequency, the scenario is Risk Class III for the worker and Risk Class III for the public (see Table 20).

Spills are considered *anticipated*. No controls have been identified that would reduce the frequency of this event.

Crediting one stage of HEPA filtration reduces the consequences to *low* for both the public and the worker.

For the immediate worker, the dose consequences without active safety control measure are qualitatively assessed as *low*. When evaluated at an *anticipated* frequency, the event represents Risk Class III. No controls to specifically protect the immediate workers require elevation to Safety Measures.

TABLE 20 GLOVEBOX DROP/IMPACT (SCENARIO H2)

Glovebox Drop/Impact (Spill)										
Radidose Parameters										
Contributor	**Material**	**χ/Q**	**Breathing Rate**	**Form of Material**	**DCF ICRP-68**	**DR**	**LPF**	**Release Duration**	**ARF**	**RF**
a. glovebox	Pu-E	95th	Heavy	Unconfined Noncombust.	Moderate	0.1	1.0	10	1 x 10^{-3}	1.00

Radidose Results							
Contributor	**Without active safety control measure**			**With active safety control measures**			Notes:
	Pu-E	**Dose Consequences [Sv]**		**Pu-E**	**Dose Consequences [Sv]**		
		WORKER	**PUBLIC**		**WORKER**	**PUBLIC**	
a. glovebox	442	1.5 x 10^{-3}	2.1 x 10^{-5}	442	1.5 x 10^{-1}	2.1 x 10^{-3}	
Total:		1.5 x 10^{-3}	2.1 x 10^{-5}	**Total:**	1.5 x 10^{-1}	2.1 x 10^{-3}	

Risk Class Evaluation		Dose Consequences with Active Safety Control Measures [Sv]							
		0 HEPA		**1 HEPA**		**2 HEPA**		**0.001 LPF**	
		WORKER	**PUBLIC**	**WORKER**	**PUBLIC**	**WORKER**	**PUBLIC**	**WORKER**	**PUBLIC**
Dose Consequences:		1.5 x 10^{-3}	2.1 x 10^{-5}	2.2 x 10^{-4}	3.0 x 10^{-6}	4.5 x 10^{-7}	6.1 x 10^{-9}	1.5 x 10^{-2}	2.1 x 10^{-4}
Consequence Level:		Low	Low	Low	Low	Low	Low	Low	Low
Risk Class	**Anticipated**	III	III	III	III	III	III	III	III
	Unlikely	III	III	III	III	III	III	III	III
	Extremely Unlikely	III	IV	IV	IV	IV	IV	IV	IV

Specifically Credited Controls	Controls Selected for Defense in Depth
Confinement (1 stage HEPA)	No additional controls for defense in depth are available.

(d) Ventilation reversal (Scenarios I)

These scenarios evaluate variations of ventilation reversals. A ventilation reversal can be caused by loss of power to air moving equipment including fans and portable air movers, or by malfunction of ventilation equipment including fans, portable air movers, dampers, and interlocks.

Scenario description

A ventilation reversal is modeled as a loss of flow/differential pressure with a momentary backward pressure gradient that has the potential to disperse airborne material, and to resuspend some percentage of unconfined particulate material, if such material is present. A ventilation reversal could be initiated by loss of power or equipment failure. This scenario affects all facility operations, in all stages of decommissioning, although the consequences of this scenario will necessarily decrease as the building inventory as residual decreases, and as the contamination level of components in size reduction decreases. The ventilation reversal, although modeled for Zone I (within the most contaminated primary system, includes glove box and ducting), is applicable to all areas of the building as it will be used to bound a Zone II (room or building) ventilation reversal.

Assumptions

It is assumed that the majority of building residual is not easily dispersible. It is further assumed that the motive force imparted to residual on metal surfaces by the ventilation reversal is significantly less than that imparted by a component drop or impact.

Scenario I1: Zone I ventilation reversal

This scenario evaluates a Zone I ventilation reversal, modeled as a loss of Zone I ventilation with a momentary pressure gradient that favors airflow outward from Zone I confinement structures. The material-at-risk for this event is assumed to be the inventory as residual for the worst-case Zone I confinement structure, including straight-line ventilation pathways to the Zone I plenum that serves the laboratory. In addition, the flow reversal is assumed to pressurize gloveboxes. While this release would actually be confined by the Zone II system, no credit is taken for the confinement provided by the Zone II ventilation system or Building confinement.

The total duct residual is estimated as 120 g Pu-E. This quantity is used as the material-at-risk for a ventilation reversal. The majority of this residual exists as powder on metal surfaces. For evaluation purposes, a damage ratio of 0.1 is conservatively used. Since gloveboxes connected to the ventilation system may experience some pressurization, residual in these gloveboxes is also included in the material at risk. The residual in gloveboxes is estimated at 442 g Pu and is evaluated as Pu-E. Due to the size of the ventilation connection relative to the size of gloveboxes, the gloveboxes would experience much less of a pressure transient. The residual in the gloveboxes is evaluated as powder with a damage ratio of 0.01. This scenario bounds the case of a Zone II ventilation reversal.

Control set and risk class

The dose consequences without active safety control measures for this scenario are *low* to the worker (0.42 mSv) and *low* to the public (5.6 x 10^{-3} mSv). When evaluated at an *anticipated* frequency, the scenario is Risk Class III for the worker and Risk Class III for the public (see Table 21).

Since the scenario is Risk Class III to both receptors and the dose consequences are below the evaluation guidelines, no controls are specifically required to reduce the frequency or consequences of the event.

For the immediate worker dose consequences without active safety control measure are qualitatively assessed as *low*. When evaluated at an *anticipated* frequency, the event represents Risk Class III. No controls to specifically protect the immediate workers require elevation to Safety Measures.

TABLE 21 ZONE I VENTILATION REVERSAL (SCENARIO I1)

Zone I Ventilation Reversal (Spill)										
Radidose Parameters										
Contributor	**Material**	**χ/Q**	**Breathing Rate**	**Form of Material**	**DCF ICRP-68**	**DR**	**LPF**	**Release Duration**	**ARF**	**RF**
a. duct residual	Pu-E	95th	Heavy	Powder (Oxide)	Moderate	0.1	1.0	10	2 x 10^{-3}	3 x 10^{-1}
b. glovebox residual	Pu-E	95th	Heavy	Powder (Oxide)	Moderate	0.01	1.0	10	2 x 10^{-3}	3 x 10^{-1}

Radidose Results							
Contributor	**Without active safety control measure**			**With active safety control measures)**			Notes:
	Pu-E	**Dose Consequences [Sv]**		**Pu-E**	**Dose Consequences [Sv]**		
		WORKER	**PUBLIC**		**WORKER**	**PUBLIC**	
a. duct residual	120	4.2 x 10^{-4}	5.6 x 10^{-6}	120	4.2 x 10^{-4}	5.6 x 10^{-6}	
b. glovebox residual	442	1.5 x 10^{-4}	2.1 x 10^{-6}	2 232	1.5 x 10^{-4}	2.1 x 10^{-6}	
	Total:	5.7 x 10^{-4}	7.7 x 10^{-6}	**Total:**	5.7 x 10^{-4}	7.7 x 10^{-6}	

