IAEA International Project

Use of Safety Assessment Results in Planning and Implementation of Decommissioning of Facilities Using Radioactive Material

(FaSa Project)

Annex Report on the FaSa Project Test Cases

Annex 3 – The Fuel Fabrication Facility Test Case

(Working Material)

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FOREWORD

Annex 3 presents the results achieved by the working group on the Fuel Fabrication Facility Test Case of the FaSa project. It addresses aspects of the application of the safety assessment methodology, as proposed during DeSa project, and aspects of the use of safety assessment results in planning and implementation of decommissioning of a fuel fabrication facility, as discussed during FaSa project.

The report particularly focuses on the use of safety assessment as described by the Decommissioning Conduct Chapter (chapter 4 of the main volume of this publication), to identify examples of good practice that may be of assistance to other member states.

The IAEA would like to express its gratitude to all the members of the FFF Test Case working group, who contributed to the development and review of the Annex 3 and, in particular, to the chairperson of the working group, A. Halle (United Kingdom), and to the Sellafield Sites Ltd (United Kingdom), for providing detailed information about the decommissioning project for the MOX fuel fabrication facility in Sellafield, which served as a basis for this test case.
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1 INTRODUCTION

1.1 BACKGROUND

This annex 3 of the FaSa Annex Report describes the fuel fabrication facility (FFF) test case for the FaSa Project for which safety assessments are carried out in order to illustrate the DeSa methodology and FaSa methodology on safety assessments for decommissioning.

During the FaSa project a number of test cases have been carried out, aiming to illustrate the specific guidance provided in the main volume of this publication. The test cases covered decommissioning of facilities of different complexity, one of them being a “fuel fabrication facility”.

The MOX fuel fabrication facility on the Sellafield Site, United Kingdom, served as a basis for this test case and detailed information about its decommissioning was made available to the FaSa project by the Sellafield Sites Ltd.

This Annex 3 has been produced by the Fuel Fabrication Facility Test Case Working Group (FFF TC WG), using the materials provided by the Sellafield Sites Ltd. as a basis. The WG meetings were organized as part of the four annual meetings of the FaSa project (2008-2011), and on two additional occasions, where a limited number of WG members participated. In addition, individual work was done by several WG members between the meetings.

The actual FFF test case pre-dates the DeSa and FaSa projects and the regarding terminology is not always an exact match regarding FaSa terminology. Therefore, an emphasis in this Annex is put onto correlating the used terminology with the FaSa terminology to demonstrate the consistency in approach. All the information presented in this Annex reflects the situation as of 2012 and has been provided by the Sellafield Sites Ltd. or developed by the participants of the FFF TC WG of the FaSa project.

1.2 SCOPE

This document is the fuel fabrication facility (FFF) test case for the FaSa Project. Information has been selected from the FFF decommissioning safety submission and included within this Annex in order to illustrate the FaSa methodology.

1.3 OBJECTIVES

The aim of this test case report is to provide an example of the FaSa methodology to member states regarding the use of safety assessments in support of decommissioning of nuclear Fuel Cycle Facilities, in this instance a fuel fabrication facility (FFF).

This test case is considered to be a good example of FaSa because it

- Demonstrates how a safety assessment is used to carry out optioneering, hazard analysis (both radiological and conventional) and the identification of the appropriate controls (both managerial and engineering);
- Demonstrates the use of safety assessment in the identification and practical implementation of safety control measures in decommissioning;
Demonstrates the graded approach to safety assessment through the lifecycle of the facility,

Demonstrates how the safety assessment is implemented and incorporated into the working documentation,

Provides recommendations on demonstration that exposures to workers and public as low as reasonably achievable (ALARA) and the application of the concept of ‘defence in depth’.

1.4 STRUCTURE OF THE TEST CASE REPORT

The test case is described in the following 2 parts of this Annex:

- Part 2 Introduction and description of the test case facility
- Part 3 Illustration of the FaSa methodology

Each part is supported by appendices where more detailed information is presented.

It is to be noted that the safety assessments and supporting data presented in this test case report are provided for FaSa illustrative purposes only and are not intended to be used for any other application.

1.4.1 Parameters for Choosing a Safety Assessment to be Examined in the Test Case

The scope of the test case is to examine aspects of the safety assessments prepared for decommissioning of a nuclear FFF and will be limited to two phases of the facility final decommissioning plan.

In order to best achieve the identified objective, the information included in the test case was selected against a number of parameters such that it is:

- Be recognisable to many member states;
- Be applicable to many decommissioning applications;
- Discuss radiological fault sequences which have the potential to result in a significant consequence;
- Discuss radiological fault sequences which have an impact on the facility ‘limits and conditions’ for safety;
- Include an example of the impact of safety assessment results and the interaction with facility engineered safety systems, structures and components (SSCs);
- Include brief examples of decommissioning safety assessments which discuss hazardous events which are initiated both internal and external to the facility.

1.5 DEFINITIONS

There are several definitions and terms used specifically in the FFF test case which are given in the following paragraphs.

1.5.1 Workers

Individuals in paid employment of the site licence company or its associated sub-contractors who are engaged in tasks on the nuclear licensed site. Workers may be
internal to the facility (actively engaged in the decommissioning project) or external to the facility. An external worker is someone perhaps employed on adjacent parts of the site but not directly involved with the particular decommissioning project or task being assessed.

1.5.2 Public
Any individual who is not in paid employment of the site licence company or its associated sub-contractors and therefore are not actively engaged in tasks on the nuclear licensed site. An example might be a local resident who lives in a village or town adjacent to the nuclear site.

1.5.3 Normal or Anticipated Operating/Decommissioning Conditions
These are considered to be those routine or anticipated activities which are undertaken on a day-to-day basis. Normal operations/decommissioning describe the mode of activities were equipment is functioning as intended by the manufacturer, within its design specifications. In addition, individuals may be said to be behaving normally within the facility if they are performing as desired, as per their instructions and within the routine requirements.

1.5.4 Abnormal conditions
Abnormal conditions or accident conditions are usually considered to be the opposite of normal conditions. In that equipment is not responding or functioning as per its design intent, the facility or operating environment has been compromised by an external force e.g. extreme weather, such as 1 in 1,000 years wind speeds, or individuals have not adequately performed their required duties. Such an abnormal event or accident condition may not always result in radiological consequences, and it is the task of the safety assessor to identify those events which do.

1.5.5 Fault Set
Abnormal or accident condition events which have been identified by the safety assessor are collated together and presented as the full fault set for the facility. The fault set describes a suite of initiating events, their associated fault sequence progression and resultant consequences. A fault set may be generated with varying degrees of detail. For example, a fault set which is compiled for the overall project (overarching safety assessment) would not be expected to contain as much detail as a fault set presented for a specific decommissioning phase (final safety assessment) or for a specific decommissioning activity. The level of detail within the fault set is to be commensurate with the level of knowledge of the decommissioning phase or activity being examined.

1.5.6 Safe Operating (or Decommissioning) Envelope
The Safe Operating or Decommissioning Envelope (SOE) defines the boundary within which the facility can be operated/dismantled in a safe manner without unacceptable risk to either workers, members of the public or the environment. The SOE has to confidently demonstrate that a significant nuclear event will not occur. The boundary of the Safe Operating/Decommissioning Envelope includes a set of ‘Limits and Conditions’.
• A ‘Limit’ defines a maximum justifiable safe state of operations. The limits are normally parameters (such as temperature or mass limits).
• A ‘Condition’ describes the circumstances necessary to keep operations within the SOE.

1.5.7 Limits and Conditions Document

The SOE for a facility is described in the ‘Limits and Conditions Document’. The aim of the Limits and Conditions document is to provide a concise worker friendly description of the safety case and environment limits and conditions, contingency and replacement arrangements.

The Limits and Conditions include identified engineering and administrative control mechanisms and measures. These are necessary to ensure that the anticipated decommissioning activities can be performed safely (normal operating levels), and that fault escalation is prevented or mitigated. This information is communicated to the workforce via working level documentation such as Operating or Decommissioning Instructions.
2 DESCRIPTION OF THE FACILITY

2.1 STATE OF THE FACILITY

The FFF test case facility is located on the multi-facility Sellafield site in the United Kingdom. The facility primarily comprised a process line. It was designed and constructed during the late 1960s. It is a two storey building, constructed of concrete block exterior walls, which are clad with aluminium corrugated sheeting. The facility began manufacturing fuel in 1971 and this continued until 1992. The operational areas were located on the ground floor and all the service feeds located on the 1st floor, such as the ventilation system.

As illustrated by Figure 1, the test case facility consisted of five key areas:

- Fuel Line Cubicles.
- Pellet Load Area and Vibro-compaction Suite.
- Active Canning Line.
- Final Assembly & Can Preparation Area.
- Ventilation system (not shown in Figure 1).

FIG 1: The Operational areas of the FFF.

The facility has completed a number of decommissioning stages. The potential for spread of radiological contamination presented by the facility progressively decreased as fuel manufacture was completed. At the beginning of the process line the gloveboxes were highly contaminated (loose MOX powders), these levels decreased as the material proceeded through the process and it was formed into fuel pellets and placed into fabricated fuel pins.
2.1.1 Decommissioning Strategy and Declared End State

Immediate dismantling was the decommissioning strategy chosen for the facility. The aim was to enable the equipment, structures, components and parts of the facility containing radioactive material to be removed or decontaminated to a level that permitted the facility to proceed to demolition.

2.1.2 Facility Operational History

The test case facility processed Mixed Oxide (MOX) powder into reactor fuel elements for the Dounreay Prototype Fast Reactor (PFR). The process feed material was provided by an on-site facility as powder and latterly from another onsite facility as granules. The feed material was processed into either sintered granules or pellets which were loaded into fuel pins and then assembled into fuel elements.

The active plant comprised mainly of equipment housed in a series of high integrity gloveboxes housed within cubicles requiring respiratory protection for access located on the West side of the building. Remote operation of the Fuel Line and pin finishing were conducted in the central free breathing area of the building, which contained a line of lower integrity gloveboxes. Part of the facility is currently (2012) undergoing decommissioning, and another part is used as an operational Plutonium Contaminated Material (PCM) drum receipt and export facility.

2.1.3 Radioactive Inventory

The radiological inventory of the test case facility was predominantly alpha nuclide based. Only notional values taken from the safety assessment are presented within this test case. These notional values are sufficient to enable the FaSa objectives to be illustrated.

The radioactive inventory forms the basis for the safety assessment for the decommissioning activities and is a prerequisite for the effective planning of working level documentation. The radioactive inventory was determined for each of the gloveboxes contained within the fuel line. Dismantling and removal of these gloveboxes formed one phase of the decommissioning project. Characterisation activities have been undertaken to assess the radiological conditions of the dismantling tasks that are to be addressed as part of later phases of the decommissioning project.

2.2 FACILITY LOCATION AND SURROUNDING AREA

The facility is located on a multi-facility site (see figure 2) which is located in the North-West of the United Kingdom, on the West Cumbria coast adjacent to the Irish Sea on the western outskirts of the Lake District National Park.

The nuclear site licensed boundary encompasses an approximate area of 276 Ha and is located at 54°N, 3°W (see figure 3). More details of the facility and its surroundings; regarding metrology, geology and hydrology etc. is presented in Appendix A.
FIG 2: Photograph of the test case facility. The FFF is the building in the lower right corner of the picture.

FIG 3: Location of the test case facility

The major local towns of Whitehaven, Workington and Barrow are approximately 14 km to the North, 25 km to the North and 38 km to the Southeast respectively. There are about 200 people living within 2 km of the site; the nearest settlement of any size is Seascale 2.5 km distant, with a population of about 1800. The countryside around the site is mainly utilised for farming. A more detailed description of other activities is provided in Appendix A.
2.3 REGULATORY FRAMEWORK APPLICABLE TO THE TEST CASE

2.3.1 Legislative Requirements (Targeted and Proportionate Approach)

The final safety assessments produced in support of the FFF decommissioning project are based on a number of applicable regulatory requirements. These requirements include not only radiological and nuclear criteria but also criteria resulting from the requirement to control conventional safety and environmental hazards. The legislative requirements include, but are not limited to:

- UK Health and Safety at Work Act, 1974 (as amended)
- UK Ionising Radiations Regulations, 1999
- UK Nuclear Installations Act, 1965 (as amended)
- UK Radiation Emergency Preparedness and Public Information Regulations (REPPIR), 2001
- UK Management of Health and Safety at Work Regulations, 1999
- UK Environmental legislation including the Radioactive Substances Act, 1993

All relevant legislative requirements are translated into a set of internal company owned criteria (standards, procedures and methodologies) which the decommissioning safety assessments are produced to and independently reviewed against. It is accepted practice in the UK for the site licence duty holder to demonstrate compliance by comparison with their own procedures. The regulator in the UK, the Office for Nuclear Regulation (ONR), permissions decommissioning activities based upon the information presented and where appropriate following detailed examination of the information in the safety submission.

A separate demonstration of compliance is completed at a company level to assure the regulator that the company administrative and approval processes have been constructed and are being maintained in an appropriate way, which enables the company to achieve competent compliance with all of the relevant legislative requirements. This regulatory due process review involves examination of the duty holder’s management arrangements and organisational structure as well as a review of the site operational safety performance. When the regulator is content that appropriate administrative arrangements are in place, the site management organisation is granted a site licence under the Nuclear Installations Act. The nuclear site licence has 36 licence conditions attached to it. These licence conditions cover all modes of the nuclear lifecycle, from new build/construction to decommissioning. The UK nuclear industry is regulated via a ‘permissioning regime’, which is targeted and proportionate. Applications to undertake a task have to be made to and received from the Regulator depending on nuclear safety significance.

