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CRAFT

The International Project on Complementary Safety Reports (2011–2014)

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CRAFT

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CRAFT

THE INTERNATIONAL PROJECT ON COMPLEMENTARY SAFETY REPORTS (2011–2014)

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2022

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FOREWORD

The International Project on Complementary Safety Reports: Development and Application to Waste Management Facilities (CRAFT) was developed to assist in illustrating the application of the graded approach to safety cases.

The objectives of the CRAFT project were to apply the methodology set out in IAEA Safety Standards Series No. GSG-3, The Safety Case and Safety Assessment for the Predisposal Management of Radioactive Waste, to representative predisposal radioactive waste management facilities and activities; to oversee the development of complementary reports illustrating the application of the methodology; to provide a forum to support its use and application for the safety case and supporting safety assessment; and to share experiences and identify lessons learned.

This publication presents the results of the CRAFT project. It provides input that can supplement current IAEA safety standards to address the demonstration of safety for facilities and activities associated with the predisposal management of radioactive waste.

The IAEA is grateful to the project participants and consultants for their contributions in drafting and reviewing this publication, in particular S. Virsek (Slovenia), A. Smetnik, D. Murlis (Russian Federation), F. Ledroit (France), C. Drobnewski (Germany) and M. Sneve (Norway). The IAEA officers responsible for this publication were M. Kinker and A. Guskov of the Division of Radiation, Transport and Waste Safety.

CONTENTS

1. INTRODUCTION

1.1. PROJECT DESCRIPTION

1.1.1. Project background

IAEA Safety Standards Series No. GSG-3, The Safety Case and Safety Assessment for the Predisposal Management of Radioactive Waste [1], published in 2013, provides recommendations for the development and review of the safety case and supporting safety assessment for facilities and activities dealing with the predisposal management of radioactive waste and spent fuel storage facilities. GSG-3 covers all waste management facilities and activities, which are varied in nature, size and complexity, and have different hazards associated with them, both from operational states and from accident conditions. Whilst there are many similarities in the approach and methodology used in the demonstration of safety, GSG-3 emphasizes the importance of ensuring that the extent and complexity of the assessment is commensurate with the nature of the activity or facility and its associated risk (graded approach). When the IAEA presented the draft of GSG-3 at the 29th meeting of the Waste Safety Standards Committee (WASSC) for approval in June 2010, during discussions it was proposed that the use of the graded approach be illustrated through the development of supporting safety reports for a range of facilities.

In October 2014, the IAEA organized the final plenary meeting of the International Project on Safety Assessment Driving Radioactive Waste Management Solutions (SADRWMS), which had been launched in 2004 to examine international approaches to safety assessment for predisposal management of radioactive waste. The results of the SADRWMS project were:

- Input to a harmonized version of GSG-3 [1] that includes the SADRWMS framework and flowcharts;
- IAEA-TECDOC-1777 [2] that describes the methodology for safety assessment of predisposal radioactive waste management activities developed under the SADRWMS project;
- The SAFRAN software tool [3] for applying the SADRWMS methodology to safety assessment for predisposal management of radioactive waste.

During the final SADRWMS plenary meeting, it was agreed that the completion of complimentary reports illustrating the use and application of GSG-3 [1] methodology and the SAFRAN tool [3] would be a significant part of the follow-up project. To oversee the development of these complimentary reports it was agreed to establish the International Project on Complementary Safety Reports: Development and Application to Waste Management Facilities (CRAFT). The objectives of the CRAFT project were:

- To apply the GSG-3 methodology and SAFRAN tool to representative radioactive waste management facilities and activities;
- To oversee the development of complementary safety reports illustrating the application of GSG-3 methodology and SAFRAN tool;
- To provide a forum for supporting the use and application of the GSG-3 methodology and SAFRAN tool for the safety case and safety assessment;
- To share experiences and identify lessons learned.

Exchange of information among the Member States was fostered by participating in the various CRAFT project meetings and through dissemination of the material developed during the project. The project considered that predisposal radioactive waste management facilities and activities are varied in nature, size and complexity, and have different hazards associated with them. Furthermore, a radioactive waste management facility or activity could be one of several facilities or activities on a site and might be independent of the other facilities, might be connected to other facilities or might be an integral part of a larger facility.

The results of the CRAFT project illustrate the application of the IAEA safety standards by providing foundation material to clarify requirements related to the safety case and safety assessment for the predisposal management of radioactive waste. This publication aims to support expert missions, training events, and peer reviews carried out under the IAEA's Technical Cooperation Fund.

1.1.2. Project organization

The CRAFT project was implemented through technical meetings and the working groups activities performed between the plenaries. The first meeting of the 3-year project was held in May 2011 and was attended by 30 specialists from 23 countries. The outcomes of the SADRWMS project (methodology report [2] and SAFRAN software tool [3]) and the results of the SADRWMS test cases (Thailand Institute of Nuclear Technology and Studsvik facilities) were reviewed; the Terms of Reference for the CRAFT project were developed, and the following working groups were formed:

- Storage Facility Application Case;
- RADON-Type[1] Facility Application Case;
- Regulatory Working Group.

The working groups collaborated on the development of the application cases for the IAEA illustrative report to complement GSG-3 [1], focusing on the national illustrative examples of those countries that had shown an interest in hosting an application case.

Materials developed by the working groups were reported and discussed at annual plenary meetings in 2012 and 2013. The final (fourth) plenary meeting of the International CRAFT project was held in October 2014 at the IAEA's Headquarters in Vienna, Austria. The meeting was attended by 18 specialists from 12 countries. Based on the review of the status of the CRAFT project and work that had been performed by the working groups, a work plan for finalization of the illustrative report and preparation of this report for publication was developed, and two consultancy meetings took place in 2016 and 2017.

1.1.3. Project participation

The CRAFT project was open to professionals from Member States who undertook technical activities related to safety assessment or predisposal management of radioactive waste. Participants represented regulatory bodies, facility operators, technical support organizations, and research organizations. They contributed actively to the project by participating in technical discussions, applying methodologies to real problems, and taking part in the development of

[1] The RADON-type facilities took their name from the RADON system that was established in the former Soviet Union for collecting, transportation, processing and near surface disposal of low and intermediate level institutional radioactive waste including disused sealed radioactive sources (DSRS).

the complementary safety reports. Participants were able to engage themselves in CRAFT working groups. In addition, during the topical sessions of coordinating meetings, they had the opportunity to give oral or poster presentations describing the safety assessment related work they had undertaken within their own national programmes or related projects.

1.2. OBJECTIVE AND SCOPE

The objective of this publication is to illustrate the demonstration of safety in the predisposal management of radioactive waste using the methodology outlined in GSG-3 and using the SAFRAN tool. The secondary objectives are, for certain predisposal radioactive waste management facilities and activities, to highlight the key components of the safety case and supporting safety assessments within the context of predisposal waste management, to describe what is needed in the way of safety justification for establishing the context and contents of the safety case and safety assessment, and to explain the implementation of the GSG-3 methodology and the use of the SAFRAN tool.

The scope of this publication covers the results of the CRAFT project. Specifically, it addresses the development of illustrative safety cases for the storage of low and intermediate level radioactive waste (LILW) at a dedicated/centralized storage facility, as well as for the retrieval of LILW from RADON-type facilities.

This publication does not intend to provide detailed guidance for the safety case for any type of predisposal waste management facility or activity. Instead, this publication identifies possible ways of addressing the methodology presented in GSG-3 [1] and highlights where differences might occur between facilities or activities.

1.3. STRUCTURE

Section 1 provides an introduction to the CRAFT project, as well as the objective and scope of this publication. Section 2 describes the work carried out by the application working groups within the CRAFT project. Section 3 presents the main outcomes of the application working groups. Section 4 identifies the lessons learned by the application working groups during the development of their illustrative safety cases.