Risk Class Evaluation		**Dose Consequences with Active Safety Control Measures [Sv]**							
		0 HEPA		**1 HEPA**		**2 HEPA**		**0.001 LPF**	
		WORKER	**PUBLIC**	**WORKER**	**PUBLIC**	**WORKER**	**PUBLIC**	**WORKER**	**PUBLIC**
	Dose Consequences:	5.7 x 10^{-4}	7.7 x 10^{-6}	8.30 x 10^{-7}	1.13 x 10^{-8}	1.65 x 10^{-9}	2.31 x 10^{-11}	5.7 x 10^{-5}	7.7 x 10^{-7}
	Consequence Level:	Low	Low	Low	Low	Low	Low	Low	Low
Risk Class	**Anticipated**	III	III	III	III	III	III	III	III
	Unlikely	III	III	III	III	III	III	III	III
	Extremely Unlikely	IV	IV	IV	IV	IV	IV	IV	IV

Controls	**Classification of Controls**
No specifically credited controls are warranted	N/A

(e) Explosions (Scenarios J)

Scenario description

These scenarios evaluate the potential explosive hazards of flammable gases. Acetylene gas is lighter than air so, if it is released into a room, it will tend to rise to the ceiling. Acetylene is unstable when pressurized, so it is dissolved in acetone inside its storage cylinder. When pressure is released from the top of the cylinder, acetylene comes out of solution from the acetone. Propane and welding gas are heavier than air, and will tend to sink to the floor when released in a room. Both propane and welding are liquids in a storage cylinder with a pressurized vapor space at the top of the tank. As pressure is released from the top of the cylinder, liquid vaporizes to restore the pressure. The ignition of a flammable gas under most postulated accident conditions will result in a deflagration. The deflagration creates an overpressure that may cause damage to building structures and containers or confinement systems containing nuclear materials.

Two general types of explosions are evaluated. The first condition evaluated is a jet release, of the type that would result if the regulator assembly were sheared off a tank. Although the ignition of the gas jet would result in a deflagration, the mixing in the jet results in a turbulent reaction that is modeled like a detonation. The resulting overpressure is a function of distance. Because welding gas and propane are in liquid form in the storage cylinders, if the cylinder were to tip over in conjunction with having the regulator sheared off (which is not unlikely), a mixture of gas and liquid would be released. Since the liquid portion of the release would immediately vaporize, this configuration allows the material in the bottle to be released more quickly. The consequences of ignition of this type of release are significantly greater than for the corresponding gas jet deflagration. If an acetylene tank were tipped on its side, some liquid acetone would be released from the tank along with the acetylene gas. This acetone release actually reduces the rate of release of acetylene gas. The consequence of a turbulent gas jet deflagration is theoretically independent of bottle size. A turbulent propane jet could cause an overpressure up to 110 kPa near the center of the deflagration, which is not sufficient to fail waste containers. Overpressures approximately 2 m from the center of the deflagration could reach 15 kPa which would not damage reinforced concrete walls.

The second condition evaluated is a vapor cloud explosion. If flammable gas is not ignited in a jet as it escapes from its cylinder, it could accumulate in a vapor cloud and deflagrate. The resulting overpressure is a function of the room volume and cylinder size. Standard cylinder sizes in use are:

— For acetylene, the MC bottle (0.28 m^3); the B bottle (1.1 m^3); and the WSL bottle (3.68 m^3); and

— For propane or MAPP gas the sizes are 0.45 kg (for a hand-held torch), 2.2 kg, and 9 kg.

Standard industrial equipment powered by propane (e.g., fork trucks) typically use a 13 kg or larger cylinder. A 0.45 kg bottle of propane or MAPP could theoretically cause an overpressure of approximately 8 kPa in a 360 m^3 room. This is not expected to damage a waste container or reinforced concrete walls.

While cinder block walls or standard studded wall-board walls might be damaged by such an explosion. While a collapsing cinder block wall might damage waste containers staged on the collapsing side of the wall, there is insufficient mass involve with standard studded wall-board walls to cause such damage. No areas inside the facility where an explosion could cause a collapse of a cinder block wall that would result in damage to waste containers were identified. Consequently, such an event is not evaluated for the facility.

Assumptions

Torch cutting using acetylene is expected to be used for size reduction of essentially uncontaminated equipment in the Offices, and the Control Room Area. Acetylene may also be used in these areas for maintenance applications. MAPP gas is not currently used at the facility but is addressed by the analysis for any future proposed uses. Propane may be used for cutting torches, for maintenance applications, or to power industrial equipment.

At 14 kPa overpressure, corrugated metal panels fail. Non-reinforced concrete or cinderblock walls are assumed to shatter at 20 kPa overpressure. Based on the results of studies performed at Rocky Flats on the consequences of explosions on gloveboxes, it will be assumed that gloveboxes also fail at approximately 14 kPa overpressure. This event is not postulated to result in fire, as the flame front from a deflagration moves with such velocity that objects of any significant thermal inertia would not be ignited. However, damage caused by falling debris can cause releases from waste containers or residual.

Waste containers are assumed to be metal containers approved for use in on-site shipping. Radioactive waste is not packaged in wooden containers. The effect of explosive overpressures on 200 ℓ drums has also been analyzed. It has been concluded that 200 ℓ drums will fail at 151 kPa overpressure in deflagrations. This analysis will assume that the same is true for standard waste boxes and metal LLW crates. For a vapor cloud explosion in larger rooms, a damage ratio of 0.1 could be used for drums since the pressure rise would be uniform on the drums and the dynamic pressure is not expected to be sufficient to overturn the drums.

Scenario J1: Oxy-acetylene vapour cloud explosion in OPS clean area

This scenario evaluates an acetylene vapor cloud explosion that involves LLW in metal crates and ILW in 200 ℓ drums. The case is postulated to occur in any Operationally Clean area of the 5 laboratory rooms where waste containers are aggregated. This case represents waste staging activities in the facility during the cold strip-out stage (e.g. source term has been removed, building debris from scabbling operations, removal of electrical fixtures, and other facility support systems) of decommissioning activities. Waste staged in areas declared operationally clean, and undergoing the process of cold strip-out, is limited to waste produced and packaged in that area.

LLW waste is modeled as contaminated noncombustible material. ILW is modeled as confined material; combustible wastes produced during cold strip-out activities (contaminated protective clothing, plastic used for contamination control, fiberboard ceiling tiles, etc.) will be packaged as LLW. Significant amounts of noncombustible LLW will also be produced (sprinkler piping, light fixtures, cinder block and concrete); the choice to model all LLW as combustible is therefore conservative.

For this scenario, six LLW waste crates at 3 g Pu-E and two ILW waste drums (one at 200 g and one overloaded at 250 g Pu-E) are assumed to be involved. This is partially based on the control that drums over 200 g Pu-E shall not be staged/stored outside confinement. Although this quantity of ILW waste is not anticipated to remain in areas declared operationally clean, this allows the introduction and use of a second container prior to removal of a filled first container, as necessary. This results in a material-at-risk of 468 g Pu-E. A damage ratio of 0.1 is use for the drums since the over-pressurization is not expected to overturn the drums. Airborne release fraction (ARF) of 5.0×10^{-3} and a respirable fraction (RF) of 0.3 is used. The material in drums is modeled as confined material while the LLW waste is conservatively modeled as unconfined noncombustible material with a damage ratio (DR) of 1.