Guidance is provided to the ONR inspectors to inform their regulatory decision making in the nuclear permissioning process. The guidance is the ‘Safety Assessment Principles (SAPs) for Nuclear Facilities’, last updated and published in 2006. The SAPs provide a benchmarking framework for making consistent regulatory judgements on nuclear safety submissions.
2.3.2 Regulation of Hazards Which Result in an Environmental Impact

Hazards which result in an environmental impact (including non-nuclear) are also considered as part of the overall safety assessment process. The Nuclear regulators within the UK seek advice from the environmental regulators prior to granting permission for the requested activity. As a result, it is necessary for the licence holder to demonstrate that they have undertaken a suitable and sufficient review of the potential environmental consequences that may result from their proposed activities. It is considered that this can be most efficiently achieved by expanding the hazard identification process employed for identification of nuclear hazards to include environmental hazards and consequences.

2.3.3 Regulation of Conventional Hazards

Conventional hazards such as dropped loads, working at height, electricity, or chemical toxicity are also considered during the United Kingdom nuclear safety assessment process, if they could lead to a nuclear safety consequence. In this context, the conventional safety hazard will be included if it is identified as being either notable as an initiating event for the fault sequence or as a consequence of a radiological incident. An example might be a crane fault which results in a dropped load, e.g. drop of scaffolding tubes, which impacts upon pipe work containing radioactive liquor, and a resultant loss of radiological containment event is caused.
3 ILLUSTRATION OF FASA METHODOLOGY

3.1 INTRODUCTION

The site licence company is allocated funding by the United Kingdom Nuclear Decommissioning Authority (NDA) in order to deliver an agreed programme of work. This includes the completion of specific declared decommissioning tasks. Resources, including provision of personnel and allocation of funding for decommissioning, are managed by the site licence company in order to support the delivery of the agreed site strategy. The overall site strategy is known in the United Kingdom as the site’s Lifetime Plan.

For the purposes of the FFF test case it is notable that sufficient resources including funding and provision of personnel have been allocated to the decommissioning project team in order to support delivery of the agreed facility strategy.

Sufficient historical records existed to support the selection of the decommissioning strategy. All required waste streams, waste storage and disposal routes have been identified and are available for use by the decommissioning project. No additional challenges were identified regarding provision of information about the facility.

The level of knowledge management regarding the facility’s extant operations and associated inventory was adequate, however it was recognised that in some areas the information was less detailed (for example on previous events) so additional characterisation studies were completed.

The FFF is located on a multi-facility site. The site is a mixture of operational facilities, deferred decommissioning facilities, facilities undergoing Post Operational Clean Out (POCO) and facilities completing dismantling and demolition tasks.

The following sections each refer to chapters of the main volume of this publication and will discuss:

- Decommissioning planning (chapter 3 of the FaSa report);
- Decommissioning conduct (chapter 4);
- Implementation (chapter 6);
- Regulatory review (chapter 7) and
- Decommissioning Termination (chapter 5).

As the FFF test case pre-dates the DeSa and FaSa projects, the regarding terminology is not always an exact match regarding FaSa terminology. At the end of each section, a table correlates the used terminology with the FaSa terminology to demonstrate the consistency in approach.

3.2 DECOMMISSIONING PLANNING

Decommissioning Planning concerns the selection of an appropriate decommissioning strategy for the facility (see chapter 3 of the main volume of this publication).

Such a strategy may include deferred decommissioning, agreement of a phased approach or to aim to achieve a ‘safe state’ which involves the removal of radioactive material but stops the task before demolition (this is sometimes referred to as an interim end state).
Further details regarding decommissioning planning are available in Chapter 3 of the main volume of this publication. Figure 4 illustrates the relationship between decommissioning planning and safety assessment for the FFF Test case.

3.2.1 Site Decommissioning Plan (known in the UK as Site Lifetime plan)

The FFF is one facility on a multi-facility site and therefore, its decommissioning strategy needs to be considered in the context of the wider site decommissioning plan. In the United Kingdom the Site Lifetime Plan is shared with the regulator.

The business strategy of the overall nuclear site in which the FFF resides is twofold; firstly to safely and securely achieve overall radiological hazard reduction and secondly to treat and store waste materials arising from the UK civil nuclear power generation programme.
Implementation of safe and secure hazard reduction includes the decommissioning, dismantling and ultimately demolition of redundant nuclear facilities.

3.2.2 Facility Decommissioning Plan

A Safety Case Strategy Overview Report (SCSOR) was developed for the FFF Test case. The purpose of the SCSOR was to present the facility decommissioning plan at a strategic level. Sufficient information was presented in the SCSOR to enable the stakeholders (regulators, internal due process) to understand the bounding hazards of the facility decommissioning and dismantling tasks, to explain the decommissioning phases that were selected and to provide strategic level information relating to the control of hazards. The SCSOR allowed for early regulatory engagement and an increase in stakeholder confidence in the overall decommissioning project.

The aim of the SCSOR was to identify where there were uncertainties in the project either with selection of decommissioning tools and techniques or with inventory and location of radioactive material. As such uncertainties invariably do exist during decommissioning, the SCSOR adopted a bounding case approach to the safety case strategy.

3.2.3 Generic Safety Assessment

The FFF test case had a generic safety assessment. This generic safety assessment assessed those tasks that were going to be carried out through all phases of the decommissioning project. These are referred to as ‘through life tasks’. An example of such a task is the operation of the facility ventilation system. By necessity the generic safety assessment was ‘bounding’ in nature – the details were developed in the safety assessments completed for the specific phases.

3.2.4 Final Safety Assessment for Phase 1

The final safety assessment for the first decommissioning phase was not presented in the generic safety assessment for the FFF test case. In this test case, the final safety assessment for the first phase was produced after the generic safety assessment. This final safety assessment for phase 1 of the decommissioning project was implemented as a modification to the extant facility operational safety case.

3.2.5 Consultation with the Regulator

For the FFF test case, although there was consultation with the regulator, there was no need to seek formal endorsement of the decommissioning plan, because the licensee’s own arrangements had already been approved.
3.2.6 Selection of the Facility Decommissioning Plan

As part of the decision making process for the strategy of this multi-facility site, a number of options for decommissioning of the FFF were considered by the site’s senior management. The following questions were addressed:

- Is the FFF needed to support existing decommissioning without modification?
- Is the FFF needed to support existing decommissioning with modification?
- Is the FFF needed as a building at all?
- Is the FFF building location (or building ‘footprint’) needed for placement of a future facility to support decommissioning?
- Are there services associated with or supplied by the FFF which are needed to support current site operations or future site decommissioning tasks?

These questions are typical of the selection process used on any UK multi-facility decommissioning site.

Note that the information collated for such site considerations is also submitted to the United Kingdom Nuclear Decommissioning Authority (NDA). This enables national level discussions regarding funding allocation for the various UK decommissioning liabilities to occur.

An immediate decommissioning strategy was selected for the FFF. This selection was a result of a number of factors which included:

- To minimise the potential increase in hazard as a result of in-growth of Plutonium daughters such as Am-241.
- The facility was redundant (no longer a business requirement to manufacture PFR fuel).
- The area that the facility was located upon was confirmed as being a suitable position for future re-use.
- There were no safety significant site services (e.g. liquor transfer lines, ventilation duct work) associated with this strategy that would impact on the decommissioning of the facility, or the future decommissioning (or other adjacent on-going operational areas) of the site.
- A new business need could be addressed by completion of this task. It was identified that part of the FFF building could be used as an interim Plutonium Contaminated Material (PCM) waste handling facility to support future site and facility decommissioning.

For the multi-facility site on which the FFF is located the overall site end state will be a combination of green field (unrestricted release), brown field (restricted release) and new waste stores on a smaller licensed site. A restricted use end point was selected for the FFF decommissioning plan, as it was in accordance with the overall decommissioning plan for the site end state.
3.2.7 Transition from ‘Operational Facility’ to Decommissioning Activities

Fuel manufacturing operations ceased in the early 1990’s. The FFF was in transition for four years prior to commencement of decommissioning. Dismantling tasks did not commence immediately after cessation of the fuel manufacturing campaign as the strategy for the facility and the allocation of funding had not then been agreed.

At the end of the fuel manufacturing campaign a number of preparatory activities, known as Post Operational Clean Out (POCO) were completed by the extant operations team. These POCO tasks included:

- Removal of hazardous materials (readily removed items such as bulk chemical storage);
- Maintenance of the required SSCs such as the ventilation system, in order to continue its operation.

The facility was then managed by the decommissioning team via a care and maintenance regime. This involved maintenance of safety SSCs and continued provision of services to the facility.

3.2.8 Phased approach to the Facility Decommissioning Strategy

The original strategy selected for the FFF was immediate decommissioning undertaken using a phased approach.

The aim of the phased approach to dismantling of the facility was to accrue decommissioning skills by completion of dismantling tasks in less radiologically hazardous (less contaminated) areas before progressing to the more contaminated areas. This phased approach also allowed the decommissioning processes, procedures, tools and techniques to be developed and tested prior to being deployed in the more radiologically contaminated and hazardous areas.

In terms of radiological inventory, phase 1 is the ‘cleanest’ with phase 4 being the most radioactive. It was during the implementation of phase 4 where it was expected that the most significant amounts of loose radiological material would be encountered (this is as it was anticipated that material has ‘accumulated’ in this location during the operational life of the facility).

The original FFF decommissioning phases comprised the following main plant areas as shown on Figure 5:

- Phase 1 - Clean out of the general facility preparation area and dismantling of the active canning line;
- Phase 2 - Decontamination and dismantling of the pellet vibration and vibro-compaction area;
- Phase 3 - Decontamination and dismantling of the fuel pellet production line;
- Phase 4 - Decontamination and removal of the building ventilation and services.
3.2.9 Desired Final Decommissioning End State

On completion of all the decommissioning phases, a radiological survey of the facility will have to be performed to demonstrate that the agreed end state has been achieved. The survey will be carried out in phases relating to the specific plant areas in which the decommissioning work is completed.

It was agreed that the facility would be dismantled and decommissioned to an ‘interim end-state’, which would stop short of demolition of the facility. The facility decommissioning interim end point was agreed as being a building that could receive PCM drummed waste and securely store it until the UK National disposal strategy was implemented. The change of use of part of the facility to become a PCM store was also identified as requiring specific detailed safety assessment and necessary internal and external approvals.

The development of the safety assessment to support the facility’s new role as a PCM store was progressed in parallel to the safety assessment of the dismantling and decommissioning phases. In effect the partial re-use of the facility became an additional phase of the overall facility decommissioning plan.

3.3 DECOMMISSIONING PLANNING CONCLUSIONS

The FFF test case pre-dates the DeSa and FaSa projects and as such is not always an exact match regarding terminology. Table 1 correlates the FaSa terminology to the activities completed during the test case to examine if there is consistency in approach.

It is considered that the concepts proposed by the FaSa project (chapter 3 of the main volume of this publication) regarding use of safety assessment results have been illustrated by the test case. It is to be noted that in the test case, the decommissioning plan is a suite of documents and that the overarching safety assessment does not include phase 1 – however this is provided before phase 1 commences.
<table>
<thead>
<tr>
<th>FaSa terminology Decommissioning Planning</th>
<th>Activities completed as part of the FFF Test Case</th>
<th>Consistency between FaSa and FFF TC</th>
</tr>
</thead>
<tbody>
<tr>
<td>Initial Decommissioning Plan</td>
<td>Site Life Time Plan &amp; SCSOR (Safety Case Strategy Overview Report)</td>
<td>Yes</td>
</tr>
<tr>
<td>Design for decommissioning(^1)</td>
<td>Not illustrated by this test case</td>
<td>No</td>
</tr>
<tr>
<td>Decommissioning Strategy</td>
<td>Site Lifetime Plan</td>
<td>Equivalent</td>
</tr>
<tr>
<td>Final Decommissioning Plan</td>
<td>Facility Decommissioning Plan</td>
<td>Yes</td>
</tr>
<tr>
<td>Application of multi-phased approach</td>
<td>Multi phases approach applied</td>
<td>Yes</td>
</tr>
<tr>
<td>Concept of Overarching Safety Assessment</td>
<td>Presented in Facility Decommissioning plan and generic safety assessment</td>
<td>Equivalent</td>
</tr>
<tr>
<td>Periodic Review of the Decommissioning Plan</td>
<td>It is not illustrated by this test case</td>
<td>No</td>
</tr>
<tr>
<td>Transition (between operation &amp; decommissioning)</td>
<td>Post Operational Clean Out (POCO) then short period of Care &amp; Maintenance</td>
<td>Yes</td>
</tr>
</tbody>
</table>

Table 1: Mapping of the FaSa terminology of Decommissioning Planning to Activities completed as part of the FFF test case

\(^1\) Considered only to be applicable to New Facilities – not relevant to FFF
3.4 **DECOMMISSIONING CONDUCT**

This section refers to chapter 4 of the main volume of this publication. In the middle of phase 3 the dismantling tasks have stopped. The safety assessment is now being revised (2011) – to ensure appropriate limits and controls for ‘reduced operations’ are identified. The FFF test case followed the approach to Conduct as outlined in Figure 6.