The publication includes one appendix and two annexes. The Appendix deals with the use of the graded approach during the evolution of the safety case. The illustrative safety cases developed within the working groups are provided as Annexes to this publication. Annex I provides the illustrative safety case for the centralized storage facility for LILW in Slovenia. Annex II provides the illustrative safety case for the retrieval of radioactive waste from legacy RADON-type facilities typically encountered in countries that were formerly part of the Soviet Union. The SAFRAN files related to the two annexes are provided in the online supplementary files which accompany this publication.

2. SAFETY CASE APPLICATION STUDIES IN THE CRAFT PROJECT

The safety case is the collection of scientific, technical, administrative and managerial arguments and evidence in support of the safety of a facility or activity, covering the suitability of the site and location and the design, construction and operation of the facility, the assessment of radiation risks and assurance of the adequacy and quality of all of the safety related work. The safety case and supporting safety assessment provide the basis for demonstration of safety and for licensing; they evolve with the development of the facility or activity, and assist and guide decisions on siting, location, design and operations. The safety case will also be the main basis on which dialogue with interested parties will be conducted and on which confidence in the safety of the facility or activity will be developed.

The IAEA has set out a framework of internationally agreed standards for demonstration of safety of the predisposal management of radioactive waste:

- IAEA Safety Standards Series No. GSR Part 5, Predisposal Management of Radioactive Waste [4], establishes requirements for the predisposal management of radioactive waste, including the preparation, scope and documentation of the safety case and supporting safety assessment.

- IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), Safety Assessment for Facilities and Activities [5], establishes the generally applicable requirements to be fulfilled in safety assessment for facilities and activities, with special attention paid to defence in depth, quantitative analyses and the application of a graded approach to the ranges of facilities and of activities that are addressed.

- GSG-3 [1] provides recommendations on the components, development, and other aspects to be considered in the safety case and supporting safety assessment for facilities and activities dealing with predisposal waste management. The components of the safety case, indicated in Fig. 1, include the context; safety strategy; facility description; safety assessment; limits, controls and conditions; iteration and design optimization; uncertainty management; and integration of safety arguments. Guidance on these components is provided in Sections 4 and 7 of GSG-3.

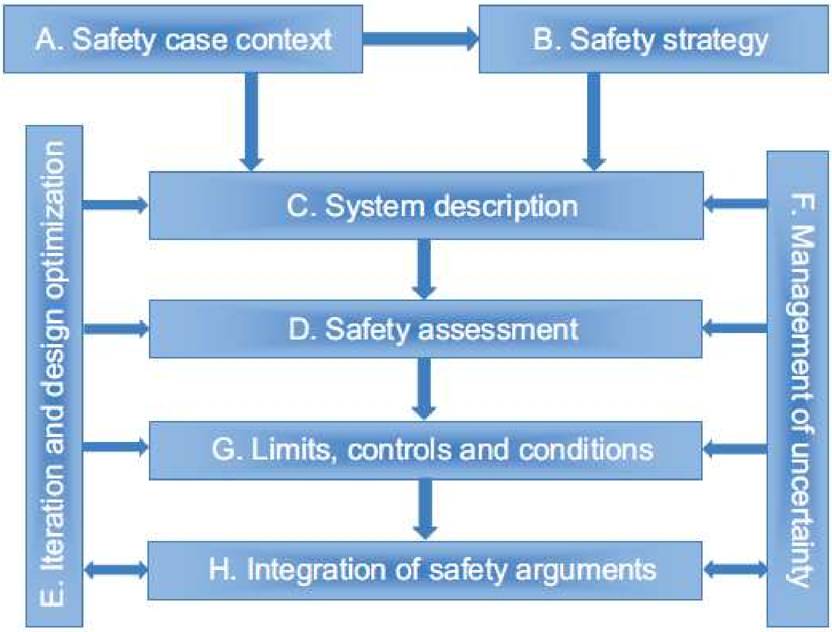

FIG. 1. Components of the safety case (adapted from GSG-3 [1]).

Safety assessment is the main component of the safety case and involves assessment of a number of aspects as illustrated in Fig. 2. The fundamental element of the safety assessment is the assessment of the radiological impact on humans and the environment in terms of both radiation dose and radiation risks. The other important aspects are site and engineering aspects, operational safety, non-radiological impacts and the management system.

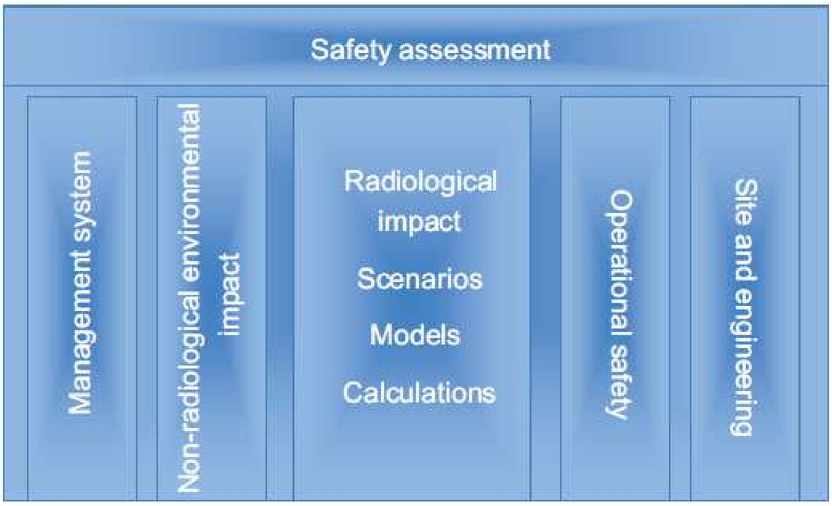

FIG. 2. Aspects included in the safety assessment (adapted from GSG-3 [1]).

GSG-3 [1] acknowledges that the extent and complexity of the safety case and supporting safety assessment will differ according to the facility or activity, and will also evolve through its lifetime (e.g. construction, commissioning, operation). In view of these considerations, a graded approach is applied to the development and review of the safety case and supporting safety assessment.

In order to address the general guidance for safety cases provided in GSG-3 and taking into consideration issues that are facility and/or activity specific, the CRAFT application working groups applied GSG-3 and the SADRWMS methodology [2] to the development of safety cases for two types of existing facilities. Sections 2.1–2.3 present a short description of the activities undertaken by the three working groups.

2.1. STORAGE FACILITY WORKING GROUP

2.1.1. Context

In many countries, the storage of radioactive waste is performed in a dedicated facility which is designed to store radioactive waste (including disused sealed radioactive sources (DSRS)) generated within the country. Such radioactive waste might be processed or unprocessed, and might also be packaged and unpackaged, with storage periods lasting up to several decades. As outlined in para. 1.4 of IAEA Safety Standards Series No. WS-G-6.1, Storage of Radioactive Waste [6], the reasons for storing radioactive waste and DSRS at these facilities can include inter alia the following:

(a) "To allow for the decay of short lived radionuclides to a level at which the radioactive waste can be released from regulatory control (clearance) or authorized for discharge, or recycling and reuse;

(b) To collect and accumulate a sufficient amount of radioactive waste prior to its transfer to another facility for treatment and conditioning;

(c) To collect and accumulate a sufficient amount of radioactive waste prior to its disposal."

To ensure the safety of storage of radioactive waste and DSRS, the radioactive waste to be stored needs to be properly characterized, treated and conditioned for the type of storage envisioned and taking into consideration the characteristics of the storage facility. Requirements for the types and characteristics of radioactive waste that can be accepted for storage are typically included in the waste acceptance criteria (WAC) for the storage facility, which is derived from the safety case that has been developed for the facility.