Control set and risk class

The dose consequences without active safety control measure for this scenario are *low* to the worker (7.80 mSv) and *low* to the public (0.16 mSv). When evaluated at an *anticipated* frequency, the scenario is Risk Class III for the worker and Risk Class III for the public.

Since the scenario is Risk Class III to both receptors and the dose consequences are below the evaluation guidelines, no additional controls are specifically required to reduce the frequency or consequences of the event (see Table 22).

For the immediate worker, the dose consequences without active safety control measures are qualitatively assessed as *high* since an explosion has the potential to cause a fatality, even though the fatality would be due to an industrial hazard (acetylene explosion) and not radiological uptake. When evaluated at an *anticipated* frequency, the event represents Risk Class I. The potential for serious injury or significant radiological exposure can be reduced by evacuating the immediate area of the event. Various aspects of the Safety Management Programmes such as training to evacuate the immediate vicinity of the event and emergency response protect the immediate worker. The dose consequences of the event for workers in other areas are qualitatively assessed as low due to the limited radiological hazard. Although the attending immediate worker could be seriously or fatally injured due to the explosion even in the case with active safety control measures, this event is not carried forward as a risk dominant scenario for the immediate worker since the injury is due to an industrial hazard and not a radiological hazard. With the immediate worker protection afforded by the Safety Management Programmes, the consequences with active safety control measures are qualitatively assessed as *low*. When evaluated at an *anticipated* frequency, the event represents Risk Class III. Since the programs governed by the Safety Management Programmes provide adequate protection for the immediate worker, no additional controls to specifically protect the immediate workers require elevation to Safety Measure.

TABLE 22 OXY-ACETYLENE VAPOR CLOUD EXPLOSION IN OPS CLEAN AREA (SCENARIO J1)

Oxy-acetylene Vapor Cloud Explosion in OPS Clean Area (Spill)

Radidose Parameters

Contributor	Material	χ/Q	Breathing Rate	Form of Material	DCF ICRP-68	DR	LPF	Release Duration	ARF	RF
a. LLW fraction	Pu-E	95th	Heavy	Unconfined Noncombust.	Moderate	1.0	1.0	10	1 x 10^{-3}	1.00
b. drum fraction	Pu-E	95th	Heavy	Confined Materials	Moderate	0.1	1.0	10	1 x 10^{-3}	1 x 10^{-1}

Radidose Results

Contributor	Without active safety control measure			With active safety control measures			Notes:
	Pu-E	Dose Consequences [Sv]		Pu-E	Dose Consequences [Sv]		b. ARF/RF for explosion involving confined material is 1.0x10^{-3}/0.1
		WORKER	PUBLIC		WORKER	PUBLIC	
a. LLW fraction	18	6.24 x 10^{-3}	8.47 x 10^{-5}	18	6.24 x 10^{-3}	8.47 x 10^{-5}	
b. drum fraction	450	1.56 x 10^{-3}	2.12 x 10^{-5}	450	1.56 x 10^{-3}	2.12 x 10^{-5}	
	Total:	7.80 x 10^{-3}	1.06 x 10^{-4}	**Total:**	7.80 x 10^{-3}	1.06 x 10^{-4}	

Risk Class Evaluation		Dose Consequences with Active Safety Control Measures [Sv]							
		0 HEPA		1 HEPA		2 HEPA		0.1 LPF	
		WORKER	PUBLIC	WORKER	PUBLIC	WORKER	PUBLIC	WORKER	PUBLIC
	Dose Consequences:	7.80 x 10^{-3}	1.06 x 10^{-4}	1.14 x 10^{-5}	1.55 x 10^{-7}	2.29 x 10^{-8}	3.10 x 10^{-10}	7.80 x 10^{-4}	1.06 x 10^{-5}
	Consequence Level:	Low	Low	Low	Low	Low	Low	Low	Low
Risk Class	**Anticipated**	III	III	III	III	III	III	III	III
	Unlikely	III	III	III	III	III	III	III	III
	Extremely Unlikely	IV	IV	IV	IV	IV	IV	IV	IV

Controls	Classification of Controls
Drums over 200 g Pu-E shall not be staged/stored outside confinement (R)	Administrative Control

Scenario J2: Hydrogen deflagration in a 200 ℓ drum

This scenario postulates a spill due to the ignition of hydrogen that has accumulated in a 200 ℓ drum. These drums are vented and the vents are inspected. However, there is potential for container vents to fail through plugging with corrosion products.

Waste containers generated in the laboratory will be shipped out for counting, and subsequently to the onsite waste storage facilities, as expediently as possible. The decommissioning mission of the facility does not allow for long-term waste storage. Therefore, hydrogen generation in waste packaged at the facility and the potential for plugging of container vents will be reduced through minimization of storage time.

This scenario involves a hydrogen deflagration in a 200 ℓ drum containing waste. Assuming a drum contains 200 g bounds the largest drum to be packaged. The contents of the drum are modeled as confined material. Default parameters have been altered, as follows: Airborne Release Fraction: (ARF) 2.0 x 10^{-2}; Respirable Fraction: (RF) 7.0 x 10^{-1}. The damage ratio (DR) for this scenario is 0.1.

The case of hydrogen deflagration in a 200 ℓ drum of transuranic waste is expected to bound that of a deflagration occurring in the head space of a standard waste box, as the standard waste box is not expected to experience lid loss due to the internal deflagration. Low-level waste is not considered, as containers of low-level waste are not expected to generate hydrogen in sufficient quantities to over-pressurize the container or form a mixture in the explosive range.

There is also a small potential for hydrogen accumulation in hydrogen-generating tanks. These tanks are vented to reduce the potential for accumulation of hydrogen in the head space. The consequences of a hydrogen deflagration in an operationally empty tank are bounded by this scenario.

Experimental work documented in DOE STD 5506-2007 [41] has determined that ignition of mixtures of 14.5 % hydrogen in air in a sealed 55-gal[8] steel ILW waste drums does not lead to lid loss. However, for a drum with a hydrogen concentration > 15%, the overpressure within the drum due to a deflagration explosion is assumed to be sufficient to separate the lid from the drum and release a fraction of the drum contents. Based on complex-wide operational experience, the frequency for an explosion in a 200 ℓ drum without active safety control measures is *unlikely*.

Control set and risk class

The dose consequences without active safety control measure for this scenario are *moderate* to the worker (971 mSv) and *low* to the public (1.32 mSv). When evaluated at an *unlikely* frequency, the scenario is Risk Class I for the worker and Risk Class II for the public.

Crediting the drum vents reduces the frequency of the event from *unlikely* to *extremely unlikely*.

Crediting one stage of HEPA filtration reduces the consequences to *low* for both the public and the worker. For the scenario with active safety control measures, the *low* dose consequences to the worker (1.42 mSv) evaluated at an *extremely unlikely* frequency results in a Risk Class IV category. The *low*

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[8] The physical characteristics of the 55 gallon drum are similar to the 200 ℓ drum in the critical dimensions ℓ/d.

dose consequences to the public (0.019 mSv) evaluated at an *extremely unlikely* frequency results in a Risk Class IV category (see Table 23).

