IAEA-TECDOC-1644

BOSS: Borehole Disposal of Disused Sealed Sources

A Technical Manual

IAEA
International Atomic Energy Agency

BOSS
BOREHOLE DISPOSAL OF DISUSED SEALED SOURCES: A TECHNICAL MANUAL

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FOREWORD

In response to requests from IAEA Member States involved in the regional technical cooperation project RAF/4/015 (AFRA), the Nuclear Energy Corporation of South Africa (Necsa) has developed a system, under IAEA supervision, known as the borehole disposal facility (BDF) for the disposal of disused sealed radioactive sources in boreholes. The BDF is a final step of a technological process called the borehole disposal of radioactive sources (BOSS). It also includes pre-disposal activities, in particular characterization and conditioning of disused sources, including high activity sources. The BDF has been designed to provide a safe, economic, simple and permanent solution for the long term management of disused sealed radioactive sources (DSRSs). Owing to its relative simplicity, it can be implemented in countries that own the sources but do not have the infrastructure needed to make them permanently safe.

In 2005, the IAEA convened an international review to assess the status of the BDF concept within the Waste Management Assessment and Technology Review Programme (WATRP). The WATRP team reviewed the documents produced by Necsa and the South African National Nuclear Regulator (NNR), saw a non-active demonstration of the disposal method, and made recommendations for further development and implementation of the BDF. The review team confirmed that the IAEA–Necsa BDF is a safe, economic, practical and permanent means of disposing of DSRSs. Accordingly, the BDF should be considered a viable waste management option for present day management of these sources.

To enhance acceptance of borehole disposal in countries lacking the nuclear infrastructure for DSRS management the WATRP team recommended the creation of an information package. The package should: (i) inform decision makers; (ii) describe the BDF to organizations that might wish to implement it; and (iii) assist regulators and implementers by describing the contents of a generic safety report. The current publication covers the second objective.

The IAEA wishes to express its appreciation to the external consultants involved in compiling and assessing the documents, in particular I. Crossland (United Kingdom) who had the leading role in drafting and revising the text.

The IAEA officer responsible for this publication was L. Nachmilner of the Division of Nuclear Fuel Cycle and Waste Technology.

EDITORIAL NOTE

CONTENTS

PREFACE

The BOSS (borehole disposal of disused sealed radioactive sources) system is a detailed, engineering level system which allows the safe and permanent disposal of disused sealed radioactive sources (DSRSs) in specially created boreholes. BOSS has been developed as part of the IAEA/AFRA project and is intended for use in countries that own DSRSs (e.g. redundant medical therapy sources), but do not have the necessary infrastructure to manage them.

This publication is the second of four IAEA reports (see diagram below) that describe the BOSS system:

- Overview — explains what BOSS is and what it involves. It is written for decision makers (for example, government departments) who will need to know what it is that they are being asked to accept. A short flyer provides a high level summary of the overview.
- Technical manual — explains how to implement BOSS on a site specific basis. This publication is written for would-be implementers of BOSS and their regulatory body.
- Project management plan — presents a template and an outline of a report that describes how the project is to be organized and monitored. This document can be found in Appendix VI of the technical manual. It is written primarily for would-be implementers of BOSS, though others may find it helpful to see how such a project will be managed.
- Specific Safety Gguide — provides guidance on the design, construction, operation and closure of borehole disposal facilities for the disposal of radioactive waste in accordance with the relevant safety requirements. This guide is written for regulatory bodies and would-be implementers.

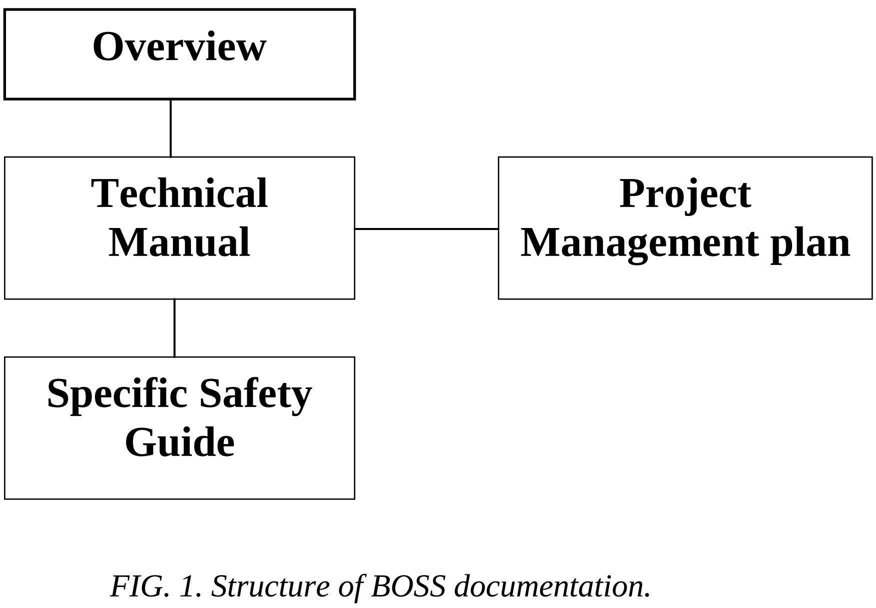

FIG. 1. Structure of BOSS documentation.

SUMMARY

The IAEA BOSS system aims to provide for the long term management of disused sealed radioactive sources (DSRSs) through:

- Collection, characterization and conditioning of disused SRS in stainless steel capsules, a process that makes it easier and safer to transport and store the SRS;
- Safe interim storage of the DSRS;
- Selection of a suitable disposal site;
- Containerization of the capsules into a disposal package consisting of a robust, stainless steel container with an concrete inner lining;
- For more radioactive sources (e.g. spent high activity radioactive sources known as SHARS), containerization is performed inside a specially designed portable hot cell;
- Permanent disposal of the containerized DSRS, including SHARS, in a specially constructed borehole that is at least 30 m deep.

This publication, the second of three describing the BOSS system, is a technical manual for regulatory bodies and implementers. It presents a comprehensive description of all aspects of the BOSS system, including:

- Preparatory work such as compilation of an inventory of sources and allocation of responsibilities to appropriate bodies;
- Source collection, characterization, conditioning and interim storage;
- Selection, characterization and recommendation of a suitable disposal site;
- Safety assessment for operation, transport and post-closure;
- Procedures for constructing, operating and closing a BOSS disposal borehole;
- An outline of the key regulatory requirements.

The scope of the BOSS system is illustrated in Fig. 2.

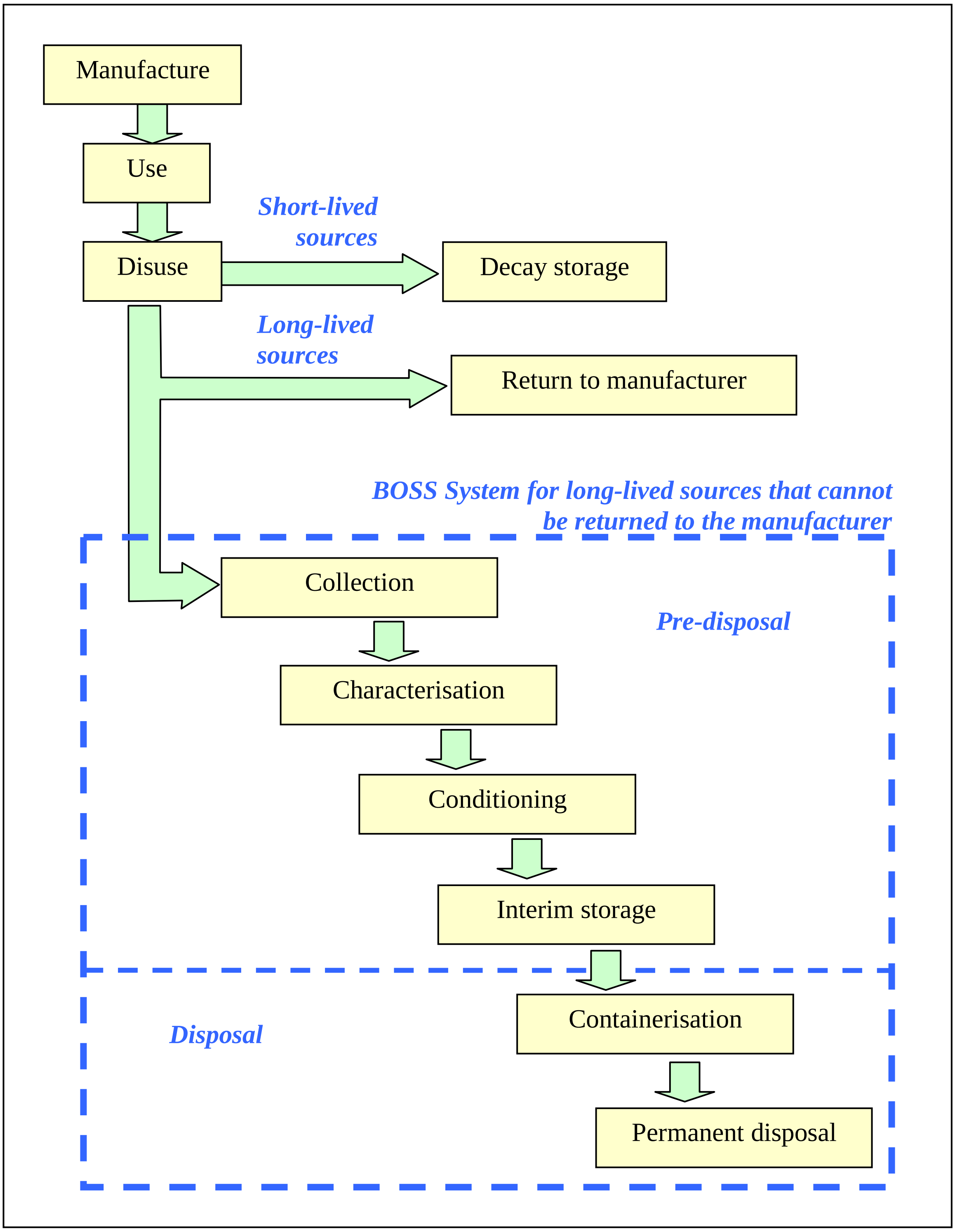

FIG. 2. Place of the BOSS system in the life cycle of sealed sources.

1. INTRODUCTION

1.1. THE TECHNICAL MANUAL

This publication is the second of four describing the BOSS system:

- Overview;
- Technical manual;
- Project management plan;
- Specific safety guide.

The purpose of the technical manual is to explain how the BOSS system may be implemented on a site specific basis. The technical manual provides much of the detailed technical information that will be required to do this. Where greater detail is needed, it refers to other documents; principally, the suite of technical reports produced by Necsa of South Africa (Appendix I). Note also that the third of the reports listed above, the project management plan, can be found in Appendix VI.

The target audience for the technical manual consists of would-be implementers of BOSS and their regulatory bodies.

1.2. BACKGROUND: USE OF SEALED RADIOACTIVE SOURCES

Since the discovery of radioactivity in 1896, radioactive materials have been put to many peaceful uses. It is estimated [1] that at least 10 million sealed radioactive sources have been produced worldwide: currently, 2 million sealed sources are in use in the USA alone. Sealed radioactive sources (SRS) are widely used in medicine, industry, agriculture, education and research. In medicine, for instance, SRS are primarily used for the treatment of cancer. In industry, SRS are needed for the non-destructive testing of welds, which is essential to ensuring the safety of pressure vessels, pipelines and even airplanes. Hundreds of other applications could be listed including smoke detectors, lightning conductors, food and instrument sterilization, and oil and mineral exploration.

Strict safety standards aim to prevent the sources being a danger to the workers who use them or to the general public and the IAEA has recently devised a categorization system for SRS [2] that is based on the level of danger that they represent (see Box 1). This categorization is primarily concerned with the immediate potential hazard that is presented by a source. As noted in Box 1, different considerations come into play when disposal is to be considered.

Eventually, all these sources come to the end of their useful life. This may be because a source is no longer radioactive enough for its original use or it may be that the equipment in which the source is housed falls out of use for some other reason. Typically, this may be because the equipment is faulty or has been superseded, or that the company using the source has gone out of business. Whatever the reason, the SRS becomes disused. This does not mean that the source has lost all its power: it is still potentially hazardous to people — as is evident from a number of incidents and fatalities [3] — and therefore, requires careful management.

Box 1. Categorization of sources and classification of waste

Source categorization aims to provide a simple, logical system for ranking SRS based on their potential to cause harm to human health. Five categories are defined by IAEA-TECDOC-1344 where Category 1 is the most dangerous and Category 5 is the least dangerous.

Description of the five categories, as in IAEA-TECDOC-1344, with examples of SRS in the various categories:

- Category 1: personally extremely dangerous — RTGs, teletherapy irradiators;
- Category 2: personally very dangerous — industrial radiography, brachytherapy;
- Category 3: personally dangerous — fixed industrial gauges, well logging;
- Category 4: unlikely to be dangerous — bone densitometry, level gauges;
- Category 5: not dangerous — permanent implant sources, lightning conductors.

SHARS (spent high activity radioactive sources) are not specifically defined by IAEA-TECDOC-1344 but they broadly correspond to Category 2 and the lower half of the activity range for Category 1. The most active Category 1 sources (>PBq quantities) are either radioisotopic thermoelectric generators (RTG) or food sterilizers, both of which are small in number and confined to a few developed Member States. Where necessary, the IAEA is making separate arrangements for their recovery.

Categorization is very helpful in defining the potential danger of a sealed radioactive source if, for example, it were to go astray. Categorization in the sense of IAEA-TECDOC-1344 is less helpful if, say, one is trying to decide the disposal route for a disused SRS. Here the main determinants are source strength and half-life.

In general, disused SRS will be suitable for decay storage if their half-life is less than one year. SRS with longer half-lives than this will usually need some form of regulated disposal.

1.3. WHAT HAPPENS WHEN SRS BECOME DISUSED?

Where an SRS is owned by a government backed organization, resources will usually be available for the management of the disused SRS. That said, there is always a possibility, however remote, that government backed control could lapse. The situation tends to be worse in the private sector, especially if regulation is lax or absent. This is simply because, for a private company, disused sources represent a liability and a drain on resources. Consequently, unscrupulous owners may be tempted to dispose of the source illicitly. Similarly, if a business is liquidated, an SRS may not be recognized for what it is and control may be lost. In all these circumstances, an SRS can be lost, stolen, vandalized and, more generally, orphaned i.e. be left with no owner with the capability to maintain safety and security. Concern over these matters is evidenced by the recent International Conference on the Safety and Security of Radioactive Sources [4] and the introduction by the IAEA of a Code of Conduct for the Safety and Security of Radioactive Sources [5].

The Code of Conduct makes recommendations regarding the safety and security of SRS with the aim of protecting individuals, society and the environment. The Code recommends, for instance, that States should have in place an effective national legislative and regulatory system of control over the management and protection of radioactive sources. This requires the establishment of a national register of radioactive sources and a regulatory body with powers to control the use of radioactive sources through the issuing of approvals. As a result of this strengthening of control, most countries now require an approval to be issued before a radioactive source can be imported. Further, before allowing importation, it is usual for the regulatory body to insist that, for all but the shortest lived sealed sources (whose radioactivity level will be negligible after a few years) the manufacturer should agree to take back the source at the end of its useful life.

As a result of these measures, it is anticipated that for newly acquired SRS the problem of orphan sources will be virtually eliminated. Unfortunately, however, there are many old SRS that cannot be returned to their manufacturer. There are several possible reasons for this: the sources may be too expensive to ship, the manufacturer may not be traceable, the special form certification is lost etc. As a result, many countries have a legacy of disused SRS in storage.

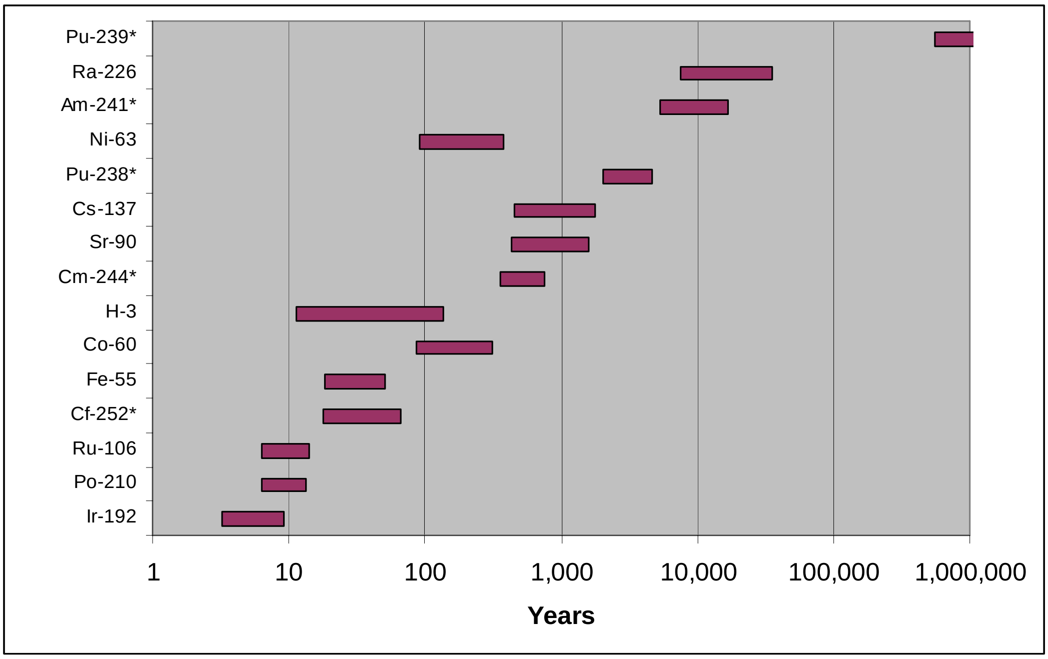

FIG. 3. Time required for a sealed source to decay to exemption levels. Radionuclides marked with an asterisk have radioactive progeny that are more long lived than the parent.

When an SRS has a short half-life, storage for a few years will be enough for the radioactivity to decay to safe levels. Figure 3 illustrates this for a range of common radionuclides. It shows the length of time needed for commonly used radionuclides to decay to exemption levels (the level at which, for the purpose of regulation, the material is no longer considered to be radioactive). The ranges correspond to the maximum and minimum radioactivity levels normally found for each radionuclide. In the case of Ir-192, for example, it is clear that less than ten years in storage will be sufficient to allow even the most radioactive SRS to decay to negligible levels. But for Ra-226, which until recently was widely used, the source can remain potentially dangerous for tens of thousands of years. Note that the figure (based on data given in Ref. [6]) does not allow for the ingrowths of radioactive progeny that are longer lived than the parent. Consequently, for those radionuclides that are marked with an asterisk the time ranges shown will be underestimates.

If a disused SRS cannot be returned to its manufacturer and if it is too long lived for decay storage, this immediately presents a difficulty. This is because the achievement of adequate standards of safety and security in the storage of long lived SRS (such as Ra-226) implies a duty of care that will persist for thousands of years. Even in the most organized societies, there can be little confidence that such long term storage is achievable — the history of human society is not one of continuous progress and there is no guarantee that future generations will have either the knowledge or the resources for the task. Further, even today, storage facilities in some countries are poor and the rise of global terrorism heightens the need to find some way of permanently placing these SRS out of harm's way.

1.4. HISTORY OF THE DEVELOPMENT OF BOSS

The BOSS system has been developed as a potential solution to the problem of disused SRS that are too long lived for decay storage and yet cannot be returned to their manufacturer. The BOSS system draws on and integrates a number of IAEA projects that have been developed through the IAEA/AFRA project (see Box 2).

The first of these projects has been operational since the early 1990s; it is concerned with the conditioning of disused radium sources in various African countries. Conditioning is a process in which SRS are sealed inside robust stainless steel capsules. This has the immediate benefit of containing the radioactive material — even if the SRS itself is leaking — and providing a fixed geometry that is convenient for subsequent handling, transport and storage.

Recognizing that interim storage is not a permanent solution for long lived SRS (such as the radium sources just mentioned) IAEA/AFRA, next commissioned the development of the borehole disposal facility (BDF) [7, 8]. This provides a specially designed and constructed borehole disposal facility where SRS can be disposed permanently, economically and, above all, safely. The BDF is designed so that it can be deployed in a wide range of geological and climatic environments. In the BDF, containerizing of the SRS for disposal (i.e. the production of a disposal package) uses a similar system to that used in the conditioning of radium sources.

Both these projects place limits on the activity of the SRS that can be handled. This is because the radiation shielding used in the conditioning/containerizing unit is fairly light. Consequently, because of the need to protect the operators, both systems are only capable of dealing with SRS of (roughly) Category 3 and lower. To facilitate conditioning of more powerful SRS, therefore, a third IAEA/AFRA project was established: the SHARS (spent high activity radioactive sources) project. This third project entails the design and construction of a portable remote handling device (generally called the BOSS hot cell) that extends the range of SRS that can be conditioned up to Category 1.

A peer review of the BDF was held in 2005. This recommended that the BDF should be made compatible with the SHARS project so that it too could deal with SHARS; this suggestion was taken up and the BOSS hot cell was developed to be part of the overall system. Some work remains to be done however, notably a system for remote welding and leak testing of disposal containers and their subsequent loading into a borehole.

Box 2. AFRA regional cooperation project — Overview

AFRA is an IAEA technical cooperation programme that started in 1990. It provides a framework for African Member States to collaborate in projects that aim to meet the shared needs of the members. In the field of radioactive waste management, projects have focussed on the safety and security of disused sealed sources through improvements in the regulatory and waste management infrastructures. The work has included the collection, transport, conditioning and storage of disused sources, the development of the borehole disposal concept and the development of a portable hot cell that will allow the handling of spent high activity radioactive sources (SHARS). These three projects provide key inputs to the BOSS system.

The AFRA Member States are Algeria, Angola, Benin, Botswana, Burkina Faso, Cameroon, Côte d'Ivoire, Democratic Republic of the Congo, Egypt, Eritrea, Ethiopia, Gabon, Ghana, Kenya, Libya, Madagascar, Mali, Mauritius, Morocco, Namibia, Niger, Nigeria, Senegal, Sierra Leone, South Africa, Sudan, Tunisia, United Republic of Tanzania, Uganda, Zambia, Zimbabwe.

Two smaller projects should also be mentioned. The first is the development and manufacture of a shielded device for storing conditioned SRS. The second is the design of a transport/transfer flask for moving SRS between the store and the disposal site or between the conditioning/containerizing unit and the borehole. At present this exists as a design only.

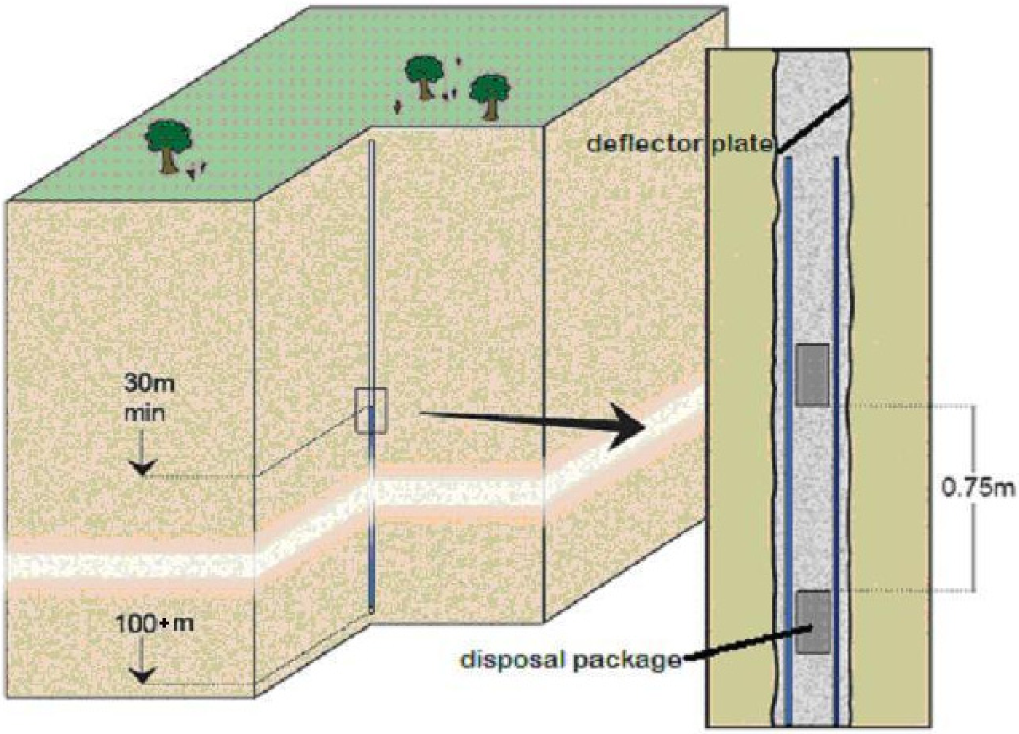

FIG. 4. Scheme of a borehole disposal facility.