![FIG 6: Decommissioning Conduct](image)

### 3.4.1 Generic Safety Assessment

As described in the decommissioning planning chapter of the main volume of this publication (chapter 4), the purpose of the facility generic safety assessment is to identify the minimum engineering measures and managerial controls required to guarantee safety of the facility until the commencement of the dismantling and decommissioning tasks.

In addition, the FFF generic safety assessment provided detailed safety assessments for those activities that were identified as being common throughout all phases of the decommissioning project. An example of such a common task was the consignment and despatch of waste arising from the dismantling tasks (see Appendix C, 7.1).
3.4.2 Major Hazards identified by the Generic Safety Assessment

The formal approach to the identification of hazards and their analysis is illustrated by the documents in Appendix B. Significant hazards include criticality, external and internal radiological doses as well as non-radiological hazards. Examples of significant hazards identified by the generic safety assessment for the FFF are

**Criticality** including fault sequences such as:
- Exceeding fissile limits;
- Uncontrolled recovery, storage, transportation and migration of MOX;
- Uncontrolled introduction of liquids;
- Uncontrolled breach of containment and subsequent accumulation of material;
- Failure to maintain required separation distances, disturbance/collapse of storage arrays.

**External radiological dose uptake (worker):**
- External radiological dose through radiation transmission as a result of loss of shielding.

**Internal radiological dose (Public/Worker) from activity migration as a result of:**
- Uncontrolled breach of containment;
- Human Error during PCM waste removal operations;
- Hazards initiated external to the facility ‘External hazards’ (e.g. seismic events);
- Catastrophic failure of self-contained breathing unit – worker’s radiological protective equipment (known locally as a ‘Windscale Suit’);
- Spread of Contamination - Size reduction operations co-incident with failed Mobile Filtration Units and Building ventilation systems;
- Loss of Containment (including spread of contamination from dropped loads and impacts, including dropped waste drums containing PCM) resulting in an internal dose uptake to a worker.

**Non-radiological hazards** which were identified as initiating events that could result in radiological consequences were also considered.

3.4.3 Phased Approach

The FFF decommissioning project was executed using a phased approach. As the facility comprised discrete operational areas it was a logical extension for the decommissioning project to group and bound the decommissioning activities around the operational areas.

The assessment outputs that were examined for the FFF test case comprised of the following:
- An overarching safety assessment (including generic/common tasks ‘through life’ safety issues assessment) which supports the Final decommissioning plan;
- Detailed safety assessments for phase 3 and phase 4 decommissioning activities;
• Updated facility Limits and Conditions documentation.

The FFF decommissioning project required numerous safety analyses that could each be referred to as a safety assessment. The safety analyses ranged from a top-level generic safety assessment which bounded the scope of the overall decommissioning project, to more detailed safety assessments which supported each decommissioning phase and the tasks contained therein.

The decommissioning phases progressed from areas of the facility where there are very low levels of radioactive contamination towards those areas with the greatest levels of contamination and also the highest potential risk to the operator.

In this test case the decommissioning of each area of the facility was treated as a discrete phase of the decommissioning plan. The safety assessments related to two specific phases of the decommissioning project were examined during the production of this test case. Phase 3 & 4 were selected for examination as they were the most recent information available.

It is noted that the facility generic safety assessment (which was developed in support of the facility decommissioning plan) was used as the starting point for the development of the phase specific final safety assessments.

3.4.4 Description of Tasks Completed for Each Phase

Phase 1 and 2 Decommissioning

Phase 1 and 2 included the removal of bulk materials and preparation of the area to enable lay down of equipment and to free up space for waste packaging tasks.

Starting point of Phase 3

• Phases 1 and 2 of the decommissioning plan have been completed.

Phase 3 Decommissioning

Phase 3 decommissioning is identified as the dismantling of the fuel line gloveboxes and decontamination of the area. An example of the tasks involved in phase 3 decommissioning of the FFF is presented below:

• Manual and remote size reduction activities of the Fuel line gloveboxes;
• Removal and size reduction of the equipment installed within the Fuel line to support remote operations;
• Removal and size reduction of the local size reduction area ‘TEDAK™’ ventilation system;
• Decontamination of the Fuel line area;
• Removal and size reduction of the 1st floor services pipework;
• Removal and size reduction of the Low Active Drain system and collection tank.

End point of Phase 3

• All plant and equipment, including the low active drain system has been removed from the Fuel line area;
The fuel line area has been decontaminated to ‘free breathing’ levels and contamination has been removed or encapsulated to levels that meet the agreed demolition handover criteria.

Starting point of Phase 4:

- Phase 3 of the decommissioning plan has been completed.

**Phase 4 Decommissioning**

Phase 4 decommissioning consists of the decontamination and removal of the building ventilation and services. An example of the tasks involved in phase 4 decommissioning of the FFF is presented below:

- Removal and size reduction of the mobile filtration unit (MFU);
- Removal and size reduction of the air-fed suit shower facility;
- Removal of the breathing air compressor;
- Removal and size reduction of the building extract system.

End point of Phase 4:

- The active ventilation and auxiliary equipment has been removed from the facility, and contamination has been removed or encapsulated to levels that meet the agreed demolition handover criteria.

**3.4.5 Evolution of the Safety Assessments**

Figure 7 provides an illustration of the information flow from the facility Decommissioning Plan, through the final safety assessment and includes the production of phase specific detailed safety assessment. This results in the identification of safety limits and conditions that are suitable for implementation in the working level documentation.
3.4.6 Incorporation of Lessons learned from Phase 1 & 2

As this is a phased decommissioning project, a review was undertaken of the existing safety assessments to identify those areas from previous phases (phase 1 and 2) that could be taken forward and utilised in the compilation of the new safety assessments for the next phases (phase 3 and phase 4).

This was completed in the test case facility by performing a gap analysis where the previous phase activities were compared and contrasted with the proposed Phase 3 activities.

This approach enabled a comprehensive hazard identification to be completed but avoided duplication of effort. This approach helped to ensure that the safety assessment for phase 3 was performed with the same rigour as the previous phases. In addition, it helped to ensure that all planned decommissioning tasks have been subject to an appropriate level of safety assessment.

This approach to safety assessment also provided an opportunity to capture Operational and Experience Feedback (OEF) from previous decommissioning phases (phases 1 and 2). The incorporation of OEF assisted in the selection of improved decommissioning tools and techniques, for use during dismantling activities. Such an approach also assisted with overall risk reduction.

Phase 3 of the decommissioning strategy incorporated lessons learned from other on-site decommissioning projects. An example of this was the selection of appropriate cutting tools. As a result of the variety of tasks completed on the nuclear licensed site, the workforce has now acquired a high level of decommissioning experience.
3.4.7 Selection of Decommissioning Tools and Techniques

Selection of suitable decommissioning techniques was completed as part of the safety assessment process. New decommissioning techniques were justified and demonstrated as capable of obtaining the desired outcome from both a performance and safety perspective during the assessment process.

The decommissioning project was phased based upon the specific areas of the facility, the potential consequences of hazards and to enable the decommissioning team to learn how to manage the radiological hazard. As the decommissioning operatives’ experience grew they gained more skills in this area and therefore were more competent to address the more hazardous activities which were present in the later phases. Operational feedback was carried forward to the next phase of the decommissioning project, such as tooling efficiency.

In the FFF test case, a number of benefits were provided by completing decommissioning trials, e.g. from physical mock-ups. These were particularly useful in assisting the design of dismantling methods and to train personnel. Before any decommissioning technique was selected, an evaluation of its suitability was conducted.

The following factors influenced the suitability of the decommissioning techniques:

- Potential impact on the workers and the environment, for example giving preference to techniques that do not generate airborne radioactivity;
- Types and properties (size, shape, contamination limits and accessibility) of the equipment and structures to be dismantled;
- Decontamination factor and cutting rate likely to be achieved;
- Impact on existing SSC’s for compatibility with decontamination solutions and processes to ensure they will not be degraded and become ineffective;
- Methods available for controlling radiological and non-radiological hazardous materials;
- Reliability of the dismantling equipment and tools and its simplicity to operate, decontaminate and maintain;
- Availability of waste containers and the associated handling systems;
- Waste management including storage and disposal;
- Cost–benefit analysis comparing the radiological benefits and waste management benefits of the decommissioning technique with the expected costs;
- Time, funding and schedule constraints.

Techniques Selected for Phase 3 of the decommissioning plan

The decommissioning phases are split into specific activities, each adopting differing techniques and these are detailed in the sections below:

A mixture of manual and remote dismantling techniques were selected for phase 3:

- Manual and remote size reduction operations of the fuel line gloveboxes:
  - Remote BROKK (remotely operated vehicle) size reduction machine;

- Removal of the plant and equipment installed to decommissioning operations:
  - Manual plasma cutting torch and grinding;
  - Use of a manually operated reciprocating saw.

- Decontamination of the Fuel line area:
  - Tie-down and fixing of contamination (by application of proprietary fixants), known as ‘spraying out’ of the area.

Techniques Selected for Phase 4 of the decommissioning plan

At this phase of the overall project a mixture of manual and remote dismantling techniques were selected.

- Removal of the plant and equipment installed to decommissioning operations.
  - Manual reciprocating saw.

- Removal of the Building Extract System.
  - Manual dismantling.

These techniques were used as an initial assumption in the safety assessment process. Those that achieved the agreed safety criteria proceeded to implementation and inclusion in the detailed safety assessment. Those tools and techniques which did not achieve the required safety criteria (e.g. due to excessive worker or public consequences) were subject to further development.

3.4.8 Aspects recorded in the Generic Safety Assessment

In the FFF test case a number of aspects were identified as being applicable to all phases of the decommissioning strategy. These were assessed and recorded in the Overarching Safety Assessment. The following is a summarised list of the aspects that were included:

Data:
- Facility Radioactive inventory;
- Plant records and data – e.g. knowledge management regarding operational throughput and likely accumulation of material.

Known plant configuration:
- Ventilation;
- Building structure and crane (handling systems);
- Plant isolations especially process feeds;
- Radiometrics (e.g. alpha in air monitoring).

Safeguards applicable across all phases of the decommissioning strategy:
- Training and the site management system;
- Security and access control;
- Waste management e.g. from routine housekeeping;
• Emergency preparedness;
• The requirements for maintaining and testing of safety controls (engineered and operational);
• Equipment care in regard to ageing and obsolescence;
• Plant operators’ routine inspections and readings.

The following **hazards** were identified in the generic safety assessment as being common to all phases of the decommissioning strategy:

• Fire;
• Criticality;
• Receipt and storage of plutonium contaminated waste;
• Externally initiated hazards such as seismic, flooding, extreme temperatures, extreme weather etc.;
• Internally initiated hazards such as a steam line leak.

### 3.4.9 Production of the Final Safety Assessment for the First Decommissioning Phase

Production of the final safety assessment of the first decommissioning phase of the FFF was achieved by producing a detailed safety assessment (hazard analysis and engineering analysis) of the phase 1 tasks. The hazard analysis and engineering analysis were completed as per the DeSa methodology. Further details on the use of the tools and techniques described by the DeSa project are available in the DeSa report.

It is notable in the FFF test case, that in addition to the detailed safety assessments for the first decommissioning phase, the Generic Safety Assessment also provided an overview of the safety assessments, facility major hazards and the likely consequences of the subsequent decommissioning phases.

The initial or outline decommissioning assessments for phase 1 were produced at an early stage so that stakeholder engagement could begin and to provide an early opportunity to incorporate learning and experience.

### 3.4.10 Update of the Safety Assessment as a Consequence of a Change in the Decommissioning Plan

As a significant amount of hazard reduction had been achieved by the tasks accomplished in Phases 1, 2 and 3, there was no longer a ‘business need’ to continue funding the on-going decommissioning and dismantling tasks.

Therefore, it was concluded that Phase 4 (removal of the ventilation system) and the subsequent demolition of the building demolition would not be progressed. As a result of this change in scope, the required managerial and engineered controls were reviewed and revised to facilitate leaving the building in a safe-state.
3.4.11 Management of Interfaces Between Phases

In order for the FFF decommissioning (and the phase specific dismantling activities) to proceed, it was necessary to maintain a number of facility-wide systems in an operational mode, e.g. the facility fire alarm system, the facility ventilation system.

The interface between these operational systems and the phase specific dismantling tasks had to be carefully planned and managed. The management and control of the identified interfaces was described in the Generic Safety Assessment.

The following is an example of how such an interface between the Generic Safety Assessment and the phase specific decommissioning activity was managed in the FFF test case.

The Generic Safety Assessment recognised that the overall facility ventilation system would normally be operational or ‘running’ – and the phase specific safety assessment confirmed that there is a requirement to provide sufficient ventilation to support glovebox dismantling. The role of the decommissioning team was to manage the identified interface. In this particular instance, the required level of ventilation could be achieved either by installation of additional local ventilation or by enhancement or modification of the overall facility ventilation.

In the case of the FFF, the decommissioning management team chose to supplement the existing facility ventilation system to assure themselves that the necessary containment and ventilation extraction rates were available. The FFF decommissioning team achieved the required safety function by the construction of an additional containment around the gloveboxes which was supported by a dedicated mobile filtration unit (MFU). The MFU supplemented the building ventilation system.

It is notable that in the actual FFF upon which this test case is based, that part of the facility was able to function as a waste facility whilst decommissioning was undertaken in adjacent areas.