For the immediate worker, the dose consequences without active safety control measures are qualitatively assessed as *moderate* since while the event has the potential to cause serious injury or a significant radiological exposure, the lid blowing off a drum is not expected to cause prompt death. When evaluated at an *unlikely* frequency, the event represents Risk Class I.

The potential for significant radiological exposure can be reduced by evacuating the immediate area of the deflagration. Various aspects of the Safety Management Programmes such as training to evacuate the immediate vicinity of the event and emergency response protect the immediate worker. With the immediate worker protection afforded by the Safety Management Programmes, the consequences with active safety control measures are qualitatively assessed as *low*. When evaluated at an *extremely unlikely* frequency, the event represents Risk Class IV. Since the programs governed by the Safety Management Programmes provide adequate protection for the immediate worker, no additional controls to specifically protect the immediate workers require elevation Safety Measures.

TABLE 23 HYDROGEN DEFLAGRATION IN A 200 ℓ DRUM (SCENARIO J2)

Hydrogen Deflagration in a 200 l Drum (Overpressurization)										
Radidose Parameters										
Contributor	**Material**	**χ/Q**	**Breathing Rate**	**Form of Material**	**DCF ICRP-68**	**DR**	**LPF**	**Release Duration**	**ARF**	**RF**
a. drum	Pu-E	95th	Heavy	Confined Materials	Moderate	0.1	1.0	10	2 x 10^{-2}	7 x 10^{-1}

Radidose Results							
Contributor	**Without active safety control measure**			**With active safety control measures**			Notes: ARF and RF based on Rocky Flats Safety Analysis and Risk Assessment Handbook Section 6.3.6.8 Item 1.
	Pu-E	**Dose Consequences [Sv]**		**Pu-E**	**Dose Consequences [Sv]**		
		worker	**public**		**worker**	**public**	
a. drum	2 000	9.71 x 10^{-1}	1.32 x 10^{-2}	2 000	1.42 x 10^{-3}	1.93 x 10^{-5}	
	Total:	9.71 x 10^{-1}	1.32 x 10^{-2}	**Total:**	1.42 x 10^{-3}	1.93 x 10^{-5}	

Risk Class Evaluation		**Dose Consequences with Active Safety Control Measures [Sv]**							
		0 HEPA		**1 HEPA**		**2 HEPA**		**0.1 LPF**	
		WORKER	**PUBLIC**	**WORKER**	**PUBLIC**	**WORKER**	**PUBLIC**	**WORKER**	**PUBLIC**
Dose Consequences:		9.71 x 10^{-1}	1.32 x 10^{-2}	1.42 x 10^{-3}	1.93 x 10^{-5}	2.85 x 10^{-6}	3.86 x 10^{-8}	9.71 x 10^{-2}	1.32 x 10^{-3}
Consequence Level:		Moderate	Low	Low	Low	Low	Low	Low	Low
Risk Class	**Anticipated**	I	III	III	III	III	III	III	III
	Unlikely	II	III	III	III	III	III	III	III
	Extremely Unlikely	II	III	IV	IV	IV	IV	IV	IV

Controls	**Classification of Controls**
Drum vents	SSC Category 2
1 stage HEPA filtration	SSC Category 2

(f) External events (Scenarios K)

A discrete number of events initiating outside the facility have the capability to impact the facility from a nuclear safety perspective. These events are:

- Loss of power;
- Crane load drop;
- Surface vehicle impact; and
- Aircraft crash.

Scenario description

Because the external events are diverse in nature and similar only in that they initiate outside the facility, the scenario description for each scenario is given with the discussion of that individual scenario. The external events are applicable in all areas of the facility, and to all facility operations. The consequences of these events will tend to decrease as facility inventory as residual is decreased through the decommissioning process.

Scenario K1: Loss of power

"Loss of power," a condition caused by an external event, could cause a release of Pu. Loss of power postulates a loss of all AC power and has been evaluated for commercial nuclear facilities.

Accident scenario

Initiators of loss of AC power include severe weather (e.g., snow, floods, lightning) or birds and animals that disrupt offsite power supplies. The turbine-generator power system is assumed to fail because of a malfunction or because it is out of service for maintenance. The UPS system, supplying control power to the Utilities Control Room and other safety systems, would normally remain functional but is not relied upon to mitigate the consequences of the event.

A loss of power will disable all equipment requiring AC power. This includes all HVAC fans, HP vacuum pumps, normal (but not emergency) lighting, air compressors, breathing air compressors, vacuum pumps, and process equipment requiring AC power. Equipment that is expected to remain functional, because it is passive or is supplied by UPS, includes the following:

- Fire Detection: The fire alarm panel backup batteries provide power to the heat detection circuits;

- Continuous Air Monitors (CAM): The effluent CAMs and their emergency blowers are powered by uninterrupted power supply (UPS) System;

- The emergency warning system is supplied by UPS;

- The UPS System: The system provides power to HVAC System 1 and System 2 isolation valves and indication of their status. Controls for the HVAC systems, which are supplied by UPS System, including the pressure differential controllers, I/P converters, and pressure transmitters; and

- Instrument Air/Plant Air: The system will continue to provide compressed gas to equipment because of stored air in the IA/PA receiver tanks. When the air pressure is reduced sufficiently (approximately 206 kPa), nitrogen gas from the nitrogen tanks will provide a source of compressed gas.

The primary release mechanism resulting from a loss of power is contamination in gloveboxes and plenums as negative pressure is lost. See the evaluation of flow reversal evaluated above as bounding.

Scenario K2: Crane load drop

The postulated crane load drop event occurs in the facility and involves the drop of a crane load, impacting material staged for waste handling. The crane load of concern could be a large component (e.g., exhaust or supply fan, large glovebox). The glovebox is chosen for analysis, as its footprint (approximately 15 m^2) is expected to bound that of any crane-lifted single component. The cargo container is postulated to be dropped on thirty 200 ℓ drums of ILW waste, awaiting transfer from the facility. Drums are chosen because standard waste boxes are not expected to be affected as significantly by the impact of the glovebox. The number of drums impacted is assumed to be the size of one waste shipment. Although the glovebox is assumed to have all loose material removed or fixed to meet transport requirements, this material is not included in the spill, since the glovebox is not assumed to be substantially breached by the drop. The contents of the highest thirty drums is bounded by 16 kg based on the inventory at the time this document was prepared. A damage ratio (DR) of 1 is used to represent the release expected from a crushing blow. The waste in containers is modeled as confined material. Assuming all of the work has been completed and the total inventory is staged for transfer to the interim waste store, the source term evaluated is 1 347 g.

Control set and risk class

The dose consequences without active safety control measures for this scenario are 46.7 mSv *for a* worker and 0.643 mSv for a member of the public. Based on the number of lifts performed and the lifting requirements associated with the movement of heavy loads, the frequency for a crane load drop without active safety control measures is evaluated at an *unlikely* frequency. When evaluated at an *unlikely* frequency, the scenario is Risk Class III for the worker and Risk Class III for the public.