The BOSS system is the integrated output of all of these projects. It aims to supply a comprehensive, fully engineered solution for dealing with almost all types of disused SRS. Box 3 contains a list of terms that are commonly used in the BOSS system.

Box 3. Definition of main BOSS terms

Anti-intrusion plate: a thick steel plate placed above the disposal zone to prevent subsequent drilling into the borehole.

Borehole sealing: the closing of a disposal borehole with a thick layer of concrete above the disposal zone.

Capsule: a 3 mm thick 316 L stainless steel container into which one or more SRS are placed. The capsule is closed by seal welding the lid into place so converting the SRS into “special form radioactive material” as defined by IAEA Transport Regulations.

Characterisation (of sources): the determination, by direct measurement, of the power (amount of radioactivity), nature (type of radionuclide) and condition (whether leaking or intact) of an SRS.

Concrete buffer: the pre-cast concrete lining on the inside of the disposal container.

Conditioning: the sealing of an SRS into a capsule. Conditioning is usually followed by interim storage.

Conditioning unit: a small building where conditioning and containerization of lower power SRS may be performed.

Containerization: the placing of a capsule into a disposal container followed by seal welding of the lid.

Disposal borehole: a borehole that is specially located, drilled and constructed for the disposal of disused SRS and other small volume wastes.

Disposal container: a 6 mm thick 316 L stainless steel container that completes the disposal package.

Disposal package: the combination of capsule, buffer concrete and disposal container that allows disused SRS to be disposed.

Disposal zone: the part of the borehole where the disposal packages are located.

Emplacement: the task of placing the disposal packages, surrounded by concrete, into the disposal borehole.

Portable hot cell: a shielded facility where SHARS may be handled, conditioned, containerized, etc. The portable hot cell is designed to be easily disassembled, transported and re-assembled.

RPO: radiation protection officer.

SHARS: spent high activity radioactive source. Typically, SHARS fall into Categories 1 and 2 of the IAEA sealed source categorization system (see Box 1).

SRS: sealed radioactive source.

Transfer/transport flask: a Type A transport container designed for the public domain transport of conditioned SRS and for transferring disposal packages to the disposal borehole. The transfer/transport flask exists only as a design at present.

1.5. BOSS SAFETY FEATURES

The whole purpose of the BOSS system is to improve the safety and security of SRS. It incorporates many safety features of which the most important are:

- Robust, fully welded stainless steel capsules into which the SRS are placed. The capsule prevents radioactive material from escaping while the SRS are stored or transported.
- Portable handling facilities (conditioning unit and BOSS hot cell) where the SRS can be safely manipulated.
- Purpose built transport packages so that SRS may be safety transported from conditioning unit to storage site to disposal site.
- Thick walled, fully welded stainless steel disposal containers into which the capsules are placed.
- A specially constructed disposal borehole where the disposal containers are placed — always at least 30 metres below the ground — and surrounded by concrete. The host rocks for the borehole are chosen so that they will preserve the integrity of the disposal containers for tens of thousands of years — enough to allow the radioactivity to decay to negligible levels.

1.6. STRUCTURE OF THE TECHNICAL MANUAL

The major part of this technical manual — Sections 3 to 5 — consists of a description of how the BOSS system may be implemented through a series of phases and activities. Important inputs to any radioactive waste disposal are the decisions by the government and regulatory body that initiate and control the repository development. If the implementation is to be described, these decisions, and how they are reached, also need to be included. To do this, it is necessary to make some assumptions about (a) the nature of these decisions, (b) who takes them and (c) their timing with respect to other programme activities. In this technical manual, these assumptions are expressed in terms of the BOSS road map. This is presented in Section 2. Then, using the BOSS road map as a template, Sections 3 to 5 describe how the BOSS system may be implemented. Finally, Sections 6 and 7 deal with management systems and training, respectively.

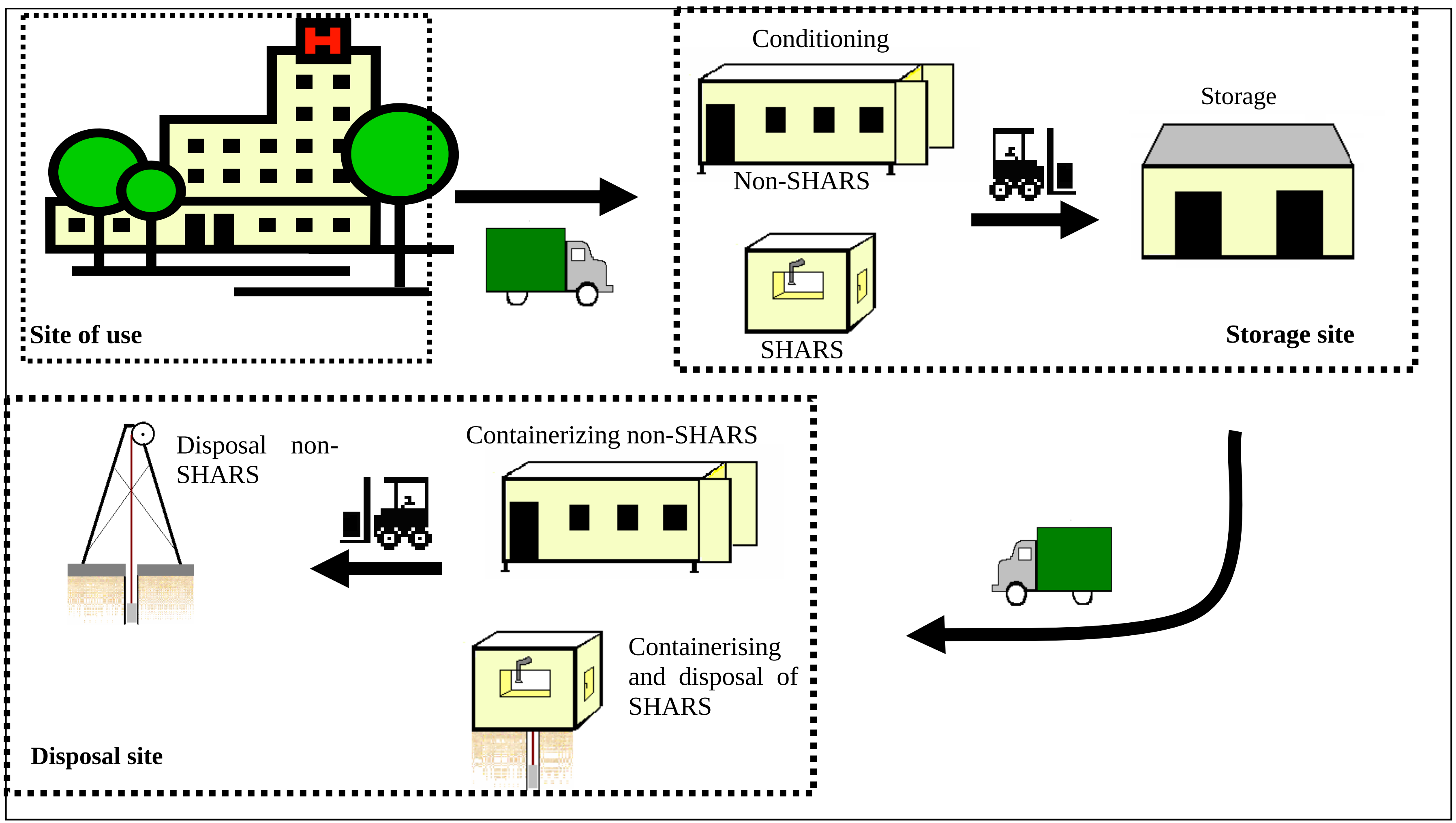

FIG. 4. Transports and operations of the BOSS system.

2. BOSS ROAD MAP

The IAEA requirements for geological disposal [9] and the corresponding safety guide for borehole disposal [10] both call for a step by step process in which a repository development moves forward in stages, each stage being separated from the next by a governmental or regulatory decision point. For a large development, four or five such decisions may be taken during a 30 year project to develop a repository for spent nuclear fuel. The BOSS system is many orders of magnitude smaller than these large scale projects in terms of cost and the amount of radioactivity being disposed of. Furthermore, a BOSS implementation might only take a few years. Key questions, then, for the implementation of the BOSS system are “how many decision points?” and “where do they occur?” The need is for an arrangement that gives reasonable assurance that the programme is being carried out conscientiously and competently but, at the same time, does not over-regulate, which would increase costs and timescales without appreciably improving safety. The BOSS road map (Fig. 5) provides one suggestion. The road map shows the disposal programme as occurring in three main stages separated by two major decision points.

This scheme is based on the view that, in a small scale disposal project, there are two key decisions. The first decision confirms that a BOSS type disposal is an appropriate solution by giving permission for the pre-disposal work to begin. The second decision approves the site selection and the site specific design and allows the disposal itself to go ahead. An important consideration in developing the scheme was the disruption to operations at the borehole and the increased costs that would be introduced by separate approvals for construction, operation and closure. In lieu of separate approvals, Fig. 5 suggests that the regulatory body would have a presence on site during safety sensitive activities.

An inbuilt assumption of the BOSS road map is that the number of boreholes to be constructed, operated and closed is sufficiently small for the work to be completed in a single campaign. A larger scale programme might require the disposal site to remain operational for an extended period (i.e. several years). In such a case, it may be convenient to place the final decommissioning and closure of the site (as opposed to closure of the boreholes) into a separate step that requires a separate approval.

Ultimately, regardless of the assumptions, Fig. 5 presents just one view of how a borehole disposal programme might be implemented. It is not prescriptive: other schemes (see for example Ref. [11]) could be devised that would work just as well or, by adapting to local circumstances and values, might work even better.

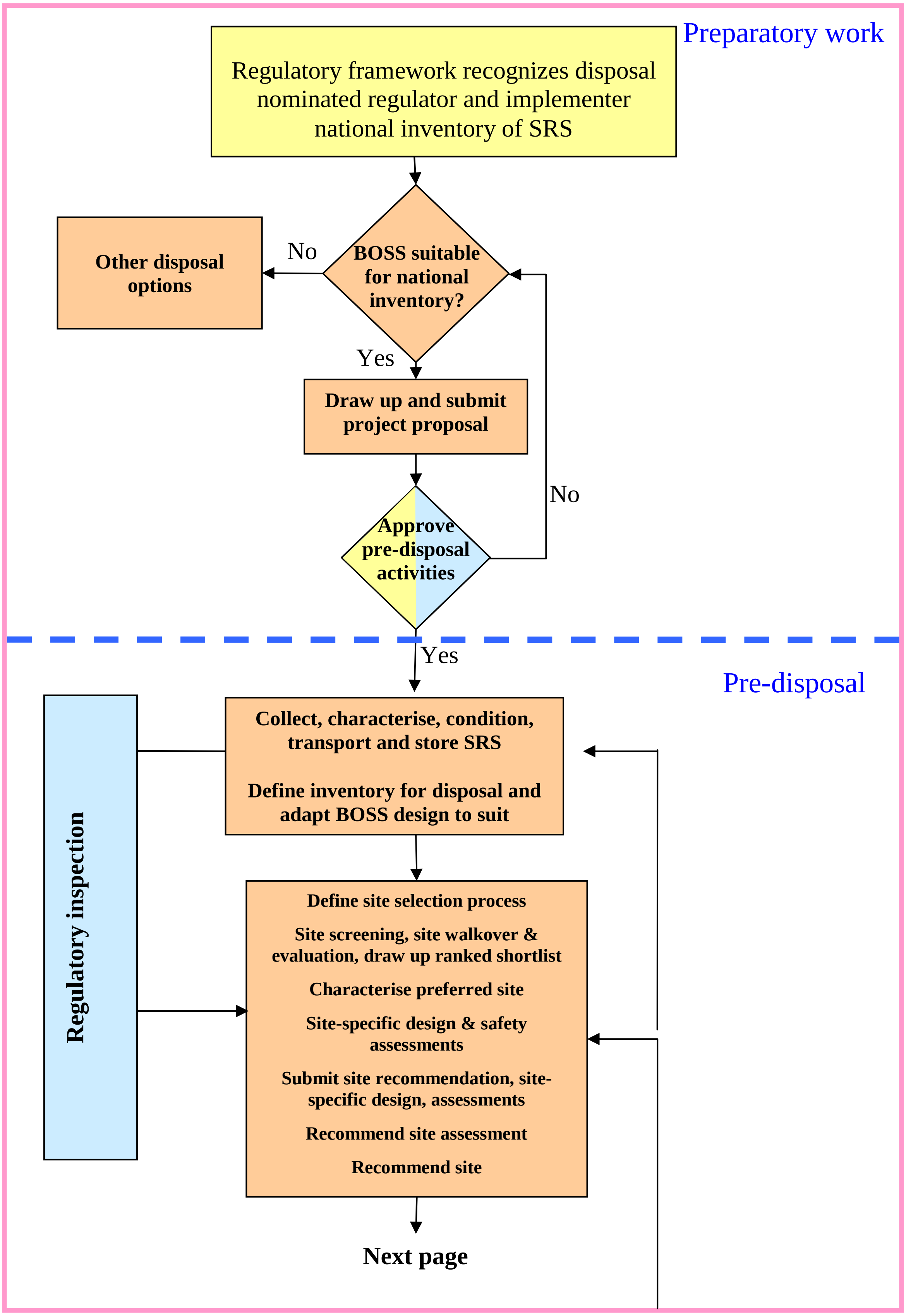

FIG. 5. BOSS road map.

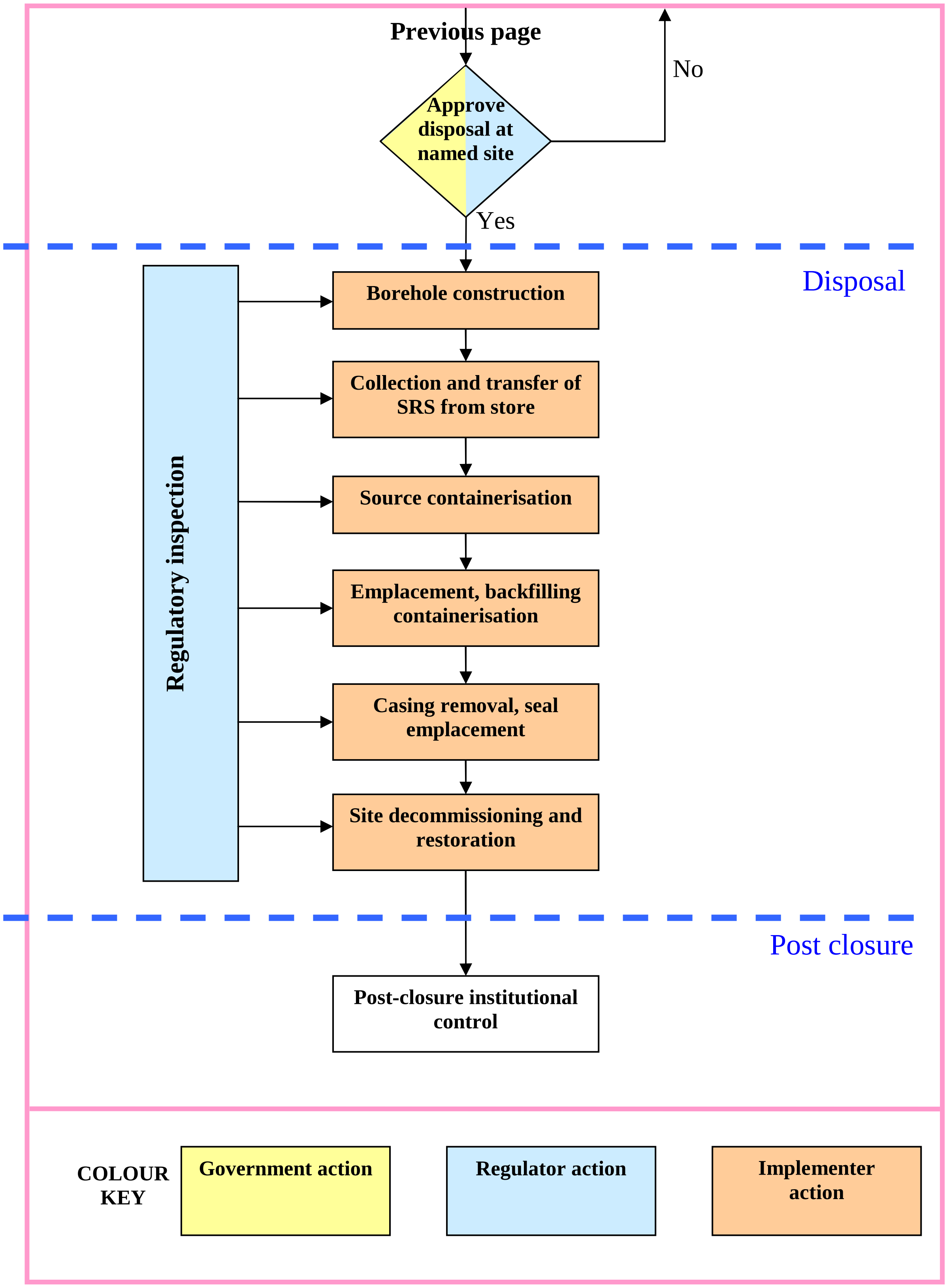

FIG. 5. BOSS road map (cont).

3. PREPARATORY WORK

It is envisaged that IAEA Member States will wish to implement BOSS in collaboration with the IAEA. In these circumstances, the main actors in the implementation of BOSS — government, regulatory body, implementer and supporting contractors — may all be entitled to technical assistance from the IAEA and other international bodies. This could, for instance, consist of training courses, workshops, visits to facilities, and access to the BOSS documentation. Eligibility for technical assistance and training will be decided on a case-by-case basis. IAEA will, in particular, seek to assure itself that:

- A suitable legal and regulatory framework is in place;
- Responsibilities for regulation and implementation of BOSS have been allocated to appropriately competent bodies;
- A national inventory of radioactive waste (including disused SRS) has been compiled;
- There is a clear commitment from national authorities to support the siting, approval and implementation of BOSS through an appropriate national radioactive waste management policy or strategy.

These four requirements may be seen as prerequisites for the involvement of IAEA in the deployment of the BOSS system. They are discussed in Sections 3.1 to 3.4 below.

3.1. LEGAL AND REGULATORY FRAMEWORK

Most IAEA Member States already have regulatory frameworks for managing radioactive materials including radioactive waste. Following IAEA requirements [12], these arrangements usually nominate a specific body — often a government department — to regulate these matters and provide the appropriate powers so that the import, export, transport and use of radioactive material are controlled in ways that promote the safety and security of workers and the general public. In many cases, however, these arrangements do not extend to the issue of radioactive waste disposal. If the complete BOSS system is be implemented, it will be necessary to close this gap so that the legal and regulatory framework allows the possibility of disposal in boreholes.

Where the wastes primarily consist of disused SRS, the legal and regulatory arrangements could be quite simple. Issues needing consideration are discussed in Sections 3.2 to 3.4. The IAEA can provide model legislation and regulations that may be helpful in formulating the legal and regulatory framework.

3.2. ESTABLISHING THE MAIN ACTORS

3.2.1. Regulatory body

In a country with small quantities of radioactive material, the responsibility for controlling the use and movement of this material will often be allocated to a government ministry, typically the ministry for the environment or the ministry for public health. The main duty of the regulatory body is to ensure that the government's radioactive waste management policy is implemented with due regard to safety and security. This duty is performed by writing and enforcing regulations and providing guidance to implementers. In the case of borehole disposal of disused SRS, the regulations will be primarily concerned with the radiation protection objectives (e.g. dose targets and limits for workers and the general public during the storage, transport and operational phases, and dose targets and limits for the general public during the post-closure phase). The regulatory guidance will advise the implementer on issues such as:

- How the regulations may be met;
- The procedures to be followed when seeking approval for pre-disposal or disposal activities;
- The time scales likely to be required by the regulatory body when considering an application for approval;
- The implementer's obligations (if any) with respect to post-closure institutional control of the site.

The regulatory body's duties with respect to disposal primarily include:

- Reviewing and assessing applications from the implementer;
- Giving approval for the pre-disposal and disposal phases;
- Inspecting the activities of the implementer (following a pre-defined plan).

More detailed requirements and guidance are provided later in this document and elsewhere [10]. It is unlikely that the regulatory body will have all the expertise necessary for doing this work. Assistance may be obtained from independent experts and from the IAEA.

3.2.2. Implementer

As with the regulatory function, responsibility for implementing the BOSS system could be allocated to a government ministry. It could be the same ministry as the one appointed to be the regulatory body but, if it is, there needs to be a clear separation of responsibilities to avoid any conflict of interest.

The main duty of the implementer is to realize government policy on the disposal of disused SRS while, at the same time, observing all the relevant legal and regulatory requirements. Normally, the government will allocate a budget to the implementer to allow it to recruit, employ and train its own staff and, as necessary, draw on external expertise. The in-house establishment needs not be large: it could consist of little more than a project manager who would be responsible for interfacing with the regulatory body and employing and coordinating a series of expert teams to perform the various activities. In these circumstances, separate teams would be introduced to condition the SRS, construct the borehole, operate the disposal site, prepare the safety documentation, etc. An important part of the project manager's responsibility would be to ensure that the contractors understand and fully comply with the regulatory requirements.

3.3. NATIONAL INVENTORY OF RADIOACTIVE WASTE

The national inventory will describe all the radioactive wastes held by a country. It should also include any radioactive material that is currently in use that will become waste at some time in the future. Some of these wastes may not be suitable for disposal or, if suitable, may be unsuitable for disposal in a small diameter borehole. It is necessary, therefore, to classify the wastes into different management classes such as:

- SRS to be returned to the manufacturer;
- Wastes suitable for decay storage;
- Disused SRS and other small volume wastes suitable for disposal in a borehole facility;

- Low level wastes suitable for near surface disposal;
- Long lived wastes needing deep disposal but with volumes or dimensions that make them unsuitable for borehole disposal.

If the wastes in the fifth category are significant, a borehole facility, which could only take part of the wastes, may not be a cost effective solution. Note that the presence of wastes suitable for near surface will not preclude the use of the BOSS system because the two could easily be deployed together. Clearly, to avoid the disruption of having to change the disposal plans at a late date, it is important for the inventory to be comprehensive.

The national inventory of radioactive waste should include information on:

- A general description of the nature of the waste (e.g. disused SRS, ion exchange resins, etc.);
- The activity (Bq) at a given date and the radionuclide(s) concerned;
- Physical size;
- Chemical nature;
- Brief history (date of manufacture, manufacturer, subsequent use);
- Present condition (e.g. packaging arrangements);
- Current shielding arrangements, including weight of shield;
- Present location and owner.

The amount and nature of the waste will largely determine the size, difficulty and cost of the national disposal programme. It is important that the inventory should be as comprehensive as possible so that adequate resources and facilities can be allocated to the work and to avoid late design changes needed to accommodate unforeseen waste.

3.4. NATIONAL COMMITMENT TO BOREHOLE DISPOSAL

Implementation of the BOSS system requires a clear commitment from policy makers to borehole disposal of disused SRS. Such a commitment should be made formally, either through the national radioactive waste management policy or strategy, or through some other means such as an agreement with the IAEA.

Confirmation that the four prerequisites are in place will be probably be achieved through an IAEA pre-mission. This will include verification of the national inventory of disused SRS for disposal.

3.5. PROJECT PROPOSAL

The next step is for the implementer to construct a project proposal. This is an important document that (in the road map used here) initiates the implementation of the BOSS system. The project proposal provides, to the extent possible, explanations of:

- Why the facility is needed;
- The disused SRS to be disposed of;
- The design of the facility;
- Why the facility will be safe (the generic safety assessment — GSA, see Section 4.8.1)will be an important source of information here);
- The activities that will be required to do the work safely.