2.1.2. Objectives and approach of the working group

The objective of the CRAFT Storage Facilities Working Group was to apply the GSG-3 [1] methodology and the SAFRAN tool [3] to existing facilities for the storage of radioactive waste.

The facility considered in the development of the illustrative safety case was the Central Storage Facility in Slovenia, which is currently in operation for the storage of institutional radioactive waste generated during research activities, medicine and industrial uses.

The Storage Working Group structured the illustrative safety case following the template provided in Section 4.1 of GSG-3 [1]. In order to develop the illustrative safety case, the working group undertook the following tasks:

Task 1 – Assemble and collate input data (e.g. site, facility, activities, waste streams).

Task 2 – Develop safety assessment:

a. Input facility structure information in SAFRAN utilizing facility specific data;
b. Define normal operations;

c. Define and agree on the postulated initiating events (PIEs) that could lead to accident scenarios;
d. Identify the accident scenarios (derived from the PIEs) for quantitative analysis;
e. Complete data entry into SAFRAN utilizing information from above tasks;
f. Perform analysis of normal operations and accident scenarios (SAFRAN);
g. Review the safety assessment (facility-specific SAFRAN file) to verify consistency with the methodology in GSG-3;
h. Review the safety assessment (facility-specific SAFRAN file) and modify as required, e.g. based on findings from review.

Task 3 – Incorporate pertinent information from the safety assessment into the safety case for the facility.

Task 4 – Review the safety case and modify as required, e.g. based on findings from reviews.

2.2. RADON-TYPE FACILITY WORKING GROUP

2.2.1. Context

Historical radioactive waste. The RADON-type facilities took their name from the RADON system that was established in the late 1950s by the former Soviet Union for collecting, transportation, processing and near surface disposal of low and intermediate level institutional RW including DSRS generated outside of the nuclear fuel cycle. These facilities were constructed in various regions of the former Soviet Union according to a standard design, with specific modifications to address local conditions of the storage locations and predicted volumes of radioactive waste.

In the context of this working 'group, 'historical radioactive waste' is considered to be radioactive waste that was disposed of in accordance with national regulations that were in place at that time, but which do not meet current requirements for characterization programmes or quality management systems. Key characteristics of historical radioactive waste are:

- Incomplete or improper characterization/treatment (waste streams might be mixed, and might be conditioned, partially treated, or raw);
- Poor or no information/traceability (cannot conclusively identify characteristics or originating process or location);
- The quality management system did not cover the whole lifetime of the waste at the time of its generation and does not meet current requirements for addressing the whole lifetime.

Several countries (e.g. Belarus, Bulgaria, China, Germany, Lithuania, Russian Federation, Ukraine, United Kingdom, United States of America) currently have disposal facilities for historical radioactive waste which were designed and constructed before management systems and WAC were established. These facilities no longer meet current requirements for the safe disposal of radioactive waste and, consequently, many countries are in the process of retrieving these wastes.

Legacy facilities and sites. Because historic activities were typically related to the radium industry, uranium mining, and/or military programmes, there are numerous sites that contain, or are contaminated with, radioactive materials. Disposal facilities for historical radioactive waste can be considered as legacy facilities.

Some legacy disposal facilities have never been licensed for disposal; some of those that had been originally licensed no longer meet international safety standards and even national requirements. Due to such non-compliance, some of these disposal facilities are now considered 'storage facilities' and operators of these facilities currently face serious problems in the recovery of the radioactive waste and remediation of the site. In some cases, unfavourable conditions inside the facilities have caused corrosion and degradation of the waste packages and the engineered barriers. In many cases, inadequate conditions have resulted in disappearance of original markings, labels, and signs that could help identify the origin and characteristics of the waste; this is further compounded by the lack of adequate records or record keeping. Accumulation of water within the vaults is often identified and, depending on the radionuclide content and specific activity, this water can often be classified as liquid radioactive waste. In some cases, radionuclides migrate into the vicinity of the sites.

Decision making for retrieval. This topic is addressed in Technical Reports Series No. 456, Retrieval and Conditioning of Solid Radioactive Waste from Old Facilities [7]; relevant paragraphs are quoted below:

> *"Safety assessments and environmental measurements have demonstrated that some of these repositories may represent an unacceptable risk or hazard to the environment, workers and the public, therefore requiring remediation actions.*
>
> *Similarly, some old interim storage facilities contain waste and waste containers that have deteriorated, or the general storage conditions no longer meet the requirements for safety. Again, this indicates a need for remediation of the facilities. In some cases, inadequate waste storage practices continue to be applied, due to:*
>
> *(a) A lack of appropriate knowledge and practical experience in radioactive waste management in general;*
> *(b) A lack of appropriate technologies for waste processing (treatment and conditioning);*
> *(c) A lack of well defined requirements for waste quality and acceptance criteria for long term storage or disposal;*
> *(d) Inadequate storage or disposal conditions, and unacceptable impact of external conditions on waste and waste packages;*
> *(e) Poor quality of waste forms, waste containers or other engineered barriers;*
> *(f) Storage or disposal of waste in its original form and without appropriate packaging.*
>
> *A decision to retrieve radioactive waste from some old storage or disposal facilities could be made if the present status of safety and security does not correspond to current standards or requirements, or if the existing social, political or economic situation requires such remediation actions. The cost of waste retrieval and facility or site remediation — both in terms of radiation exposure and financial expenditures resulting from the remediation — is normally justified by the improved safety and security of the facility or site after remediation, the availability of the facility or site for other purposes, etc. In all steps of waste retrieval and site remediation, safety of the staff, protection of the environment and waste security should be given the highest priorities."*

2.2.2. Objectives and approach of the working group

The main objective of the RADON Working Group was to apply the methodology presented in GSG-3 [1] and the SAFRAN tool [3] for retrieval of radioactive wastes from a typical near surface RADON-type facility. The secondary objective is to support decision making for planned operational waste retrieval operations and predisposal radioactive waste management activities at a near surface historical disposal facility for solid institutional radioactive waste.

In developing their illustrative safety case, the RADON Working Group followed a similar approach as the Storage Working Group; their illustrative safety case was structured following the template provided in Section 4.1 of GSG-3 [1].

The application case demonstrates the application of the GSG-3 methodology and SAFRAN tool to assess activities and technologies for waste retrieval procedures as well as other decision making related to the operation of the RADON-type facility.

2.3. REGULATORY WORKING GROUP

The Regulatory Working Group provided guidance and support to the other working groups during the drafting process of their reports. Discussions focused on the application of the graded approach as presented in GSG-3 [1] during the different stages of the lifetime of facilities for predisposal waste management (from site selection to operation and decommissioning). The results of these discussions are summarized in the Appendix.

3. MAIN OUTPUTS OF THE CRAFT PROJECT

The main outcomes of the CRAFT project are the illustrative safety cases and supporting safety assessments developed by the working groups. These illustrative reports followed the structure outlined in Section 4.1 of GSG-3 [1], which was agreed and refined during Technical Meetings of the CRAFT project.

3.1. STRUCTURE OF THE ILLUSTRATIVE SAFETY CASE REPORTS

For preparation of the illustrative safety case reports a template was developed and was later used by the working groups. The template considers the guidance given in GSG-3 [1] as well as the specific elements that would need to be addressed during the preparation of a safety case for the centralized storage of radioactive waste as well as for the retrieval of radioactive waste from legacy facilities.

A. Safety case context:

- Purpose of the safety case;
- Scope of the safety case;
- Demonstration of safety;
- Graded approach.

B. Safety strategy

C. Description of facility or activity and waste:

- Site conditions;
- Facilities and activities;
- Inventory of radioactive waste.