For the immediate worker, the dose consequences without active safety control measures are qualitatively assessed as *low*. When evaluated without active safety control measures at an *unlikely* frequency, the event is Risk Class III. When the case with active safety control measures is evaluated at an *extremely unlikely* frequency, the event is Risk Class IV. No controls to specifically protect the immediate workers require elevation to Safety Measures (see Table. 24).

Vehicular traffic also presents a potential kinetic energy hazard. The Laboratory Complex is surrounded by barriers which prevent interactions with normal traffic. Traffic within the immediate area of the laboratory is limited to security vehicles, cargo transportation vehicles, service vehicles, and various types of construction equipment. The laboratory structure provides considerable protection against an impact from vehicular traffic. Therefore, a release caused by a vehicular impact with the facility is not considered. Since waste shipments may be staged outside the facility, a vehicular impact with such a staged shipment must be evaluated. The Crane Load Drop scenario assumes a cargo container impacts a shipment of staged drums. Since this scenario already evaluates such an occurrence, a vehicular impact into such staged drums is bounded by this scenario.

TABLE 24 CRANE LOAD DROP (SCENARIO K2)

Crane Load Drop (Spill)										
Radidose Parameters										
Contributor	**Material**	**χ/Q**	**Breathing Rate**	**Form of Material**	**DCF ICRP-68**	**DR**	**LPF**	**Release Duration**	**ARF**	**RF**
a. drums	Pu-E	95th	Heavy	Confined Materials	Moderate	1.0	1.0	10	1 x 10^{-3}	1 x 10^{-1}

Radidose Results							
Contributor	**Without active safety control measure**			**With active safety control measures**			Notes:
	Pu-E	**Dose Consequences [Sv]**		**Pu-E**	**Dose Consequences [Sv]**		
		WORKER	**PUBLIC**		**WORKER**	**PUBLIC**	
a. drums	1 347	4.67 x 10^{-2}	6.34 x 10^{-4}	1 347	4.67 x 10^{-2}	6.34 x 10^{-4}	
	Total:	4.67 x 10^{-2}	6.34 x 10^{-4}	**Total:**	4.67 x 10^{-2}	6.34 x 10^{-4}	

Risk Class Evaluation		Dose Consequences with Active Safety Control Measures [Sv]							
		0 HEPA		**1 HEPA**		**2 HEPA**		**0.1 LPF**	
		WORKER	**PUBLIC**	**WORKER**	**PUBLIC**	**WORKER**	**PUBLIC**	**WORKER**	**PUBLIC**
Dose Consequences:		4.67 x 10^{-2}	6.34 x 10^{-4}	2.58 x 10^{-5}	3.51 x 10^{-7}	5.17 x 10^{-8}	7.01 x 10^{-10}	4.67 x 10^{-3}	6.34 x 10^{-5}
Consequence Level:		Low	Low	Low	Low	Low	Low	Low	Low
Risk Class	**Anticipated**	III	III	III	III	III	III	III	III
	Unlikely	III	III	III	III	III	III	III	III
	Extremely Unlikely	III	IV	IV	IV	IV	IV	IV	IV

Controls	Classification of Controls
Critical crane lifts shall be performed in accordance with a hoisting and rigging checklist	Administrative Control
ILW waste containers shall be moved from below the lift path	Administrative Control

5.2.3. Worker safety evaluation

Hazards that only lead to occupational injuries or illnesses, do not contribute to accident source terms, and are not accident precursors, initiators, or propagators are considered standard industrial hazards (SIHs). For the immediate worker’s safety, three levels of protection are addressed:

(a) Physical barriers around or dealing with the hazard that can protect the worker (e.g., primary containers, shielding);

(b) General classes of protective equipment for the worker (e.g., protective clothing, breathing devices); and

(c) Administrative imposed requirements to protect the worker (e.g., postings, lockout/tagout).

Each of the specified protective features has a corresponding Safety Management Programmes identified as the credited program (Quality Assurance, Industrial Safety, Radiological Protection, etc.). The set of protective features covers those aspects of the hazard that are considered to place the worker at most risk and is not intended to be a complete list.

This evaluation concludes that standard industrial hazards are sufficiently controlled by the Safety Management Programmes. Hazards that are not standard industrial hazards were further evaluated in the accident analyses.

No controls were identified that warrant elevation to Safety Measure exclusively for the protection of the immediate worker. Elements/attributes of the Safety Management Programmes that are important to the protection of the worker in the event of fires, spills, and facility explosions are discussed below. Recirculation HEPAs are available and while not specifically credited to protect the immediate worker, provide filtered air, which limits exposure to dispersion events during worker evacuation.

— *Fires*

Some fire scenarios applicable to the Laboratory Decommissioning scope present some risk to the worker due to the potential for burns, smoke inhalation, and radioactive material inhalation. In the event of a fire in the either the decommissioning area, the primary means to protect the work is to assure that they evacuate areas in a timely manner. Building evacuation may be initiated by an occupant(s) recognizing accident indicators (fire, smoke, heat, etc.), notifying others, and/or activating manual fire alarm systems

— *Spills*

Spill scenarios in the laboratory also present some risk to the worker due to the potential for radioactive material inhalation. Operational spill events may result in personnel injury due to falling containers or a compressed gas cylinder missile. Operational spill events occur when personnel are involved/present (*e.g.,* forklift operator punctures or drops a waste container, personnel inadvertently allows a compressed gas cylinder to topple, etc.). Because operational spill events are “attended”, personnel will apply the “SWIMN” process for which they are trained. The SWIMN (Stop, Warn, Isolate, Minimize, and Notify) process results in work Stoppage, Warning others in the immediate area, Isolating the spill, Minimizing personnel exposures, and Notification. Due to dispersion of the release during an operational spill event the area of concern is in the direct vicinity of the accident, lessening the threat to personnel removed from the accident. In the event of an operational spill, evacuating immediate areas in a timely manner provides the most effective protection to the worker’s safety.

— *Explosions*

A facility explosion presents Immediate Worker risks similar to fire events including burns, smoke inhalation, and radioactive material inhalation. Additionally, an explosion event may result in serious injury or death due to impacts, collapsing structure, falling equipment, flying debris/shrapnel, etc. Evacuating the building by whatever means available provides the most effective protection to the worker. Building evacuation may be initiated by an occupant(s) recognizing accident indicators (explosion sound, fire, smoke, heat, etc.), notification by others, and activation of manual fire alarm systems.

— *External Events*

External events are those events that are not initiated by activities within the facility, such as a loss of power and surface vehicle impacts. No external events were identified where the risk to the immediate worker was significant. The dose consequences associated with a crash involving a large aircraft were high but the probability of such an impact is sufficiently low to not warrant implementation of additional protective measures beyond those already available.

— *Natural Phenomena*

Natural phenomena such as earthquakes and heavy snow present significant risks to the immediate worker. No controls to reduce the frequency of natural phenomena are available and the largest risk to the immediate worker involves impacts from falling debris or a collapsing structure. In these cases, evacuating the building by whatever means available provides the most effective protection. Natural phenomena and hazardous events were not included in this evaluation as the activities proposed by this analysis reduce the amount of hazardous materials previously evaluated in the operational safety analysis and the proposed activities do not alter the structure of the facility.