All the activities are described — including both the pre-disposal and the disposal phases. Clearly, some of these activities — especially those towards the end of the project — will not be known in detail but it is important, nonetheless, to describe them as fully as possible because the regulatory body will need assurance that the whole project is capable of being safely and effectively implemented. This means that the project proposal will also need to include a project management plan (see Appendix VI) that will detail:

- The proposed project schedule;
- The need for, and availability of, resources (i.e. cost estimate and budget);
- How assurance is to be provided that the work will be completed to the required budget, timescale and quality.

With respect to the third of these points, an important way forward is first to identify those aspects that are crucial to safety. Examples for worker safety are the activities in which workers need to perform operations on disused SRS. Here the key to safety is the use of suitably qualified and experienced people and a comprehensive radiation protection plan. Another example, this time related to post-closure safety, is the integrity of the waste capsule and container. In this case, assurance is derived from traceable material and quality control during manufacture, welding and leak testing.

Once completed, the project proposal is forwarded to the regulatory body for independent scrutiny and, if satisfactory, approval. If the regulatory body rejects the project proposal, the plan’s shortcomings should be described so that it may be re-worked.

4. PRE-DISPOSAL PHASE

Regulatory approval of the project proposal allows the pre-disposal phase to start. The pre-disposal phase includes both non-site specific and site specific activities. These are described in roughly chronological order, beginning with the non-site specific activities. Radiation protection activities are described in Appendix I.

4.1. COLLECTION, CHARACTERIZATION AND CONDITIONING

Disused SRS will be collected from their place of temporary storage and, following the transport procedures described in a later section, taken to the conditioning unit (Fig. 6) or (for SHARS) the BOSS hot cell (Fig. 6), which will have been established in a convenient and secure location. General processes for the collection, segregation and characterization of disused SRS are described in Refs [13, 16].

4.1.1. SRS characterization

For any source to be properly managed, information is needed about the:

- Strength — amount of radioactivity at a given date;
- Type — purpose, identity of radionuclide;
- Condition — physical condition, dimensions, nature of seal, leaking or intact.

This information should be contained in the documents accompanying the SRS. A unique identifier should allow the SRS to be unambiguously linked to the documents.

Where the source documentation is deficient, the IAEA International Catalogue of Sealed Sources may be helpful [14]. This is a comprehensive and still growing database of radiation devices (e.g. radiotherapy and radiography machines) that allows devices to be quickly and accurately identified. Having identified the device, the Catalogue can then provide much useful information. This includes the name, address and telephone number of the maker (or its successor), the isotope involved, its initial activity, likely date of manufacture and physical and chemical form. The Catalogue cannot, of course, provide information on the current condition of the source. Access to the International Catalogue is through an individual nominated by each Member State.

If it is necessary to re-characterize an SRS for the purpose of conditioning, a certificate specifying the method used to obtain the results and the name and address of the responsible person must be provided.

On receipt into the conditioning unit, the identity, nature and strength of the SRS will be confirmed by direct measurement prior to conditioning. Any SRS that do not tally with the documentation will be returned to the transport/storage package pending investigations by the radiation protection officer.

4.1.2. Conditioning

What conditioning is

In the BOSS system conditioning is the placing of one or more disused SRS within a 316 L stainless steel capsule that is subsequently seal welded. Prior to 2001, some capsules for radium conditioning were made from 304 stainless steel, which is similar to 316 L, but with slightly inferior properties with respect to localized corrosion. Capsules are 3 mm thick and currently manufactured in one length and two diameters to accommodate different physical sizes of source.

Conditioning provides:

- Conversion of the SRS to a special form radioactive material (in the sense of the IAEA transport regulations);
- Proper documentation for the SRS;
- Physical protection from damage and radionuclide release;
- A standard sized package that is more easily handled.

All this, greatly facilitates, and improves the safety of, subsequent transportation, storage and disposal.

For Category 3 sources and below (e.g. Ra-226 brachytherapy sources) characterization and conditioning may be performed in the lightly shielded conditioning unit. The operation of the conditioning facility, including procedures for welding, leak testing and radiological protection, is described below and in Section 6 of Ref. [11]. More powerful gamma emitters (up to 40 TBq Co-60 equivalent) will require the use of the BOSS hot cell. At the time of writing, the hot cell has been commissioned but has not been used for containerization of SRS.

FIG. 6. Conditioning unit located at Pelindaba, South Africa.

General requirements for conditioning

SRS accepted for conditioning must meet the relevant waste acceptance criteria (see Section 4.8.4).

The following guidelines should be observed when setting up a supervised area where sources are manipulated:

(a) Continuous control of contamination must be carried out. Storage containers should be subjected to contamination control before, during and after transfer of this content to another container (e.g. the conditioning capsules).
(b) The transfer zones should be covered with double sheets of polyethylene wherever contamination may be expected to take place. This should include covering of the lead bricks (Fig. 7) used for shielding. Confirmatory air sampling will be performed in this area when leaking sources are handled.
(c) The operational area should be made as small as practical, so that all operations may be carried out within a limited area.
(d) Suitable ventilation should be maintained, extracting to the outside air via HEPA filters. Ventilation should be controlled in such a way so as to avoid spread of contamination. The ventilation hood will be installed to cover the area with the biggest potential for loose contamination. This normally is the transfer area where leaking sources could be transferred into a stainless steel capsule.
(e) The operation should be optimized to limit the number of manipulations of the sources for various operations.
(f) Special attention must be given to the receiving area where the incoming transport packages are handled.

FIG. 7. Lead brick shielding standing in front of the leak testing unit.

The areas should be subjected to the following:

- All containers within the supervised area should be checked for contamination before the campaign and at regular intervals during it.
- Radiation levels within the supervised area should be established prior to the intended work (e.g. encapsulation operation). The radiation level in the transfer zone should be checked before and after the transfer of the contents of a transport or storage container to a capsule.

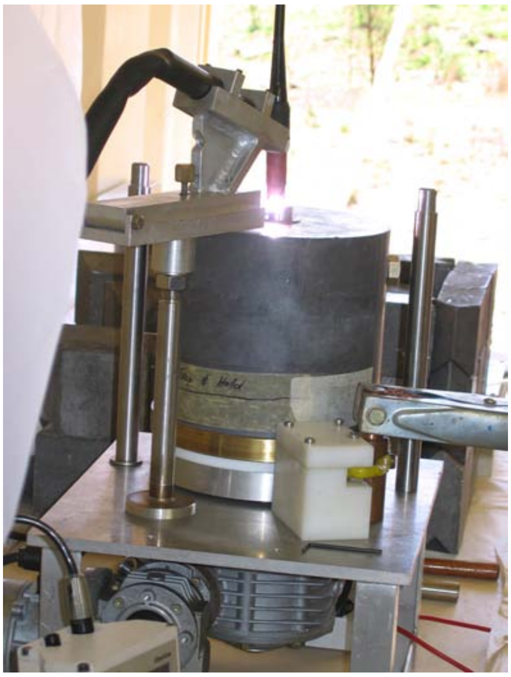

FIG. 8. Capsule welding in progress (note the cylindrical lead shield).

Permissible contents of a capsule

The amount of radioactivity that it is permissible to place inside a capsule is mainly determined by the operational safety assessment (described in Section 4.8.2) and is defined by the waste acceptance conditions described in Section 4.8.4.

Operations in the Conditioning Unit

The supervised area for the conditioning operation should consist of the following five zones (Fig. 9):

- Receiving zone;
- Transfer zone;
- Welding zone;
- Leak testing zone (Fig. 7);
- Container filling/storage zone.

The receiving zone is the area where SRS in their transport or storage container (which may include the original device) are received. At this point, the team should know the total inventory of the received package, as much as possible about its content (e.g. number of sources, their activities and geometry), external contamination, if any, and any other useful information (e.g. ease or difficulty of opening of the shield, approximate weight of the shield). During the removal of the SRS the workers will wear protective clothing and dust masks in case the source is leaking. The masks will also contain activated carbon to reduce exposure to Rn-222 that may emanate from Ra-226 sources. The extraction hood will further limit the potential inhalation dose to the workers.

Removal of an SRS from a device will usually take place behind the lead shield. In some cases however, this may not be possible and here the radiation protection officer (RPO) will advise on the need for temporary shielding. In any event, dismantling will stop when the component that contains the source is small enough to fit into the capsule.

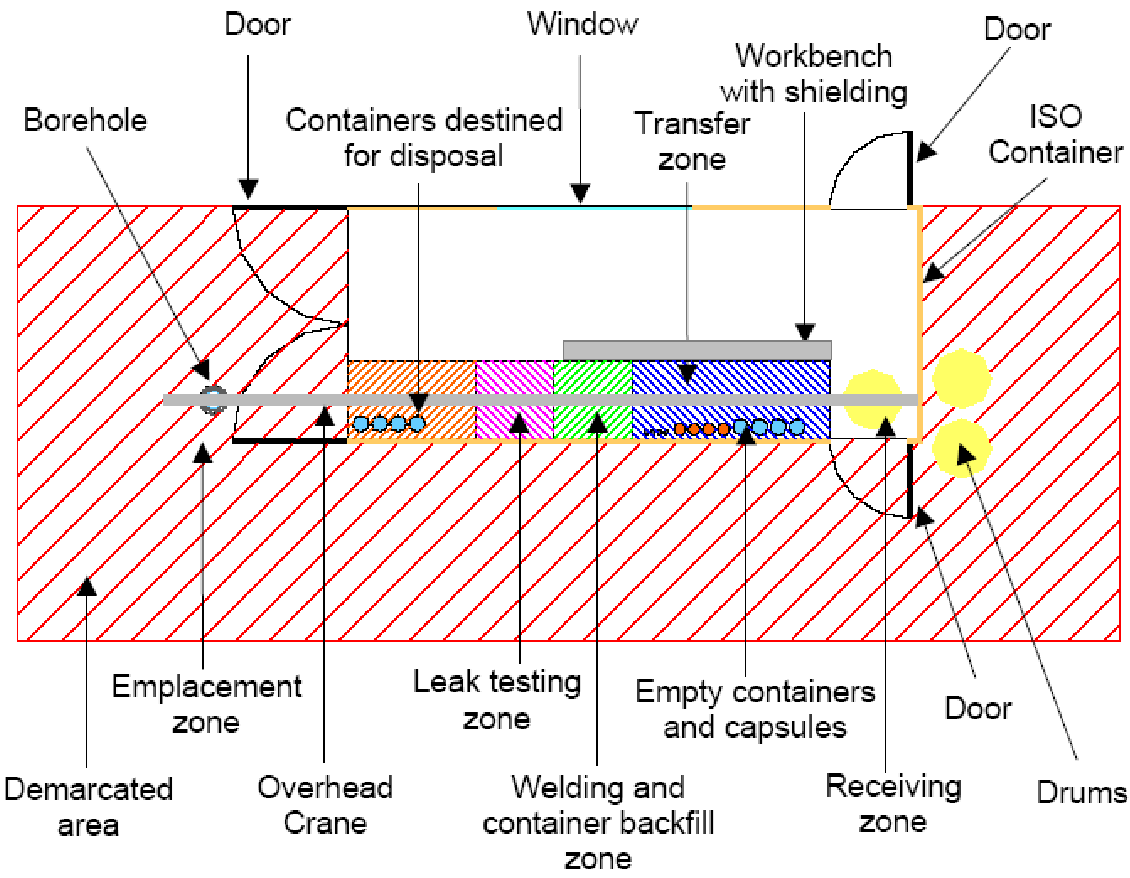

FIG. 9. Conditioning and containerization facility situated adjacent to a disposal.

Transfer zone is the area where the SRS is transferred to the capsule. The zone has a calibrated radiation monitor that allows the level of radioactivity to be checked. Shielding needs to be arranged so that the monitor readings are not affected by radiation from other SRS that may be present in the conditioning unit. Workers are protected using a lead brick wall (see Figs 7–9).

The welding zone is the area where the capsule is welded (Fig. 8). Tungsten inert gas (TIG) welding is used for the capsules. The zone contains a special radiation shield for welding and an area designated for the cooling of the welded capsules.

The leak testing zone is the zone where the welded capsules are tested for any leakage. The zone contains the leak testing equipment and is shielded with lead bricks in front of the leak testing unit (Fig. 7).

Container filling/storage zone is the area where the welded and leak tested capsule is transferred to its temporary shielded container for storage or transport to the storage facility. This zone is shielded with lead bricks in front.

Operations in the BOSS hot cell

For SRS in Categories 1 and 2 (i.e. SHARS) the radiation fields will be too high to allow time and distance methods of controlling dose. Here, the BOSS hot cell or something similar will be needed to perform tasks such as removal of the SRS from the working shield (i.e. the device) or the transport package, characterization, conditioning (if needed) and containerization. Heavily shielded transport packages will also be required as will more heavily shielded storage arrangements. For disposal, it will be necessary to position the hot cell over the borehole so that the disposal package (containing the SRS) can be lowered directly into the hole.

Conceptually, the BOSS hot cell (Fig. 10) consists of two steel boxes, one inside the other. The inner box is the cell itself while the outer box defines the external dimensions. The annulus in between the two boxes contains the biological shield — a 1.55 m wide, body of sand. If the sand has a density of at least 1.6 kg/L, this should allow the radiation from a 1 000Ci Co-60 source to be reduced to 0.045 $mSv.h^{-1}$ on the surface of the outside wall of the cell [15]. 90 m^3 of sand is required to fill the cavity. The working volume of the cell is 2.5 m long, 1.6 m wide and 3.0 m high. This gives enough space to allow a SHARS to be admitted with its working shield while still allowing free movement of the manipulators. It also accommodates the internal crane and the other tools and equipment needed for dismantling of the working shield (i.e. the device) containing the SRS.

The BOSS hot cell includes a window — essentially a water tank made from shatterproof polycarbonate — that provides radiation shielding equivalent to that of the walls. This allows most of the cell to be viewed directly. Areas that cannot be seen through the window can be viewed using the CCTV cameras.

Leaking sources

Some SRS may be leaking when received. It is expected that this will be known in advance from smear tests on the transport package or during dismantling of the device to allow removal of the SRS. Nonetheless, protective equipment will be worn when removing SRS from devices or shielded transport packages. Clean-up following discovery of a leaking SRS is likely to generate secondary wastes (see below).

Records

The following records should be maintained:

- Details of the make-up of each capsule;
    - Source identities and radionuclide contents together with confirmatory measurements (e.g. gamma spectroscopy results);
    - Unique capsule number;
- Confirmation that all the steps in the procedures were completed as planned (with dates and signatures);
- Description of any non-conformances to the procedures and an explanation of the likely consequences;
- Leak test results on the capsules.

Timing of conditioning

It is likely that conditioning of the disused SRS will be completed months or even years in advance of disposal. SRS will then be held at a central, secure storage location.

FIG. 10. BOSS hot cell during commissioning.

4.1.3. Secondary radioactive wastes

While it is not expected that secondary wastes will be generated by the conditioning processes, it is possible that this could occur if a leaking SRS is encountered and it is necessary to do some decontamination. The waste could be solid and liquid.

The basic waste management principles should be adhered to when radioactive contamination is encountered during conditioning namely waste prevention and minimization. The basic steps in waste management will also be applicable to the secondary waste, namely pre-treatment, treatment, conditioning, storage and eventual disposal. The steps described below are intended to allow the secondary wastes to be disposed to a borehole along with the conditioned sources.

Solid secondary radioactive waste

It is assumed that solid radioactive waste will consist of lightly contaminated materials but that no other disposal (e.g. near surface) facility is available. For example, where a leaking SRS is removed from a device, loose contamination will be removed from the conditioning unit and the device with tissues.

The following steps will be required:

- Collect the solid waste. Establish which radionuclides are present and the level of contamination. If this is very low the regulatory body may allow the waste to be sent to a municipal landfill. Otherwise it should be placed in a suitable storage container. In most cases, contamination levels will be too low to justify encapsulation in concrete.
- Measure the dose rate on the outside of the container. If it exceeds 2 mSv/h on contact, then place the container in a shielded over pack that will reduce the dose rate below this figure while the waste is in storage.

- Mark the outside of the container as follows:
  - Radiation warning sign;
  - Nuclide composition of the waste;
  - Nuclide activity;
  - Container number.

Liquid secondary radioactive waste

The following steps will be required:

- Collect the liquids and characterize by taking samples for radiochemical analysis.
- Where the waste is lightly contaminated the regulatory body may allow discharge to a municipal sewage system.
- More highly contaminated liquids should be solidified in a prepared cement–vermiculite mixture inside a storage container or containers with a diameter of less than about 130 mm.
- Mark the outside of the container as described above for solid waste.

Records

For secondary wastes the following records should be preserved:

- Original physical form (solid or liquid) of secondary waste;
- Method of encapsulation (if any);
- Physical dimensions of the waste package;
- Nuclide identity and activity;
- Number and dimensions of secondary waste containers.

4.2. TRANSPORT OF SRS

Transport of sources will be performed in accordance with the IAEA Regulations for Safe Transport of Radioactive Material [17]. The regulatory body should always be informed of movements of radioactive material through the public domain.

4.2.1. Unconditioned SRS

Unconditioned SRS will often be kept at a central storage site where, for instance, radium sources may have been encapsulated into 200 L, concrete filled drums [16]. In many cases, difficulties with transport can be eliminated by locating the conditioning unit at the storage site. In this way, movement of unconditioned SRS only takes place on the licensed site i.e. public domain transport can be avoided so that, under the supervision of the RPO, lower standards of shielding and containment are permitted. Once the SRS has been characterized and conditioned, transport through the public domain will be much simpler because the SRS will have special form certification and source documentation.

Where it is necessary to transport an unconditioned SRS through the public domain, problems — well described in Ref. [13] — can arise if:

- Source certificates/information is missing;
- Special form certification is missing or invalid;
- Even if the original transport package is available, it may not meet modern standards.

If information about the SRS is missing, this may be obtainable from the original manufacturer or it may be re-generated by characterizing the source. This will need suitable equipment, of course. The method of transport then depends on (i) the amount of radioactivity, (ii) the identity of the radionuclide, and (iii) the existence (or not) of a valid special form certificate issued by the manufacturer. Where a valid special form certificate exists, and the amount of radioactivity falls below the appropriate limit [17], the SRS can be transported in a Type A package. If the amount of radioactivity exceeds the Type A limit, a suitable Type B transport package will be needed. In some circumstances a 200 L concrete filled drum may serve as a Type B package. If the source cannot be characterized, it can only be transported under special arrangements agreed with the regulatory body.

In many cases the special form certification for an unconditioned SRS will have expired. The radioactive material may still be contained, however, so that some Type B packages may be suitable. Alternatively, the regulatory body may agree a special arrangement or may renew the special form certificate for long enough to allow the SRS to be moved to the conditioning facility in a Type A package (or Type B for higher amounts of radioactivity). Where the special form certification is missing, it may be possible to obtain a copy from the manufacturer; if not, transport will require a special arrangement agreed with the regulatory body.

If the original transport package (the one used to bring the SRS to the site of use) is available this may simplify the transport arrangements. In some cases however, these packages may not meet modern standards. Where this is the case and no suitable alternative transport package exists, the regulatory body may permit the use of the original package on a special arrangement basis.

4.2.2. Conditioned SRS

SRS that have been conditioned qualify as special form material. Provided that the radioactivity in the capsule does not exceed the limits for a Type A package (i.e. the A_1 values listed in Table 1 of Ref. [16]), it can be transported in the BOSS transport/transfer flask. Where the radioactivity exceeds the Type A limit, a suitable Type B package will be needed.

Unusually, an SRS in storage may have been conditioned and containerized already. Tests on the disposal container [18] demonstrate that it meets the requirements of the IAEA transport regulations for a Type A package. Provided the amount of radioactivity does not exceed the Type A limit, therefore, it will be possible to transport the SRS without an additional transport package. If the amount of radioactivity exceeds the Type A limit, a suitable Type B container will be needed.

See also Section 4.8.3 on transport safety.

4.3. INTERIM STORAGE

Transport packages are carefully engineered, often with heavy radiation shielding, so that, when radioactive material is transported through the public domain, workers and members of the public are protected, even in the event of accidents. The amount of shielding and detailed engineering means that transport packages can be expensive so that they are almost always designed to be re-used. Consequently, when a transport package arrives at its destination e.g. an interim store, the radioactive material will usually need to be removed from the transport package and placed in a shielded storage container. The radiation shielding on the storage container does not need to be as heavy as that on the transport package because a store will operate under closely controlled conditions with, for instance, restricted access areas. The design of storage facilities is described in Refs [13, 19, 20] while handling, conditioning and storage of disused SRS are described in Ref. [16].

To promote the safe storage of disused SRS, the AFRA programme is providing assistance with the construction of storage facilities and has designed and constructed a storage container for disused SRS (Fig. 11).

FIG. 11. BOSS interim storage container.

Interim storage of SRS is relatively straightforward and the requirements to achieve acceptable levels of safety and security have been comprehensively described in the references just cited. It is not proposed, therefore, to elaborate on these requirements here.

4.4. DEFINE INVENTORY FOR DISPOSAL

Having compiled a complete inventory of radioactive waste, the implementer will then confirm the inventory for borehole disposal. This will be a subset of the complete national inventory if, for instance, there are other wastes that can be scheduled for near surface disposal.

If secondary wastes are being held in storage pending disposal, the implementer should consider whether any additional treatment or packaging is needed to allow them to be disposed along with the SRS.

Having defined the SRS for borehole disposal, the detailed information in the inventory can be used to settle some of the design details, in particular, the number of disposal containers that will be needed and, hence, the length of the disposal zone. The disposal inventory will also indicate whether there are any outsize sources that could require extra long capsules or disposal containers to be procured.

According to the present design, only one capsule may be placed in each container. If the number of capsules is known, therefore, the number of containers will be known also. It is conceivable, however, that the design could change to allow more than one capsule per container. The operational safety assessment (see Section 4.8.2) will generally determine the activity limit per container. This information will allow individual SRS to be allocated to specific capsules (with specific dimensions) and so allow the number of disposal packages and their lengths to be specified. Assuming that the spacing between disposal containers is fixed, the length of the borehole disposal zone can be calculated. If the borehole depth is known, this will allow an estimate to be made of the number of boreholes that will be required.

4.5. SITE SELECTION

4.5.1. Define the process, shortlist sites

Always, it is necessary to remember that site selection is never about finding the safest site (an impossible task) but, rather, about finding one that meets the requirements for safety. Given that the BOSS disposal borehole is designed to tolerate a wide range of geological and climatic conditions, there should be an expectation that most sites will be capable of meeting the safety requirements and, where this is indeed the case, site suitability is then likely to depend on issues such as transport distances and accessibility and non-technical matters such as local acceptability and proximity to important cultural or religious sites. In the identification of individual sites, land ownership is often an important factor with preference being given to government owned land.

So far as technical issues are concerned, a good site should have stable geology, suitable geochemistry, low water flow at depth, and reasonable accessibility; it should not be subject to rapid erosion or flooding and there should be no natural resources.

Site selection can be done in many ways: three are outlined here. The first starts with a small number of sites of convenience i.e. sites that, for a variety of reasons, are preferred. Typically, existing radioactive waste storage sites, existing nuclear facilities and suitable government owned land may fall into this category. If there is only one preferred site, and provided this shows a reasonable prospect of meeting the long term safety requirements (see Section 4.5.2), it will be possible to proceed directly to the site characterization stage.

TABLE 1. POSSIBLE FACTORS THAT MAY BE USED TO HELP DEFINE A SHORTLIST OF SITES

Technical factors	
Suitable	Unsuitable
Geological stability	Volcanic or tectonic activity
Geomorphological stability	High erosion/steep topography
Arid areas	Flood plains/surface salt deposits/sea inundation
Absence of natural resources	Minerals, gas, geothermal or (where water is in short supply) water resources present
Non-technical factors	
Government owned land	National parks/nature reserves
	Land close to disputed boundaries
Land far from habitation	Areas of high population density
Sufficient information available	Comprehensive lack of information

A second method of site selection is based on the screening out of unsuitable areas. This method will usually start with the whole country and then, using whatever relevant information is available (see Section 4.5.3), will eliminate regions that, for technical and non-technical reasons (see Table 1) are likely to be unsuitable. Having established potentially suitable areas, individual sites within these areas are identified — normally from lists of sites in government ownership.

A third method of site selection is, in effect, the inverse of the second in that it starts by looking for areas that meet a list of suitability criteria (see Table 1). Again, once suitable areas have been located, individual sites within these areas may be identified from lists of sites in government ownership.

Any of these three methods will yield a shortlist of potential sites. Whatever method is used, it is essential that all the steps in the decision making process should be defined in advance and that they should be clear, logical and justified.