D. Safety assessment:

- Radiological impact assessment;
- Site and engineering aspects:
  - Engineering analysis;
  - Passive safety;
  - Defence in depth;
  - Scientific and technical / engineering principles;
  - Quality of the site characterization.
- Operational safety aspects;
- Non-radiological environmental impact;
- Management systems.

E. Management of uncertainties

F. Iteration and design optimization

G. Identification of safety measures

H. Limits, controls and conditions

I. Integration of safety arguments:

- Comparison against safety criteria;
- Plans for addressing unresolved issues.

J. Interacting processes

3.2. SAFETY CASE FOR A STORAGE FACILITY

The safety case report developed for the storage facility application case is provided in Annex 1. The SAFRAN file which captures the safety assessment for the storage facility safety case is provided in an online supplementary file which accompanies this publication.

3.3. SAFETY CASE FOR A RADON-TYPE FACILITY

The RADON-type Working Group developed two documents: the illustrative safety case for retrieval activities, and guidance that can be considered in the future development of such safety cases. This set of recommendations is included in Section 4.2 of this publication.

The safety case report developed for the RADON-type facility application case is provided in Annex 2. The SAFRAN file which captures the safety assessment for the RADON-type facility safety case is provided in an online supplementary file which accompanies this publication.

4. LESSONS LEARNED BY THE WORKING GROUPS DURING THE DEVELOPMENT OF THEIR ILLUSTRATIVE SAFETY CASES

4.1. STORAGE FACILITY WORKING GROUP

While preparing the safety case for the Slovenian Central Storage Facility (CSF) and adopting the GSG-3 [1] methodology for the safety case, some lessons learned were identified and are addressed in Sections 4.1.1–4.1.6 below.

4.1.1. Graded approach

Under Slovenian law, the CSF meets the criteria to be classified as a nuclear facility (Slovenia also operates a nuclear power plant). Licensing requirements in Slovenia for nuclear facilities refer to the use of the graded approach, but do not give specific guidance on their practical application. Prior to the development of the safety case for the CSF, the Slovenian Agency for Radioactive Waste Management (ARAO) communicated with the regulatory body about their application of the graded approach for the development of the safety case.

4.1.2. Strategy for safety

When the safety case was being developed for the CSF, the strategy for safety was discussed with the regulator and was identified as a useful tool for communicating with the regulator.

The safety functions of the various structures, systems and components (SSCs) of the predisposal waste management facility need to be defined and addressed in order to demonstrate that the defence in depth concept is adequately implemented for the facility.

4.1.3. Description of the facility or activity and the waste

It is necessary to understand the site, the facility and the waste and their interdependencies at a detailed level in order to ensure that the safety assessment is aligned with the lifetime stage of the facility and the purpose of the safety case.

Building design and construction records and information form an important part of the records for the facility. The CSF is an existing facility that met general building construction standards at the time that it was constructed (in 1986). However, some building standards were either not in place or were not descriptive (e.g. seismic building codes at that time only required that structures be “seismically safe”). Other original facility design information was missing. During the preparation of the safety case for the CSF, additional investigation activities were necessary in order to enable ARAO to understand the characteristics of the facility. Records that could have helped ARAO to understand decisions that were made in the past about the building design were also missing.

The inventory of radioactive waste and DSRS needs to be verified and characterization of the waste (waste streams) needs to be focused on reducing uncertainties in the results of the assessment. While ARAO already had records and information regarding the radioactive waste and DSRS in storage at the CSF, further characterization and verification activities were needed in order to enable ARAO to prepare the safety case.

ARAO will need to consider aligning the WAC for the CSF with the anticipated future disposal of the waste. As no disposal facility exists, then acceptance requirements can only be anticipated.

4.1.4. Safety assessment

Two methods were used to identify scenarios for normal operation as well as accident scenarios:

- Hazard and operability study (HAZOP) method;
- Screening of PIEs listed in Annex I of GSG-3 [1] and the SAFRAN tool [3].

The results from using the HAZOP method and the screening of PIEs were compared and it was found that both methods resulted in a comprehensive identification of scenarios covering anticipated (normal) operations as well as accident scenarios. Both methods require a team of experts from different backgrounds who understand the scope and the objectives of the safety case.

The HAZOP method resulted in the identification of scenarios that are not specifically within the scope of safety assessment (e.g. terrorist attack, stealing of a package); these will need to be assessed through a security assessment.

Safety assessment is more than "just" calculating potential doses to the workers and the public; it requires the developer to understand the methodology used to calculate the doses and to correctly interpret the results.

The focus of the safety case development is typically on the radiological impact assessment; other aspects (e.g. site and engineering aspect, engineering analysis, non-radiological environmental impact) need to be included.

4.1.5. Management of uncertainties

The goal of ARAO in managing uncertainties in their safety case was to increase the confidence in the safety of the facility and activities, as well as enabling the developer to show compliance with regulatory requirements. It is important to develop an approach for managing uncertainties in the safety case that ensures that they are identified, assessed, and reduced where possible. This has to be taken into consideration during the development of the safety case and subsequent (independent and regulatory) reviews.

4.1.6. Iteration and design optimization

Typically, periodic safety reviews are performed every ten years. In the case of the CSF, it was determined that iterations of the safety case need to be performed more frequently, in order to prevent the loss of data and knowledge caused by staff changeover, as well as changes in knowledge, methods and computer tools.

It is important to optimize and upgrade the facility during the operational lifetime stage. In the case of the CSF, the safety of the facility was improved (specifically, the risk of a fire inside the facility was reduced) by replacing the wooden pallets used to stack waste containers with metal pallets (and other upgrades).

4.2. RADON-TYPE FACILITY WORKING GROUP

4.2.1. Framework for specific guidance

While developing the RADON-type facility illustrative safety case using the GSG-3 methodology and the SAFRAN tool (which utilized the SADRWMS methodology), it was recognized that there is a need for specific guidance regarding waste retrieval operations from historical/legacy facilities. A framework for such specific guidance is presented in Sections 4.2.1.1–4.2.1.9 below.

4.2.1.1. Context of the safety case

In the case of historical waste facilities, a safety case might not have been developed at earlier stages and might now only be performed for the first time to support decision making (e.g. whether to retrieve the waste from the facility or to improve the safety of facility). The safety case for decision making can differ from the safety case that is developed for licensing of the facility. In general, the safety of waste retrieval operations is assessed and demonstrated either within the operational licensing or within specific licensing requirements specified in national regulations. Of special importance are the application of management systems for ensuring the quality of all safety related work, and arrangements to facilitate the involvement of interested parties in the development and use of the safety case.

In the case where a safety case is already in place for the facility, there might still be uncertainties due to the lack of information (due to data being lost or unrecorded) if it was not required according to the former safety regulations. In this case, a complete review would need to be performed against the latest national regulations and international recommendations.

4.2.1.2. Purpose of the safety case

Retrieval activities from historical waste facilities might be a part of the preparation for facility decommissioning or part of another activity aimed to improve safety. As such, it is important that the purpose of the safety case is clearly established; examples include:

- Testing of initial ideas for safety concepts;
- Demonstration of the safety of the facility or activity;
- Optimization of the activity arrangements;
- Evaluation of clearance and discharge activities;
- Assessment of the maximum inventory of waste that can be managed, or secondary waste generated as a result of the waste retrieval;
- Definition or revision of limits, controls and conditions.

4.2.1.3. Scope of the safety case

The scope of the safety case might be limited to retrieval of waste from the facility, or it might include other activities such as preparatory investigations (e.g. of the engineered barriers and/or of the waste inventory) and post-retrieval processing and conditioning. At a minimum, the scope of the safety case for retrieval of waste from RADON-type facilities will include the following:

- Retrieval of the waste or waste packages from historical storage or disposal units;
- Waste characterization;
- Packaging, repackaging or overpacking of retrieved waste and interim storage at the site (e.g. temporary holding of the waste pending the next steps).