A summary of the accident analysis results is presented in Table 25.

TABLE 25 SUMMARY OF ACCIDENT ANALYSIS RESULTS

Scenario	Dose Consequence [Sv] Without Active Safety Control Measures		Dose Consequence [Sv] With Active Safety Control Measures		Safety Control Measures
	WORKER	**PUBLIC**	**WORKER**	**PUBLIC**	
Small Drum Fire Inside Confinement (Scenario A1)	4.4 x 10^{-2}	6.0 x 10^{-4}	6.4 x 10^{-5}	8.7 x 10^{-7}	Combustible material controls, Confinement (1 stage HEPA), hot work controls
LLW, and TRU Waste Fire in Ops Clean Area (Scenario A2)	2.40 x 10^{-2}	3.25 x 10^{-4}	2.4 x 10^{-2}	3.25 x 10^{-4}	No Controls required
Medium Drum Fire Inside Confinement (Scenario B1)	1.07 x 10^{-1}	1.49 x 10^{-3}	1.5 x 10^{-4}	2.17 x 10^{-6}	Combustible material controls, Confinement (1 stage HEPA), hot work controls, fire suppression
Medium Fire in Return Plenum Rooms (Scenario B2)	1.1 x 10^{-1}	1.55 x 10^{-3}	4.3 x 10^{-3}	5.82 x 10^{-5}	Combustible material controls, Confinement (1 stage HEPA), hot work controls, fire suppression
Large Fire in Return Plenum Rooms (Scenario C1)	Same as Medium Fire	Same as Medium Fire	Same as Medium Fire	Same as Medium Fire	
Small Glovebox Fire during Size Reduction of Process Piping (Scenario D1)	1.3 x 10^{-1}	1.81 x 10^{-3}	1.94 x 10^{-4}	2.07 x 10^{-6}	Combustible material controls, Confinement (1 stage HEPA), hot work controls
Small Fire Involving a Glovebox in a Size Reduction Enclosure (Scenario E1)	9.20 x 10^{-3}	1.20 x 10^{-4}	1.30 x 10^{-5}	1.80 x 10^{-7}	Combustible material controls, Confinement (1 stage HEPA), hot work controls
Large Fire Involving a Glovebox in a Size Reduction Enclosure (Scenario F1)	4.28 x 10^{-2}	3.83 x 10^{-4}	3.35 x 10^{-2}	2.63 x 10^{-4}	Combustible material controls, Confinement (1 stage HEPA), hot work controls, Plenum deluge, Fire suppression (sprinklers)
Transuranic Waste Spill Outside Confinement (Scenario G1)	2.8 x 10^{-2}	3.8 x 10^{-4}	2.8 x 10^{-2}	3.8 x 10^{-4}	Drums over 200 g Pu-E shall not be staged/stored outside confinement, Drums/ standard waste boxes s must not be stacked
Duct Drop/Impact (Scenario H1)	4.16 x 10^{-3}	5.65 x 10^{-5}	4.16 x 10^{-3}	5.65 x 10^{-5}	No Controls required
Glovebox Drop/Impact (Scenario H2)	1.50 x 10^{-3}	2.10 x 10^{-5}	1.50 x 10^{-3}	2.10 x 10^{-5}	Confinement (1 stage HEPA)
Zone I Ventilation Reversal (Scenario I2)	5.70 x 10^{-4}	7.70 x 10^{-6}	5.70 x 10^{-4}	7.70 x 10^{-6}	No Controls required
Oxy-Acetylene Vapor Cloud Explosion in OPS Clean Area (Scenario J1)	7.80 x 10^{-3}	1.06 x 10^{-4}	7.80 x 10^{-3}	1.06 x 10^{-4}	Drums over 200 g Pu-E shall not be staged/stored outside confinement
Hydrogen Deflagration in a 55-Gallon Drum (Scenario J2)	9.71 x 10^{-1}	1.32 x 10^{-2}	1.42 x 10^{-3}	1.93 x 10^{-5}	Drum vents,1 stage HEPA filtration
Crane Load Drop (Scenario L1)	4.67 x 10^{-2}	6.34 x 10^{-4}	4.67 x 10^{-2}	6.34 x 10^{-4}	Critical crane lifts outside the facility shall be performed in accordance with a hoisting and rigging checklist,

6. ENGINEERING ASSESSMENT

6.1. ENGINEERING ASSESSMENT METHODOLOGY

The process for identifying engineered control measures (SSCs) has been outlined in Section 2.6.2 and Section 5 above. The safety assessor has to specify the necessary safety related functions, and any performance requirements, of each SSC. There then engineering evaluation needs to be performed to demonstrate that the safety and performance requirements assumed by the safety assessor will be provided by each SSC as expected.

It is normal practice to categorize SSCs in accordance with the importance of the safety function that they will be required to provide. This allows a graded approach so that engineering expertise and effort can be applied proportionately to the safety significance of the SSCs. The operator may devise his own engineering assessment process, as there is no universal international standard in this area, but an example is given below for information and consideration:

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

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

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

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

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

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

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

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

If a SSC is provided by new facility engineering assessment by the operator is not needed. However, the design documentation needs to be in accordance with the appropriate national engineering codes or standards, together with a demonstration that the safety and functional requirements of the SSC specified in the safety assessment are satisfied. The detail in the engineering assessment demonstrating compliance with functional and performance requirements needs to be proportionate to its SSC Category.

6.2. ENGINEERING MEASURES (SSCs) DERIVED FROM NUCLEAR LABORATORY SAFETY ASSESSMENT

The results of the safety assessment were presented in Section 5 in a series of summary tables; one for each significant accident scenario selected for assessment. The tables identify the engineering and administrative control measures necessary to ensure that the radiological consequences of each accident scenario are within the requirements of the accident risk criteria (see Section 3) and are also ALARA. On this basis the identified SSCs for the nuclear laboratory decommissioning are summarized in Table 25 “Summary of Accident Analysis Results”.