One task of the facility decommissioning project was to receive, monitor and despatch PCM waste. The PCM waste arose from other decommissioning projects in adjacent facilities on the licensed site. The waste operations were completed in an area of the facility that had been decontaminated and decommissioned during phase 1 of the facility decommissioning plan.

The Generic Safety Assessment and the phase specific detailed safety assessments identify safety mechanisms, devices, controls and managerial arrangements. Together these documents define the limits and conditions to enable safe decommissioning of the facility. It was the responsibility of the safety assessor to compare the phase specific detailed safety assessment with the generic safety assessment to ensure the complete safety envelope was described. The generic safety assessment and the phase specific safety assessment have to be implemented in a timely manner to enable the full safety envelope to be adequately controlled.

It was the responsibility of the decommissioning project team to ensure that the arrangements described in the generic safety assessment and in the phase specific safety assessments were implemented and understood by the decommissioning operatives, before the decommissioning task began. Further details regarding the implementation of safety mechanisms limits and conditions are provided in Chapter 5 of the main volume of this publication and are referred to in section 3.6 of this Annex.
3.4.12 Role of the Regulatory Body During Decommissioning Conduct, Including Multi-Phased Approach

Where regulatory approval was requested (e.g. before commencing phase 3) the detailed task plan supported by its associated final safety assessment and the independent review of the safety assessment was submitted to the regulator for consideration.

The regulator did not choose to complete a further detailed examination of the final safety assessment or the associated supporting documentation. Regulatory review depends upon a number of factors, not least of which are the potential consequences of the decommissioning phase. The regulatory consultation was completed prior to agreement to proceed was given to the decommissioning operators.

3.5 DECOMMISSIONING CONDUCT CONCLUSIONS

The FFF test case pre-dates the DeSa and FaSa projects and as such is not always an exact match regarding terminology. Table 2 correlates the FaSa terminology with the activities completed during conduct of the test case to examine if there is consistency in approach.

It is considered that the concepts proposed by the FaSa conduct chapter (chapter 4 of the main volume of this publication) regarding use of safety assessment results have been illustrated by the test case.

The safety assessments completed in support of decommissioning were updated and revised to become detailed safety assessments that reflected operational experience feedback from earlier decommissioning phases.

<table>
<thead>
<tr>
<th>FaSa terminology Decommissioning Conduct</th>
<th>Activities completed as part of the FFF Test Case</th>
<th>Consistency between FaSa and FFF TC</th>
</tr>
</thead>
<tbody>
<tr>
<td>Optioneering</td>
<td>Option selection, design aspects and detailed SA</td>
<td>Yes</td>
</tr>
<tr>
<td>Evolution of safety assessments</td>
<td>Iterative development of safety assessment was used for FFF.</td>
<td>Yes</td>
</tr>
<tr>
<td>Initial safety assessment</td>
<td>From initial (scoping) calculations, to final safety assessment.</td>
<td></td>
</tr>
<tr>
<td>Preliminary safety assessment (for the next phase, phase 1+n)</td>
<td>Final safety assessments for each phase were available prior to the phase specific decommissioning tasks commencing.</td>
<td></td>
</tr>
<tr>
<td>Detailed Safety Assessment</td>
<td>Use of Operational Experience Feedback to influence next phase.</td>
<td>Yes</td>
</tr>
<tr>
<td></td>
<td>Update as a result of the change in overall facility strategy was also illustrated.</td>
<td></td>
</tr>
</tbody>
</table>

Table 2: Mapping of the FaSa terminology of Decommissioning Conduct to Activities completed as part of the FFF test case
3.6 IMPLEMENTATION OF SAFETY ASSESSMENT RESULTS

This test case demonstrates how the resultant safety assessment derived limits and conditions were implemented and incorporated into the decommissioning task working documentation, including the appropriate compliance activities (both managerial and engineering) according to chapter 6 of the main volume of this publication. Illustrating Documents are included in Appendix B and Appendix C.

The safety assessments prepared as part of the FFF planning and conduct work stream were used to define and derive the decommissioning task limits and conditions. The FFF managerial team had the opportunity to review the safety assessments and consider the practical implications of implementing the identified Safety SSCs, before the detailed safety assessments were finalised.

The benefit of such an approach was that the common tasks and their associated safety controls were identified by the FFF managerial team and were implemented across the whole decommissioning project ahead of the completion of detailed safety assessments for a particular decommissioning phase. This ‘early’ implementation of common controls and safety mechanisms enabled early release of personnel to support the implementation of the phase specific tasks.

One disadvantage associated with the early adoption of the generic safety assessment was the management of the subsequent phases and their interfaces with the generic safety assessment. The interaction between the phase specific detailed safety assessments needed to be carefully monitored to understand the impacts of completion of decommissioning phases upon the assumptions and safety mechanisms and arrangements identified in the generic safety assessment.

Figure 8 provides an overview of the information flow from the overarching (or generic) safety assessment down through the detailed safety assessments to the point of decommissioning work control.
3.6.1 Derivation of Decommissioning Safety Limits and Conditions -

The aim of the completion of the phase specific safety assessments was to identify and list those control measures (engineered and managerial) considered to be most effective and significant in ensuring safe conditions against the identified fault sequences.

As identified in DeSa and FaSa, safety controls can comprise engineered controls (e.g. SSCs) and/or administrative controls, which mitigate the consequences and/or frequency of faults or prevent them from happening.

In the United Kingdom the ONR nuclear site licence conditions No. 23 and 27 require that the safety assessment explicitly identifies:

- The safety controls needed to keep risks to workers and others as low as reasonably practicable (ALARP);
- The contingency arrangements to be followed when controls are known to be unavailable, i.e. during planned or unplanned outages of equipment or when administrative controls have been breached.
3.6.2 Methodology Used by the FFF Test Case for Identifying Safety Controls

For each fault sequence leading to worker or public dose, the safety assessor recorded the safety controls available to mitigate the effects. This was presented in a tabular form known as a fault schedule. The fault schedule identified the set of controls that the safety assessment claimed credit for in keeping risks ALARP. These controls were identified as candidate safety controls.

The demonstration of safety often depended upon many assumptions, some of which, (such as inventory limits), were more obvious and important than others. In the case of the FFF test case any protection (an inventory limit for example) needed to ensure the safety assessment assumptions remained valid were included at this point as a candidate safety control.

In the United Kingdom it is also a requirement to ensure that sufficient protective measures against significant faults to onsite workers and members of the public are also designated.

During the production of the FFF safety assessments the fault sequence groups (see Appendix B, section 6.4) were examined to identify the facility’s significant faults, and understand if the unmitigated fault sequence required further more detailed assessment (see DeSa project).

Where reasonably practicable the safety assessor:

- selected controls that were as close to the start of the fault sequence as possible;
- chose engineered controls which were preferable to administrative controls;
- chose controls that prevented fault sequences from escalating.

The DeSa project provides further examples on the identification of safety controls.

3.6.3 Implementation of Administrative Limits and Conditions Identified from the Final Safety Assessment

The first step that was completed was to confirm that a suitable and sufficient overall management system was in place for the FFF decommissioning project. It was already confirmed by the regulator that the site licence company had a suitable and sufficient management system (generic safety assessment) which ensured the following list was covered:

- Organisational structure for decommissioning with clear roles and responsibilities;
- Process and procedures for management of change (change control process);
- Process and procedures for the preparation and execution of work tasks (works control);
- Maintenance and testing procedures;
- Operator or worker training and testing programmes;
- Radiation protection programmes and procedures;
- Occupational health and safety programmes;
- Emergency preparedness;
• Quality assurance programme,
• Record retention, e.g. documentation and record keeping procedures.

It was confirmed that these site-wide and generic safety management measures would remain in place until the final decommissioning stage was completed.

Each administrative control that was defined in the final safety assessment was embedded into the working procedures associated with the specific decommissioning activity. The ‘decommissioning limits and conditions’ were developed in order to define how in practice the results of the final safety assessment were to be implemented.

Linked with this document are the detailed working procedures that were developed to allow the dismantling activities to be performed by the workers. Arrangements were put in place to ensure that the safety related activities were adequately authorised and controlled, with the level dependant on the safety significance of the activity. These arrangements enabled the following:

• Authorisation of the decommissioning limits and conditions;
• Approval of the working procedures;
• Training of personnel;
• Changes to the decommissioning limits and conditions;
• A single point of responsibility.

All the facility working procedures have a standard format that is specified by the management system. Thus, they followed a suitable sequential presentation that has a clear, concise and logical text.

The decommissioning activities were undertaken in accordance with approved working procedures and these defined how the decommissioning activity was carried out and identified the steps to be taken in the event of an abnormal occurrence. These working procedures were issued and controlled in accordance with the management system.

3.6.4 Implementation of Engineering Limits and Conditions identified from the Final Safety Assessment

The following are examples of engineering limits and conditions identified by the FFF test case:

• Improved ventilation systems including interlock to the existing building ventilation system;
• Defined size reduction areas with local ventilation system that are interlocked to the size reduction tooling;
• Improved containment and additional radiometrics, including links to the building alarm monitoring system;
• Fire exit routes / emergency egress, including additional emergency lighting and fire points.
3.6.5 Assembling of the Identified Decommissioning Limits and Conditions

As detailed in conduct (chapter 3 of the main volume of this publication), one of the principle outputs of the safety assessment process is to determine the limits and conditions necessary for safety. The DeSa project provides many examples of the various criteria used to select appropriate safety controls. These criteria are not repeated in this test case. The FaSa project seeks to understand the implementation of safety assessment results and one key area of FaSa is the use of safety assessment results (managerial and administrative controls) at the work face.

In order to ensure that the limits and conditions applicable to each phase of the FFF decommissioning project were understood and applied by the decommissioning operatives, the first stage of implementation was to collate them all into one document.

Part of this involved the safety assessor recording the derivation of each identified limit and condition. In the case of the FFF test case, this information was presented as a summary table produced against each fault sequence group (see Appendix B, 6.3).

Within the summary table, for each engineered control, the safety assessor also identified the action on outage (e.g. breakdown or maintenance) the substitution arrangements and the performance criteria.

For both engineered and managerial controls the facility limits and conditions schedule identified the required compliance arrangements. The compliance arrangements explicitly identified how the facility would demonstrate compliance with the identified controls. The improved visibility of the safety assessment results enabled the decommissioning operators to better understand the safety controls required for each decommissioning task.

To achieve full understanding and clear visibility of the various safety systems and components the FFF operating instructions which record the plant conditions necessary prior to commencement of operations, see extract in Appendix B, 6.5, were reviewed, revised and re-issued.

3.6.6 Implementation of Engineered Safety Controls

As discussed in chapter 6 of the main volume of this publication, implementation, it is necessary to implement the output of the decommissioning safety assessment results in the working level documentation.

The placement of the safety limits and conditions in the working level documentation ensures that at the point of work the decommissioning operator is fully aware of the hazards associated with the decommissioning task and the important safety controls that will keep him safe.

For engineered systems it is important to not only record their safety function in the operating documentation but it is essential to ensure that the required safety function continues to be delivered. Safety function and safety performance is monitored by the use of maintenance. The FFF test case ensured that for those engineered items that has a safety function in the decommissioning phase that they were recorded on the facility maintenance schedule, see extract in Appendix C, 8.3.

During the FFF decommissioning project and transition between decommissioning phases (e.g. Phase 1 to Phase 1+n) the facility maintenance schedules were updated.
This was to record if a piece of equipment has been removed or if an additional item has been brought into the facility. One of the challenges presented by the use of a phased decommissioning approach was to manage the interfaces between the phases. An up to date maintenance schedule was helpful in assisting the plant management in controlling the decommissioning safety envelope.

The plant maintenance schedule (PMS) is the ‘top level’ document and was used in the FFF to define the examination, inspection, maintenance and testing requirements for all equipment which might have had an effect on radiological or nuclear safety.

The FFF PMS document also specified the appropriate examination, inspection, maintenance and testing requirements to ensure safety, availability and reliability. The PMS provided a clear link to the associated working procedure that utilised the particular safety control.

The mechanism for prompting and recording the examination, inspection, maintenance and testing requirements of the identified safety controls was and continues to be a computerised maintenance management system.

3.6.7 Good Practice Approach to Implementation of the Working Procedures

During production of the working level procedures, the FFF test case identified a number of good practices that helped to ensure relevant decommissioning task safety information was conveyed to and understood by the workforce. The FFF test case good practices are outlined in the following paragraphs:

Organisation of Work

- A clear understanding and managerial arrangements in place enables understanding and control of the interfaces between the decommissioning work packages & tasks;
- If parallel working is to be used (as in the FFF Test case where receipt of PCM material to one area of the facility is ongoing during decommissioning) then it proved useful to put in place clear managerial controls to clearly demarcate where decommissioning activities start and finish;
- Control of contractors, where contractors are being used to complete a specific decommissioning task (such as removal of asbestos contaminated material), there clear arrangements can ensure appropriate managerial control is exerted.

Safety parameters

- Identification of safety parameters which are necessary to monitor compliance with the defined safety envelope;
- Arrangements enable comparison of the specific decommissioning phase with the facility Safety Envelope / Basis of safety as described in the Overarching Safety Assessment;
- Arrangements to enable an update of the Site Emergency Plan such as changes to access/ egress routes and the update of the overall site risk data, as the facility moves from one decommissioning phase to the next.
3.6.8 Creation of Working Level Documents (including OEF)

For the FFF test case a good practice was noted whereby the facility operatives were included in the production and review of the revised decommissioning instructions. As part of the planning for the next decommissioning phase, the decommissioning operatives were encouraged to record operational experience and feedback that could shape and influence the detailed safety assessments and method statements for subsequent decommissioning phases. This enabled operational experience feedback to be captured and was particularly useful for the selection of dismantling tools and techniques.