4.5.2. Select one site

Having established a shortlist of sites, one of these must be selected. The first task is to decide whether any of the shortlisted sites may be incapable of meeting the long term safety requirements: if there are, they must be eliminated. For the BOSS borehole concept, factors that could eliminate a site for reasons of post-closure safety are mostly related to (i) geological and geomorphological stability, (ii) geochemistry and (iii) hydrology/hydrogeology as elaborated in Table 2. Some of these factors will have already been addressed in drawing up the shortlist.

TABLE 2. FACTORS THAT COULD ELIMINATE A SITE ON GROUNDS OF POST-CLOSURE SAFETY

Stability	Geology is unstable i.e. active fault close to the site Geomorphology suggests erosion down to the top of the disposal zone over the timescale of interest
Geochemistry	Unsuitable* groundwater or porewater geochemistry Evidence that, on the timescale of interest, the geochemistry could change to an unsuitable* state *Suitability in terms of geochemistry is discussed in Section 4.8.1
Hydrology	High rates of rainwater percolation, groundwater flow, and weathering so that, over the timescale of interest, the weathered zone would extend down to the disposal zone

The “timescale of interest” referred to in Table 3 will depend on the level of radioactivity and the half-lives of the radionuclides in a similar way to Figure 1 but with an upper limit of about one million years, which is widely recognized as a maximum timeframe for post-closure safety assessment.

The information required to support this preliminary assessment of post-closure safety comes from documentary evidence (e.g. maps, see next section) and from site visits. The latter will provide information through site walkovers, interviews with local experts, examination of potential host rocks at outcrop (which may not occur at the site itself) collection of hand samples and so on. These visits will enable the documentary evidence to be confirmed, corrected and supplemented.

Having established a shortlist of potentially suitable sites, i.e. ones that appear to be capable of meeting the required standards of safety, they can be ranked into a preferred order. Preference criteria broadly fall into three groups:

- Issues related to the difficulty or ease of demonstrating compliance with the long term safety requirements; examples are geological and hydrogeological complexity, and significant, difficult to fill knowledge gaps about a site;
- Issues related to cost and convenience e.g. accessibility, travel distance from the waste storage sites, availability of utilities at the site;
- Social factors such as sites with cultural, religious, social or scientific value.

Where it falls to the implementer to select a site, it is essential that (i) all the steps in the decision making process — especially the chosen preference criteria and the weight given to each during ranking — should be clear, logical and justified; and (ii) the process defined at the outset should be strictly followed; where this becomes impossible for unforeseen reasons, these reasons need to be carefully explained.

4.5.3. Identification of data sources for siting

There are many information sources that may be useful to site selection. These include maps, databases, satellite imagery and aerial surveys showing information about:

- Topography;
- Geology;
- Climate/rainfall;
- Hydrology/drainage;
- Meteorology (especially long term data);
- Population;
- Soil type;
- Land ownership and use;
- Conservation areas (e.g. national parks).

The results of mineral and water resource exploration, where these are available, may provide useful information about the geology and hydrogeology of the rocks at depth. Other potentially useful sources of information are previously drilled wells and boreholes and areas of hydrothermal activity.

For the environmental, social and cultural aspects of siting, in addition to population and land use (listed above), information may be obtained from:

- Environmental (e.g. wildlife) surveys;
- Studies of cultural history, anthropology/archaeology, etc., which may also provide information about earthquakes and earlier climate states.

4.6. CHARACTERIZATION OF PREFERRED SITE

4.6.1. General

The overall aim of site characterization is to gain a general understanding of the site in terms of its regional setting, its past evolution and likely future natural evolution over the assessment time frame. This will include investigating the site characteristics especially with respect to geological and geomorphological stability, geochemistry and hydrogeology. It is expected that site characterization for a borehole facility could be performed using standard mineral and water resource exploration techniques. To provide information for the environmental impact assessment, baseline data should be collected on hydrology and hydrogeology, fauna and flora, noise, traffic levels, human activities, etc.

The post-closure safety of a BOSS disposal borehole relies on long term physical containment of the radionuclides in the SRS. Because certain types of groundwater could cause the cement to degrade or the steel to corrode, geochemistry is a potentially critical determinant of site suitability. Water is usually the most important medium for radionuclide transport, so that the hydrogeology of a site will always need to be understood but the high level of physical containment that is key to the BOSS system should allow hydrogeology to be explored in less detail than would otherwise be the case. Overall, the characterization programme should seek to establish whether the site falls within the envelope of conditions investigated by the generic safety assessment (GSA) that is described in Section 4.8.1.

An important part of the site characterization programme is the drilling of one or more investigatory boreholes [21]; these have three main purposes:

- To determine the nature of any chemical reactions (especially unwanted ones) between the engineered barrier system and the surrounding environment;
- To establish the position of the water table and gather sufficient hydrogeological data to construct a simple model of groundwater movement through the disposal zone and the surrounding rocks;
- To gather data relevant to the feasibility of constructing the facility and the need, for instance, for borehole casing.

The following sub-sections describe how a potentially suitable site might be characterized. An alternative procedure is outlined in Ref. [21].

4.6.2. Geology

A key part of the geological investigations is the drilling of at least one investigatory borehole. This should extend some tens of metres below the anticipated maximum disposal depth or (at unsaturated sites) down to the water table unless this is known to be exceptionally deep i.e. more than 200 m.

Rock samples

At least one investigatory borehole should be designed to extract core for examination; a typical core diameter could be 25 mm (typically using a 75–100 mm drill). Core should be labelled and boxed and sent for expert examination and characterization. Where percussion drilling is used, a spade full of drill chips should be collected for every 1 m drilled from the top to the bottom of the borehole. This must be placed in sequential piles to allow for a suitably qualified and experienced geologist/hydrogeologist to record a drill log of the borehole petrology. A removable sample (marked plastic bag) must be taken from each 1 m pile of rock to allow for a physical record of the borehole and for further analysis. In general, it is unlikely that laboratory tests will be needed to establish the radionuclide transport properties of the rock (radionuclide sorption, rock matrix diffusion). However, in a case where post-closure safety relies on retardation of radionuclides in the geosphere and the sorption characteristics of the rock are uncertain, it may be necessary to use some rock samples to evaluate these radionuclide retardation properties.

Penetration rate

The drill penetration rate is indicative of the hardness of the rock type, which in turn can be associated with rock types or structural changes in rock types. The advance time for every meter drilled should be recorded with a stopwatch. One meter interval markings on the drill rods will assist with this determination.

Water strike depth

During drilling the depth at which any water is encountered and the estimated yield of water should be recorded. This may occur more than once. The water level in the finished borehole must also be measured and a sample taken.

Blow yield

Water which is encountered in a borehole during rotary air percussion drilling is blown out of the borehole during drilling. The amount of water being blown from the borehole provides an indication of the yield of the borehole and can be estimated fairly accurately with the use of a 900 V-notch weir. The water must be channelled away from the borehole and through the V of the V-notch. The height of the water flowing through the V is translated into a flow rate or yield. If more than one water horizon is struck, each water yield should be measured individually by this simple field method. These depths and yields must be recorded.

Other drilling information

Any other drilling information which could assist in the characterization of the borehole geology/hydrogeology must also be recorded. This includes:

- Loss of compressor air (indication of joints or fissures);
- Minor or sudden changes in penetration rate (indicates changes in lithology or structure);
- Sharp colour changes (lithological changes, or indications of weathering);
- Sharp increase in size of drill chips (indication of fractures).

The weathered or unstable horizon at the surface of the boreholes must be cased to ensure a stable borehole with no collapse. This will also allow water sampling equipment and down-hole geophysical equipment to be used if necessary.

Information obtained from the drilling of characterization and disposal boreholes should be collected into a borehole log showing soil and rock types and their respective depths, depth to water table, faults, depth of weathering, etc. The overall aim should be to confirm and complement existing geological information about the site — especially evidence about the geological stability of the site. A geological map should be constructed to show the different rock types and formations together with any faults.

In some circumstances, fracturing or creep of the host rock could impede the operation of the facility so that casing becomes essential. In these cases, monitoring of rock movements (rock breakout, creep) in the investigatory boreholes could be a useful guide in planning the construction of the disposal borehole (remembering that, if the investigatory boreholes are much smaller diameter than the disposal borehole, this will need to be allowed for). It may also be helpful to measure rock stress from within an investigatory borehole.

Boreholes used for site characterization will normally be sealed after use. Exceptionally, they may, if suitable, be converted for waste disposal and thus become part of the facility. This would require plugging of the hole below the disposal zone and increasing the diameter of the hole in the disposal zone and above.

Other geological data

In addition to work in the investigatory borehole(s), standard geophysical investigations at the site may provide data on faults which, in exceptional circumstance, could act as fast pathways for groundwater flow. Geologically recent (<100 000 year old) faults are of particular interest because they may suggest that fault activity could occur within the timescale of interest. In general, disposal boreholes would aim to avoid crossing a fault. Information on the geology of the site should be supplemented by, where possible, examination of the rocks of interest at outcrop — these outcrops will not necessarily occur within the area immediately surrounding the site, of course.

4.6.3. Landform evolution/climate change

A geomorphological study of the site and its surrounding area should be conducted to examine evidence for erosive processes and past land movements (landslip, faults and earthquakes). The effect of past climate states on landform development may be helpful in assessing the likely impact of any future states.

4.6.4. Geochemistry

The geochemistry of the disposal environment is an important part of the site characterization programme for a number of reasons:

- Groundwater/porewater chemistry can affect the longevity of the stainless steel and concrete near field;
- It can provide insights into the origins of sub-surface groundwater;
- It can affect the portability of sub-surface waters (which may affect the biosphere model);
- It can affect the migration of radionuclides through the geosphere.

Oxidation/reduction (redox) potential is probably the most important parameter because of its effect on metal corrosion and radionuclide migration. Unfortunately, this is a difficult parameter to measure directly with any accuracy so that it will usually have to be inferred from observations on the host rock. These observations will include the identification of mineral assemblages in the host rock and on surfaces that have been in contact with groundwater. The observation of iron sulphide, for instance, usually indicates a reducing environment. Analysis of solute speciation in groundwater samples will also help to determine the redox conditions. In general, there is an expectation that the host rock environment will become more reducing with depth, which will usually make it necessary to estimate the redox potential at the top and bottom of the disposal zone. Where it is necessary to take groundwater samples at different depths, this will require sections of the borehole to be isolated using inflatable packers or similar.

Groundwater samples should be extracted from the borehole at different horizons and analysed for solutes and colloids. Samples should be placed in sealed containers with as little air space as possible for transport to the analytical laboratory. Anions in solution whose concentration should be measured include the halogens (especially chloride which is likely to be the most abundant), sulphate, carbonate, bicarbonate and nitrate. In conjunction with geological mapping and an understanding of the surrounding hydrology, these measurements may allow conclusions to be drawn about the hydrogeology of the site. The implications of various solutes for the integrity of the near field are discussed later.

The amount and nature of the solutes in groundwater will affect its portability. If radionuclides were to leak from the waste containers, the most likely impact on human populations would be through contamination of an aquifer. If the solute level in the groundwater makes it undrinkable, this effectively rules out this pathway, adding to the safety of the disposal.

4.6.5. Hydrology and hydrogeology

In this document hydrology and hydrogeology refer to the study of surface and sub-surface water respectively. If radionuclides were to leak from the containers, groundwater would be an important agent for radionuclide transport in the geosphere. Insofar as surface hydrology may dictate the boundary conditions for any model of the hydrogeology, the hydrological information that was obtained during the earlier stages of site selection may need to be confirmed and supplemented by, for instance:

- Meteorological measurements, especially precipitation;
- Estimates of evapotranspiration;
- Indications of any previous flooding of the site.

The disposal zone can be located in either the saturated or the unsaturated zone but, for reasons explained later, it is recommended that the interface between the saturated and unsaturated zones should be avoided. The position of any sub-surface water bodies and their seasonal variability is therefore an especially important feature of the hydrogeology. Depending on where the disposal zone is placed — above or below the water table, for instance — this will largely determine the conceptual model for contaminant migration in the event of failure of a container.

Where disposal will be in the saturated zone, the investigatory borehole(s) should be used to identify the flow characteristics of the sub-surface water body. Pump tests and flow recovery tests can be used to estimate the flow properties of the surrounding host rocks. In general, 75 mm boreholes will be adequate for this purpose. Where there is to be more than one investigatory borehole, these will normally be drilled along the line of the anticipated head gradient so that the magnitude of this can be confirmed. Information about the hydrological properties of the intervening rock can be obtained by monitoring pressures in one borehole while pumping water from the other. In general, low hydraulic conductivity, low groundwater flux and good radionuclide retention capacity are favourable properties for the host rocks of a disposal borehole.

4.6.6. Biosphere, land use, human activities

Site characterization should also include characterization of the biosphere of the site and areas where groundwater from the vicinity of the facility could discharge in the post-closure period. The information collected should include land use, local population habits (especially consumption of foodstuffs) and sources of drinking water. The nature of the current day biosphere will help to set the context for the biosphere model used in post-closure safety assessment. Similarly, data on food consumption is likely to be required for defining critical groups and estimating doses.

4.7. SITE SPECIFIC DESIGN

General

Having selected and characterized the site, the next step is to adjust the design of the borehole so as to make good use of the qualities of the site. Some aspects of the borehole design are standardized. These include the disposal package design, the borehole diameter, the thickness of backfill/grout between the disposal packages and the borehole wall, and the backfill and grout material.

Borehole casing

The function of borehole casing is to aid the facility operations by retaining the borehole wall, providing a smooth inner surface for the borehole, and keeping the borehole dry. This means that borehole operations will usually require a casing to be fitted. Post-closure safety, on the other hand, does not require a casing but, rather, requires the casing to be removed above the disposal zone to prevent its acting as a channel along which radionuclides could migrate undiluted to the surface.

The borehole design allows a choice between steel or high density polyethylene (HDPE) which, being lighter and more easily handled may be preferred from the viewpoint of ease of construction. On the other hand, mild steel may be preferable from an operational point of view because down-hole tools are less likely to snag by digging into the casing surface and removal of the top section could be easier because of its higher strength.

Package spacing

The reference design for the disposal borehole and the generic safety assessment (GSA, Ref. [22]) assumes that the waste containers are placed in the borehole at 1 m intervals. It may be convenient, however, for this to be reduced because this would enable more disposal packages to be emplaced in a given length of disposal zone. This change in geometry is not expected to affect the longevity of the containers and capsules and, so long as the containers remained intact, post-closure safety will be unaffected. If containers were leaking, however, the fact that they are closer together will produce a more concentrated plume of contaminants and this will affect post-closure safety. The effect is not expected to be profound but it will require the post-closure safety assessment to deviate from the GSA Ref. [22] so that new calculations will be needed.

Other parameters that will vary from one disposal to another are:

- Radionuclide inventory;
- Length of the disposal zone;
- Minimum depth of water table allowing for seasonal and longer term variations;
- Depth of the local erosion base;
- Depth of suitable host formations;
- Groundwater flow regime;
- Geochemical conditions.

Depth of disposal

The BOSS borehole is designed to provide a distance of at least 30 m between the top of the disposal zone and the surface. The purpose of this is to avoid the possibility of human intrusion by excavation and tunnelling. In undulating country, however, one can imagine that major road or railway construction could involve cutting or tunnelling into hillsides to a depth of more than 30 m. In this type of country, therefore, the depth requirement is that there should be at least 30 m between the top of the disposal zone and the local erosion base (which will usually be the elevation of the nearest water course to which the site drains). If the local erosion base is E m below the borehole position, this constraint fixes the minimum depth of the disposal zone as E + 30 m. The length of the disposal zone is the product of N, the number of packages, and ℓ the interval between containers in the borehole. In the most straightforward borehole geometry, the bottom of the borehole will then be at a depth of $E + 30 + N\ell$ metres.

This depth should next be compared with the depth of the water table, W. While the disposal zone can be positioned in either the unsaturated or the saturated zones, the interface between the two is best avoided because (1) corrosion performance is always impaired by wet–dry cycles, which can cause salts to be concentrated on the metal surface and (2) it is difficult to model the near field behaviour under these conditions. Consequently, if the water table lies above the disposal zone, i.e.

$$W < E + 30 \text{ metres}$$

the disposal will be situated in the saturated zone. If, on the other hand, the water table lies below the base of the disposal zone, i.e.

$$W > E + 30 + N\ell \text{ metres}$$

the disposal will be situated in the unsaturated zone. If the minimum depth of the water table falls between E + 30 and $E + 30 + N\ell$ metres, however, the borehole design will need to be adjusted from the most straightforward case. There are two possibilities:

(i) Increase the depth of the borehole so that the disposal zone is always fully saturated;
(ii) Reduce the length of the disposal zone by increasing the number of boreholes so that the disposal zone is always unsaturated.

Other things being equal, the GSA [22] suggests that disposal in the unsaturated zone will usually result in an improved level of post-closure safety. But unsaturated rocks may not always be preferred. If, for example, there was a very low permeability saturated rock, such as plastic clay, at an accessible depth, it would probably be preferable to locate the waste there. Overall, the aim should be to choose the host rock and hydrogeology that give the highest level of post-closure safety at reasonable cost.

4.8. SITE SPECIFIC SAFETY ASSESSMENT

4.8.1. Post-closure safety

Physical containment and geochemistry

In broad terms, corrosion of stainless steel may be classified into two types: uniform and localized. The latter includes pitting and crevice corrosion, stress corrosion cracking and microbially induced corrosion. These tend to be prevalent at above ambient temperatures and can occur at rates of up to tens of μm per year. Uniform corrosion rates in stainless steel are, on the other hand, usually extremely low. Depending upon temperature, pH and chloride concentration, the rate of general corrosion will typically be in the range 0.01–2 $\mu m \cdot a^{-1}$ (see Appendix I of the GSA [22]). The lower figure will apply under the most favourable environmental conditions, which include those present inside the disposal container before it is breached. With a total thickness of stainless steel (capsule plus container) of 9 mm, uniform corrosion from one side only gives a time of penetration of at least 4500 years. Comparing this with FIG. 3, we see that this is sufficient to allow decay to negligible levels of all the radionuclides shown with the exception of Ra-226 and the actinides.

To achieve long container lifetime localized corrosion must be avoided. Appendix I of the GSA [22] provides a useful decision tree, reproduced here as Fig. 12. This illustrates the key geochemical parameters that enable localized corrosion to be eliminated. It shows that the most important geochemical parameters are the oxidation potential, pH and chloride concentration. If anaerobic conditions can be guaranteed, localized corrosion will not occur and the combined container/capsule lifetime will be very long indeed.

Appendix I of Ref. [22] considers the possibility that, even under anaerobic conditions, localized corrosion could be promoted by the radiolysis of water to produce oxidizing species such as O_3. So long as the container remains intact, such accelerated corrosion will not occur at the capsule or on the inside of the container because, as indicated by Fig. 12, the low chloride and high pH conditions there will prevent it. It follows that, so long as the container remains intact, radiolysis can only produce localized corrosion on the outside of the container and only then if geochemical conditions favour it — if, for example, the groundwater contains chloride at suitably high concentrations.

According to Appendix I of Ref. [22], radiolysis could produce localized corrosion if the dose rate exceeds 2.8 $Gy.h^{-1}$. At 57 mm distance from the source (the outer radius of the container, and allowing for 40 mm of concrete between the capsule and the container) this dose rate could be produced by a Ra-226 source of around 80 GBq (~ 2 Ci). For Ra-226 this would be a very large and unusual source. For Co-60 a dose rate of 2.8 $Gy.h^{-1}$ would be produced by a source of around 10 GBq (~ 0.3 Ci) which is relatively small (for Co-60). But here, because of the shorter half-life, the required endurance of the container will be much less. Similarly, a large alpha source would not constitute a difficulty in this respect because the alpha radiation would not penetrate the steel capsule. From this brief survey it seems that, in practice, radiolysis may not have a significant effect but it will need to be addressed on a case-by-case basis. It would be helpful to have a firmer definition of the dose rate that would lead to localized corrosion.

As indicated by Fig. 12, an oxidizing environment is a necessary but not sufficient condition for localized corrosion because, if localized corrosion is to occur, the critical pH for localized corrosion must not be exceeded and, in addition, aggressive anions, especially chloride, need to be present. If either of these additional conditions is absent (i.e. if the pH is high or the chloride is low), corrosion will occur at the uniform rate. Appendix I of Ref. [22] calculates the degree of physical containment for different combinations of groundwater chemistry, flow and degree of saturation. Groundwater flow and degree of saturation have an effect because they determine the rate at which alkaline components are leached out of the cement and, therefore, the rate at which the pH falls. Typical times to initial capsule perforation for a range of reference groundwater conditions are shown in Table 3. All of these reference conditions exclude the possibility of localized corrosion and we see that there is a factor of ten difference between the shortest and the longest times but even the shortest time (72 000 years) would be sufficient to allow the most powerful Ra-226 SRS to decay to exemption levels.

Heat generation

Radioactive decay generates heat so that, especially for the most powerful sources, there may be an increase in the temperature of the SRS. Reference [28] recommends that the heat output of SRS should be less than 50 W so as to limit the temperature increase at the capsule to about 50°C. With an ambient temperature at depth of, say, 25°C, the capsule could reach 75°C. The temperature reached at the container will be 10–15°C lower.

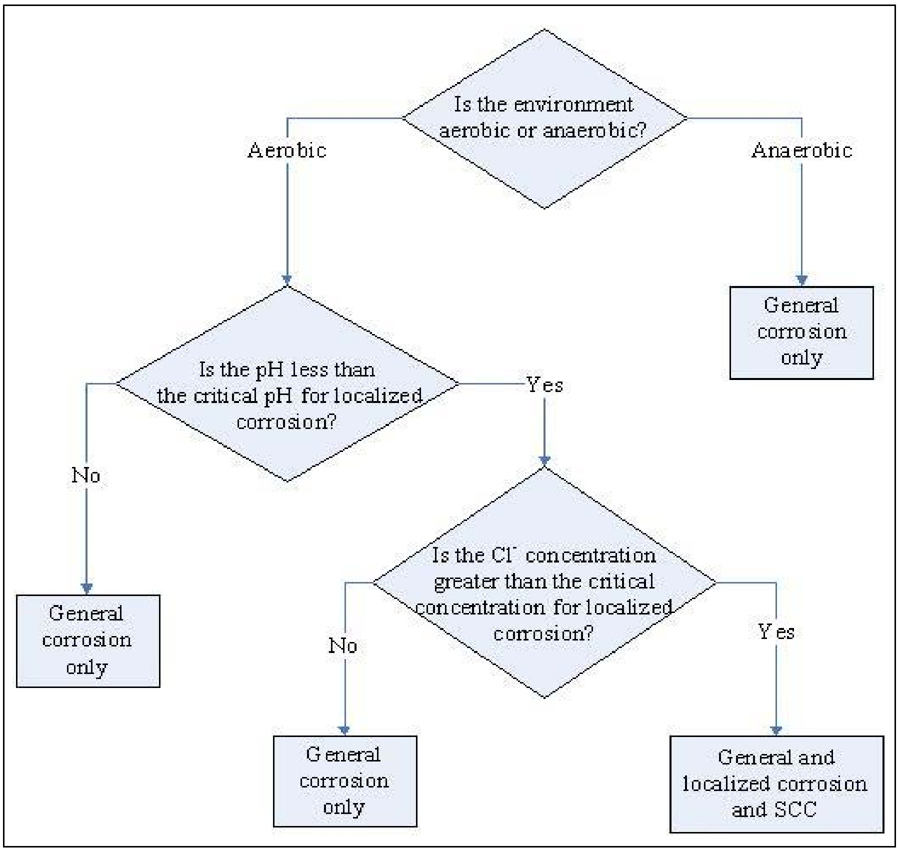

FIG. 12. Corrosion decision tree indicating the conditions that favour general and localized corrosion.