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

TABLE 26 ENGINEERING SCHEDULE FOR THE NUCLEAR LABORATORY

Description of SSC	SSC Safety Class	SSC Safety and Performance Requirement and Identified Shortfalls	Action to Address Shortfall
VENTILATION FANS **Space and cell extract fans (duty and standby fans)**	3	Safety function: To minimize concentration of air-borne activity and control spread of contamination	Separate cell /fume cupboard extract from space extract Install separate cell extract Install separate fume cupboard extract
Fans fitted to Lab. 3 Beta Gamma Line Cell	3	There is uncertainty whether the fans can produce the specified cell depression	Start one of the extract fans and verify the operating depression of the Cell.
		The extract fans do not meet current standards, are over 40 years old and the Performance Requirements are not fully satisfied.	Replace both extract fans with units constructed and tested in compliance with current standards
		No instruments are fitted to the Cell or the fan to monitor extract fan performance.	Fit instrumentation to comply with current standards.
Actuated isolation dampers at the Space and Cell extract fans	3	Safety function is to ‘Provide means of remotely isolating the fans’	The safety function is compromised by the inability of the dampers to provide means of remotely isolating the fans upon loss of electrical or compressed air services. The dampers have been demonstrated to provide the safety function during ‘normal’ operation (all services functioning.) The safety function can only be met fully by reversing the action of the damper actuators to fail-close instead of fail-open.
Space and cell extract booster fans	3	Cell extract booster fans in Lab. 3 are not currently in use. However the ventilation System Manual’ that these fans are necessary to maintain required depression levels in the cells.	Reinstate these fans to provide the required safety function.
Fire Suppression Sprinklers	3	To suppress fire in Laboratory rooms by activation of fusable bulbs	No action required as system is inspected routinely as part of Facility Surveillance Programme
Mechanical Handling Equipment **Gaitered Tongs**	4	Laboratory 2 gaiters were found not to be serviceable.	Laboratory 90 gaiters require replacement.
Master-Slave Manipulators (MSM) and gaiters in Lab. 1	4	The MSM in Lab 1 has had 4 breakdowns within a year. The MSM has not been modified in accordance with general recommendations for all site MSMs.	Upgrade MSMs to recommended operational standard for decommissioning purposes.
Active Drains and Facility Washings Drains **Low Active Drain Pipework**	4	Underfloor systems consisting of polyethylene lines in asbestos containment gullies. Asbestos is porous and absorbent and as such is an unsuitable material for the containment of radioactive fluids	Investigate for the presence of asbestos in secondary containment gullies and remove in a manner compliant with asbestos regulations. Replace with stainless steel or HDPE secondary containment.
Facility Washings Drain Pipework	4	Underfloor systems consisting of polyethylene lines in asbestos containment gullies. Asbestos is porous and absorbent and as such is an unsuitable material for the containment of radioactive fluids.	Investigate for the presence of asbestos secondary containment gullies and remove in a manner compliant with asbestos regulations. Replace with stainless steel or HDPE secondary containment.
Cell Structures **Cell and cell sealing**	3	Lab 2 cells have a potential for alpha contamination but cells are not constructed to alpha containment standards	Remove contamination from cell or upgrade cell sealing arrangements to prevent possible spread of alpha to general cell areas
Ventilation Ducting – Desirable on ALARA Grounds Only **Underfloor Ducting**	4	The condition of the brickwork walls of the ducts is not known. In particular it is desirable to know if all bends and perpendicular joints are fully filled to ensure gas tightness.	Recor Carry out an inspection of the condition of the duct brickwork, as part of a survey of the inside of the vent duct, in the central corridor.

7. EVALUATION OF RESULTS AND IDENTIFICATION OF SAFETY CONTROL MEASURES

7.1. COMPARISON OF ANALYSIS RESULTS WITH CRITERIA

The results of the safety assessment are presented in the tables of Sections 4 and 5. Those accident scenarios selected, evaluated without active safety control measures, consequences for analysis ranged between Risk Class II and IV. Safety control measures were identified (engineered and administrative) that will reduce risk for all scenarios to not higher than Risk Class III, which represents an acceptable level of risk. Further consideration is given to additional safety control measures if they can be justified on ALARA grounds.

Identified engineered safety control measures (SSCs) were subject to engineering assessment, as described in Section 6, to demonstrate that the selected SSCs can deliver their specified functional and performance requirements. The results of the assessment are summarized in Table 26 the Engineering Schedule, in Section 6. Once the recommendations in the schedule are completed the engineered and administrative control measures will be included in the facility/ projects Surveillance Programme as the Technical Safety Requirements.

7.2. TYPES AND TREATMENT OF ASSUMPTIONS AND UNCERTAINTIES

Due to the limited activities associated with this decommissioning project (i.e., the decontamination and removal of glove boxes without concurrent decommissioning and demolition of the buildings/structures), the uncertainties in the safety assessment are limited. Specifically, the considerations described in the DeSa recommendations (see Volume I of this report), such as uncertainty about the physical facility, construction and facility aging, models/codes used and waste/waste streams are not pertinent.

The amount and type of information available about the radiological condition of glove boxes is very detailed and complete, due to the post-operational cleanup activities and the comprehensive characterization activities. In addition, the safety controls and procedures were developed using the “worst case” input parameters discussed in Section 3.2.3, and the release model used is a conservative one. Some uncertainty exists about the radiological condition of the ventilation system; however, any potential safety impacts from the isolation of the ventilation system will be mitigated by the cutting and isolation activities being conducted in the tented enclosure.

7.3. SAFETY CONTROL MEASAURES

The safety assessment results also allow radiation protection advisors to give specific authoritative advice on such matters as monitoring equipment, air sampler placement, alarms, respiratory equipment and other personal protective equipment (PPE).

The safety control measures derived from the safety assessment for the laboratory decommissioning operations are summarized in Table 27 below.

TABLE 27 SUMMARY OF ENGINEERED AND ADMINISTRATIVE CONTROLS (TECHNICAL SAFETY REQUIREMENTS)

Control No.	Engineered Safety Control Measures (SSCs)	Administrative Safety Control Measures
1	Primary HEPA filters at the extract of each Cell SSC2	Combustible Material Control SAC*
2	Fire Suppression Sprinklers SSC3	Hot Work Controls
3	Space and cell extract fans (duty and standby fans) SSC 3	Drums/ standard waste boxes must not be stacked (R)
4	HEPA filters fitted to the Lab 3 Beta Gamma line Cell SSC2	Drums over 200 g Pu-E shall not be staged/stored outside confinement,
5	Plemum deluge sprinklers SSC3	Critical crane lifts outside the facility shall be performed in accordance with a hoisting and rigging checklist,
6	Fans fitted to Lab 3 Beta Gamma Line Cell SSC3	TRU waste containers shall be moved from below the lift path
7	Fans fitted to Lab 3 Beta Gamma Line Cell SSC3	
8	HEPA filters fitted to the Lab 3 Beta Gamma line Cell SSC3	
9	Space and cell extract booster fans SSC3	
10	Cell Structure Sealing SSC 3	
11	Ventilation System Dampers SSC3	
12	Modular Containment and Portavent SSC3	

*SAC Specific Administrative Control

The analysis results comply with the criteria, demonstrating that the proposed decommissioning operations can be carried out as planned.

The purpose of the selection of the Technical Safety Requirements is to identify the set of engineered and administrative controls needed to ensure the safe operation of facility to protect the public, the workers, and the environment. These are sometimes referred to as the ‘limits and conditions for safe operation’ In general, protection for the environment is provided by the protection afforded to the above groups. The primary output of the safety assessment concludes that the radiological dose/risk to a member of a critical group is estimated to be 33.5 mSv to the worker and 0.3 mSv to the public under accident conditions. The expected worker dose for completion of decommissioning activities is estimated to be 0.375 mSv for the 6 month period. The associated set of ‘limits and conditions’ have been derived that must be applied to ensure that the decommissioning work can be conducted safely and in accordance with the ALARA principle.

8. GRADED APPROACH

The safety assessment identified those reasonably foreseeable accident conditions that could occur during planned decommissioning operations. The evaluation then grouped these accidents into categories to assess the maximum radiological exposure that could result without active safety control measures. The graded approach adopted here for the safety assessment is to keep the assessment as uncomplicated as possible to limit assessment effort, while at the same time ensuring that the assessment results are sufficient to evaluate risk and identify safety measures that will ensure risk to workers and public are optimized and ALARA. It is important to ensure that by application of the graded approach that it does not compromise safety and compliance with relevant safety requirements and criteria.