This operational, experience feedback was captured in the revised phase 4 detailed safety assessments.

3.6.9 Training of Operators in Revised or Updated Safety Limits and Conditions

The FFF decommissioning project adopted a phased approach to improve skills – particularly with respect to training in use of specialist tools and techniques. The use of a phased approach to decommissioning enabled the FFF decommissioning operators to be trained in a controlled manner and avoided the introduction of error producing conditions. The decommissioning operatives moved from working in areas of relatively low contamination (relatively low hazard) to high contamination, increasing their decommissioning skills and experience as the project progressed.

As the FFF decommissioning project advanced through the various phases it was necessary to update the decommissioning operatives in the revised procedures and to up-skill them with respect to the use of particular dismantling tools and techniques.

The training of the decommissioning operators consisted of use of in-active trials or mock-ups as well as review of the revised phase specific decommissioning method statements and risk assessments. Much use was made of point of work discussions at the start of each work day.

It was imperative that the decommissioning management team organised the decommissioning tasks to suit the skills profile of the available personnel. This ensured that the potential for radiological dose uptake or other injury to personnel was minimised.

3.6.10 Works Authorisation

Authorisation was received from the appropriate authority at the site level. Daily work authorisation (work permits) were given and understanding of the key limits and conditions discussed during the pre-job brief.

Graded approach may be necessary in terms of undertaking a ‘readiness review’ prior to carrying out the work, these was not deemed necessary for FFF proportionate to the hazards associated.

3.6.11 Creation of Working Level Documentation.

Working level documentation was produced for each phase of the decommissioning project. The aim of the phase specific working level documentation was to convey in
a meaningful way the safety significant limits and conditions that have been identified by the safety assessment process.

Such working level documentation can include operator instructions, supervisor instructions, point of work risk assessments and point of work control documents known as ‘permit to works’.

3.6.12 Validation

Validation is recognised as important for high hazard activities as a graded approach; however, this was not required in this test case as was the local responsibility of the project / facility manager to ensure compliance. During this test case, the safety assessors carried out reviews to ensure that the intent of the final safety assessment continued to be met in the practicality of the decommissioning activities.

3.6.13 Decommissioning Implementation Lessons Learned

Lessons learnt from the implementation phase was that in some areas the controls identified by the radiological and criticality assessments contradicted one another. This was picked up by the project review carried out by the facility operators. Such events can be avoided by involving operators in a cross-check of the safety assessment assumptions.

‘Mock-ups’ took place early in the FFF project. The ‘mock-ups’ enabled the safety assessors to fully understand the decommissioning task and allowed the operators to become familiar with the required safety controls. This enabled the overall risks associated with the decommissioning task to be reduced as well as achieving timely training of the operators.

Pre-job briefs were carried out which increased the visibility of the limits and conditions to the operators. The pre-job briefs used an ‘active listening system’ whereby the management team and operators used listening and feedback to check the operators’ level of understanding.

Lesson learnt regarding post validation of project review was not undertaken other than the business drivers of cost and time taken. Of benefit and with hindsight would have been a review of how the safety assessment was implemented and what lessons could be learnt.
3.7 IMPLEMENTATION CONCLUSION

The FFF test case pre-dates the DeSa and FaSa projects and as such is not always an exact match regarding terminology. Table 3 correlates the FaSa terminology with the activities completed during implementation of the test case to examine if there is consistency in approach.

It is considered that the concepts proposed by the FaSa Implementation chapter regarding use of safety assessment results have been illustrated by the test case.

<table>
<thead>
<tr>
<th>FaSa terminology Decommissioning Implementation</th>
<th>Activities completed as part of the FFF Test Case</th>
<th>Consistency between FaSa and FFF TC</th>
</tr>
</thead>
<tbody>
<tr>
<td>Limits and Conditions</td>
<td>Decommissioning Limits and Conditions specified</td>
<td>Yes</td>
</tr>
<tr>
<td>Management System</td>
<td>Site Management System</td>
<td>Yes</td>
</tr>
<tr>
<td>Working Procedures</td>
<td>Working procedures for each decommissioning action</td>
<td>Yes</td>
</tr>
<tr>
<td>Works Authorisation</td>
<td>Simple illustration of the actions and also daily authorisation process</td>
<td>Yes</td>
</tr>
</tbody>
</table>

Table 3: Mapping of the FaSa terminology of Decommissioning Implementation to Activities completed as part of the FFF test case
3.8 REGULATOR ENGAGEMENT

The regulator was periodically provided with updates regarding the project (regulatory review is referred to in chapter 7 of the main volume of this publication). Agreement to commence Phase 3 (a permission) was provided from the regulator, post submission of the safety assessments for phase 3 (for a correlation with the FaSa terminology see table 4).

Then where appropriate (proportionate to the hazard and according to the licensor’s own due process) the specific facility decommissioning strategy is submitted to the regulator for information or permissioning. This is an example of the graded approach.

<table>
<thead>
<tr>
<th>FaSa terminology Regulatory Review</th>
<th>FFF Test Case Activities completed as part of the</th>
<th>Consistency between FaSa and FFF TC</th>
</tr>
</thead>
<tbody>
<tr>
<td>Review of the operator’s written safety assessment results</td>
<td>A graded approach was applied. Regulatory permission was received before the commencement of works covered by the TC</td>
<td>Yes</td>
</tr>
<tr>
<td>Review of the Implementation of safety assessment results in the final decommissioning plan</td>
<td>For the FFF Test case the Regulator was engaged in regular discussions to provide an update on decommissioning progress</td>
<td>No</td>
</tr>
<tr>
<td>Review of the implementation of safety assessment results during inspections</td>
<td>Due to Regulatory confidence from past inspections of the facility there was no formal inspection carried out as per UK’s graded approach to regulation</td>
<td>Equivalent</td>
</tr>
</tbody>
</table>

Table 4: Mapping of the FaSa terminology of regulatory review activities completed as part of the FFF test case

3.9 DECOMMISSIONING TERMINATION

Decommissioning termination is addressed in chapter 5 of the main volume of this publication. The original endpoint of this FFF test case was going to be demolition of the facility. This has not taken place and therefore termination has not taken place for this project.

As noted in earlier sections, the FFF decommissioning plan now (2012) includes the use of the facility as a PCM receipt and limited holding area prior to consignment. The reduction in hazard / inventory in the facility has resulted in a deferral of the
phase 4 of the decommissioning project and deferral of demolition of the facility. As such an ‘interim end state’ has been reached as agreed with the site management.

- Phase 1 – Completed
- Phase 2 – Completed
- Phase 3 – Completed
- Phase 4 – Deferred

3.9.1 Strategy Change

Due to strategy changes at site level during the implementation of the decommissioning operations in the FFF, the decision was taken to postpone the achievement of the proposed end state. After dismantling of all equipment and removal of waste of the facility during phase 3, the decision was taken to:

- not perform the decontamination of the building walls and floor;
- continue to operate the interim storage of PCM drums;
- keep the ventilation systems of the building operational.

The final decontamination and dismantling will be performed at a yet undetermined date.

3.9.2 Additional Considerations as a Result of Deferred Termination

The impact of the deferred decommissioning is captured in the periodic review of the facility safety assessment which is scheduled to be completed every 10 years. The periodic review of the facility safety assessment examines the basis of safety of the facility and seeks to update the overarching safety assessment.

There are now administrative and physical aspects to be considered due to the deferral:

- Asset management;
- Record keeping;
- Periodic review;
- Review of the interdependencies now that the facility is not being demolished;
- Mobilisation of the decommissioning operators during the care and maintenance phase.

Completion of the facility periodic review enables the impact of deferred decommissioning to be quantified. The review process identifies if there are any life-limiting features or if there are any safety functions which are likely to be challenged by the deferral of their removal. The periodic review process re-examines the functionality and operability of safety related equipment and essential services to the facility. This is part of the engineering substantiation process and its outcomes shape and influence the asset management strategy that is adopted for the facility (ageing of equipment, maintenance of the building).

During the planning phase, the end state described in the decommissioning plan was the following:
Physical state: all equipment, active ventilation and ancillary equipment have been removed (all rooms are empty);

Radiological state: decontamination of walls and floor has been performed to reach ‘free breathing level’ and the criteria to perform handover and demolition have been met.

The impact of the deferral is not a straightforward one, as the asset management of the building remaining in a ‘care & maintenance’ state is an important factor in terms of on-going safety assessment, and this review has to be incorporated into the safety assessment.

**3.10 DECOMMISSIONING TERMINATION CONCLUSION**

As the achievement of the end state has been postponed, decommissioning termination as defined in chapter 5 of the main volume of this publication cannot be reached at end of the original time frame, and cannot be addressed in the scope of the FFF test case (see table 5). However, at the end of phase 4, important data such as radiological characterization of the facility (radiological survey, hot spots location) has been collected to allow and facilitate the implementation of the phase 5, when the site is ready to proceed.

Table 5: Mapping of the FaSa terminology of decommissioning termination completed as part of the FFF test case

<table>
<thead>
<tr>
<th>FaSa terminology Decommissioning Termination</th>
<th>Activities completed as part of the FFF test case</th>
<th>Consistency between FaSa and FFF TC</th>
</tr>
</thead>
<tbody>
<tr>
<td>Approaches to safety assessment</td>
<td>Termination is not wholly illustrated by the FFF Test case.</td>
<td>No</td>
</tr>
<tr>
<td>Approaches for restricted use.</td>
<td>A Restricted use end-state was defined (re-use of the building location for other purposes).</td>
<td>Yes</td>
</tr>
<tr>
<td>Multi-facility site\</td>
<td>Due to changes in strategy the test case facility has deferred the latter decommissioning stages.</td>
<td>Yes</td>
</tr>
<tr>
<td>End state definition</td>
<td>As the facility is part of a complex multi-facility site, the end state is restricted use</td>
<td>Yes</td>
</tr>
</tbody>
</table>
4. **SUMMARY**

The FFF test case pre-dates the DeSa and FaSa projects and as such is not always an exact match regarding terminology. Tables 1-5 correlate the FaSa terminology to the activities completed during the FFF test case with regard to decommissioning planning (table 1), decommissioning conduct (table 2), implementation of the safety assessment result (table 3), regulator engagement (table 4) and termination of decommissioning (table 5). The FFF test case illustrates that despite the pre-dating there is a wide consistency of the used approach with the DeSa and FaSa methodology.
5 APPENDIX A: DETAILED FACILITY DESCRIPTION

5.1 SITE DESCRIPTION AND LOCAL INFRASTRUCTURE

The site is located on the West Cumbria coast adjacent to the Irish Sea on the western outskirts of the Lake District National Park. The site licensed boundary encompasses an approximate area of 276 Ha and is located at 54°N, 3°W. The site is mainly in the Parish of St Bridget Beckermet, within the Copeland District of the County of Cumbria.

The major local towns of Whitehaven, Workington and Barrow are approximately 14 km to the North, 25 km to the North and 38 km to the Southeast respectively. There are about 200 people living within 2 km of the site: the nearest settlement of any size is Seascale 2.5 km distant, with a population of about 1800.

5.1.1 Land Use

The countryside around the site is mainly utilised for farming. A brief description of the other activities is given below:

5.1.2 Visitor Centre

The Sellafield Visitor Centre, which is situated approximately 1 km from the site has attracted on average about 170,000 visitors per year since refurbishment. Visitors spend typically 2-4 hours at the centre. While the daily attendance has averaged 467 over the same period, the maximum day attendance has been 2600. The maximum number allowed in the centre at any time is 750.

5.1.3 Combined Heat and Power Plant (CHP)

The Fellside CHP plant is authorised under the Pollution Prevention and Control (England and Wales) Regulations 2000: it is rated at 176 MW, has three gas turbines and one steam turbine, each turbine driving its own alternator. Multiple units are provided to ensure the steam supply is highly reliable. The gas supply is provided by pipeline direct from the British Gas Main at Ulverston, which is supplied from both the North Sea and the Morecambe Bay fields.

5.1.4 Local Industry

There are no significant industrial establishments within 5 km of the site. The nearest significant establishment in the chemical and allied industries is a contract manufacturer and processor of custom chemicals at Workington (32 km north). The gas platforms in Morecambe Bay are, at nearest, 50 km away. Gas from the field is landed at Barrow (38 km south-east). The nearest military site is a firing range at Eskmeals (15 km south).

5.1.5 Traffic

The nearest main road is the A595 to the east of site: the smaller approach roads to the site are used almost exclusively by site traffic. The railway from Whitehaven to Barrow passes close to the site. A branch line which runs onto the site is used to receive spent reactor fuel from power stations, bulk chemicals and export Low Level Waste (LLW) to the Low Level Waste Repository. The volume of aerial traffic in the area is low. The nearest airports are at Carlisle (70 km north) and Barrow (40 km...
south). All aircraft (commercial, military and general) are restricted from flying at a height of less than 2200 ft within a circle of radius 3.7 km around the site.
5.2 GEOLOGY OF THE LICENSED SIZE

The FFF is located on a nuclear licensed multi-facility site, adjacent to the Irish Sea Coast at an altitude of between 5-50 m Above Ordnance Datum (AOD). The site falls within the West Cumbria Coastal Plain and is identified as an area of varied open coastline of mudflats, shingle and pebble beaches with localised sections of dunes, sandy beaches and sandstone cliffs.