TABLE 3. TIMES TO INITIAL PERFORATION OF THE CAPSULE FOR A RANGE OF REFERENCE GROUNDWATER CONDITIONS (FROM APPENDIX I OF REF. [22])

Conditions	Initial perforation y
Unsaturated zone, aerobic groundwater, low chloride, pH 4.1	7.20E + 04
High flow (porous and fractured), groundwater, anaerobic, low chloride, pH 8.5	7.20E + 05
Medium flow, groundwater anaerobic, low chloride, pH 8.5	7.20E + 05
Low flow, groundwater, anaerobic, high chloride, pH 8	2.90E + 05

4.8.1.1. *Hydrogeology and chemical containment*

For actinide-containing SRS, even the very long container lifetimes produced by uniform corrosion are insufficient to allow them to decay to negligible levels. Am-241 for example, has a half-life of only 432 years but decays to Np-237, which has a half-life of 2.1 million years. The resulting Np-237 will, of course, have a much lower activity than the parent radionuclide but, nonetheless, these long time periods, and the need to consider the consequences of premature container failure, make it necessary to consider the fate of radionuclides if they were to escape from the disposal package. This is done in the generic safety assessment (GSA).

The GSA [22–23] presents post-closure performance calculations for a standardized borehole geometry and a very wide range of possible site specific conditions. References [22, 23] are complementary and should be read in conjunction; reference [22] will supersede both of these but only exists as a draft at the time of writing.

The scenarios used in the GSA broadly follow the conceptual models for unsaturated and saturated flow described in Fig. 13 and Fig 14, respectively. Both assume that one or more disposal packages are leaking. In the unsaturated case, rainwater percolates down from the surface, causing the radionuclides to migrate downwards, through the disposal zone and the rock below, into an underlying aquifer. The contaminated aquifer water is pumped up via an extraction borehole at 100 m distance and is used for drinking, irrigation of crops, etc. This produces a radiation dose to the people living on the site.

In the saturated case, groundwater moves parallel to the ground surface leaching radionuclides from the disposal zone. The contaminated groundwater is pumped up via an extraction borehole at 100 m distance and the radiation doses occur as before.

Actinide species sorb strongly to concrete and rock so that their rate of migration is retarded. This is sometimes known as chemical containment. The effect produces very slow leaching of these radionuclides into the groundwater or the aquifer. As a result, the GSA [22] indicates that greater than TBq quantities of the actinides can be accommodated even if the disposal package is assumed to fail early and hydrogeological conditions at the site are relatively unfavourable.

In deriving an appropriate site specific safety assessment from the GSA, the key factors are:

- Whether the disposal zone is saturated or unsaturated;
- If unsaturated, the rate of water infiltration, the water filled porosity and the height of the disposal zone above the water table;
- If saturated, the hydraulic conductivity, the hydraulic gradient and the water filled porosity;
- The sorption coefficients of the relevant radionuclides on the surrounding rocks.

Estimation of these parameters and comparison with the GSA will allow an appropriate GSA calculational case to be chosen. The results for this GSA case will be found in the form of tables that show the total borehole inventory of each radionuclide that would produce a dose rate to the critical group of 0.3 $mSv.a^{-1}$. Because the GSA has been devised so that the dose rate for each radionuclide is linearly proportional to the inventory of that radionuclide, the GSA limiting inventory can be used to calculate the dose rate per Bq for each radionuclide (as is done in Table 4). Using the actual inventory, a dose rate for each radionuclide can then be calculated. Summing these dose rate values over all radionuclides then produces the dose rate for the whole facility and this can be compared with the local regulatory target or constraint.

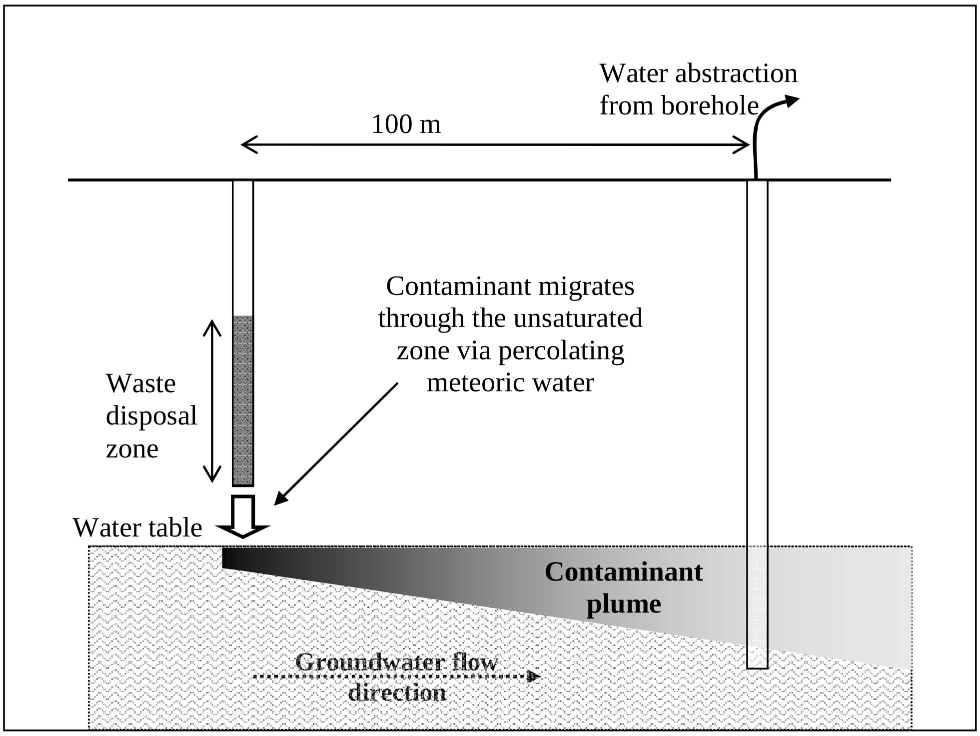

FIG. 13. GSA [22] conceptual model for a leaking disposal package when disposal is located in the unsaturated zone.

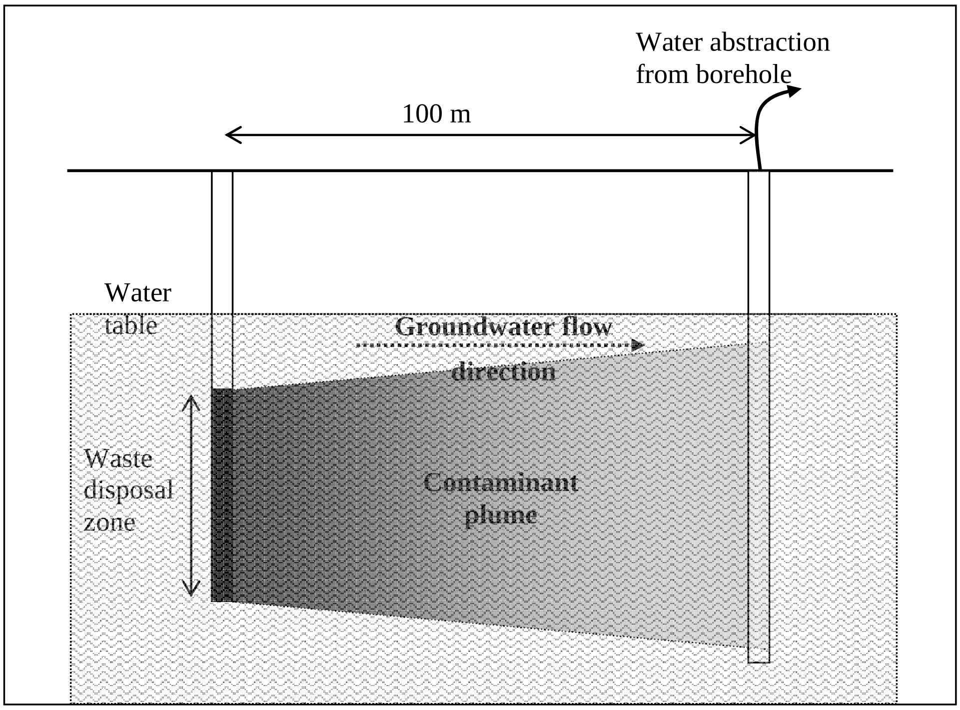

FIG. 14. GSA [22] conceptual model for a leaking disposal package when disposal is located in the saturated zone.

TABLE 4. DOSE RATE (MSv.a^{-1}) PER Bq FOR THREE RADIONUCLIDES DERIVED FOR THE LIQUID RELEASE (I.E. LEACHING) SCENARIO IN THE GSA FOR DIFFERENT GROUNDWATER FLOW REGIMES

Disposal zone	Groundwater flow rate	Groundwater flow type	Dose rate (mSv.a^{-1}) per Bq Pu-238	Pu-239	Am-241
Unsaturated	High	Porous	3E-15	3E-15	3E-15
		Fractured	<3E-19	1E-17	6E-17
	Medium	Porous	1E-16	3E-15	1E-15
	Low	Porous	<3E-19	3E-18	<3E-19
Saturated	High	Porous	8E-14	3E-15	6E-14
		Fractured	1E-15	4E-17	2E-15
	Medium	Porous	2E-16	4E-16	6E-16
	Low	Porous	<3E-19	<3E-19	<3E-19

For the design scenario used in the GSA, we find that liquid, gaseous and solid releases impose no constraints on the total activity of H-3, Co-60, Ni-63, Kr-85, Cs-137, Pb-210 or Ra-226 that can be placed in the borehole (dose rate per Bq was less than the limiting value of 3E-19 mSv.y^{-1}.Bq^{-1}). Constraints do apply to Pu-238, Pu-239 and Am-241 as indicated for different groundwater flow regimes in Table 4, which shows the GSA data re-presented in the form of dose rate per Bq of each radionuclide.

4.8.2. Operational safety

An operational safety assessment has been performed on the detailed BOSS design to demonstrate compliance with safety criteria and to identify activities and equipment with safety significance. The safety assessment consists of two components: a hazard and operability study (HAZOP) [24] which presents a detailed analysis of the operations and possible fault scenarios and a prospective hazard assessment [25] which calculates doses under normal and off-normal conditions. The studies consider all operations from conditioning through to closure of the borehole.

Exposure of workers to radiation from normal operations and from accidents is estimated based on conservative assumptions and assumes that, for all normal operations within the conditioning unit, the operator is shielded by a 100 mm thick wall made from standard lead shielding bricks (see Figs 9–11). The estimates of some activities are compared with actual dose measurements taken during previous missions where radium sources were pre-conditioned for safe storage. The current procedures for SRS conditioning [26] limit the activity within a capsule (and therefore, within a disposal container produced in a lightly shielded unit) to one tenth of the special form material activity listed in the ST-1 transport regulations [17] for a type A transport package. For Co-60 this amounts to a maximum of 40 GBq (~ 1 Ci).

An omission from the documentation at the time of writing is the safety assessment for the various activities to be carried out in the BOSS hot cell. In addition, when the two hazard assessment studies were conducted, there were significant uncertainties regarding the form and packaging of the sources, the form of transfer to the conditioning and containerization facility, the transfer method from the facility to the borehole and the backfilling of the borehole. Furthermore, it was considered that sources will often need to be handled on an ad hoc basis making it difficult to set up fully comprehensive procedures and to describe the actions required. Safe operation will, therefore, strongly depend on the experience of the RPO and radiation workers.

The hazard assessment study [25] estimated doses to workers and the general public during normal and off-normal operations. This work is summarized in Appendix III, which shows that worker doses/risks during normal and off-normal operations are acceptable. Most of the normal worker dose is accumulated during conditioning. For off-normal operations the highest worker dose occurs if a source is dropped and then breaks. There are no circumstances in which doses are received by members of the public.

4.8.3. Transport safety

Transport of sources will be performed in accordance with national regulations for safe transport of radioactive material. Often, these are based on the IAEA Regulations for Safe Transport of Radioactive Material and, where no national regulations exist, the IAEA regulations will be used in their place. The IAEA regulations define the activity limits that can be placed in transport packages according to:

- The individual radionuclides;
- The form of the radioactive material;
- The type of transport package.

When manufactured, all sealed radioactive sources are classified as special form radioactive material. This certification is time limited, however, so that placing the SRS into a sealed capsule provides a new certificate. Since the conditioning procedures for the conditioning unit limit the amount of radioactivity that can be placed in an individual capsule to one tenth of the special form radioactivity permitted in a Type A package, it is clear that Type A packages will always be suitable for SRS conditioned or containerized in the conditioning unit.

For SHARS, which will be conditioned and containerized within the BOSS portable hot cell, more heavily shielded transport packages will be needed.

See also Section 4.2 on transport of SRS.

4.8.4. Waste acceptance criteria

Generic waste acceptance criteria (WAC) for the BOSS system are derived in Refs [27, 28]. Other criteria arise from the generic post-closure safety assessment (GSA [22]). It is not expected that transport of SRS will impose any additional constraints on what can and cannot be accepted for conditioning, containerization and disposal although transport must all times comply with the IAEA Transport Regulations [17] or national regulations, if these are different. See Sections 4.2 and 4.8.3.

Generic WAC

The amount and type of radioactivity in an SRS (including SHARS) accepted for conditioning and/or containerization must fall below the operational limit for the facility being used (i.e. conditioning unit or portable hot cell) as described in more detail in the next sub-section. SRS accepted for containerization must also have an associated and correctly completed application for disposal form.

In addition:

- The document that underwrites the stated radionuclide identification and activity value will only be accepted if there is traceability between the document and the SRS.
- Where an undocumented source has been newly characterized for the purpose of allowing it to be conditioned and disposed, a certificate specifying the method used to obtain the results and the name of the responsible and accountable authority that characterized the source must be provided.
- Leaking sources must be identified but will be accepted provided that they fulfil all other conditions.
- SRS must be physically small enough to fit inside a capsule (capsules are available in two diameters).
- Sealed gas and liquid sources are acceptable provided they are identified as such and comply with the other WAC.
- Prior to disposal, all radioactive material must be conditioned within a capsule and containerized in a disposal container as described elsewhere in this technical manual. Documentation specifying the procedure and the amount and type of radioactive material should form part of the disposal record.
- Containers must be manufactured from 316 L type stainless steel according to an approved QA plan (see, for example Ref. [11]). Capsules may be manufactured from 316 L or 304 stainless steels — wherever possible according to an approved QA plan.

Mixing of different radionuclides (i.e. different types of SRS) in one capsule is allowed provided that the foregoing conditions are met and also provided that:

- When the SRS activities are expressed as a fraction of the appropriate limit from Appendix VII and added together (i.e. using the summation rule prescribed in Ref. [17]) the value of the sum is less than one;
- It can be shown that mixing the different SRS will not have unwanted effects.

Operational limits for conditioning unit and portable hot cell

Reference [28] calculates the activity limits for individual capsules. These are derived from consideration of the operational limits, specifically, the operator doses that arise during conditioning and containerization. It is assumed that two kinds of facility are available: the relatively lightly shielded conditioning unit (Fig. 6 and Fig. 9) and the heavily shielded BOSS hot cell (Fig. 10) and WAC have been developed separately for these.

The shielding provided by the conditioning unit and the portable hot cell and the handling procedures that will be used in these facilities largely determine the strength of the SRS that can be accepted by these facilities. A useful rule of thumb is that the conditioning unit and the hot cell were designed to take gamma sources with a power equivalent to 4×10^{10} and 4×10^{13} Bq of Co-60 (i.e. ~1 and ~1 000 Ci), respectively. Separate calculations [28] have confirmed that these design limits are appropriate and have extended the calculations to cover alpha, beta and neutron sources while also considering limits derived from a maximum allowable heat generation of 50 W.

Tables showing the maximum permissible activities for a wide range of radionuclides are contained in Appendix VII These tables show that for the conditioning unit the limiting factor is usually the gamma dose. For the hot cell the limiting factor is usually the heat generation limit of 50 W. This limit is derived by applying a limiting temperature increase of 50°C. The calculation in Ref. [28] is conservative and approximate, however, and more detailed calculations might allow this limit to be increased.

Reference [28] assumes that the post-closure safety assessment (i.e. the GSA [22]) will only impose whole facility limits i.e. the total amount of radioactivity that can be placed in a borehole or group of boreholes. While the GSA does derive limits for single packages, these cannot be considered as definitive because they merely represent the total activity divided by 50 (the assumed number of packages) i.e. they are not based on any considerations of safety. It is expected that future revisions of the GSA will investigate this aspect more thoroughly. For the moment it is recommended that these GSA derived package limits be ignored.

Site specific waste acceptance criteria for disposal

For individual disposal packages, the site specific WAC will be the same as the generic WAC. There will be, however, a limit on the total activity that is permitted in an individual borehole or group of boreholes. This total activity limit will be calculated on a site specific basis from the site specific post-closure safety assessment. Provided that the chosen site can be shown to correspond to one of the disposal environments considered by the GSA [22]. Section 4.8.1 provides an explanation of how the GSA may be used to establish that an actual proposed inventory for disposal could be emplaced in a borehole facility.

4.9. APPLICATION TO DISPOSAL SITE RECOMMENDATION

At the end of the pre-disposal phase (according to the BOSS road map) the implementer will make an application to construct, operate and close a disposal facility for disused SRS at a recommended site. It is important that the content of the application should have been agreed with the regulatory body well in advance. It is possible that the regulatory body may require the whole application to be presented as an environmental impact statement (EIS). Alternatively, the EIS may simply focus on non-radiological impacts and constitute just one part of the application.

Howsoever the application is organized, the information supplied should be well presented and comprehensive to give the regulatory body reasonable assurance that:

- The site selection process was appropriately defined and implemented;
- The facility, located at the recommended site under the supervision of the implementer:
    - Will be safely and efficiently constructed, operated and closed;
    - Will meet the requirements for post-closure safety.

The first of these points — information about site selection — is only needed if the implementer was required to select the site (as opposed to its being prescribed at the outset). Here, it will be necessary to present a justification for the process and a demonstration that the process was adequately implemented. This will require the process and its implementation to be described in detail explaining, in particular, the decision making criteria and the weights given to them.

The remaining points are addressed by the safety report template shown in Appendix IV. The present intention is that this template will be expanded into a model safety report that will then form Volume 3 of the suite of documents describing the BOSS system. In Appendix IV, the EIA forms a part of the safety report, rather than vice versa.

Further guidance on the content of regulatory submissions and their review is available in the IAEA Safety Guide [10] and from documents prepared by the South African National Nuclear Regulator (NNR) [29, 30].

4.10. REGULATORY APPROVAL AND INSPECTION

The regulatory body may give approval for disposal if it is satisfied that:

- The recommended site was appropriately chosen;
- The proposed design, implemented at the recommended site, is capable of providing an adequate level of safety and is ALARA with respect to radiological protection;
- The implementer (and its contractors) are sufficiently competent and possess the resources necessary to carry out the work.

Borehole construction should not start until regulatory approval has been received.

Before granting the approval, the regulatory body should be confident that the implementer will carry out these activities competently and conscientiously. In coming to this judgment, the regulatory body will consider the implementer's performance during the activities running up to the application. If there are any aspects that the regulatory body feels should be improved, it could place conditions in the approval to effect this improvement.

Approval to proceed with the disposal phase is an acknowledgement by the regulatory body that it considers the implementer to be capable of performing the activities competently and conscientiously. This does not mean, of course, that the implementer must have all the necessary skills in-house but it does mean (a) that suitable resources — equipment, materials and people — are available and (b) that the implementer has (or will have) the financial resources to procure them.

The regulatory body will need to decide the extent and intensity of any on-site inspections during this phase and should update its regulatory inspection plan accordingly.

5. DISPOSAL PHASE

5.1. PREPARATION FOR DISPOSAL

5.1.1. Identification of necessary resources

The implementation of radioactive waste management and disposal requires a very wide range of disciplines and skills. Consequently, it is common practice throughout the world for the implementer to employ expert contractors to perform the work and for the implementer to become, in effect, the project manager. This strategy is particularly suitable for small programmes where the implementer may have relatively few permanent employees. It follows that, if the work is to be progressed efficiently and economically, there needs to be a choice of contractors to perform the various tasks.

The BOSS system has been designed to use commonly available materials and expertise which will help the aim of identifying a broad contractor base. Similarly, borehole drilling and completion will follow procedures similar to those used for deep water abstraction. Because of the high level of reliance on the integrity of the stainless steel capsule and container, their manufacture will need to be subject to a strict quality control and a quality management regime. This and the economies of scale may make it desirable for the container and, to a lesser extent, the capsule to be manufactured by a single supplier. Section 4 of Ref. [11] describes the procedures for manufacture and qualification of capsules and containers.

Areas where nuclear training will be required are associated with the handling of radioactive materials: conditioning, containerizing and disposal package emplacement. For all of these tasks, staff will require adequate training and experience. The most demanding post in terms of qualifications and experience is that of the RPO, who is charged with ensuring that work with radioactive materials, is suitably assessed and then properly controlled and monitored.

The range of resources required during the disposal phase is likely to include:

- Suitably qualified, experienced and equipped radiation workers, including an RPO;
- Earth moving equipment (and staff) to level the site (if necessary);
- Fencing and fencing contractor for security fencing at the site;
- Drilling contractors and equipment available to drill, characterize and grout the borehole;
- Materials needed for borehole construction i.e. drilling fluid, sand, cement and borehole casing (see below);
- Transport for moving personnel, SRS, equipment and materials;
- Trained staff to operate the conditioning unit and, if required, the boss portable hot cell;
- Suitable containers for SRS transport;
- Other consumables (e.g. water of suitable quality);
- Accommodation for disposal team;
- Portable power supplies.

From this it is clear that a number of different teams will be required at different stages of the work. Some of these teams may be recruited locally. Others may need to be brought in from other countries.

5.1.2. Identification of materials sources

It is intended that, so far as possible, bulk materials used for the construction of the disposal borehole and the hot cell will be sourced locally. These will usually consist of sand, cement, water and casing. The aim is to identify suitable local sources for these materials and means of transporting them to the site.

Other materials such as capsules and containers, and specialized equipment for welding, leak testing and radiological protection will be procured in advance and imported to the site of use.

5.2. PREPARATION OF THE SITE

Depending on the topography of the site, it may be necessary to level it, at least to provide a platform for the drilling rig and the portable hot cell. The suitability of the approach road for the drilling rig etc. will also need to be considered. Security fencing will be installed once the groundwork is complete. Unless they are already available, portable water and electricity supplies will also be needed.

5.3. CONSTRUCTING THE FACILITY

Construction of the borehole repository can only start after regulatory authorization has been issued. The main construction activities, discussed in more detail below, are:

- Mobilization of suitable construction and operating equipment (e.g. conditioning and containerization unit, drilling rig) to the site;
- Installation of a site security system;
- Drilling of the borehole;
- Insertion of the casing;
- Grouting of the casing;
- Plugging of the bottom of the casing;
- Commissioning tests to verify that the borehole is constructed in accordance with the approved design.

5.3.1. Mobilization

Mobilization costs are often a significant part of the total costs so, if costs are to be controlled, it will usually be preferable to mobilize the drilling rig and the conditioning and containerization unit just once. Clearly, a site that allows easy access for vehicles will also reduce time and costs and improve safety and security.

5.3.2. Site security

In addition to a security fence, an administrative system (i.e. guards) may be needed to prevent unauthorized access.

5.3.3. Drilling of the borehole

The most common method employed for the drilling of larger diameter (more than 100 mm) boreholes is rotary air percussion using a down-hole hammer (Fig. 15). This type of drilling is ideally suited to hard rock formations and for the likely depths (30 m to 100 m) for the borehole disposal.

FIG. 15. Construction of a borehole using percussion drilling inset: close-up of the drill bit.

Other, more sophisticated techniques are less commonly used (i.e. mud rotary drilling, ODEX drilling, dual-tube reverse circulation) but may be needed for application in loose, soft and unconsolidated formations. Where drilling fluid/mud is used, arrangements should be put in place to collect it and prevent it from spreading over the surface of the site.

The reference dimensions are:

- Borehole depth minimum 30 m;
- Borehole diameter 260 mm (minimum at maximum depth);
- The annular space between the borehole wall and the casing should be 50 to 100 mm;
- The wall thickness of the casing should be sufficient to provide enough strength for normal duty, e.g. retaining the borehole wall and also to allow casing removal above the disposal zone; the required strength will vary depending on the likelihood that loose material could settle in the ungrouted annulus.

The disposal borehole diameter should be large enough to allow the trouble-free insertion of the casing down to the bottom of the borehole. This will normally mean that the top few metres, which will often be in weathered or unstable overburden material, will have a larger diameter than the lower parts of the borehole. The drill must penetrate at least 3 m into the more competent (non-collapsible) rock before the upper formation can be secured from potential collapse or washout, by insertion of large diameter casing (minimum 300 mm diameter casing).

Problematic geological horizons could be encountered again at greater depths (softer collapsing horizons) which could again require that the borehole be drilled to a larger diameter and reamed and cased down to this horizon. However, it is unlikely that such a geological area would be selected for the borehole disposal. The borehole must then be drilled to the bottom at a diameter of 260 mm.