4.2.1.4. Use of a graded approach

Historical waste storage facilities are varied in nature, size and complexity, and have different hazards associated with them, both from normal operation and from potential accidents. The magnitude and content of the radioactive inventory is also varied. For example, a historical waste facility might have been designed for disposal and storage of radioactive waste and might represent several different types of construction methods, designs and facility age. Further, they might be independent of or dependent on other facilities, or might be an integral part of a larger facility that also includes waste treatment and other facilities. Commensurately, the extent and complexity of the safety case and supporting safety assessment will differ according to the facility or activity and will also evolve through the lifetime of the facility (e.g. construction, commissioning, operation, and decommissioning or closure). In view of these considerations, a graded approach is required to be applied to the development and review of the safety case and supporting safety assessment.

4.2.1.5. Evolution of the safety case

The safety case is developed while the retrieval activities and the overall waste management project (including final disposal of the retrieved waste) progresses and is used as a basis for decision making (for example, optimization of waste retrieval procedures or for regulatory decision making).

4.2.1.6. Strategy for safety

The strategy for safety of the waste retrieval operations addresses a number of key elements, namely the practical realization of multiple safety functions, engineered barriers, defence in depth, shielding and confinement, and the selection of appropriate approaches to waste retrieval and processing. It also addresses how secondary waste will be minimized, how waste management will be optimized with regard to reuse, recycling and clearance of materials and, if relevant, discharge of effluents, and how interdependencies with other steps in the predisposal management and with the disposal of the waste will be taken into account.

4.2.1.7. Description of the facility, activity and of the waste

Historical waste can vary considerably in terms of inventory, activity, size, waste form, and condition of the containers. While failure of the containers during retrieval operations is a real possibility, other components of the engineered barrier systems might be in a condition that does not protect humans or the environment. In the case of RADON-type facilities, the underground vaults and the waste might have been deteriorated by groundwater, precipitation and/or other external impacts. This can bring additional hazards during the waste retrieval activity that are covered by the safety case and supporting safety assessments.

4.2.1.8. *Safety assessment*

Identification of hazards and initiating events

The historical waste might have been placed in bulk and that the waste packages might have deteriorated (e.g. due to corrosion), with a potential for dispersion or leakage of radionuclides or other types of hazardous material from the original packages. This might need to be considered during the identification of hazards and initiating events.

Management of uncertainties

There are large uncertainties in estimating the time frames associated with the specific waste retrieval operations at these types of facilities. To manage these uncertainties, there are essentially two options:

1. Simulating actual retrieval activities (without radioactive material) to get a better estimate of the time needed for specific operations;
2. Making conservative assumptions of the time needed to conduct each activity.

Option 1 might be limited by a lack of specific knowledge of the actual conditions of the facility and the waste.

Assessment models

The models need to give special attention to background radiation originating from adjacent areas during retrieval operations (e.g. including vaults in the case of RADON-type facilities). Consideration of potential worker doses under varying retrieval scenarios can be used to determine and refine retrieval strategies; specifically, this can be useful in informing decisions on defining the sequence of operations in order to optimize doses to the workers and the public.

4.2.1.9. *Specific issues*

Reliability

While the safety assessment takes into consideration the reliability of components over the lifetime of the RADON-type facility, for legacy waste facilities, it is also important to consider the age of the facility at the time of retrieval (including the condition of the waste packages and potential for their degradation).

Interdependencies

There might be other facilities on the same site which might have been constructed and used to store or dispose of other types of waste. Possible interdependencies might exist between other activities and other facilities located on the site and these will also need to be considered.

4.2.2. Application of the SAFRAN tool

The safety assessment performed for the RADON-type facility retrieval activities has demonstrated the potential application of the SAFRAN tool for this purpose. The general sequence of work performed consisted of the following steps:

- Description of the facilities;

- Creation of the area structure, where work is to be performed, and parameters of exposure (e.g. external dose rate) in work areas;
- Description of operations performed in the course of the activity;
- Establishment of the dose constraints according to the national regulations;
- Description of the regulatory framework for normal and accidental situations;
- Input of personnel job positions who are assigned to perform aforesaid operations;
- Identification and assessment of potential impacts (dose rates to the worker and the public) during normal operations and under abnormal conditions, and estimation of time parameters for each operation;
- Analysis of the results for normal operations and under abnormal conditions;
- Identification and assessment of postulated accident scenarios and calculation of the relevant dose rates to the worker and the public by means of applying the SAFRAN tool's SAFCALC (**SAF**RAN **Calc**ulation Tool) module;
- If necessary, revision of the area structure and working zones resulting from the operations and personal involved (optimization) and iteration of the safety assessment.

The application of the SAFRAN tool allows processing of the input data, creation of the safety assessment structure and analysis of the alternative options for personnel response actions under normal operation, abnormal conditions, and postulated accidents.

APPENDIX.
GRADED APPROACH AND THE EVOLUTION OF THE SAFETY CASE

GSR Part 4 (Rev. 1) [5] identifies the following criteria to be taken into consideration in the application of a graded approach:

- Safety significance (most important);
- Complexity;
- Maturity of the facility or activity.

Using the graded approach provides flexibility for both the regulatory body and the operating organization in ensuring safety of workers and the public.

The regulatory working group provided guidance and support to the other working groups during the drafting process of their reports. This Appendix summarizes the results of the regulatory working group discussions that were focused on the application of the graded approach. The Appendix is structured to address the following stages of the lifetime of predisposal waste management facilities: siting, design and construction, commissioning, and operation.

A.1. SITING

The safety case for the siting stage presents the strategy for safety and how safety will be met. It is generally not possible to provide a detailed description and assessment of the facility (or activity). In the absence of any quantitative demonstration, emphasis is placed on qualitative justifications for the strategy for safety adopted. These include initial approaches for radiological impact assessment, the management system, and management of uncertainties. The output of the safety case at this stage of development is justification that the site is viable for the proposed facility (or activity) (para. 6.13 of GSG-3 [1]).

Factors include:

- Proposed activities at the facility (e.g. storage, long term storage, conditioning, thermal treatment);
- Other activities at the site and their impact on the proposed facility or activity;
- Isotopic activity levels and form: solid, liquid and/or gas;
- Site features:
    - Surface water (e.g. flooding, tsunami);
    - Seismology;
    - Depth to groundwater;
    - Geology;
    - Topography (e.g. for surface water run-on);
    - Weather impacts (e.g. snow, rain, wind).
- Building structure (e.g. in the desert with occasional fires);
- Nearest neighbours and surrounding industries (e.g. zoning, representative persons);
- Demographics;
- Access to facility;
- Transportation routes (e.g. aircraft flight paths, major highways);

- Additional land for expansion of the site;
- Anticipated WAC.

Concurrently, additional data will be gathered, including:

- Environmental impact assessment;
- Stakeholder involvement.

A.2. DESIGN AND CONSTRUCTION

In the design and construction stages, the safety case will provide justification that the (as designed) facility or activity:

- Is needed;
- Will meet all safety requirements;
- Can be safely constructed and operated.

The safety case will demonstrate that:

- The likelihood that safety-related SSCs are failing is low.
- In the event of degradation, the loss of a safety function of one component does not jeopardize the safety of the whole system (defence in depth).
- It is a mature assessment of the engineering and the impact of the facility or activity.