In the safety assessment the radiological inventories and other hazards were well characterized thus avoiding the need for overly conservative assumptions. The identification of fault grouping was also used to reduce the extent of safety assessment by grouping faults with similar initiating events. For example dropped loads onto the glove boxes have a number of initiators but the consequences and control measures necessary to mitigate the radiological consequences are similar. Therefore a bounding fault type with a conservatively chosen frequency that represents the overall risk was chosen

The initiating event frequency and consequence evaluation allocated accident scenarios into four Risk Classes. Consequences are divided in 3 Categories as follows:

- High consequence for public (100 mSv to 1 000 mSv) and for workers (> 1 000 mSv);
- Moderate consequence for public (10 mSv to 100 mSv) and for workers (100 mSv to 1000 mSv); and
- Low consequence for public (1 mSv to 10 mSv) and workers (10 mSv to 100 mSv).

The initiating event frequencies are graded as follows:

- Anticipated 10 x 10^{-1} to 10 x 10^{-2} per year);
- Unlikely (10 x 10^{-2} to 10 x 10^{-4} per year);
- Extremely unlikely (10 x 10^{-4} to 10 x 10^{-6} per year); and
- Beyond extremely unlikely (< 10 x 10^{-6} per year).

The risk classes are read from the matrix as depicted in Table 2.1. For example, a high consequence event which is anticipated at beyond extremely low frequencies would be classified as a risk Class III. Similarly, a moderate consequence event at extremely low frequencies would be classified as a risk Class IV. For risk Class IV no further assessments is required, since the Safety Management Programme is regarded as adequate to optimize and control the risk to be as reasonably low as achievable, taken cost into consideration. For Category 1 and 2 events, a detailed safety assessment is required.

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

9. CONFIDENCE BUILDING IN THE SAFETY ASSESSMENT

9.1. QUALITY MANAGEMENT

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

Furthermore, management systems such as the regulatory authorization of activities and modification to these actions, design control, justification of release of land from regulatory control, clearance levels, material accounting programme, storage of waste, supporting facilities, safe guards programme, waste minimization programme, radiation protection programme, environmental programme, security programme, work permit system, access – egress control systems, waste management systems, transport of radioactive material programme, in-service inspection programme, maintenance programme, care and maintenance programme (for the period of institutional control after the completion of decommissioning), staffing and training programme, emergency preparedness and response programme, fire protection programme, etc. have all to be controlled by the quality assurance system. Some of these systems have been mentioned but none of them had been included in the test case in any detail.

In accordance with the decommissioning project plan a safety assessment project team comprised of qualified operations and expert safety personnel was assembled to plan and evaluate the safety of the proposed decommissioning activities. Facility characterization was included in preparatory work in the form of a radiological survey, material sampling, and review of operational history.

9.2. INDEPENDENT REVIEW AND PPROVAL PROCESS

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

(a) The input data and assumptions are valid;

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

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

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

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

In the case of the DeSa project the nuclear laboratory test case was produced by a group of experts. The document was reviewed to the Regulatory Review and Graded Approach working groups for independent review in order to verify completeness but also to ensure consistency in approach. With the other test cases, the comments received have been incorporated into this document.

10. SUMMARY AND LESSONS LEARNED

The aim of this report is to illustrate the application of the safety assessment methodology developed as part of the phase 1 of the DeSa project to a small facility by applying the graded approach. This report is aimed at presenting an assessment of proposed decommissioning activities in order to demonstrate to the Regulatory Body that the activities can be conducted safely and work could then be authorized.

The facility used for the illustration of the assessment methodology consists of 5 individual laboratories within a larger laboratory suite, some of which will remain operational after these laboratories have been decommissioned. It covers immediate dismantling with the aim for release of the building for unrestricted use after a limited period of institutional control. Demolition will be deferred until all laboratory operations are completed at adjacent laboratory facilities within the same building.

The practical use and benefits of the graded approach was clearly demonstrated through the application of the of the safety assessment using risk classes. Risk classes where defined through a combination of consequence of initiating events and frequency of these expected events.

The methodology developed in the DeSa project for the assessment and evaluation process was followed in this test case. It was quantitatively demonstrated what the effects of the application of individual safety significant components, administrative measures and limiting conditions of operation on the resulting effective dose would be. It was also quantitatively demonstrated how the implementation of individual mitigating actions could optimize the detrimental effects associated with the decommissioning action. The results are illustrated by in the assessment in terms of without active safety control measure and dose to both workers and the public and further with active safety control measures.

It was demonstrated how through the application of the graded approach, effort applied for analyzing the consequences associated with decommissioning could be minimized when the bounding criteria is well established and applied. This was, for example illustrated in the case of a fire inside a Modular containment, where the reduction of dose could be observed after the inclusion of an additional set of HEPA filters.

The importance of input data was demonstrated in each safety analysis. On the one hand it is important to use conservative input data to over estimate the results in calculations in order to demonstrate confidence in uncertainties. (Uncertainties are also limited due to the limited amount of activities.) On the other hand, unrealistic and over conservative data could lead to unnecessary effort in the amount of assessment work required to demonstrate compliance with the basic safety criteria. This was illustrated by the analysis of various layers of mitigation.

The amount and type of information available about the radiological condition of glove boxes is very detailed and complete, due to the post-operational cleanup activities and the characterization activities. In addition, the safety measures and procedures were developed using conservative input parameters. Some uncertainty exists about the radiological condition of the ventilation system; however, any potential safety impacts from the isolation of the ventilation system will be mitigated by the cutting and isolation activities being conducted in the tented enclosure

The depth of safety assessment required depends on the complexity of the facility and the hazards associated with the decommissioning activities. By defining appropriate risk class criteria it was adequately demonstrated how the individual hazards were assessed with and the effort of assessment required for compliance. In the case of this test case, the assessments were fairly uncomplicated because of the nature of the facility. For similar types of facilities but with mixtures of nuclides and unknown source terms the assessments and evaluation can become much more complex.

The safety assessment needs to demonstrate that the decommissioning of the facility does not impose unacceptable hazards (e.g. leading to effective doses in excess of relevant constraints, criteria and limits) or undue burdens on future generations. For this specific facility, the waste is entered into the existing waste management system. Institutional control will be determined by the demand for the adjacent laboratory facilities in use, when according to the existing plan; the rests of the facilities will be decommissioned to green field levels.

This test case did not deal with materials management as this was beyond its scope, other that specifying the waste acceptance criteria and stating that it was added to existing site waste inventories.

The applied method demonstrated that various engineering measures, administrative measures and safety management programmes can be applied. It demonstrated that through the application of the methodology the most appropriate and effective mitigating factors could be identified and implemented, thereby optimizing the amount of decommissioning activities and the associated effects. The assessment demonstrated that the DeSa methodology could be applied effectively to facilities of various types, sizes and complexities to identify safety significant components and structures, to evaluate safety measures and demonstrate compliance with specific regulatory requirements.

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