5.3.4. Insertion of borehole casing

The purpose of the casing is to aid operation by stabilizing the rock wall, avoiding snagging of disposal packages and keeping the borehole dry, recognizing that disposal boreholes may be situated in saturated or unsaturated conditions. If a borehole is very stable and dry, casing may not be needed at all. Calliper tests will be needed on the borehole before inserting the casing to ensure the casing will fit.

The most widely used casing consists of steel; HDPE (high density polyethylene) has lower weight but also lower strength. To avoid snagging of disposal packages and ingress of water, the individual sections of casing need to be joined (welded in the case of steel) in such a way that the joints are smooth and completely sealed to stop any ingress of water. Casing centralizers must be placed at least every 6 m (for steel casing) between the borehole sidewall and the casing in order to ensure that the casing stays in the centre of the borehole, along the length of the borehole. For HDPE casing more centralizers per unit length will be needed. This is to provide a uniform gap between the casing and the sidewall to allow access of grout into the annulus area between the borehole sidewall and the outside of the casing. The larger diameter casing that in most cases will be needed to stabilize the overburden material should stay in place and not be removed while emplacement is in operation. The section of the casing above the disposal zone needs to be removed after the emplacement of the disposal packages.

5.3.5. Placing of the casing bottom plug

Using centralizers to position the casing vertically, a free flowing grout is prepared. Using a tremie pipe, sufficient grout is then placed into the bottom of the borehole to produce a bottom seal that is at least 0.5 m thick.

5.3.6. Grouting of the casing

Grouting of the annulus between the borehole sidewall and the outside of the casing ensures that there are no cavities that could provide easy access of groundwater to the casing and the disposal packages. The grout material is the same mixture used for grouting the waste containers inside the casing (see Section 5.4.4).

There are several methods of placing grout into the borehole casing annulus. A common procedure is to use a small diameter pipe (tremie pipe) running to the bottom of the bottom of the borehole. Grout is pumped through the tremie pipe so that it fills up the annulus from the bottom. Care should be taken to ensure that it meets the specification and is well mixed to allow free flow and prevent void formation. So that the casing above the disposal zone may be removed prior to sealing of the borehole, this section should not be grouted.

5.3.7. Borehole commissioning

Test for borehole straightness and obstructions

Slow drilling will produce a straighter borehole. The straightness of a borehole is measured by the degree to which it deviates along its length from an imaginary centreline running through the borehole. This is physically determined by passing a dummy through the borehole down to the bottom. The dummy consists of a rigid hollow steel pipe that has slightly larger dimensions (diameter and length) than the disposal package to be disposed of.

The dummy will slowly be lowered via a flexible steel rope down the borehole to the bottom. If the dummy can be inserted without any obstruction it will ensure that the eventual disposal package(s) will also have free access down to the disposal depth(s). The repeated placement of backfill cement into the borehole allows the possibility — especially if a tremie pipe is used — that cement could be dropped or splashed onto the wall of the borehole. This could obstruct the passage of a disposal container. For this reason, unless it can be demonstrated that this does not occur in practice, it will probably be necessary to check for obstructions by lowering and removing the dummy before every disposal package emplacement.

Test for verticality of the borehole

The borehole verticality represents the plumbness of the borehole as measured by the deviation of the centreline of the borehole from the vertical at any depth within the hole. Verticality is less important than straightness.

Absolute verticality of a borehole is often dependent on the geology as well as on how the drill is operated. To ensure that the borehole is as vertical as possible it is important that verticality checks are done on the drill rig after the final set-up is done and the drill rig has been secured in position. This test can be done by measuring the verticality in various directions on flat surfaces on the drill tower with an ordinary level.

This test should be repeated at least once after drilling has commenced (preferably when the final diameter (260 mm) drilling has commenced). This is to ensure that no deviation from the vertical has taken place during the emplacement of the overburden casing or overburden drilling.

5.3.8. Data recording and reporting

It is important to keep a detailed and accurate record of all information arising from the borehole drilling activity. The same drilling information that was collected during site characterization (Section 4.6.2) should also be recorded during construction of the disposal borehole.

Other records

The following information should also be preserved to provide a record of the location, construction and the as-built design. The amount of detail should be sufficient to allow anyone to understand the borehole design and how it was constructed. The documentation should also provide evidence (by way of material certificates and signed task sheets) that the borehole was constructed to the approved design, in particular:

- Coordinates of the borehole using high precision GPS (global positioning by satellite).
- Inspection certificates for the casing and the thread dimensions if the casing sections are threaded. Inspection certificate for the weld quality of the casing centralizers.
- Measurements of borehole diameter(s) and depth.
- Casing type, length and diameter of the various casing sections. Leak test results for the casing joints and the frequency of the casing centralizers.
- Results of borehole verticality and straightness tests.
- The actual sources and mix of components used for casing grout. Volume of grout placed in the annulus.
- Geological drill logs, penetration rates, water strike depths and yields including any additional drilling information.

5.4. OPERATING THE FACILITY

The operations to be carried out at a disposal facility are described in Ref. [26]. In summary, operation consists of three main activities:

- Containerization;
- Transfer;
- Emplacement.

Radiation protection procedures for containerization follow those shown in Appendix II.

5.4.1. SRS containerization

What containerization involves

Containerization is the placing of a stainless steel capsule (containing one or more SRS) into a stainless steel disposal container which is cylindrical, 114 mm outside diameter, 250 mm long, 6 mm thick and made from type 316 L stainless steel. The purpose of containerization is to convert the conditioned SRS into a disposal package. As with conditioning, containerization will be carried out in either the conditioning unit or in the BOSS portable hot cell. At present, the BOSS design allows only one capsule per container. It is possible that this could increase in future.

As suggested by Fig. 5, containerization will probably occur just prior to disposal. For this reason, a description of how containerization will be performed is likely to form part of the site specific design. Containerization will need to be done in a controlled environment with strict quality control and quality assurance by trained and qualified personnel. For SHARS a heavily shielded facility, such as the BOSS hot cell, will be required. The site specific design will then need to describe how the facility will be constructed, commissioned and operated. For SRS with lower activities, a facility that is similar to the conditioning facility is likely to be suitable.

A disposal container will be prepared to receive its capsule by first labelling it indelibly so that it, and its contents, can be identified uniquely. Also, the concrete buffer insert is cast into place: capsules are made in two diameters and the insert will be designed to fit whichever one is appropriate. If the portable hot cell is being used, the disposal container is then posted into the cell.

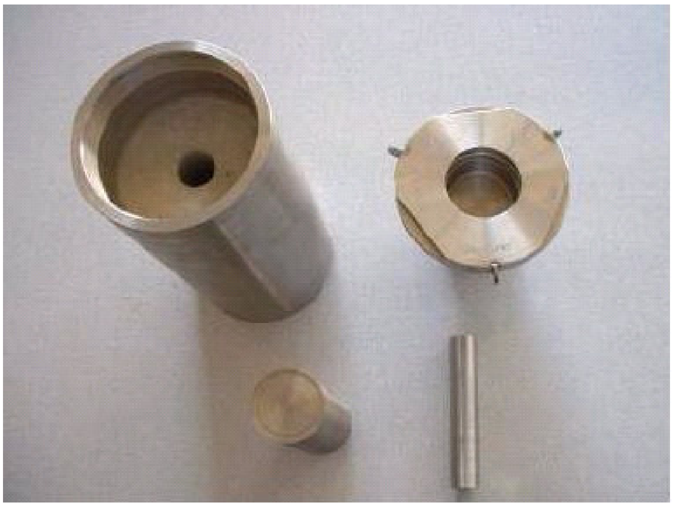

FIG. 16. Disposal container and lid (top row) and two sizes of capsule with lids in place. The scale may be judged from the diameter of the container, which is 114 mm. Note also the pre-cast concrete buffer that lines the inside of the container.

General requirements

The following guidelines should be observed when setting up a supervised area where capsules and containers are manipulated:

- Capsules should be checked for contamination immediately after removal from their temporary shields.
- The receiving area and the first transfer zone should be covered with double sheets of polyethylene so that contamination, should it occur, does not spread to the equipment. This should include covering of the lead bricks.
- Ventilation should be maintained in the receiving area/transfer zone extraction hood.
- The operational area should be made as small as practical, so that all operations may be carried out within a limited area.

- Operations should be optimized to limit the number of source manipulations.
- Radiation levels within the supervised area should be established prior and after each containerization.

Operations in the conditioning unit when it is used for containerization

While there are significant differences in the dimensions of the equipment used, the conditioning and containerization procedures are very similar so that it is expected that, with some modification, it will be possible to use the conditioning unit for both conditioning and containerization. The supervised areas for the containerization operation will be essentially the same as for conditioning (Fig. 9).

The receiving zone is the area where capsules are received and removed from their temporary storage or transport shields. It is expected that the number and content of the capsules for disposal will be well established and that there should be no possibility of contamination leaking from capsules. Nonetheless, the extraction hood will be used in the receiving area and capsules will be subjected to a smear test to confirm the absence of contamination. Workers should not be required to wear protective clothing and dust masks, though these should be available.

Transfer zone is the area where the capsule is transferred to the disposal container. Prior to the transfer, a calibrated radiation monitor will confirm (so far as possible) that the contents of the capsules are as expected. The zone where the capsule is placed into a disposal container and the zone where loaded disposal containers are temporarily placed awaiting welding are shielded so that they will not influence the monitor reading. The shielding also provides adequate protection to the worker. With the capsule in place, more concrete is added (this could be fresh concrete or a pre-cast disc) and the container lid is put in place.

The welding zone is the area where the disposal container is welded. The zone contains a special shield for welding and another area where welded containers can cool. The thickness of the disposal container will require metal inert gas (MIG) welding in which metal weld rods are consumed. Welding will be an automated process. The welded container is then allowed to cool in a designated (and shielded) area.

The leak testing zone is the zone where the welded containers are tested for any leakage. This is done by immersing the container in a bath of ethylene glycol and subjecting the bath to reduced pressure. Leaks are shown by air bubbles emerging from the container. The zone is shielded with lead bricks.

Container filling/storage zone is the area where the welded and leak-tested container will be loaded into a transfer flask or, alternatively, placed directly onto an overhead hoist. This zone is shielded with 100 mm lead bricks.

A more detailed description of containerization, as it would be carried out in the conditioning unit, is contained in Ref. [11].

Operations in the BOSS hot cell when used for containerization

It is envisaged that the operations in the BOSS hot cell will closely follow those just described. At the time of writing, however, these have yet to be developed.

Leaking capsules

For contamination to leak from a capsule requires both the SRS and the capsule to be leaking. If contamination is found on a capsule the most likely reason is that it arose from contact with contaminated material during storage. The capsule should, therefore, be cleaned and checked for leaks using the leak testing equipment. If the capsule is intact, containerization can proceed as usual. If a leaking capsule is confirmed, protective equipment will be donned. If the leaking capsule is the smaller size, it should be overpacked into the larger size capsule. If it is a larger size capsule that is leaking, this will need to be returned to the central store for removal of the SRS and re-conditioning.

Records

The following containerization records should be maintained:

- Details of the make-up of each disposal package consisting of:
  - Unique capsule number;
  - Unique disposal container number;
- Confirmation that all the steps in the procedures were completed as planned (with dates and signatures);
- Description of any non-conformances to the procedures and an explanation of the likely consequences;
- Leak test results.

5.4.2. Secondary wastes

It is considered very unlikely that secondary wastes will arise during the containerization and disposal operations because all the SRS will have been sealed into stainless steel capsules before arriving on site. It is possible, however, that secondary wastes, above clearance levels, could have been produced during conditioning and placed into storage (Section 4.1.3). If so, these will need to be disposed to the borehole. Such wastes are likely to have low levels of radioactivity such that, if a facility had been available, they would have been suitable for near surface disposal. In these circumstances, it would be reasonable to arrange for these wastes to be placed at the top of the disposal zone immediately below the anti-intrusion plate. Suitable treatment and packaging should be agreed in advance with the regulatory body. In most cases, it should be sufficient to pack the waste into a steel drum and fill the void space with concrete.

5.4.3. Transfer to borehole

Following containerization, the disposal container must be transferred to the borehole. The conditioning unit has an overhead crane that extends from the working area to the adjacent borehole. In container filling/storage zone, the disposal container can be placed on the crane and then run out of the conditioning unit so that it hangs directly over the borehole from where it can be emplaced (as in Fig. 9). This will allow a containerized SRS to be lifted quickly from its temporary shield then, with the operators standing at a distance, moved directly to the borehole and lowered into it. With this arrangement, it is expected that any source that can be containerized within the conditioning unit can be transferred to the borehole without need of a transfer flask.

The most powerful SRS that can be conditioned in the conditioning unit is a Co-60 source of 0.04 TBq (about 1 Ci). This produces a dose rate of about 50 $mSv.h^{-1}$ at 0.5 m. If the crane cannot be located over the borehole (as just described), it is clear that, to avoid excessive worker doses, a transfer flask will be needed to transfer the more powerful sources. The decision as to which sources need the transfer flask and which do not will be taken on a case-by-case basis by the RPO.

Figure 17 shows the design of the BOSS transfer/transport flask which only exists as a design at present. This is designed [26] for a 40 GBq Co-60 source. The intention is that it could also be used as a Type A transport package for transporting disposal packages to and from the disposal facility. The transfer flask is designed to be loaded from the top or the bottom. Using the gate at the bottom, the disposal container may be lowered into the borehole.

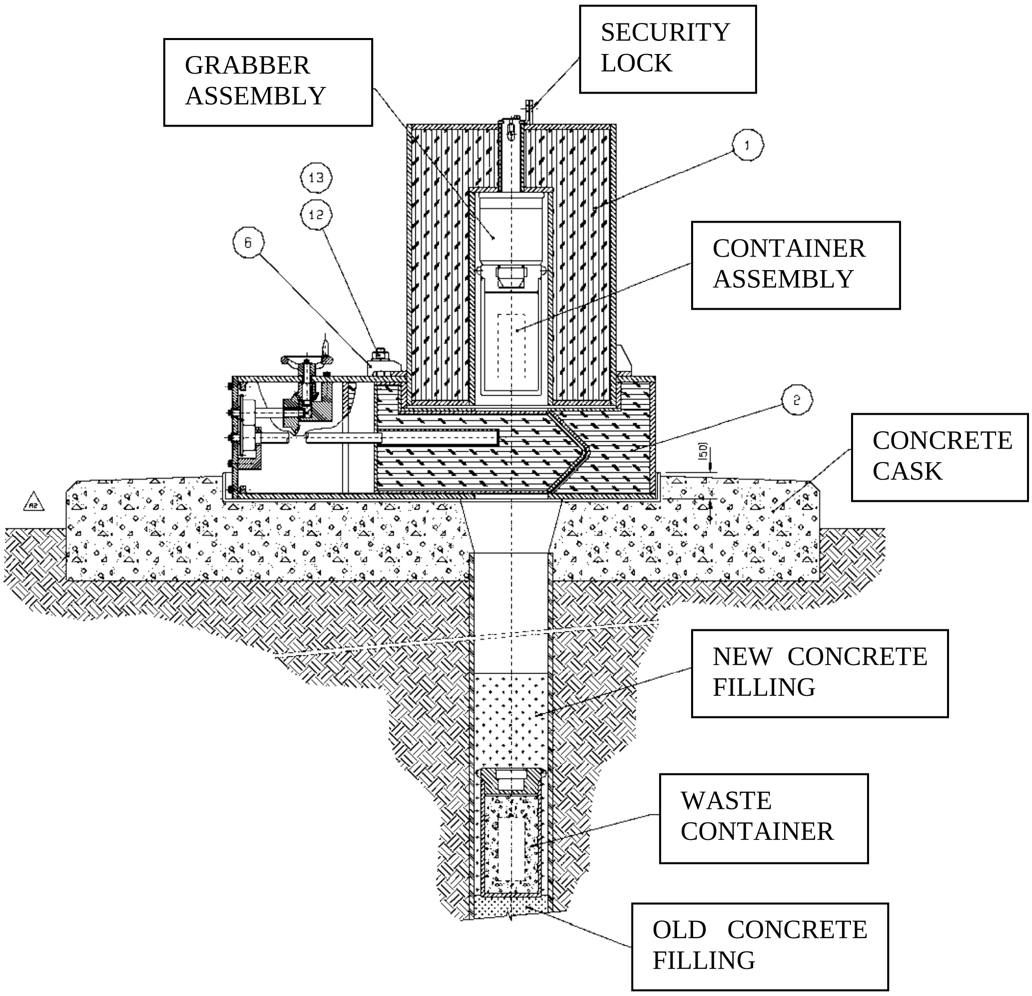

FIG. 17. BOSS transport/transfer flask [26].

When the hot cell is used for conditioning and/or containerization, it is envisaged that it will be constructed directly above the previously constructed borehole. A port in the base of the hot cell will allow the disposal container to be lowered directly into the borehole. Alternatively, the source will be transferred from the hot cell to a heavily shielded transfer flask (not yet designed) that can be moved into position above the borehole. Because of the weight of the shielding this will usually require a mechanical handling device e.g. a fork lift truck.

5.4.4. Emplacement

The number of disposal packages per borehole, their allowable radioactivity content and the depth and length of the disposal zone will be determined in advance from the site specific safety assessment and approved by the appropriate authorities.

Before the start of any emplacement action is it necessary to prepare the borehole by verifying that:

- The borehole is empty and dry;
- A dummy disposal package can be lowered to the bottom of the borehole;
- The borehole has the correct depth and diameter.

In the reference design the repeat distance between adjacent disposal packages is set at 1 m. The implications of reducing this are discussed in Section 4.7.

Information/procedures for the following are contained in Appendix V:

- Backfill preparation;
- Disposal package and backfill placement;
- Inspection and maintenance,
- A list of needed equipment;
- Records to be kept.

5.5. BOREHOLE CLOSURE, DECOMMISSIONING AND REHABILITATION

5.5.1. Closure

Removal of casing

Closure of the borehole disposal facility takes place when the last disposal (or secondary waste) package has been emplaced. The section of casing above the disposal zone or, at least, the top 30 m is to be removed. In the case of HDPE casing it may be possible to separate the sections by unscrewing them. Steel casing may require down hole cutting.

The force required to remove the top section of casing will depend on the extent to which loose material has lodged in the annulus between the casing and the borehole. This will depend, in turn, on the geology and the degree of care taken to prevent this from happening. Where the forces are likely to be high, steel casing may be advantageous.

Anti-intrusion plate

When the casing has been removed, enough grout should be placed in the borehole so that the top of the remaining casing is covered. The steel anti-intrusion plate should be fitted. This should be placed above the casing so that any drill bit hitting the plate is re-directed into the surrounding rocks. The anti-intrusion plate is rectangular with dimensions related to the inside diameter (D) of the borehole. If it has a length D and a breadth of 0.71D it should be possible to place it so that it rests at an angle of about 45° inside the borehole. A plate with a thickness of 10 mm is likely to be sufficient. The anti-intrusion plate should then be surrounded by backfill grout using a tremie pipe or similar.

Borehole sealing

With the anti-intrusion plate in place, the top section of the borehole above the disposal zone is then backfilled (sealed) with the same grout mixture until it is a few metres below ground surface. The remaining section may then be backfilled with soil.

5.5.2. Decommissioning

The decommissioning activities include the decontamination of the conditioning unit and/or the BOSS portable hot cell, the decontamination of the disposal equipment and the removal of any temporary buildings.

5.5.3. Rehabilitation

The rehabilitation activities will usually include the removal of waste/surplus materials and debris, repair of construction scars, re-levelling of the site to natural grade and restoration of natural vegetation. Depending on the post-closure requirements, the security fencing may be removed or left in place.

5.5.4. Records

A closure, decommissioning and rehabilitation report should be prepared. This should include details of the composition, properties, quantities and number of batches of grout used for sealing. It should also contain clearance certificates for the facilities and equipment showing them to be free of significant radioactive contamination.

5.6. OPTIONAL IMPLEMENTATION OF INSTITUTIONAL CONTROL

One of the main aims of geological disposal of radioactive waste is that the system should be passively safe and have no need of human intervention to maintain safety. It follows that institutional control is unnecessary. That said, some governments or regulatory bodies may require some degree of institutional control in the post-closure period. Such controls are generally separated into two types: active and passive. Active controls include monitoring of the surrounding environment for leaks of radioactivity. If such measures are to be introduced it is vital that pre-disposal (so-called baseline) levels should be measured to allow before and after comparisons. Other active controls could include the maintenance of a fence around the facility and a permanent guard. Such measures are expensive and, it is repeated, not strictly necessary.

Passive controls include restrictions on land ownership, the presence of warning markers and the lodging of information about the facility in a national or international archive. In the case of disposal boreholes, the design allows the top few metres of the borehole to be filled with natural soil. This is a security measure that is intended to make the precise borehole position difficult to find without specialized equipment.

6. QUALITY MANAGEMENT SYSTEM

The basis of any quality management system is a set of procedures that prescribe what tasks are needed to achieve a given goal, and how and by whom these tasks are to be performed. In the case of a highly regulated activity, such as radioactive waste management, the procedures are likely to be detailed and to include the keeping of records that can be used to confirm that the various tasks were completed satisfactorily, what, if any, problems occurred, and how these were resolved [31, 32].

In principle, there are two main ways of establishing a quality management system for a specific project. The first is where an already established company management system contains pre-approved procedures for all the tasks that are required by the project. This is only likely to occur where the tasks are routine and where one company does everything. More usual is the second way — through the use of project specific quality plans. Reference [11] contains a quality plan for the manufacture of the capsule and container.

6.1. QUALITY PLANS

Quality plans provide a means of achieving appropriate quality standards where the tasks are non-routine and where different companies (e.g. contractors) may be used to carry out specialist tasks. A quality plan will list all the tasks needed to complete the project and will show who is responsible for performing each task and what procedures (if any) are to be used. These procedures may belong to different companies and some may be specially written for the project. They will include requirements for record keeping at an appropriate level of detail.

The quality plan should aim to show (i) that all safety relevant activities are covered by procedures; and (ii) that the procedures are fit-for-purpose i.e. that tasks that are safety sensitive are given more weight than those that are not.

The regulatory body may carry out quality audits to demonstrate that the implementer and/or its contractors are following the prescribed procedures. Similarly, the implementer may audit its contractors. Where a company (implementer or contractor) is formally accredited to ISO 9001 (2000), this may reduce the need for independent audits.

6.2. QUALITY ASSURANCE RECORDS

The procedures should allow for the recording of key information about the disposal as described under each of the procedures outlined above. This information will need to be preserved as agreed with the regulatory body. The measures could include the sending of copies to the national archive and IAEA.

7. TRAINING

The personnel that perform the conditioning and disposal operations will need to be trained and qualified to do this. Training should cover all aspects relevant to the work to be performed and should include comprehensive radiation worker training. The person responsible for the welding must be qualified to do this but no formal welding training is needed because the welding is an automatic process and the person operating the machine only needs to know how to set it up correctly. Training may be supplied under the auspices of IAEA and would be open to candidates from countries identified as potential users of the BOSS system. Candidates would need to have received a basic scientific education so that they will be able to understand what radioactivity is, how it can be harmful and how can be measured.

The RPO should be a formally qualified and experienced health physicist with a sound knowledge of radiological protection including:

- Uses of radioactivity;
- Radiation quantities and units;
- Basic radiobiology;
- Dose calculation;
- Current standards for radiological protection;
- Dose management techniques.

APPENDIX I
SCOPES OF RELEVANT SOUTH AFRICAN DOCUMENTS

The April 2005 WATRP (waste management assessment and technical review programme) review of the Necsa Borehole Disposal Facility (presented in this technical manual under the name BOSS) examined a total of 19 documents. Of these, 14 were issued by Necsa, two were reports written for Necsa by consultants, two were reports by the South African regulatory body (NNR) and the last was a licensing guide that was also produced by NNR. Of these 19 documents, 5 were considered to be “not needed or inappropriate in [their] current form for Phase IV”. The list below shows the 14 reports that were considered to be both needed and appropriate and includes an outline of the scope of each. Also included in the list is a further document that was not reviewed (GEA-1627). Note that these documents refer to pre-conditioning and conditioning rather than the terms conditioning and containerization used in this technical manual.

GEA-1620 development philosophy

The document provides a useful summary of all the BDF project documents. The key components of the philosophy are:

- The approach will be ‘pragmatic’;
- The relevant safety requirements will be met;
- Operational safety will be ALARA.