Factors include:

- Doses for workers and the public are safely below regulatory limits:
    - Individual and collective dose optimized.
- Site security (this is typically addressed in the security plan for the facility):
    - Physical barriers and other passive or active access controls;
    - Monitoring and response (e.g. camera to an offsite guard).
- Building layout:
    - Building codes, fire codes, electrical codes;
    - Access (personnel and vehicle doors, location of access points);
    - Sufficient and separate areas to segregate waste by dose rate;
    - Internal layout optimized for processes (e.g. receiving, storage, processing);
    - Lighting (natural and provided);
    - Optimized layout for package handling;
    - Shielding (e.g. engineered overpacks, hot cell);
    - Labyrinth passageways and cableways to prevent streaming;
    - Floor and walls sealed for ease of decontamination.
- Air circulation and temperature control:
    - Natural circulation or forced ventilation with controls such as HEPA filters;
    - Lowest ambient pressure at highest contamination levels;
    - Moisture control.
- Radiation protection programme:

  - Overarching policy statement;
  - Optimization programme;
  - Procedures;
  - Zoning (for dose rates, waste type, and processing priority);
  - Personnel monitoring;
    - External and internal monitoring.
- Processing and storage of large volumes of liquid waste and effluents:
  - Compatible storage containers (e.g. proper for non-radiological hazards);
  - Radioactive waste collection reservoir (e.g. low point sump) with sampling capability;
  - Containment with thick liner below concrete floor;
  - Volume and time limits placed on storage.
- Pre-operational environmental monitoring:
  - Thermoluminescent dosimeters for ambient radiation levels;
  - Media sampling to establish baselines;
  - Continuous stack monitoring.
- Effluents and permitted discharges to the environment:
  - Air emissions limited: radiological, sulphur oxides (SO_x) and nitrogen oxides (NO_x);
  - Conditions suitable for liquid releases (e.g. consideration of water table).
- Management system (consistent with the activities inside the facility):
  - Quality assurance programme (e.g. records management, ensuring performance of components);
  - Independent review of the safety case and safety assessment;
  - Action in place to identify and report reversible and irreversible non-conformances.
- Non-radiological hazards (e.g. chemicals and/or combined chemical/radiological)
- Processing of retrieved and secondary RW: decontamination or packaging for storage?
- Consideration that the design facilitates decommissioning
- Other government agencies involved
- Safety assessment
  - PIEs:
    - External natural events (e.g. lightning, extreme temperatures, offsite fires);
    - External human induced events (e.g. fire, accidental aircraft crash);
    - Internal events (e.g. inappropriate processing, arcs and sparks, explosions, gross incompatibilities, failure of component or systems).
  - Use of computer codes;
  - Normal scenarios and abnormal conditions;
  - Accident scenarios (including design basis accidents and design extension conditions).

A.3. COMMISSIONING OF THE FACILITY

In the commissioning stage, specific attention is paid to the performance of structures, systems, and components important to safety. The aim of the safety case for cold commissioning is to justify the decision that the as-built facility is safe to operate (taking into account any design changes during construction) and also to identify potential areas for optimization. For hot commissioning, the aim is to justify the decision that the facility can accept radioactive material safely. Additionally, the safety case provides updated information on the management system (para. 6.21 of GSG-3 [1]).

Factors include:

- Develop and proof-test procedures in all safety areas (e.g. health physics, management systems);
- As-built facility meets the final design criteria for safety:
    - Walls have required density for shielding to achieve dose standards;
    - Floor strength verified and sealant coating tested.
- System line-ups (e.g. valves and breakers as needed) verified;
- Cold commissioning of equipment prior to hot commissioning;
- Graded approach used for activity levels at start-up: kBq to MBq to GBq;
- Finalize WAC (e.g. DSRS, liquids, dry solid waste);
- Operating procedures necessary to operate facility:
    - Critical procedures embedded "in license";
    - Non-critical procedures can be attached to the license.
- Record keeping from notification of incoming RW until transfer to another facility:
    - Quality assurance (e.g. nuclear quality assurance).
- Updated plan for the decommissioning of the facility;
- Emergency plan and procedures in place;
- Update of the safety assessment:
    - PIEs;
    - Normal operation and anticipated operational occurrences;
    - Accident scenarios (including design basis accidents and design extension conditions).

A.4. OPERATION

The initial safety case for operation provides evidence that the facility has been constructed in accordance with the design and that commissioning demonstrated the facility can be operated safely. Any significant differences between the actual performance and predicted performance of the facility (or activity) are identified and the reasons for the differences investigated. All discrepancies are justified.

The aim of the safety case for operation is to justify the decision that the facility can be operated safely for a specific period and can then be safely decommissioned (para. 6.26 of GSG-3 [1]). Significant changes to the facility or changes that could affect safety are addressed in updates to the safety case.

Factors include:

- Updated plan for the decommissioning of the facility;
- Documentation of facility changes/updates vs. periodic safety review update:
    - Changes might be extensive (e.g. adding conditioning or treatment) that require additional design, construction and commission stage evaluations.
- Changes in national regulations and rules;
- Change in knowledge;
- Management system (e.g. for record keeping).

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ANNEX I
ILLUSTRATIVE SAFETY CASE FOR THE SLOVENIAN NATIONAL STORAGE FACILITY FOR INSTITUTIONAL RADIOACTIVE WASTE

SUMMARY

Slovenia joined the CRAFT project in 2012 with the view to implement IAEA Safety Standards Series GSG-3, The Safety Case and Safety Assessment for the Predisposal Management of Radioactive Waste [I–1] in the preparation of the new revision of the Central Storage Facility (CSF) in Slovenia.

The Slovenian Agency for Radioactive Waste Management (ARAO), referred to herein as the operator, received a 10 year operating license for the CSF in 2008. Since the issuing of the license in 2008, a number of new documents were prepared for the facility, prompting ARAO to make the decision to update the Safety Case. The updated Safety Case will also address changes and optimization of the facility. Work on the Safety Case began in 2013 and was completed in 2017. The purpose of the safety case is to support the continued operation of the facility following requirements for the periodic safety review as prescribed under Slovenian legislation.

During the revision of the safety case, the philosophies of the graded approach and the step by step approach were used. In order to reduce uncertainties, all data used in the safety case was updated. In order to increase the confidence of the competent regulatory authority – the Slovenian Nuclear Safety Administration (SNSA) – in the assessment, ARAO followed a combination of methods and tools to prepare safety assessment in parallel: “traditional” tools (performing standard dose calculations using MS Excel and other commercially available modelling tools); and following the Safety Assessment Driving Radioactive Waste Management Solutions (SADRWMS) methodology [I–2] and using the Safety Assessment Framework (SAFRAN) tool [I-3], both of which were developed under the predecessor to the CRAFT project.

This Annex presents the work performed from 2013 through 2015 during the preparation of the safety case for the CSF. The Annex documents the work done and the results achieved in updating the Safety Case. The Safety Case includes a comprehensive safety analysis of the CSF facilities and activities, considering the current and anticipated future inventory of radioactive waste (RW). The scope of the Safety Case also includes the storage of institutional RW collected in Slovenia from various small waste producers. The Safety Case addresses issues of importance such as the management system, site aspects, facility design, RW inventory, storage capacity, and operational (storage) activities.

The methodology for the safety assessment includes:

- Engineering analysis of the facility (description of SSCs and identification of their safety functions);
- Development of scenarios (normal and accidental) using the hazard and operability study (HAZOP) method and subsequent identification and screening of postulated initiating events;
- Assessments of the exposures of the workers and the public during normal operation, anticipated operational occurrences, as well as accident conditions.

The results of the assessment indicate that the facility and activities described and performed in accordance with the provisions set out in this Safety Case comply with national and international regulations and standards and meet the relevant dose criteria for workers and the public.