The site is generally protected from coastal flooding by cliffs, the shingle spit of the river Ehen and a railway embankment. However, coastal erosion and sea level rise has the potential to affect the southern end of the licensed site within the next 100 years if existing defences are not maintained. In summary, the geological sequence at the FFF site comprises of:

- Made ground deposits;
- Quaternary fluvio-glacial deposits;
- Bedrock.

The coastline around the licensed site and further south comprises a sandy beach backed by low sparsely vegetated sand dunes and silted inlets. To the north the coast rises to St Bees Head where there are steep cliffs (about 100 m) of Triassic Sandstone. Further details are presented below.

**Made Ground Deposits**

A layer of made ground of varying thickness is present across the majority of the site due to a long history of repeated excavation, construction, backfilling and landscaping works. The majority of the made ground comprises disturbed, mixed and re-deposited natural ground and a proportion of building debris (e.g. brick, concrete, tarmac, wood, wire, plastic etc.). The thickness of made ground is variable, ranging from less than 1 m to over 5 m, most notably where the former course of the River Calder was infilled during the 1970’s and at the site’s licensed landfills.

**Quaternary Deposits (Drift)**

These deposits comprise a sequence of gravels, sands, silts and clays which are very variable in thickness and lithology. These unconsolidated deposits are predominantly of glacial and fluvial origin and have a maximum thickness of 74 m at the site.

**Bedrock**

The stratigraphic succession is presented in Figure 9 and illustrates the bedrock sequence at the Sellafield site. The Triassic Calder and Ormskirk sandstone formations of the Sherwood Sandstone Group are the only formations described here; the deeper Permian, Carboniferous and Ordovician formations are not discussed as they are not considered relevant to the conceptual model due to the thickness of the sandstone. The sandstones are generally reddish brown in colour, fine to medium grained and prone to weathering. For general interpretation purposes both of these two sandstones have been grouped together under sandstone bedrock. The sandstone bedrock dips towards the south-west with an average inclination of 25°. The strata of the Sherwood Sandstone Group beneath the regional area ranges in thickness between 650m and 1150m, averaging at about 800m thickness.
FIG 9: Stratigraphic succession of the area of the site.
5.3 HYDROLOGY OF THE LICENSED SIZE

The site is located within the surface water catchments of the rivers Calder and Ehen. The river Calder catchment, including its subsidiary stream, Newmill Beck, has a total area of 55.5 km$^2$, while the river Ehen catchment has a total area of 156.6 km$^2$. The site is located at the down-gradient end of the Calder catchment.

The Calder flows through the site in a south-south-westerly direction and forms a natural barrier separating the west and east sides of site. The Ehen flows in a south-south-easterly direction along the south-western site boundary, where it merges with the Calder before flowing across the beach to discharge into the Irish Sea. Newmill Beck flows around the south-eastern corner of the site, where it has been culverted to divert its flow around a licensed landfill and beneath the coastal railway line. Beyond the railway, it feeds two small ponds above the high water mark at the beach before discharging, via another culvert, into the Calder. The site slopes gently from its inland boundary towards the coast with a decrease in ground elevation from approximately 40m AOD to 8m AOD.

Two additional minor streams flow into the Calder. Seaburn Beck drains into the river from its western side and flows through the northern end of Sellafield. The second is an unnamed stream that drains towards the river from its eastern side and flows through the site, although it does not discharge directly into the river. Instead it is intercepted and drains to the Irish Sea via an offshore pipeline. Both streams are partly culverted within the site boundaries. The location of the culverted section of Seaburn Beck has been modified several times since 1946.

Analysis of seasonal river flow hydrographs for the rivers Ehen and Calder indicates that they respond relatively rapidly to rainfall events due to the steep topography and rapid surface runoff in the catchment headwaters. Consistent flow to the river channel from groundwater is observed throughout the year, although baseflow indices are likely to increase between catchments as a function of the proportion of sandstone within that catchment.

Flows through the Calder and Ehen vary seasonally, but typical averages are $1.5 \times 10^5$ m$^3$ per day based on 1930 mm rainfall per year for the Calder, and $5.2 \times 10^5$ m$^3$ per day based on 1750 mm rainfall per year for the Ehen.

The coastline around Sellafield and further south comprises a sandy beach backed by low sparsely vegetated sand dunes and silted inlets. To the north the coast rises to St Bees Head where there are steep cliffs (about 100 m) of Triassic Sandstone.
5.4 SEISMOLOGY OF THE LICENSED SIZE

A commissioned report concluded that seismic activity varies considerably from one area of Great Britain to another.

The seismic activity at the site of the FFF and the Cumbrian region has also been studied and is reported as higher than the average for Britain as a whole, while the offshore Irish Sea is lower than the average.

The $10^{-4}$ annual exceedance probability lies at an acceleration figure of 0.24g for the site compared with a 0.19g average for the country as a whole. The West Cumbrian fault zone includes the coastal strip in the neighbourhood of the site and consists of a complex collection of scattered Northwest trending faults.

Quantification of seismic activity was carried out on a site-specific basis for the nuclear licenced site and calculations show that most of the risk results from moderate sized shallow earthquakes close to the site.

The following annual probabilities of exceedance for various peak ground accelerations were calculated based on the seven historical earthquakes considered for the sites area (table 6).

<table>
<thead>
<tr>
<th>Peak Ground Acceleration</th>
<th>Annual Probability of Exceedance</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.125g</td>
<td>$1.1 \times 10^{-3}$</td>
</tr>
<tr>
<td>0.24g</td>
<td>$1 \times 10^{-4}$</td>
</tr>
<tr>
<td>0.25g</td>
<td>$0.87 \times 10^{-4}$</td>
</tr>
</tbody>
</table>

*Table 6:* Annual probabilities of exceedance at the site.

The design of new plants at the site are assessed for their release potential following a Design Basis Earthquake. The release potential of individual plant items or areas of plant determines the appropriate design category of seismic ruggedness. The magnitude of the Design Basis Earthquake is agreed with the UK licensing authorities (Office of Nuclear Regulation, ONR).

Existing plants have been assessed against seismic events, where there is the potential to give a dose to a member of the public in excess of 5 mSv from inhalation or direct radiation special seismic provisions are put in place.
5.5 SUPPORTING FACILITIES

There are 2 facilities that support the decommissioning operations been undertaken in the FFF:

- Engineered Drum Store – Interim storage facility for PCM;
- Site Services – Electricity, Water and Steam.

All the supporting facilities have their own specific detailed safety assessment which covers all the operations relating to their facility.
5.6 INVENTORY

The radiological inventory of the test case facility was predominantly alpha nuclide based, i.e. mainly Plutonium-239 and ingrowing daughters such as Americium-241. Only notional values taken from the safety assessment are presented within this test case (Appendix B).
6. APPENDIX B: HAZARD IDENTIFICATION AND QUANTIFICATION

6.1 HAZARD & OPERABILITY STUDY (HAZOP)

In the following, an example is given for a formal approach to hazard identification, a so-called hazard & operability study (HAZOP). The formal HAZOP approach is best summarised as a structured examination of facility layout and instrumentation drawings, flowsheets and process diagrams. In the case of the FFF it was structured around the steps and phases of the decommissioning project.

- Hazop 0 – The outline of the overall decommissioning project was reviewed;
- Hazop I – Examination of the available decommissioning phases;
- Hazop II – Review of the phase specific decommissioning tasks.

The HAZOP discussions were directed by use of the guidewords using a systematic and formalised tabular approach.

A list of the guidewords used for this FFF test case is presented below:

- External dose;
- Internal dose;
- Criticality;
- Shielding;
- Loss of containment;
- Ventilation;
- Fire;
- Explosion / detonation;
- Maintainability;
- Remote / manual handling;
- Transport;
- Loss of service;
- Effluent / washings;
- Environmental issues;
- Corrosion / erosion;
- Domino;
- Extreme weather;
- Seismic;
- Toxicity;
- Impact;
- Access / egress;
- Working environment;
• Emergencies;
• Computer systems;
• Communications / noise;
• Control / instrumentation.
### 6.2 HAZARDS IDENTIFIED IN THE SAFETY ASSESSMENT

Table 6: Examples for hazards identified from the safety assessment for the FFF test case. Each hazard is correlated with a fault sequence reference and a fault identification reference (references not included into the FFF test case report).

<table>
<thead>
<tr>
<th>Fault Sequence Reference</th>
<th>Fault Description</th>
<th>Fault Identification Reference (team/meeting/sheet for HAZOP)</th>
</tr>
</thead>
<tbody>
<tr>
<td>01</td>
<td>Accumulation of excess MOX during relocation of the MOX residue Bottles.</td>
<td>Table 2, Reference 1.12</td>
</tr>
<tr>
<td>02</td>
<td>Dropping a MOX residue bottle in the Windscale Suit Shower during relocation.</td>
<td>Table 2, Reference 1.12</td>
</tr>
<tr>
<td>03</td>
<td>Complete failure to relocate the MOX residue bottles.</td>
<td>Table 2, Reference 1.12</td>
</tr>
<tr>
<td>04</td>
<td>Accumulation of excess MOX during Cubicle 3 decommissioning.</td>
<td>Table 2, Reference 1.12</td>
</tr>
<tr>
<td>05</td>
<td>Increased moderation of MOX during Cubicle 3 decommissioning.</td>
<td>Table 2, Reference 1.12</td>
</tr>
<tr>
<td>06</td>
<td>Mishandling of MOX residue bottles generated during Cubicle 3 decommissioning.</td>
<td>Table 2, Reference 1.12</td>
</tr>
<tr>
<td>07</td>
<td>Accumulation of excess MOX during Cubicle 2 South and Cubicle 4 decommissioning.</td>
<td>Table 2, Reference 1.12</td>
</tr>
<tr>
<td>08</td>
<td>Increased moderation of MOX during Cubicle 2 South and Cubicle 4 decommissioning.</td>
<td>Table 2, Reference 1.12</td>
</tr>
<tr>
<td>09</td>
<td>Mishandling of MOX residue bottles generated during Cubicle 2 South and Cubicle 4 decommissioning.</td>
<td>Table 2, Reference 1.12</td>
</tr>
<tr>
<td>10</td>
<td>Failure to re-monitor the Cubicle 2 North Gloveboxes.</td>
<td>Table 2, Reference 1.12</td>
</tr>
<tr>
<td>11</td>
<td>Accumulation of excess MOX during decommissioning of the Cubicle 1, 2 North, 7 and 8 gloveboxes.</td>
<td>Table 2, Reference 1.12</td>
</tr>
<tr>
<td>12</td>
<td>Increased moderation of MOX during decommissioning of the Cubicle 1, 2 North, 7 and 8 gloveboxes.</td>
<td>Table 2, Reference 1.12</td>
</tr>
<tr>
<td>13</td>
<td>Mishandling of MOX residue bottles generated during decommissioning of the Cubicle 1, 2 North, 7 and 8 gloveboxes.</td>
<td>Table 2, Reference 1.12</td>
</tr>
<tr>
<td>14</td>
<td>Increased fissile material in the Cabinet Extract System.</td>
<td>Safety Memo No. 5319</td>
</tr>
<tr>
<td>15</td>
<td>Increased interaction during decommissioning of the first floor cabinet extract system.</td>
<td>Expert Judgement</td>
</tr>
<tr>
<td>16</td>
<td>Increased moderation of MOX during decommissioning of the first floor cabinet extract system.</td>
<td>Safety Memo No. 5319</td>
</tr>
<tr>
<td>17</td>
<td>Mishandling of high fissile content PCM items identified during piece</td>
<td>Table 2, Reference 1.12</td>
</tr>
<tr>
<td></td>
<td>Description</td>
<td>Reference/Sequence</td>
</tr>
<tr>
<td>---</td>
<td>-----------------------------------------------------------------------------</td>
<td>--------------------------------------------</td>
</tr>
<tr>
<td>18</td>
<td>High fissile content in a filled PCM drum.</td>
<td>Table 2, Reference 1.12</td>
</tr>
<tr>
<td>19</td>
<td>Exceeding the safe surface density limit in a PCM drum or drum array.</td>
<td>Standard Fault Sequence</td>
</tr>
<tr>
<td>20</td>
<td>Inhomogeneity in PCM drums.</td>
<td>Standard Fault Sequence</td>
</tr>
<tr>
<td>21</td>
<td>Accumulation of MOX in the Windscale Suit Shower filters.</td>
<td>Standard Fault Sequence</td>
</tr>
<tr>
<td>22</td>
<td>Flooding.</td>
<td>Standard Fault Sequence</td>
</tr>
<tr>
<td>23</td>
<td>Fire and Firefighting.</td>
<td>Standard Fault Sequence</td>
</tr>
<tr>
<td>24</td>
<td>Inadvertent consignment of PCM to Drigg.</td>
<td>Standard Fault Sequence</td>
</tr>
<tr>
<td>25</td>
<td>Chronic accumulation of MOX in the Cubicle 3 Dismantling Area or Cubicle 5.</td>
<td>Standard Fault Sequence</td>
</tr>
</tbody>
</table>
### 6.3 FAULT SEQUENCE GROUPS FOR THE FFF CASE

Table 7: Examples for fault sequence groups identified from the safety assessment for the FFF test case. Each fault description is correlated with a sequence group reference and a fault identification reference (references contained within this table are not included into the FFF test case report).