GEA-1621 requirements for the borehole disposal facility

This is an overview of the BDF as a long term management solution for disused SRS. The text discusses, evaluates and documents a wide range of policy, regulatory, engineering, logistical and scientific issues related to setting defensible and achievable requirements for safe implementation of the BDF. As an overview document, it helps to guide interested readers to other key documents in the BDF programme that provide more specific, detailed technical information.

GEA-1622 borehole disposal facility: generic site selection and characterization

This document provides guidelines for site selection and site characterization for a borehole disposal facility so that, in conjunction with the engineered barriers, the facility will provide radiological protection that complies with international standards.

GEA-1623 design for the borehole disposal facility

This document provides a clear description of the generic or reference design and how the facility would be operated. It provides detailed information on most aspects of the design and operation.

GEA-1626 road map and procedures to implement the BDF

This document contains an implementation process (road map) for the BDF and procedures for many of the associated activities such as site selection and characterization, borehole construction, container manufacture, and source conditioning and containerization.

GEA-1627 practical demonstration and qualification

This document provides a description of the various trials and demonstration tests that were carried out. It contains useful information and photographs that help to explain some of the procedures and the choice of materials.

GEA-1628 techno-economic study

This document, referred to in the overview but not technical manual, derives an estimate of the likely cost of the BDF.

GEA-1632 generic HAZOP for borehole disposal of spent radioactive sources

This document is likely to be useful to the RPO. It presents an analysis of potential hazards associated with the BDF. Various types of deviation (e.g. ventilation failures, monitoring instrument failures, welding failures, power supply failures, etc.) are listed along with their important characteristics i.e.:

- Causes;
- Consequences;
- Ranking of consequences;
- Protecting or mitigating features;
- Actions (including recovery to a normal situation).

GEA-1633 prospective hazard assessment of a borehole disposal facility

This document describes operational activities so as to demonstrate that the doses to operators of the facility can be controlled to within acceptable limits. The estimates for some activities are compared with actual dose measurements that were taken during previous missions where radium sources were pre-conditioned.

GEA-1641 practical demonstration of the borehole disposal facility

This report briefly describes Necsa's intentions with respect to demonstrating the BDF. It is largely of historical interest and is not referred to in this technical manual.

GEA-1643 generic waste acceptance criteria for the borehole disposal facility

This document presents generic waste acceptance criteria (WAC) for the BDF, which may be used to help define site specific WAC. It provides guidelines with regard to the acceptability of materials other than those used in the generic design and associated safety assessment. The document specifically excludes SHARS and neutron sources however.

GEA-1714 waste acceptance criteria for the borehole disposal facility

This document presents generic (but detailed) waste acceptance criteria for the borehole disposal facility. It includes tables of limiting activity levels for individual waste packages based on their heat output and doses arising from operation of the conditioning unit and portable hot cell.

BDF-006-NTW NNR review of borehole disposal investigation

This document presents a review of the BDF by the South African nuclear regulator (NNR). The review aims to compare Necsa’s proposals for the BDF with international best practice. The document concludes that “The studies conducted by Necsa, including the generic safety assessment, demonstrated compliance with regulatory requirements, based on international standards”. The document is likely to be helpful to other regulatory bodies.

BDF-007-NTW guidance document on the borehole disposal facility (previously published as lg-1043, licensing guide: borehole disposal facility safety case)

This presents comprehensive regulatory body guidance to the implementer of a borehole disposal with respect to compliance with regulatory criteria and, specifically, with respect to the preparation of the safety case. The document is likely to be helpful to other regulatory bodies.

QRS-1128A-6 v2 generic post-closure safety assessment (GSA)

This document is a comprehensive generic post-closure safety assessment for the BDF. Calculations are performed for three lithological environments (arenaceous, argillaceous and crystalline) that, along with other variables, serve as a proxy for a range of hydrogeological and geochemical conditions. The container corrosion rates are based on localized corrosion of stainless steel.

QRS-1128C-1 v2 (F-GSA)

This document presents further calculations on post-closure safety and should be read in conjunction with the GSA. The effect of a wider range of corrosion rates is explored but the lowest rates used are still significantly higher than those applicable to uniform corrosion.

APPENDIX II
RADIATION PROTECTION

Segregation of work areas

The basic principle to be observed in order to avoid unnecessary occupational exposure is to handle the SRS in a suitable manner and with appropriate equipment. In order to achieve the required level of radiological protection, the work area should be classified into:

- Supervised area;
- Free area.

The supervised area is defined as one in which radiological conditions could lead to an annual effective dose of greater than 1 $mSv.a^{-1}$. This area is subjected to special safety rules described below. The free area is defined as the area in which the planned activities lead to doses that are smaller than the primary limit for the general public (i.e. 1 $mSv.a^{-1}$). The free area is surrounded by a fence to separate it from the public area.

General radiation protection within the supervised area

Inside the supervised area, the radiation protection officer (RPO) will perform the radiological survey in order to ensure safe working conditions and avoid unnecessary radiological exposure. Anyone entering the supervised area should wear personal dosimeters. Access to the supervised area should be allowed only by permission of the RPO (i.e. through a list of approved personnel). The radiation doses received by anyone inside the restricted area should be consistent with the Basic Safety Standards (BSS) [33].

The conditioning unit and the portable hot cell (if used) will be located within the supervised area.

Within the supervised area, the place where the transferral and disposal operation will be done should have at least physical barriers to avoid access by non-involved persons. The RPO will take continuous readings with the dose rate monitor while the transfer takes place to verify that the ambient dose rate levels comply with the criteria for a supervised area.

Control of contamination and surface dose rate

Radiological protection requires that surface contamination and surface dose rate of the container and (if used) the shielded flask be monitored and kept as low as reasonably achievable. External, non-fixed contamination levels for accepted packages should not exceed 4 $Bq.cm^{-2}$ for beta, gamma and low toxicity alpha emitters, and 0.4 $Bq.cm^{-2}$ for all other alpha emitters. These limits are applicable when averaged over any area of 300 cm^2 of any part of the surface.

The role of shielding is to reduce external dose rates to workers to acceptable levels. When the waste container cannot provide sufficient shielding to meet these requirements, then a shielded flask or overpack must be used.

Records

The following radiological protection records should be maintained:

- All personnel dosimetry records;
- Personal dosimeters, thermo-luminescent detectors (mostly for dosimetry of the hands);
- Names of personnel;
- Exposure period;
- Individual dose (mSv);
- Extremity dose (left and right hands separately);
- Results of monitoring for contamination and area dose rates including information on any radiological incident that occurred during the operations.

Emergency preparedness

The following events may lead to a radiological emergency during the conditioning and disposal exercises:

(a) A source is crushed during the transfer and conditioning process and contaminates the facility.
(b) During handling a source is separated from its capsule or container and falls unprotected to the floor or any other surface.

The following procedure will be followed in the event of the above or any other emergency:

(a) Evacuate the facility immediately through the nearest exit. It is very important that all activities immediately be terminated to prevent overexposure of personnel.
(b) The RPO will take measurements using a dose rate monitor to determine a safe distance from the area where the source is located or where the contamination occurs.
(c) The RPO will demarcate the problem area.

The retrieval operation should be a well planned action and every person involved will be fully informed about what is expected. In planning the retrieval operation, ALARA must be the most important consideration. The retrieval operation will be regarded as completed only after final release of the area by the RPO.

The following process will be exercised to retrieve the source:

(a) Find the right equipment for the job namely lead shield or container, tongs, alarm dosimeters.
(b) Place the shield/container as close as possible to the area where the source is located. Find the source and place it into the source container.
(c) Close the container.
(d) Monitor and release the area.

APPENDIX III
OPERATIONAL HAZARD ASSESSMENT

In support of the operational safety assessment, the hazard assessment [25] examines the worker and public dose consequences of normal operation and off-normal operations such as accidents, loss of electrical supplies etc. The work draws on previous experience with the radium conditioning programme.

Normal operations

For the purpose of estimating the likely operational dose, [25] provides a list of operations and the corresponding dose for a 40 GBq Co-60 source. This source is chosen because, although other isotopes may be present in a disposal package at higher activities (e.g. 200 GBq Cs-137), the attenuation due to shielding is much less for Co-60 due to its intense, high energy gamma radiation. 40 GBq is the maximum Co-60 activity allowed in a disposal container.

The operations considered and the corresponding worker doses are:

- Conditioning 1 020 µSv;
- Containerization 230 µSv;
- Transfer to borehole 9 µSv;
- Emplacement 63 µSv.

This produces a total dose of 1 300 µSv. The conditioning doses can be compared with those observed in the field during conditioning of 1.9 GBq Cs-137, 0.268 GBq Co-60 and 3.87 GBq Ra-226 sources during a mission to Ethiopia. The doses that the four workers accumulated during the pre-conditioning of these sources are: 19, 23, 35 and 75 µSv. (The worker that received the 75 µSv was involved in an unforeseen incident in which a 0.1 GBq Ra-226 needle had to be retrieved manually by sifting through a spade full of soil.) The set of calculations in Ref. [25] was repeated for 1.9 GBq Cs-137, 0.268 GBq Co-60 and 3.87 GBq Ra-226. A dose calculation for conditioning of these sources added up to 81 µSv. If only routine conditioning activities are considered, it can be seen that the estimate is conservative by a factor of 2 to 3.

The calculations suggest that, for the complete sequence of four operations, the dose could be as high as 6.5 mSv if five such canisters are treated during a single mission and all the operations are performed by the same worker. This indicates that, even on a conservative basis, a single mission will produce acceptable worker doses. Workers might be expected to perform multiple missions in a single year on the basis of actual dose measurements.

Off-normal operations

The generic HAZOP study [24] identified conceivable deviations from normal operating conditions, the likely causes of such deviations, their consequences and the mitigating measures (if any) put in place to avoid these deviations. The prospective hazard assessment [25] estimated the radiological consequences of these off-normal operations. The off-normal situations considered (and their reference numbers) are copied into Table III-1. Some additional explanation is offered in square brackets.

Admitting the possibility that some SRS may be damaged, the SRS to be disposed of will, for the most part, be sealed. As the conditioning and containerization processes progress, the containment and the shielding of the sources will increase and no operational accidents are foreseen in which the public can be affected by a release of radioactivity. The operating deviations that could affect the worker dose are listed as Items 1.5, 1.9, 2.6 and 2.10 in Table III-1. These could lead to the worker being contaminated. For this to happen not only must the SRS be damaged but it must be in such a form as to allow re-suspension. Other reasons for increased doses could be that the worker is working without personal radiation protection equipment (PPE), or the extraction system is malfunctioning, and both the worker and RPO have failed to notice this. The combined probability of this happening is low, but to reduce it further the extraction system shall be made the subject of an operational and technical specification (OTS), as it is a system relied on for safety.

An additional possibility is that listed as Item 1.14 in Table III-1. This is a situation in which an SRS is dropped and breaks on impact. The probability of such an event is estimated as 1 in 1 000 based on a 0.01 probability that the source will be dropped and 0.1 that it will break. Most of the sources are such that, if the worker cleans this up, the doses will be similar to the 75 µSv that the worker received in Ethiopia while searching for a Ra-226 needle of approximately 0.1 GBq. Appendix A of Ref. [8] shows that less than 10% of the sources in the list of typical African country sources have activities of more than 1 GBq. This brings the probability that this could occur with a 40 GBq Co-60 source (the highest that could be accepted in the conditioning unit) down to 1 in 10 000.

The radiological consequences of such an event may be estimated from the dose that the worker received in Ethiopia under accident conditions. The worker received 75 µSv when a 0.1 GBq Ra-226 source was retrieved. It is conservatively assumed that the source that is dropped is a 40 GBq Co-60 source and, even more conservatively, that no special measures are taken to reduce the worker dose. The dose is:

$$75\ \mu\text{Sv X } 1.6 \text{ X } 400 \text{ i.e. } 48\ \text{mSv}$$

where

1.6 is the dose ratio between Ra-226 and Co-60 (see Table 1 of Ref. [25])

400 is the concentration ratio between 0.1 and 40 GBq.

Acknowledging the conservatisms and the probability of 1 in 10 000, this is acceptable.

All the other deviations in the prospective hazard assessment (1.1, 1.8, 1.11, 2.1, 2.9, 2.15, 3.1, 3.2 and 3.4) are effectively controlled with a radiation protection programme. An RPO shall be present during operations.

TABLE III-1. POSSIBLE DEVIATIONS FROM IDEAL OPERATION IDENTIFIED BY THE PROSPECTIVE HAZARD ASSESSMENT [25]

1	Conditioning
1.1a	No measurement by RPO on instrument or container to verify the activity of the source
1.1b	The capsule is not welded
1.1c	The capsule is not leak tested
1.1d	Sources not put in capsule
1.2	No electrical power
1.3	No control to welding operation
1.4	The welder does not rotate the disposal container during the welding operation
1.5	No extraction
1.6	Pressure not low enough during leak test
1.7	Welding current too high
1.8	The radioactivity of the source(s) is too high
1.9	The airflow through the hood is too low
1.10	The activity is too low
1.11	There may be more sources than anticipated
1.12	The activity in the capsule is incorrectly calculated
1.13	Inadequate time allowed for cooling of the capsule
1.14	Loss of containment (of source)
2	Containerization
2.1a	The disposal container is not topped up with cement
2.1b	The disposal container is not welded
2.1c	The disposal container is not leak tested
2.1d	RPO does not measure the dose rate on the disposal container
2.2	No Argon flow to the welder
2.3	No electrical power
2.4	No control to welding operation
2.5	The welder does not rotate the disposal container during the welding operation
2.6	No extraction through the extraction hood
2.7	Pressure not low enough during leak test
2.8	Welding current too high
2.9	The dose rate on the disposal container is too high
2.10	The airflow through the hood is too low
2.11	Temperature achieved by the welder is too low
2.12	The activity [per container] is too low
2.13	The welding may be interrupted
2.14	The filtration is incomplete
2.15	More capsules, than anticipated, may need to be disposed of
2.16a	The incorrect disposal container material may be used
2.16b	The incorrect cement may be used
2.17	Inadequate time allowed for cooling of the disposal container
3	Transfer
3.1	RPO does not measure the dose rate on the disposal container
3.2	The bottom gate of the transfer flask is not closed before it is lifted
3.3	More of [i.e. too high] pressure [in transfer flask pneumatic system]
3.4	The transfer may be interrupted
4	Emplacement
4.1a	Pre-checks on borehole not done
4.2	No compressed air available
4.3	No power
4.4	Too much rain
4.5	Pressure in hose [grab mechanism] too low
4.6	Power interrupted when the disposal container is lowered down the borehole
4.7	The next disposal container is lowered down the borehole before the last one has been capped
4.8	Cement not set when next container lowered
4.9	The container may be damaged when it falls down the borehole

APPENDIX IV
SAFETY REPORT TEMPLATE

This is a template for the preparation of a safety report that can be submitted to the regulatory body (or to international peer review supporting the competent national regulator) in order to obtain a licence for implementation. It is styled as a safety report which also contains the main elements of an environmental impact assessment (for cases where an EIA is required by national policy).

General information

- Purpose and context of this report;
- Why the BDF option was chosen and the alternatives considered;
- General facility description;
- Inventory of waste for disposal and the waste acceptance criteria;
- Regulatory framework: operation and post-closure (international, if no national regulations exist);
- Arrangements for interfacing with the regulatory body.

Safety strategy and safety case

- Approach to building the safety case;
- Preliminary statement of the safety case;
- Safety functions: multibarrier principle;
- Timescale appropriate to hazard.

Site characteristics

- Acceptable and unacceptable characteristics of a site;
- Location, geography, population;
- Environment, land use;
- Climate and climate history;
- Geomorphology and landform history;
- Geology: rock formations, structure, tectonics;
- Natural resources;
- Surface water hydrology;
- Hydrogeological properties;
- Hydrogeochemical properties;
- Target formation properties.

Design and construction of borehole

- Principal design features;
- Construction methods and equipment;
- Checking and acceptance of completed borehole;
- Quality control procedures.

Design and construction of disposal package components

- Principal design features (capsule, container, fill material);
- Construction methods and materials;

- Closure method;
- Quality control procedures.

Design of conditioning facilities[1]

- Principal operational design features;
- Principal safety and environmental protection features.

Operation of conditioning facility

- Transport of DSRS to the conditioning facility;
- Receipt of DSRS;
- Handling and interim storage of DSRS;
- Encapsulation and packaging for disposal;
- Management plan for secondary wastes;
- Environmental monitoring procedures;
- Decontamination procedures;
- Quality control procedures.

Operation of disposal borehole

- Preparation of borehole for initial disposal;
- Disposal package transfer and emplacement;
- Borehole cement matrix emplacement;
- Removal of borehole casing;
- Filling and closure of borehole;
- Quality control procedures;
- Site operational security;
- Controlling access to the conditioning and disposal areas.

Site closure plan

- Site stabilization procedures;
- Environmental restoration of site after borehole closure.

Non-radiological environmental impacts of operation

- Impacts of site preparation;
- Impacts of site investigations;
- Impacts of drilling;
- Impacts of waste transport to the site.

Occupational radiation protection

- Exposure pathways;
- Radiation protection features and procedures;

[1] Necsa conditioning unit or units (e.g. second unit for SHARS) could be used. If a national conditioning facility is built, then this section would need to be extended to describe construction.

- Estimated occupational exposures;
- Emergency procedures;
- Radiation protection programme.

Exposure of the public during operations

- Exposure pathways and expected doses from transport of sources to the site;
- Exposure pathways and expected doses to members of the public in the vicinity of conditioning and disposal operations;
- Accident scenarios and recovery measures.

Post-closure safety of the disposal facility

- Exposure scenarios and pathways from the GSA;
- Characteristics of the site in relation to those in the GSA;
- Calculated individual radiation exposures for selected scenarios and site specific conditions;
- Uncertainties;
- Compliance with national (or international) standards.

Post-closure control

- Plan for institutional control, if any.

Record keeping

- Record of site location;
- Record of DSRS conditioned and disposed;
- Record of secondary wastes produced and their disposal;
- Record of design, operation history and closure method for borehole;
- Record of non-conformances to QA procedures and responses made;
- Procedure for post-closure care of records.

Final statement

- Statement of overall environmental impacts;
- Statement of overall radiological safety performance;
- Statement of confidence in the safety case.

Annexes

- All the procedures to be undertaken;
- References to supporting documentation.

APPENDIX V
PROCEDURES/INFORMATION RELATING TO WASTE AND BACKFILL EMPLACEMENT

Backfill preparation

The reference value for the repeat distance between adjacent disposal packages is set at 1 m. This, the inside diameter of the casing and the disposal package dimensions determine the volume of backfill that needs to be added. This will need to be calculated for each site specific design. The backfill grout has four constituents: a standard, widely available flyash cement CEM II B-V 32.5N (roughly 70% Portland cement, 30% flyash), a cement additive (Conbex) for improved flow properties, water and river sand. Table V-I shows the amount of each component required to make 1 m^3. See also below and Section 7 of Ref. [11]. The water that is used should be free from impurities, especially chloride and sulphate. Water that is drinkable will suffice.

TABLE V-I. BACKFILL GROUT MIXTURE

Materials	**Quantities per m^3**			**Quantities for 1-bag mix**	
	By mass		**By volume**	**By mass**	**By volume**
	Aggregates dry	**Aggregates damp**	**aggregates damp & sand bulked**	**aggregates damp**	**Aggregates damp & sand bulked**
Cement: CEM II B-V 32,5N	810 kg	810 kg	16,2 Bag	50 kg	1 bag
River sand	825 kg	865 kg	0,68 m³	53 kg	42 ℓ
Conbex 208	4050 g	4050 g	4050 g	250 g	250 g
Water	405 ℓ (gross)	365 ℓ (net)	365 ℓ (net)	23 ℓ (net)	23 ℓ (net)

1) Bulking and moisture content should be checked on site. The net water is variable and depends on the moisture content of the aggregates.
2) For consistence a flow time of 34 seconds is recommended (based on ASTM Designation C939-97).

Disposal package and backfill placement

After the borehole and backfill preparation, the emplacement proceeds as follows:

Enough backfill grout to just envelop a disposal package is delivered to the base of the borehole using a remotely operated hopper. The disposal package is lowered into the wet grout and released when it reaches the previous measured depth. The grabber is removed. With the first batch of grout still uncured, a second batch of backfill grout of pre-determined volume is added so as to produce the required distance between the disposal packages and form a platform on which the next disposal package will stand. The identity of the disposal package and its position in the borehole is recorded. The grout is then left to cure for 24 hours or, at least until the concrete can take the weight of the next container. This waiting period largely determines the length of the disposal campaign (one container per day) so that it would be beneficial to examine ways in which it might be reduced. As explained in Section 5.3.7, before emplacing the next disposal package the dummy disposal package is lowered to ensure that the borehole still has no obstructions. This done, the next disposal package can be emplaced in the same manner as before.

Inspection and maintenance

All the lifting, handling and emplacement equipment needs to be inspected and tested before the emplacement of a disposal package. Dimension X (Fig. V-I) must be measured over the ball of the grabber before each emplacement. Minimum allowed value is 55 mm. If it is smaller than 55 mm it will require the piston of the grabber be replaced.

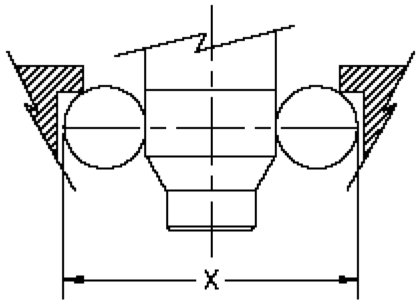

FIG. V-I. Grabber piston and stainless steel balls showing the measurement X.

Equipment needed

Radiation protection

- Portable dose rate meter.

Transfer

- Transfer flask (if required);
- Lifting and handling device, capacity 2 000 kg if transfer cask is used;
- Equipment to transport a transport cask;
- Grabber and lifting equipment to handle the disposal container.

Disposal

- Cement mixer;
- Lifting and handling device, capacity 500 kg with minimum cable travel and length corresponding to the required borehole depth;
- Cement pump and length of tremie pipe corresponding to the required borehole depth;
- Grabber;
- Dummy disposal canister;
- Downhole cement hopper;
- Nitrogen for grabber;
- Flow cone for backfill material.

Consumables

A minimum of 1.8 m^3 backfill materials is required for the backfill of a borehole of 140 mm diameter and 100 m deep; this allows for 20% wastage and requires:

- Cement: CEM II B-V 32,5N – 30 bags of 50 kg;
- Conbex 208: 7 290 gm;
- Water: 657 L;
- 1.3 m^3 sand containing 5% moisture and bulks 20%. It is recommended that natural river sand be used and that it is as close as possible to the grading shown in Table V-2. Screening, blending and washing of the sands are options that can be followed to get the grading closer to the recommended values. If the proposed consumables are not available, it will be necessary to seek advice on an alternative.

TABLE V-2. RIVER SAND GRADING

Sieve analysis (SABS method 829:1994)[14]	
Sieve size (mm)	**Percentage passing by mass**
6,70	100
4,75	99
2,360	93
1,180	73
0,600	39
0,300	12
0,150	4
0,075	2,2
Fineness modulus	2,8

Records

- Inspection reports of lifting and handling equipment;
- Inspection reports listing measurement X (Fig. V-1) measured before each emplacement;
- Report showing the depth of each disposal package with reference number in the borehole;
- Report on the composition and flow properties of backfill material used.

APPENDIX VI
PROJECT MANAGEMENT PLAN

About this document

IAEA's BOSS system (borehole disposal of sealed radioactive sources) has been developed to allow the safe management of disused sealed radioactive sources (SRS). The concept provides for the conditioning, storage and disposal of disused SRS in specially created and engineered boreholes. To support the Member States in implementing BOSS, the IAEA is developing a suite of generic (not site specific) technical reports. As shown in Figure VI-1, these consist of:

- An overview of the concept;
- A technical manual that explains how the borehole disposal system would be implemented on a site specific basis;
- A project management plan (PMP, this document);
- A specific safety guide that provides a safety assessment template — published in parallel within Safety Report Series (see also Appendix IV of the technical manual).