I–1. INTRODUCTION

Slovenia has a very small nuclear programme, with one operating nuclear power plant (Krško), one research reactor (Ljubljana), and one central storage facility for RW generated by small producers.

There are two storage facilities in Slovenia that store RW awaiting disposal. One facility is operated at the site of the Krško nuclear power plant, storing RW arising from its operation. The second storage facility is the Central Storage Facility (CSF) which is located in Brinje near the Slovenian capital Ljubljana. It is intended for the storage of low and intermediate level RW (LILW) arising from medical, industrial and research applications. The CSF is operated by the Slovenian Agency for Radioactive Waste Management (ARAO). ARAO is a non-profit organization of the Slovenian Government, providing a state-owned public service for RW management. The construction of the CSF started in 1984 and the facility was put into operation in 1986. In 1999, the responsibility for management and operation of the facility was transferred from the Institute Jožef Stefan (IJS) to the ARAO. Following refurbishment and two and a half years of trial operation, a new operating license was issued in early 2008. In 2018, the first periodic safety review of the CSF was finished and the new operating license is valid until 2028.

ARAO carries out the following activities:

- Collection of RW at waste producers' premises;
- Collection of RW on-site in the event of accidents;
- Collection of RW in the case where the waste producer is unknown;
- Storage of the collected RW in the CSF;
- Dismantling of sealed sources at producers' premises (less complex sources);
- The use of radioactive sources for calibration and testing of measuring devices;
- The treatment and conditioning of RW and disused sources in a processing facility – hot cell facility (rented) for the purpose of storage;
- The transport of radioactive materials, and transportation of nuclear materials as a part of public service.

ARAO operates within the framework of the RW management programme [I–4], which is an operational document for RW management in Slovenia that covers the organization and methods of carrying out activities, recording and reporting, definition of responsible services and persons, information on documents forming the basis for carrying out activities, information on packaging, information on RW, management procedures and methods, measures to minimize RW generation, clearance, capacities in place, consideration of interdependencies between all stages of management, alignment of the management procedures with operative programmes under the national programme of RW management.

The waste inventory in storage in the CSF has been characterized, treated and conditioned. ARAO has been performing treatment and conditioning of RW on a regular activity at the nearby processing facility (hot cell facility) since 2012. The ARAO staff carries out waste sorting, characterization, compaction, dismantling of disused ionizing smoke detectors and solidification of liquid RW. It is planned to implement dismantling of other sealed sources in the future.

During the development of the waste acceptance criteria (WAC) [I–5] for storing the institutional RW in the CSF, ARAO considered the generic WAC for the planned LILW disposal facility [I–6], IAEA-TECDOC-864 [I–7], Slovenian legislation and operators' practices. The WAC for the storage facility will be revised when the WAC for disposal are approved.

I–2. LEGAL AND REGULATORY FRAMEWORK

The legal framework for safety of radioactive waste and spent fuel management of the Republic of Slovenia is formed by the Ionizing Radiation Protection and Nuclear Safety Act, the Rules on Radiation and Nuclear Safety Factors [I–8], and the National Programme for the Management of Radioactive Waste and Spent Fuel for the period 2006 – 2015 [I–9].

A system of licensing of spent fuel and RW management is provided in the 2002 Act, while the Rules on Radiation and Nuclear Safety Factors lay down details on documentation that are required to be submitted in a particular phase of licensing. At every step of the licensing process, the investor or operator is required to attach to the license application, in addition to the design documentation, a safety analysis report and the opinion of an radiation and nuclear safety expert authorized by the competent regulatory body (the Slovenian Nuclear Safety Administration (SNSA), and other prescribed documentation required by the Rules on Radiation and Nuclear Safety Factors [I–8].

On 2 July 2013, the Parliament of the Republic of Slovenia passed the Resolution on the 2013–2023 Nuclear and Radiation Safety in Slovenia (Official Gazette RS, No. 56/2013). The Resolution, as a high level national policy paper, covers the following chapters: The fundamental safety principles, description of nuclear and radiological activities in Slovenia, description of the international cooperation in the field of nuclear and radiation safety, description of the existing legislation (including binding international legal instruments, such as conventions and other relevant international instruments), description of the institutional framework, competence of professional support (research, education, training), objectives and measures to achieve them during the period up to 2023.

I–3. CONTEXT OF THE SAFETY CASE

I–3.1. Purpose of the safety case

In accordance with the requirements associated with the licensing process in Slovenia, the investor or operator attaches to the license application design documentation, a safety report, the opinion of an authorized radiation and nuclear safety expert (authorized by the SNSA) and other prescribed documentation set by the Rules on Radiation and Nuclear Safety Factors.

Revision 0 of the safety report for the CSF was prepared in 2007. Subsequently, the term Safety Case was defined, in relation to the management of RW, in GSG-3 [I–1]. Following good international practices and requirements, a definition of the Safety Case is now included within the safety strategy for CSF [I–10] as a collection of scientific, technical, administrative and managerial arguments and evidence in support of the safety of a CSF, following the definition in GSG-3 [I–1]. Therefore, the Safety Case for the CSF includes a number of documents and reports that will be summarized in the main Safety Case report document.

As a result, a series of new documents related to the CSF and revisions of existing documents have been prepared: training programme for staff relevant to nuclear and radiation safety [I–11], facility decommissioning plan, RW management programme, physical protection plan, ageing management programme [I–12], working procedures and manuals, etc. The significant changes in the legislation in the field of physical protection of nuclear facilities came into force in 2013. The upgrades regarded the physical protection in the CSF were implemented in 2014 and 2015. In 2013, ARAO began work on the revision of the safety case for the CSF. The new revision takes into account all the above-mentioned changes (changes in documentation and upgrades in the CSF) and also the commitment by ARAO to implement GSG-3 [I–1] and SADRWMS [I–2] methodologies and to use the SAFRAN [I–3] tool.

The main purposes of the new revision of the safety case are:

- To perform a periodic reassessment of the safety of the facility;
- To apply certain changes related to optimization of the facility design.

The Safety Case for the CSF is a 'living document', with supporting references, which is developed and updated throughout the lifetime of the facility (including operation and decommissioning). This Safety Case forms the basis for phased regulatory decisions as well as operational decisions.

The CSF is currently in the operational stage. The principal purposes of this revision of the Safety Case are:

- To demonstrate the safety of the facility in its current stage;
- To support licensing (license prolongation) of the facility with the regulatory body;
- To justify continued operations and identify areas for improvement in the facility.

As a result of the iterative development of the Safety Case through the facility lifetime, the following results are achieved:

- The systematic collection, analysis and interpretation of the necessary scientific and technical data;
- The development of plans for operation;
- Optimization of protection and safety;
- Iterative studies for design optimization, operation and safety assessment with progressively improving data and comments from technical and regulatory reviews.

The following specific aspects will be addressed in this Safety Case:

- Demonstration of the safety of the CSF;
- Demonstration of the safety of various RW management activities performed by the operator. These activities include acceptance and characterization of the RW;
- Optimization of the respective waste management activities described above;
- Management systems implemented to ensure the safety of the respective waste management activities described above;
- Definition of limits, controls and conditions that will be applicable to the facilities and the respective activities described above;
- Input to the improvement of existing radiation protection programmes and activity procedures.

This Safety Case takes into consideration IAEA Safety Standards Series Nos GSR Part 5, Predisposal Management of Radioactive Waste [I-14], GSG-3, The Safety Case and Safety Assessment for the Predisposal Management of Radioactive Waste [I–1], and GSR Part 4 (Rev. 1), Safety Assessment for Facilities and Activities [I–14]. Safety criteria are taken from the Slovenian regulatory framework and from IAEA Safety Standards Series No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [I–15].