<table>
<thead>
<tr>
<th>Sequence Group Reference</th>
<th>Fault Description</th>
<th>Fault Identification Reference (team/meeting/ sheet for HAZOP)</th>
</tr>
</thead>
<tbody>
<tr>
<td>70</td>
<td>Residue bottle dropped as a result of difficulties encountered during posting out operations.</td>
<td>MSA/98/147</td>
</tr>
<tr>
<td>80</td>
<td>Dropping of a residue bottle within the shower area.</td>
<td>MSA/98/147</td>
</tr>
<tr>
<td>90 + 100</td>
<td>Windscale Suit problems (ripped or loss of air) during operations.</td>
<td>MSA/98/147</td>
</tr>
<tr>
<td>110</td>
<td>RHM and/or Brokk collision with containment.</td>
<td>MSA/98/147</td>
</tr>
<tr>
<td>130</td>
<td>During size reduction operations RHM or Brokk impact adjacent gloveboxes.</td>
<td>MSA/98/147</td>
</tr>
<tr>
<td>00 - 60</td>
<td>Failure of TEDAK extract/filtration during cutting operations.</td>
<td>MSA/98/147</td>
</tr>
<tr>
<td>120</td>
<td>Failure of MFU extract/filtration during cutting operations.</td>
<td>MSA/98/147</td>
</tr>
<tr>
<td>140</td>
<td>Significant radiation levels on equipment subsequently taken into maintenance area.</td>
<td>MSA/98/147</td>
</tr>
<tr>
<td>130</td>
<td>Impact potential between moving basket and other equipment/gloveboxes.</td>
<td>MSA/98/147</td>
</tr>
<tr>
<td>150</td>
<td>Problems encountered during drum posting out operations.</td>
<td>MSA/98/147</td>
</tr>
<tr>
<td>150</td>
<td>Drop of PCM drum during transport.</td>
<td>MSA/98/147</td>
</tr>
<tr>
<td>160</td>
<td>Glovebox is dropped or becomes snagged during removal to decommissioning area.</td>
<td>MSA/98/147</td>
</tr>
<tr>
<td>170</td>
<td>Changing a manipulator arm.</td>
<td>MSA/96/032.1</td>
</tr>
<tr>
<td>180</td>
<td>Consequences following a release of hydraulic oil.</td>
<td>MSA/96/032.1</td>
</tr>
</tbody>
</table>
6.4 EXAMPLE OF A FAULT SEQUENCE GROUP

Table 8: Example of a fault sequence group (fault sequence group No. 0204, loss of containment during posting out). References contained within this table are not included into the FFF test case report.

<table>
<thead>
<tr>
<th>Fault Sequence</th>
<th>Safety Mechanism</th>
<th>Required Safety Function(s)</th>
<th>Safety Function Class [see NOTE X]</th>
<th>Example of Safety Performance Requirements</th>
<th>Repair Time / PTI [see Note 1]</th>
</tr>
</thead>
</table>
| Internal Dose  | Containment       | To minimise migration of any airborne activity releases as a result of Failure of local, mobile filtration coincident with on-going manual grinding (size reduction activities) | 2 | To provide a protection factor of at least
  • 1E3 in the event of failure of the extract
  • 1E3 with no bag on posting out port (extract on)
  • 50 with no bag on posting out port (no extract) | Not applicable |

Description of potential consequences:
Reduction in the Decontamination Factor provided by posting port resulting in migration of activity into the PAC

Highest consequence threshold exceeded:
Worker: 20-1000 mSv
Public: < 1µSv

Safety Mechanisms and required Operating Instructions:
See Schedule
6.5 EXAMPLE OF A HAZARD ANALYSIS FOR INTERNAL DOSE

In the following paragraphs, an example of a hazard analysis for internal dose is given, as a consequence of a mobile filtration coincident with on-going manual grinding (size reduction activities).

Identification of Hazards

Table 9: source of identification of hazards covered by this assessment to correlate the fault description to fault identification references. References contained within this table are not included into the FFF test case report.

<table>
<thead>
<tr>
<th>Sequence Group Reference</th>
<th>Fault Description</th>
<th>Fault Identification Reference (e.g. team/meeting-sheet for HAZOP)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Part 2 - Fuel Line Operations</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
| FSG 0201 | Generation of airborne activity co-incident with Air-fed suit failure. Failures include: | Ref 1.2: 1.8  
Ref 1.3:6(18), 8(21)  
Ref 1.5: 2b-1.1-1.7. 2b-1.9-12  
21-1.1-1.2 |
| | • loss of air supply  
• air fed suit tears/damage | |
| FSG 0202 | Loss of Extract | Ref 1.2: 1.11, 1.12  
Ref 1.3: 1(3), 2(3)  
Ref 1.5:2b-1.11, 2b-4.1-4.2,  
2b-4.3-4.6, 2c-2.1-2.2  
2d-1.2, 2d-1.3 |
| FSG 0203 | Loss of Filtration | Ref 1.2: 1.11, 1.12  
Ref 1.3: 3(4)  
Ref 1.5: 2b-4.1, 2b-5.1, 2d-1.1 |
| FSG 0204 | Loss of containment during posting out | Ref 1.2: 1.15 |
| FSG 0205 | Fire in C5 area | Ref 1.3: 7(19)  
Ref 1.5:2b-1.8 |

Description: Fault Sequence Group 0202: Loss of Extract

The active ventilation system that provides extract to the Fuel Line is a Mobile Filtration Unit (MFU). The air that is extracted is subject to two stages of HEPA filtration (2 banks of primary and secondary filters) which is then discharged into the building extract system mixing chamber on the 1st floor. The air is then subject to further filtration by the building HEPA filters in the adjacent building prior to being discharged via a chimney.

The Building ventilation system is supplied with air by a duty and standby plenum fan. The system is extracted by a duty and stand by extract fan.
As failure of the duty fan is part of the initiating event of this fault. Failure of the duty fan would be detected by the fan failure detection system which alarms at the central control room in the facility, with repeater alarms in the building foyer and health physics and surveyor’s office.

Upon failure of the duty fan, a building evacuation is initiated. The air-fed suit worker and standby dresser (wearing a respirator) would remain in the building following a building evacuation. The building duly appointed person would also remain in the building wearing a respirator.

Identification of Initiating Events

There are therefore 2 main initiators for this Fault Sequence Group:

- Failure of the building extract and Failure of the MFU extract,
- Failure of the building extract when the MFU interlock both trips and failed is assessed below.

Consequences to the Public

If the MFU alone were to fail then the consequence to the public are bounded by the assessment in FSG 0203 (loss of filtration) or by FSG 0202a below.

FSG 0203 looks at extracting air from the active area into the building extract and out through the chimney stack without MFU filtration.

FSG 0202a calculates the consequences to the public from migration of activity from the active area outside the building at ground level.

Consequences to the Worker

The consequences to the worker are considered to be bounded by those assessed for failure of both the MFU and building extract.

The only exception to this is the dose calculated below to an operator in the operating area, where it is assumed that no motive force is present in that area and hence any activity will diffuse into the available volume. This scenario may not be the case if the building extract is available. However, as discussed above the building extract will provide some flow of air across the barrier and therefore the dose calculations made are considered to be adequately bounding of the dose to an operator in the operating area following loss of MFU extract but building extract still available.

Therefore, the following faults are considered, and consequences to the public and workforce are determined from failure of the building ventilation stand by fan or failure to start it:

- FSG 0202a – Failure of the building extract system, MFU interlock trips
- FSG 0202b – Failure of the building extract system, MFU interlock fails

Unmitigated Consequence to a Member of the Public

Unmitigated public dose as a result of inhalation is given by the following equation:

\[ Dose = \sum I \frac{RF}{DF} DPUR_{PubCat} \]

Where:
I : Inventory (Bq)

RF is the release fraction; this will vary depending on how the scenario is modelled, it is a property of the material and the method of release (e.g. spillage of powder, dropped filter)

DF is the decontamination factor, and a factor of 10 will be applied for each layer of containment (i.e. ISO container and drum) resulting in a combined DF of 100

DPUR_{PubCat} is the dose to a member of the Public per unit release (Sv/Bq or Sv/g)

Description Fault Sequence Group 0202b - Unmitigated consequences to public through building– MFU working, MFU HEPA filters failed

The worst case public consequences would occur if the MFU filters failed coincident with failure of the building extract and the failure of the interlock between the building ventilation system and the MFU system.

The consequences to the public via the chimney following loss of MFU filtration whilst the building extract is still operational is assessed in FSG 0203 below.

Airborne Activity Concentration

The airborne activity concentration within the immediate vicinity (approximately 8m$^3$) to where it is generated has been assumed to be 0.1 mg/m$^3$ and it is argued in the safety assessment that the airborne activity concentration within the rest of the air-fed suit area will be approximately 0.01 mg/m$^3$ at the time that the released activity fills the Fuel Line.

Inventory Available to be Released

The Fuel Line has a volume in excess of 2736m$^3$ and there will be approximately 0.028g Pu airborne within the area that is available to be released to the public.

\[
\text{Released Inventory} = \text{Concentration} \times \text{Available volume} = 0.1 \frac{mg}{m^3} \times 2763m^3 = 0.028g
\]

Decontamination (or Protection) Factor

A decontamination factor (DF) of 10 can be claimed for the residual DF provided by the filters and the effect of system knock out (Table 2.1, DF 2-8).

In addition, a DF of at least 10 can also be claimed for the building fabric of the building (SF) (Table 2.1, DF2-7).

Potential Public Dose Uptake from 1g of Pu (Calculated Previously)

It is known from completion of previous detailed safety assessments that the public dose from release of 1g of Magnox Plutonium, from a release height of <10m is 8.7 mSv.

Public Dose

Applying this information to the above DFs and pessimistically assuming that all of the activity is released, gives a public dose of:
The potential dose to a member of the public is therefore calculated to be of the order of 2.4 µSv.

**Comparison with Design Basis Accident Criteria**

Table 10: consequence thresholds exceed for FSG 0202

<table>
<thead>
<tr>
<th>Fault Sequence</th>
<th>Public</th>
<th>Worker</th>
</tr>
</thead>
<tbody>
<tr>
<td>0202a – (i) Worker in Air-Fed Suit</td>
<td>n/a</td>
<td>&lt; 2 mSv</td>
</tr>
<tr>
<td>0202a – (ii) Worker in Change Facility</td>
<td>n/a</td>
<td>2-20 mSv</td>
</tr>
<tr>
<td>0202a – (iii) Worker adjacent to Change Facility</td>
<td>n/a</td>
<td>&lt; 2 mSv</td>
</tr>
<tr>
<td>02a – Public</td>
<td>&lt; 1µSv</td>
<td>n/a</td>
</tr>
<tr>
<td>0202b – (i) Unmitigated, Worker</td>
<td>n/a</td>
<td>20-1000 mSv</td>
</tr>
<tr>
<td>0202b – (ii) Mitigated, Worker</td>
<td>n/a</td>
<td>&lt; 2 mSv</td>
</tr>
<tr>
<td>02b (iii) – Unmitigated Public, through building</td>
<td>1-100µSv</td>
<td>n/a</td>
</tr>
<tr>
<td>02b (iv) – Unmitigated Public, through hole in duct outside building</td>
<td>1-100µSv</td>
<td>n/a</td>
</tr>
</tbody>
</table>

**Comparison with Probabilistic Risk Criteria**

The unmitigated worker consequences of FSG 0202b have the potential to exceed 20 mSv, and therefore, this fault sequence requires an estimation of the top event frequency, as indicated below.

The initiating event is a combination of the frequency of fan failure of the building extract (estimated to be 1 yr⁻¹ (2.10)), the probability of loss of filtration and failure to evacuate.

From Reference 2.11 it can be seen that the estimated failure rate for a filter (for a normally dry system as in B277) is 1E-4/filter year. The MFU has both primary and secondary filters, with 2 of each so there are a total of 2 filters per stage.

However in identifying a failure rate for all filtration it is to be noted that a common mode factor needs to be taken into account.

Using a factor of 0.2, for duplicate systems, it can be seen that this will dominate the failure rate and will be in the order of 4E-5 y⁻¹.

The filters are tested once every year, therefore the probability of failure of cell vent filtration is
\[ 4 \times 10^{-5} \text{ yr}^{-1} \times \frac{1}{2} = 2 \times 10^{-5} \]

In order to exceed 20mSv the operator would have to fail to evacuate the area within 13 minutes.

In the event that this fault lead to an increase in activity in the operating area, then local activity in air alarms and the MFU alpha in duct alarm would alert the operator to this.

From the HAZAN references it can be seen that the probability of failing to evacuate within 10mins to 2 hours is estimated to be \(1 \times 10^{-2} \text{ yr}^{-1}\).

The sequence frequency is further reduced by the fact that it also requires failure of the building extract/MFU interlock which is un-quantified here.

Hence the actual sequence frequency is concluded to be \(<1 \times 10^{-7} \text{ yr}^{-1}\) and hence negligible compared with any relevant risk target applicable to the worker.
6.6 EXAMPLE: CONSEQUENCES TO A MEMBER OF THE PUBLIC

The following example illustrates the possible typical consequences to a member of the public for the Fault Sequence group associated with dismantling by use of Manual Grinding techniques.

The material which could be released during cutting operations is dependent upon the contamination level of the material to be cut, the type of cutting implement used, and the duration of the cutting operations.