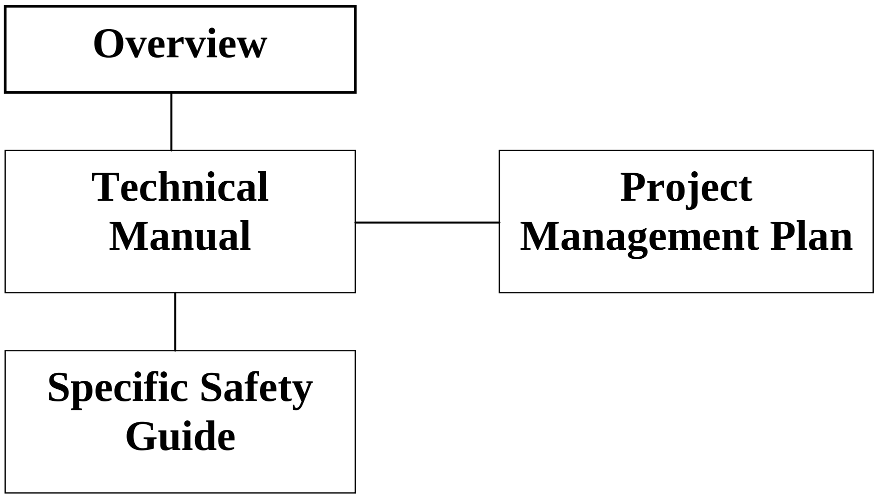

FIG. VI-1. Structure of the BOSS documentation.

The technical manual presents a BOSS road map — a sequence of activities that would end with safe disposal of disused SRS. One of the early activities is the construction of a project management plan that will be used to manage the project so that it delivers to time, budget and quality and, in so doing, will support the licensing process. This present document provides a template for such a BOSS project management plan (PMP) consisting of five main sections:

- Introduction and background;
- Goals;
- Management plan;
- Quality programme;
- Project control;
- Appendices (if needed).

The content of each section is described in turn.

1. INTRODUCTION

The Introduction should describe the background to the project e.g. a brief description of the concept and an explanation of why it was considered an appropriate technology in this case.

2. GOAL

Under this heading the PMP should provide a high level mission statement such as:

"The project will safely and securely implement all the necessary activities leading to a BOSS type disposal of all the disused sealed radioactive sources currently held in our country."

3. MANAGEMENT PLAN

This chapter should describe the major activities that will be performed to organize and manage a BOSS project as detailed below.

Project activities

For a BOSS project it is suggested that the major activities will be:

- Project control;
- Pre-disposal of disused sealed sources;
- Repository preparation;
- Regulator interactions;
- Construction at disposal site;
- Operation of disposal site;
- Site decommissioning.

These seven areas correspond to the activities of the work breakdown structure.

Work breakdown structure

A work breakdown structure (WBS) is simply a means of listing and structuring the many tasks that will need to be performed to complete a project. To help organize the project, the WBS has a hierarchical structure as illustrated for a BOSS project in Fig. VI-2. The WBS should link groups of activities to persons or organizations so as to promote ownership, responsibility and accountability and allow the project the project to take advantage of the enthusiasm and creativity of its participants.

In the WBS presented here the three hierarchical levels are given the following names: highest level — activity; intermediate level — task; lowest level — sub-task. Clearly, other names could be used and an actual BOSS WBS might well consist of more than three levels.

1. Project control
    a. *Project management*
    b. *Quality assurance*

2. Pre-disposal of disused sealed sources
    a. *Construction/ provision of storage and conditioning facilities*
    b. *Collection*
    c. *Transport*
    d. *Inventory*
    e. *Conditioning*
    f. *Storage*

3. Repository preparation
    a. *Site selection*
        i. Selection criteria
        ii. Site search
        iii. Confirmatory site characterisation
    b. *Design*
    c. *Safety assessment*
        i. Operational
        ii. Post closure
        iii. Transport
        iv. Conventional safety

4. Regulator interactions
    a. *Environmental impact study*
    b. *Pre-disposal licence submission*
    c. *Disposal licence submission*

5. Construction at Disposal Site
    a. *Site infrastructure*
        i. Hot cell for disposal
        ii. Utilities, Security Etc
    b. *Boreholes*

6. Operation of disposal site
    a. *Containerisation*
    b. *Emplacement*
    c. *Closure of borehole*

7. Site decommissioning

FIG. VI-2. Suggested work breakdown structure for BOSS implementation (three highest levels only).

Participants

This section of the PMP should list the participants who will perform the activities described in the WBS. These may be organized into teams with a team leader. Some teams may consist of only one member. Clearly, team members will need to have the qualifications and experience necessary to perform the required tasks.

External agencies, e.g. regulatory body, IAEA, are not participants in the project. Nonetheless, their activities, insofar as they relate to the project (e.g. regulatory approvals), need to be included in the planning and responsibilities for interfacing with the regulatory body should be clearly defined. Supervising ministries should be regarded as the sponsor.

Roles and responsibilities

It is most important that the project has a single leader to whom all participants report. This may be the project manager, for example. This arrangement is especially important when participants are contributed from different organizations. The project is implemented through a series of teams responsible (following the WBS example shown previously) for:

(i) Project management;
(ii) Pre-disposal;
(iii) Repository preparation;
(iv) Regulatory body interactions;
(v) Construction;
(vi) Operation;
(vii) Site decommissioning.

This section of the PMP should describe the roles and responsibilities of the teams and individual members with specific responsibilities such as the team leaders, radiation protection officer and so on.

Each team leader will be responsible for planning (including setting milestones), costing, budgeting and reporting their part of the project. The project manager will collate this information so as to provide a report on the complete project for the benefit of the sponsoring organization. The project manager will also be responsible for:

- Providing project vision;
- Securing the budget and allocating and distributing it to each activity;
- Coordination with IAEA and external experts;
- Coordinating project activities;
- Establishing whole project milestones;
- Establishing an effective communication system.

Communication

All decisions, agreements, approvals and results of all project communication must be documented. Changes to the project plan (i.e. to the schedule, the budget or the required level of quality) for individual tasks or sub-tasks should be communicated to the project manager. Changes to the project plan must be recorded and communicated to all participants. Where the changes are significant enough to affect the overall timescale or cost of the project this must be agreed with higher authority (usually with the sponsor).

The goals of the project and scope of each activity, task and sub-task (including specified times, budget, and quality) must be written and communicated to all relevant participants. Written reports on each activity must be submitted to the project manager for review on a regular basis (usually monthly). The report must provide the current status of the activity (see Section 5.5). Successful completion of work schedules should be acknowledged and celebrated.

Conflict resolution

The PMP should specify how conflicts are to be resolved. For example, where a disagreement occurs between members at the same level, it should say that the issue is to be taken to the next highest level for resolution. It is important not to let the project stall because of the lack of a decision. Often, any decision is better than no decision because, if a wrong decision is taken, it will usually be possible to recognize this quickly and rectify it.

4. QUALITY MANAGEMENT PROGRAMME

The successful and safe implementation of BOSS can only be achieved with an appropriate quality management programme. For example, disposal containers must be manufactured under a comprehensive quality management regime that extends from material procurement through manufacture, sealing and leak testing. A similar approach needs to be defined and implemented for all the other activities to ensure the eventual safe implementation of BOSS. The quality management programme forms an integral part of the project, and is also an essential part of the licence submission. The ISO9001 (2000) system is strongly recommended as the basis for the quality management programme.

Quality statement

A PMP should include a statement that expresses commitment to deliver project milestones to an agreed level of quality. The latter should be integrated into the quality management system (QMS) of the implementing organization, if such a system exists. If the implementing organization does not have a QMS, it will be necessary to develop a quality management programme specific to the project.

Scope of the quality management programme

This section of the PMP should provide an overview of the quality management programme. The quality management programme comprises a specific plan to meet the quality goals of BOSS. Such a programme should begin with an overarching statement of the overall mission or goal of the project, as suggested in Section 2. The detailed quality requirements are then derived from consideration of the desired outputs of the various activities that need to be carried out to achieve this objective.

For example the desired outcome of the collection of SRS could be that the SRS are safely stored in a specific location (or locations), and that all known information about them (e.g. identification number, manufacturer, application in which it was used, location from where it was collected, nuclide, activity, physical size, source of information, name of person who compiled the record) have been verified and recorded in a prescribed format. The quality management programme should ensure that these goals have been met through the signing off of appropriate documentation by personnel responsible for these actions. In general the allocation of responsibility and accountability to specific post holders for successful completion of tasks is a key component of any QMS or quality management programme.

As a further example, a desired output of the borehole construction would be that a borehole of the specified diameter has been drilled at the chosen location, and that a casing made of the specified material has been inserted in the prescribed manner. In this case, the quality management programme could, for example, be designed to provide documentary proof that a drilling contractor with the ability to perform the work has been contracted, that he has received clear instructions, that inspections have been carried out by responsible personnel to ensure that the instructions are followed and that the dimensions of the final product have been verified to be in accordance with the specifications.

It is important to realize that quality management programmes will vary in both scope and depth. They should, however, be designed to suit specific project requirements, and in general, the more important the issue the more detailed the programme becomes. Useful information on quality management systems for pre-disposal and disposal of radioactive waste are contained in two IAEA Safety Guides [32, 33].

The document management system is an important element that needs to be fully integrated into the quality management programme. Documents relevant to the BOSS implementation as well as their distribution information should be kept. These will eventually form a comprehensive record for future reference. Important documents i.e. those describing the location, contents, design, safety analysis, monitoring data and the licensing of the facility should be securely archived at the national library (or similar institution) and the IAEA.

5. PROJECT CONTROLS

Project controls must be implemented to insure that the project's objectives are met. Project controls can include the division of a BOSS project into a set of activities (as discussed in Section 3.2), the development of a schedule for those activities, the development of a cost estimate, the identification of milestones, the development of a quality management programme and the implementation of project management activities. Project management controls include the tracking and reporting of the milestones and the budget.

Good project management software is available at low cost. This will be of considerable help in scheduling, budget control and reporting on project progress.

Schedule

The PMP must include a schedule of project activities. A schedule is a document that lists: the name or description of each activity, the date the activity should start, the duration of the activity, the order of the performance of the activity, and the dependency of the activity (i.e. the relationship of the activity to other scheduled activities). The schedule can be linked to the budget (with each work activity having a cost estimate) and the schedule should include the identification of the critical path. The critical path is the sequence of activities that bound or limit the overall duration of the project, in other words, those activities that will immediately delay the project if they are not performed as planned. One of the advantages of project management software is that it allows the critical path to be identified quite simply.

For example, the schedule might include:

- The name of the activity (e.g. geophysical investigations, Task 32);
- The duration of the activity (e.g. 180 days);
- The order of performance of the activity, (e.g. geophysical investigations is scheduled to be performed after the desk studies, Task 31);

- The dependency of the activity (e.g. the geophysical investigations cannot begin until the desk studies have been licensed).

Figure VI-3 shows a type of schedule called a tracking Gantt Chart as applied to the two highest levels of the WBS (i.e. activities and tasks). This uses a bar to show planned duration and marks that identify the start date, the finish date and milestones.

Milestones

The PMP should identify the project milestones. Milestones are markers taken from the schedule selected so that they occur at one- or two-month intervals. They play an important role in allowing the measurement of the progress of the project, being checkpoints on the road to successful completion of the project. Milestones frequently include the completion of a set of related activities. They are selected from appropriate portions of the schedule so the manager can detect scheduling problems early, enabling positive corrective action to be taken in time to effect change.

Milestones may also be used to celebrate the completion of a major activity such as site selection or licensing.

Financial management

The PMP must include a cost estimate. A cost estimate is a document that lists the anticipated costs of scheduled activities. The cost estimate can be linked to the schedule. The cost estimate provides an estimate of the cost of each activity, the total cost of the project, the anticipated amount of money needed each month (the spend profile) and the anticipated yearly cost. Sponsors usually require a schedule and cost estimate as justification for providing funding. The funding that is actually provided is usually called a budget. The budget is allocated to activities, tasks and sub-tasks, and the actual spending is compared to the budgeted spending. Financial procedures and reporting should also respect national accountancy rules.

A typical financial report would be a table showing future projected costs and actual historic costs throughout the project. This is usually updated on a monthly basis. Often, the current financial year is shown on a month by month basis while previous and future years may be reduced to a single entry. As well as showing how the expenditure changes with time, the table may also show it broken down into categories, such as different types of labour supplied by the controlling organization and different contractors. Figure VI-4 shows a hypothetical example of a monthly financial report for a BOSS project.

Project management

Projects must be actively controlled and tracked to determine whether the project is going as planned. The following are basic steps in controlling a project:

- Evaluation of the progress, by comparing actual completion of activities (as indicated by milestones) and the actual spending, to the scheduled set of activities and the budgeted spending;
- Identifying and analysing any variances between planned and actual progress/spending;
- Determining the impact of variances and any necessary corrective actions.

Reporting

Reporting is an important method of communicating project activities to sponsors, team members and others and the PMP will need to specify this in detail. Reporting is usually in the form of a written report, prepared monthly or quarterly. Such reports are not discursive but, rather, tend to be fixed format with boxes to be filled in. Items that will certainly be included are the completion of milestones, spending, variances and corrective actions taken. Progress reports may also describe what activities are scheduled to be performed next.

6. APPENDICES

Appendices could include such things as an organization chart, contact details for staff, the project quality programme, detailed work breakdown structure, detailed Gantt Chart and so on.

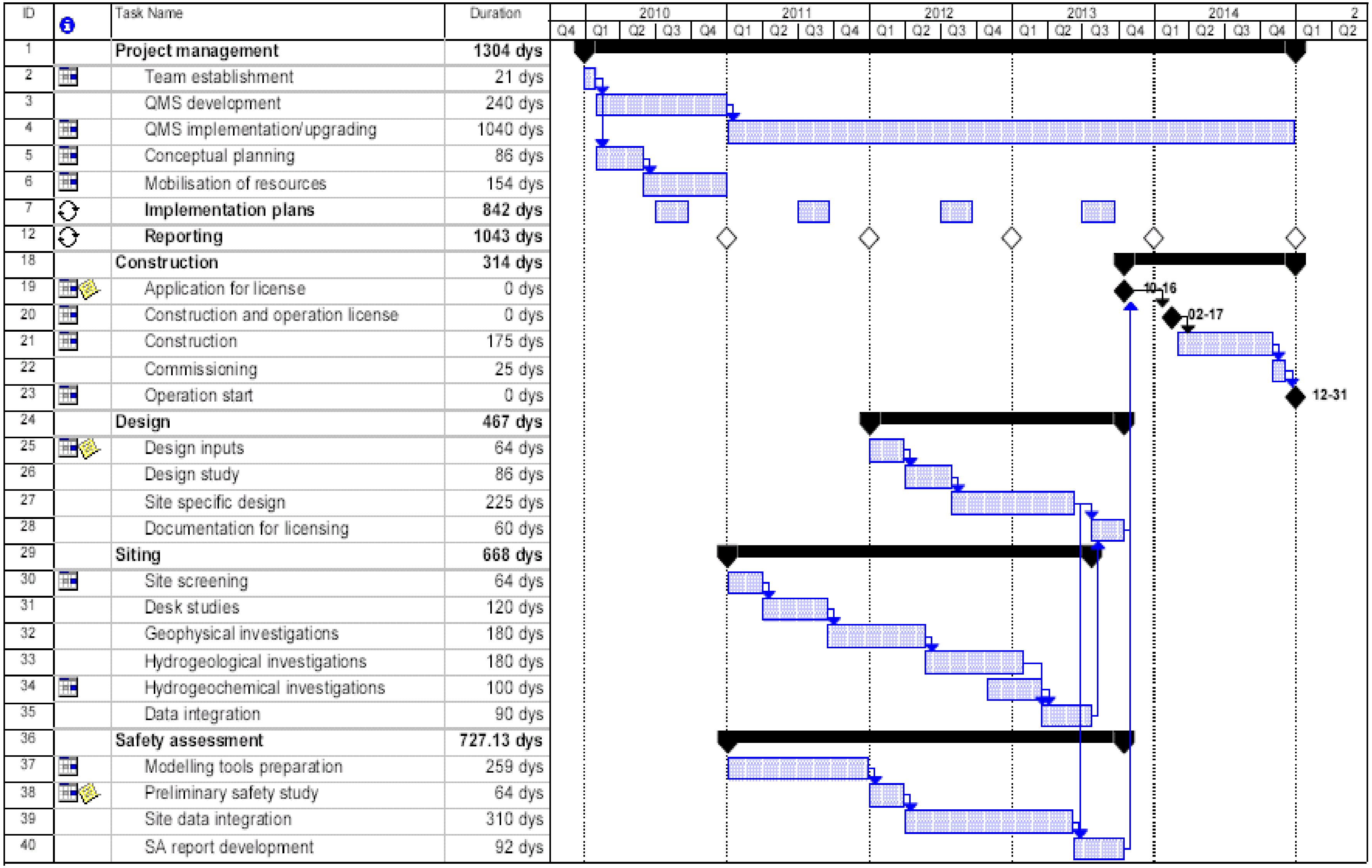

FIG. VI-3. Example Gantt Chart of BOSS development process.

BOSS PROJECT COST PROJECTIONS				DATE OF ESTIMATE: April 2010						PROJECT MANAGER					
	2009						2010						2010	2011	All
External suppliers		Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec	Jan	Feb	Mar	Total		years
Transport contractor	20													0	0	20
Site characterization	30	10	10	10	10									40	0	70
Licensing submissions	20	5	5	5	5	5	5							30	20	70
Drilling contractor									25	15	15	15	15	85		85
Decommissioning														0	15	15
Licensing fees	10							10						10	10	30
TOTAL EXTERNAL k$	80	15	15	15	15	5	5	10	25	15	15	15	15	165	45	290
Internal suppliers																
Supervision	20	2	1	2	2	1	2	1	2	2	1	2	2	20	20	60
Radiological protection	10	1	1	1	1	1	1	1	1	1	1	1	1	12	15	37
Skilled labour	30	3	2	3	2	2	3	3	2	2	2	3	3	30	30	90
General labour	10	1	1	1	1	1	1	1	1	1	1	1	1	12	10	32
TOTAL INTERNAL k$	70	7	5	7	6	5	7	6	6	6	5	7	7	74	75	219
TOTAL COSTS k$	150	22	20	22	21	10	12	16	31	21	20	22	22	239	120	509

FIG. VI-4. Hypothetical example of a monthly spending report.

APPENDIX VII
RADIONUCLIDE ACTIVITY LIMITS (WASTE ACCEPTANCE CRITERIA) FOR CONDITIONING UNIT AND PORTABLE HOT CELL

Permissible activity levels for the lightly shielded conditioning facility. The table considers heat generation and each type of radiation separately (heat generation in Column 3, alpha, beta, gamma and neutron radiation in Columns 4 to 7, respectively). Short lived radioactive progeny are assumed to be in secular equilibrium with the parent. NA=not applicable. Shaded cells show the limiting values.

Nuclide	Decay/ radiation type (includes progeny)	Activity to produce 50 W Bq	Activity to produce 10 $mSv.h^{-1}$ alpha at 1.2 m Bq	Activity to produce 10 $mSv.h^{-1}$ beta at 1.2 m Bq	Activity to produce 10 $mSv.h^{-1}$ gamma at 1.2 m Bq	Activity to produce 10 $mSv.h^{-1}$ neutron at 1.2 m Bq
H-3	β	5E+16	NA	4E+14	NA	NA
Be-10	β	2E+15	NA	2E+13	NA	NA
Na-22	γ e p	1E+14	NA	2E+13	8E+10	NA
Mn-54	EC γ e	4E+14	NA	NA	1E+11	NA
Fe-55	EC γ e	6E+16	NA	NA	2E+12	NA
Co-57	EC γ e	2E+15	NA	NA	6E+11	NA
Co-60	β γ	1E+14	NA	6E+13	4E+10	NA
Ni-63	EC β	2E+16	NA	1E+15	NA	NA
Zn-65	EC γ e pr	5E+14	NA	4E+15	2E+11	NA
Se-75	EC β γ	8E+14	NA	NA	3E+11	NA
Kr-85	β γ	1E+15	NA	1E+13	5E+13	NA
Y-88	EC γ e p	1E+14	NA	5E+15	4E+10	NA
Sr-90	β γ	3E+14	NA	2E+12	4E+14	NA
Cd-109	EC γ e	3E+15	NA	NA	3E+13	NA
I-129	β γ e	4E+15	NA	3E+14	2E+13	NA
Ba-133	EC γ e	7E+14	NA	NA	3E+11	NA
Cs-137	β γ e	4E+14	NA	2E+13	2E+11	NA
Pm-147	β γ	5E+15	NA	1E+14	3E+16	NA
Sm-151	β γ e	2E+16	NA	1E+15	3E+14	NA
Eu-152	EC β γ e	2E+14	NA	3E+13	6E+10	NA
Gd-153	EC γ e	2E+15	NA	NA	7E+11	NA
Ir-192	β γ e	3E+14	NA	2E+13	4E+10	NA
Au-195	EC γ e	2E+15	NA	NA	5E+11	NA
Tl-204	EC β γ e	1E+15	NA	1E+13	1E+13	NA
Pb-210	α β γ e	5E+13	1E+17	5E+12	4E+13	NA
Po-210	α γ	6E+13	1E+17	NA	1E+16	NA
Ra-226	α β γ e	1E+13	1E+17	2E+12	8E+10	NA
Pu-238-Be	α γ e	6E+13	1E+17	NA	2E+15	1E+13
Pu-239-Be	α γ e	6E+13	1E+17	NA	2E+15	1E+13
Am-241-Be	α γ e	6E+13	1E+17	NA	4E+12	1E+13

Permissible activity levels when using the portable hot cell. The table considers heat generation and each type of radiation separately (heat generation in Column 3, alpha, beta, gamma and neutron radiation in Columns 4 to 7, respectively). Short lived radioactive progeny are assumed to be in secular equilibrium with the parent (see Table 1). NA= not applicable. Shaded cells show the limiting values.

Nuclide	Radiation type	Activity to produce 50 W Bq	Activity to produce 0.12 mSv.h-1 alpha at 1.8 m (shielded) Bq	Activity to produce 0.12 mSv.h-1 beta at 1.8 m (shielded) Bq	Activity to produce 0.12 mSv.h-1 gamma at 1.8 m (shielded) Bq	Activity to produce 0.12 mSv.h-1 neutron at 1.8 m (shielded) Bq
H-3	β	5E+16	NA	>1E+17	NA	NA
Be-10	β	2E+15	NA	>1E+17	NA	NA
Na-22	γ e p	1E+14	NA	>1E+17	8E+13	NA
Mn-54	EC γ e	4E+14	NA	NA	1E+15	NA
Fe-55	EC γ e	6E+16	NA	NA	>1E+17	NA
Co-57	EC γ e	2E+15	NA	NA	>1E+17	NA
Co-60	β γ	1E+14	NA	>1E+17	4E+13	NA
Ni-63	EC β	2E+16	NA	>1E+17	NA	NA
Zn-65	EC γ e pr	5E+14	NA	>1E+17	6E+14	NA
Se-75	EC β γ	8E+14	NA	NA	>1E+17	NA
Kr-85	β γ	1E+15	NA	>1E+17	>1E+17	NA
Y-88	EC γ e p	1E+14	NA	>1E+17	7E+12	NA
Sr-90	β γ	3E+14	NA	6E+15	4E+16	NA
Cd-109	EC γ e	3E+15	NA	NA	>1E+17	NA
I-129	β γ e	4E+15	NA	>1E+17	>1E+17	NA
Ba-133	EC γ e	7E+14	NA	NA	5E+11	NA
Cs-137	β γ e	4E+14	NA	>1E+17	1E+16	NA
Pm-147	β γ	5E+15	NA	>1E+17	>1E+17	NA
Sm-151	β γ e	2E+16	NA	>1E+17	>1E+17	NA
Eu-152	EC β γ e	2E+14	NA	>1E+17	1E+14	NA
Gd-153	EC γ e	2E+15	NA	NA	>1E+17	NA
Ir-192	β γ e	3E+14	NA	>1E+17	4E+16	NA
Au-195	EC γ e	2E+15	NA	NA	>1E+17	NA
Tl-204	EC β γ e	1E+15	NA	>1E+17	>1E+17	NA
Pb-210	α β γ e	5E+13	>1E+17	>1E+17	>1E+17	NA
Po-210	α γ	6E+13	>1E+17	NA	>1E+17	NA
Ra-226	α β γ e	1E+13	>1E+17	>1E+17	3E+13	NA
Pu-238-Be	α γ e	6E+13	>1E+17	NA	>1E+17	2E+14
Pu-239-Be	α γ e	6E+13	>1E+17	NA	>1E+17	2E+14
Am-241-Be	α γ e	6E+13	>1E+17	NA	>1E+17	2E+14

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