The worst case contamination level at the point of cutting for gloveboxes to be dismantled in decommissioning phase 3B has been measured as 8.8E-3 g cm\(^{-2}\).

The release per unit length of material cut is known to be 200 Bq m\(^{-1}\)/Bq cm\(^{-2}\) for grinding. (This could also of course be expressed as grams rather than Bq when the contamination is a single element of constant isotopic distribution as in this case (Pu)).

The maximum cutting rates for grinding are 8cm/minute. The release rates per metre of cut and per minute of cutting can therefore be calculated:

\[ 200 \text{ g m}^{-1}/\text{g cm}^{-2} \times 8.8 \times 10^{-3} \text{ g cm}^{-2} = 1.76 \text{ g m}^{-1} \times 0.08 \text{ m min}^{-1} = 0.14 \text{ g min}^{-1} \]

The facility indicated that for manual grinding operations it would not be expected that a cut length of more than 20 cm would be undertaken in a single action (Operational Assumption). After this there would be a pause before the next cut.

This would give a cutting time of 20/8 = 2.5 minutes.

The activity released per cut would be 0.14 g min\(^{-1}\) x 2.5 min = 0.35g/cut.

Operators have also indicated that a total of no more than 60 minutes cutting time (Operating Instruction) could be achieved in two separate ‘Windscale’ Suit entries over a period of at least 3 hours in any operating day (Operational Assumption).

The total activity released in a day of grinding operations would therefore be

\[ 0.14 \text{ g min}^{-1} \times 60 \text{ min} = 8.4 \text{ g/day} \]

The lowest overall Decontamination Factor (DF) that will apply is 1E+3 for a stack release (assumed that the local TEDAK filters or extract and Mobile Filtration Unit and building filters have failed) and 1E+5 for a ground level release (TEDAK filters or extract and MFU filters and building extract failed).

The maximum activity release from a full day (60 minutes) of grinding operations is 8.4g

A release of 1g of Pu to the public at ground level has been previously calculated and is known to result in a potential public dose of 6.7mSv.

Based upon the above operational assumptions the dose resulting from a full day of grinding, assuming a ground level release with a DF of 1E+5 applied is therefore:

\[ = \text{Dose resulting from 1 gm release x (Amount released / DF)} \]

\[ 6.7 \text{mSv} \times 8.4/1E+5 = 6E-4 \text{ mSv} = 0.6 \mu\text{Sv} \]

No site consequence thresholds are breached.
If it is assumed that the same material is released from the associated 30 m chimney it would result in a potential dose of 1.6 mSv.

The consequences to a member of the public, assuming a DF of 1E+3 (assumes TEDAK extract failure or TEDAK filtration failure combined with MFU and building filtration failure) would therefore be:

| 1.6mSv x 8.4/1E+3 = 13.4 µSv |

The low consequence threshold could be breached, but not any consequence threshold requiring frequency assessment.

It is noted that any scenario which results in a DF above 1E+4 (i.e. all those which do not involve building filtration failure and MFU filtration failure) will result in no consequence thresholds being breached.

Thus the only scenarios which have the potential to breach consequence thresholds are TEDAK extract failure or TEDAK filtration failure combined with MFU and building filtration failure.
6.7 EXAMPLE: CONSEQUENCES TO AN ONSITE WORKER

The following example illustrates the typical consequences of dismantling operations to an Onsite worker during Phase 3.

Normal modelling of inhalation doses requires the radionuclide to be released in a known volume, which then becomes dispersed and the operative/exposed person inhales the material.

The model has a number of variables which include; amount of material released, volume into which material is dispersed, air flow/mixing rates (dispersion coefficient) and if the release is instantaneous/continuous.

These variables are normally explained using the following relationship

\[ Dose = \sum I \frac{RF}{DF} BE_{inh} C \]

Where

- \( I \) Inventory released (Bq)
- \( RF \) Release Fraction
- \( DF \) Decontamination Factor
- \( B \) Breathing rate of the worker, typically \( 3.3 \times 10^{-4} \text{ m}^3 \text{s}^{-1} \)
- \( E_{inh} \) Is the radio-toxicity of the nuclide (Sv/Bq)
- \( C \) Is the dispersion coefficient, which varies according to whether the release is indoors. Outdoors, rate of forced air changes and distance of exposed person from the point of release (m^3/s)

Fault scenario modelling

A technique which was considered appropriate for this scenario where high air change velocities apply was to model the release as instantaneous into an 8 m^3 breathing zone (broadly equivalent to the volume of a hemisphere of 1.5 m radius around the operator) and to then assume that the operator is exposed to it for a maximum period equivalent to the average residence time of the activity in his breathing zone, taking account of the velocity past him induced by the extract.

This approach also has the benefit of being relatively insensitive to operator position relative to the fume/extract orientation. It parallels the methodology normally used for releases outside.

It is to be noted though that in practice doses will be lower when the operator is not between the fume generation point and the extract point, and as a matter of ALARP it is an operational assumption that this will be the normal practice wherever possible.

It is assumed within the Hazard Analysis that the available activity released during cutting operations is dispersed into an enclosed space which is the containment unit in which the operator is working.
**Assumptions Made in the Model:**

Operators working inside the active containment wear ‘Windscale’ Suits – self contained breathing air suits (operating instruction).

An undamaged ‘Windscale Suit’ with a healthy air supply will provide a DF of at least 5E+4.

In the event that the suit is failed (e.g. torn) a DF of 1E+3 applies.

If the air supply fails, a DF of 1E+2 applies.

Significant Windscale Suit faults are clearly revealed to the operator, who is well trained in how to respond (operational assumption) and it is expected that cutting will stop almost immediately.

It is pessimistically assumed however that the suit fault develops toward the end of a maximum length (20cm) cut taking 2.5 minutes.

The amount of Pu made airborne by this cut has been previously assessed to be 0.35g.

The 20 second dose to an operator whose Windscale Suit is torn will therefore be:

\[
5.5E+2 \text{ Sv} \times 0.35 / (12.5 \times 1E+3) = 1.5E-2 \text{ Sv} = 15 \text{ mSv}.
\]

The above consequence also includes the assumptions that:

- the local extract provides an overall DF of 12.5 on activity reaching the operator;
- The 20 second dose to an operator from a release of 1g Pu has been previously calculated to be 5.53E+2Sv;
- A release fraction for a torn suit applies (1E+3)
## 6.8 COMPARISON WITH RADIOLOGICAL RISK AND FREQUENCY TARGETS

Table 11: Comparison of the frequency of the fault sequences with risk and frequency targets

<table>
<thead>
<tr>
<th>Fault Sequence Number</th>
<th>ON-SITE</th>
<th>Aerial</th>
<th>OFF-SITE</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Frequency of receiving a Whole Body Dose ($y^{-1}$)</td>
<td>Time Averaged Critical Group Dose ($\mu Sv \cdot y^{-1}$)</td>
<td>Frequency of receiving a dose ($y^{-1}$)</td>
<td>Frequency of Large Societal Consequences ($y^{-1}$)</td>
</tr>
<tr>
<td></td>
<td>20-1000 mSv</td>
<td>&gt;1000 mSv</td>
<td>0.1-1 mSv</td>
<td>1-10 mSv</td>
</tr>
<tr>
<td>FSG0202</td>
<td>1E-7</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>FSG0204</td>
<td>3.1E-8</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td><strong>TOTAL</strong></td>
<td><strong>1.3E-7</strong></td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Target</td>
<td>$10^3$</td>
<td>$10^5$</td>
<td>4</td>
<td>$10^2$</td>
</tr>
<tr>
<td>% of Target</td>
<td>0.01%</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
</tbody>
</table>

Note (1): Targets from Site Licence Accident Risk Criteria. Targets shown are per building (on-site criteria), or per plant concept (off-site criteria).

Note (2): Doses due to aerial releases are the sum of inhalation/direct radiation and ingestion doses, with the last assumed to be limited to 5 mSv as a result of imposed countermeasures.


Note (4): Non-contributory fault sequences: 0408, 0410 (list of appropriate Fault Sequence Numbers)
7 APPENDIX C: EXAMPLES FOR IMPLEMENTATION OF SAFETY ASSESSMENT RESULTS

7.1 WASTE MANAGEMENT ISSUES FOR PHASE 3 DECOMMISSIONING

There is an accepted Waste Management Hierarchy (WMH) that can be used to define the preferred option for management of radioactive wastes, and this is listed below:

1. Avoid;
2. Reduce;
3. Re-use;
4. Recycle;
5. Dispose.

The WMH is a way of describing the options for dealing with waste using an order of desirability; with the prevention and minimisation of waste as the most, and disposal as the least favourable options. As the plant and equipment already exist and the work is being carried out to reduce the radiological hazard of the facility, then generation of PCM waste is inevitable. However, minimisation of the production of secondary wastes is to be a major consideration during project – such as the reuse of the equipment when possible. Also, by removing the equipment from packaging before taking it into active areas and taking the minimum of equipment into the C5 area therefore minimising the amount, which will be removed as PCM as opposed to LLW.

This facility has been identified as an interim waste storage and transit area for other alpha contaminated waste that has been generated by other onsite decommissioning projects. The alpha waste received into the facility has to be be compliant with the generic safety assessment and the Conditions for Acceptance.

This enables the early start of decommissioning and allows multi-facility site to reduce the storage locations required. For the purposes of this test case the term waste management comprises the steps necessary for:

- Conditions for acceptance;
- Releasing the material from radiological control, including characterisation, segregation, decontamination and clearance levels;
- Preparing the material for final disposal, including conditioning and packaging of the wastes in accordance with the acceptance criteria.

Waste acceptance criteria have been generated for all the waste routes, including:

- Plutonium Contaminated Material (PCM);
- Low Level Waste (LLW);
- ‘Free release’ waste.

In addition it is noted that UK Department of Transport Regulations (offsite) also form conditions for this test case, but these are not examined in this document. Steps for the treatment of radioactive wastes include collection, sorting, segmenting, compaction, decontamination, drying and packaging have been assessed and described within the overarching safety assessment. These waste treatment
techniques are all available on the site, and a low level waste repository on a different site nearby.
7.2 EXTRACT FROM FACILITY WORKING INSTRUCTION

Operational Clearance Certificate B277/OCC/001 contains the following requirements:

Operating instructions
8187/OI/05 – On failure of the Tedac extract, cutting operations will stop and Operators will evacuate from the Tedac enclosure.
8187/OI/11 – On revealed high loading of the Tedak sock filters cutting operations must cease.
8187/OI/12 – On revealed high loading of the Tedak HEPA filter cutting operations must cease.

Safety Mechanisms
8187/SM/01 – Flow and pressure/DP alarms on the Tedak system.
8187/SM/06 – Interlock between grinder/plasma and Tedak system.
8187/SM/07 – Tedak extract system.
8187/SM/08 – Tedak filtration system (cyclone, sock filters and HEPA).
8187/SM/09 – Tedak sock filter DP alarm.
8187/SM/10 – Tedak HEPA filter DP alarm.

Safety Related Equipment
None.

Safety Memorandum
None.

Compliance Record Sheet (CRS)
Cat B Record Sheet.

Industrial safety requirements and other related information

Plant Modification Proposal (PMP)
None.

Personal Protective Equipment (PPE)
Air fed suit, steel toe cap Wellingtons, three pairs of Surgeons gloves, PVC oversuit and oversuit protection as identified in B277/LR/04.
Heavy duty cat 2 gloves (HANDKEVG 12 or REFLEXK+8704X1) when emptying cyclone bucket.

Risk Assessment/Work Safety Plan

- This operation has been risk assessed and the findings incorporated into this instruction.
## Section 1: Safety Mechanisms (SM) continued......

<table>
<thead>
<tr>
<th>System / Item Tag Number</th>
<th>Description</th>
<th>Description / Function</th>
<th>EIM&amp;T Activity</th>
<th>Supporting Documentation</th>
</tr>
</thead>
<tbody>
<tr>
<td>B277/TEDAK/3</td>
<td>TEDAK EXTRACT SYSTEM</td>
<td>To provide a minimum DF of 12.5 between activity releases at the cutting zone and the air-fed suit operators realthing zone &amp; to provide a min DF of 1E3 on activity released across the Tedak curtain.</td>
<td>WR&amp;D/B277/PT/019</td>
<td>12 MONTHS</td>
</tr>
<tr>
<td>B277/TEDAK/FILT</td>
<td>TEDAK FILTERS</td>
<td>To provide a min DF of 1E6 on air discharged from the Tedak</td>
<td>VTSG/HEPA/001</td>
<td>6 MONTHS</td>
</tr>
<tr>
<td>B277/MFU/5</td>
<td>MFU HEPA FILTER</td>
<td>To provide a min DF of 1E5 on air discharged into PAC.</td>
<td>VTSG/HEPA/001</td>
<td>3 MONTHS</td>
</tr>
<tr>
<td>B277/MFU/3</td>
<td>MFU EXTRACT FAN</td>
<td>To support a min DF of 1E5 across air-fed suit containment. (1E3 if one filter failed).</td>
<td>B277/PFR/OI/26</td>
<td>12 MONTHS</td>
</tr>
<tr>
<td>B277/MFU/2</td>
<td>B277 MFU LOW FLOW ALARM</td>
<td>To reveal failure of MFU extract or low flow.</td>
<td>WR&amp;D/B277/PT/009</td>
<td>12 MONTHS</td>
</tr>
</tbody>
</table>