I–3.2. Scope of the safety case

The scope of the Safety Case includes the following activities:

- Acceptance, identification, removal of waste from the transport container and placement into the storage container (further considered as packaging in the document) and handling of RW at the CSF;
- Storage of the RW at the CSF;
- Maintenance and inspection of the RW packages and their contents during their storage in the CSF.

A detailed description of these activities is given in Section I–5.3.

This version of the Safety Case specifically excludes the following activities:

- Collection and transport of RW to the CSF;
- Waste retrieval;
- Buffer storage of untreated liquid waste;
- Decommissioning of the facility;
- Non-radiological hazards.

I–3.3. Demonstration of safety

This section describes the approach to demonstration of safety, specifically the safety objectives and safety principles that are applied and the regulatory requirements that are to be met. Taking cognizance of the scope of the Safety Case and the application of the graded approach as described in Section I–3.4, the safety of the waste management and storage facilities will be evaluated and demonstrated as described hereafter.

I–3.3.1. Approach to basic engineering analysis

A qualitative assessment will form the basis of the basic engineering analysis, which will mainly cover the following:

- Basic site characteristics and credible external events considered in the design of the CSF to ensure structural stability;
- Quality assurance considered in the design, construction, maintenance and modification of the CSF;
- Application of national construction codes and standards;
- Inspection and maintenance plans;
- Formal processes for the evaluation, approval and implementation of modifications;
- Safety and security aspects.

Quantitative and qualitative assessments will be performed to assess the impact of the waste management activities and results will be assessed in terms of the safety criteria (see Section I–6.1.2).

The following specific assessments will be performed:

- For normal operations, quantitative deterministic assessments of worker dose resulting from the range of activities by workers, including determination of the allowed working hours in CSF areas;

- For anticipated operational occurrences, quantitative deterministic assessments of occupational and public doses as applicable;
- For all other credible accident scenarios, a quantitative and qualitative assessment of the impact of other occurrences with identification of specific preventative and mitigating measures.

I–3.3.2. Approach to safety assessment

The radiological assessment follows a realistic and conservative approach taking measured data into consideration where possible. Where such data are not available, the data embedded in the SAFRAN tool [I–3] calculation modules and other models are applied to model exposures based on reasonable assumptions.

Uncertainties inherent to the assumptions made in the quantitative assessments or any other uncertainties identified during the safety assessment are evaluated to determine their impact on safety. The main sources of uncertainties that might have a significant impact on safety are presented along with recommendations for their management (see Section I–6.11, Table I–57).

The above safety assessments are discussed in Section I–6. Section I–10 presents the results from the quantitative and qualitative assessments for comparison against the proposed target and objectives set for the optimization of protection.

A qualitative assessment is performed of the availability and level of implementation of an integrated management system in order to ensure a sustained level of safety. This assessment focuses on radiation protection, work procedures, quality assurance, and processes for the management of operating limits and conditions.

I–3.3.3. Overall approach to safety of the facility

A simple robust design was adopted for the construction of the CSF to make operations within the facility simple and easy to undertake. The facility design and construction provide defence in depth; the facility is designed to rely predominantly on passive safety features. The overall safety of the facility does not rely on a single design feature.

I–3.4. Graded approach

A graded approach is applied to define the extent and depth of this Safety Case by the use of qualitative assessment of hazards and deterministic analysis of doses to potential representatives (e.g. workers and the public). This takes into consideration the relative safety significance, low complexity of operations and the maturity of the facility and storage activities.

The main factors to justify a relatively simple approach to the safety assessment for the CSF are:

- The limited scope and function of structures, systems and components (SSCs) of the CSF (storage of RW packages awaiting disposal).
- The simplicity of the activities involving storage of RW and supporting activities (e.g. internal transport, monitoring, inspection). The facility is under permanent supervision (through monitoring and physical and technical protection).
- The radiological hazard when undertaking the various management activities involving stored containerized RW can be regarded as low.
- The maturity of the facility:
    - ARAO started operations at the CSF in 1999.

- Reconstruction work was finished in 2004.
    - In 2005, the trial operation license was issued.
    - In 2008, the 10 years operational license was approved.
    - For the past 7 years, the CSF has, without any unexpected events, regularly accepted RW packages, performed continuous monitoring of the radioactivity, and has ensured that all installed SSC are well maintained and inspected and that workers are all trained.
    - Storage activities at the CSF follow good practice and internationally accepted concepts.
- Inherent high level of passive safety associated with the management of disused sealed radioactive sources (DSRS) and limited reliance on active protection systems.

I–4. STRATEGY FOR SAFETY

This section describes the strategy for safety, including the approach taken during facility design and all respective RW management activities to comply with regulatory requirements and to ensure that good engineering practice has been adopted and that safety and protection are optimized.

In view of the scope of the Safety Case as defined in Section I–3.2 and the document that described the safety strategy for the CSF prepared by ARAO [I–10], the following strategies for demonstrating safety of the management of RW are adopted:

- Safety principles – all the safety principles defined by IAEA Safety Standards Series No. SF-1, Fundamental Safety Principles [I–16], and resumed in the Resolution on Nuclear and Radiation Safety in the Republic of Slovenia [I–17] are met.
- Step by step approach is used with principle that the facility and the activities performed in the facility can adopt new findings and practice.
- Defence in depth – Care is taken to ensure that multiple safety layers and safety functions are established. This principle is considered to ensure that no important safety argument is based on a single level of protection.
- Passive safety – The use of passive safety systems wherever possible.
- Shielding – Ensuring that doses to workers and the public are as low as possible. This also includes the optimization of shielding usage during all waste management activities including transportation and storage is considered.
- Selection of implemented waste management practice – Approach to waste management with regards to the following is regarded as contributing to safety.

A qualitative assessment is performed on the implemented waste management practices. In the approach to waste management the following will be regarded as contributing to safety:

- Clearly defined responsibilities for waste management;
- Implementation of the principles of waste minimization and avoidance, namely, re-use or re-processing of waste, return to supplier, safe and secure storage and conditioning and final disposal of waste.
- Hazards of the generation of secondary waste associated with all waste management operations (routine and ad hoc) are known, monitored, projected and managed by due management processes.
- Interdependencies between the various steps of waste management are known and managed.

- WAC are defined and waste management activities (and the outputs of such activities) are aligned with the set of WAC.
- Interim storage of RW will only take place inside proper containment such as the original working shields or another type of suitable containment.
- Conditioned RW will be stored in a dedicated storage area with passive safety features and adequate access control.

Engineering aspects that ensure safety are:

- The engineering characteristics of the building. Information expressed in the building design document show the engineering features. Its design ensures structural stability under extreme environmental conditions (e.g. earthquake, storm).
- The characteristics of the walls ensure a dose rate that complies with the restriction for public exposure (0.1 mSv/a) for the representative person.
- The lighting system will be adequate and permits the performance of operations in a safe manner.
- Electric power is limited to the radiological control and management operations areas.
- Each delineated area has sufficient physical space to ensure a minimal probability of accident occurrence during package management.
- Storage building areas were designed under the principle of labyrinth, which contributes to optimize and minimize the exposure of workers.
- Drums with radioactive sources are stored in such a manner that packages do not contact the interior surface of the building walls. This enables adequate control operations and ensures that the potential corrosion of packaging and/or containers is limited.
- Unconditioned radioactive sources are stored in stacking systems ensuring normal operation and minimizing probability of accidents.
- There is a vault with special shielding structure that minimizes worker exposure for the storage of radioactive sources of high activity that cannot be conditioned.

For anticipated operational occurrences and accident conditions due to internal operational factors, the engineering systems ensuring safety are:

- Floor and wall finish that allow easy decontamination.
- The segregation of the different areas limits the potential dispersion of any contamination.
- In case of a potential surface decontamination using liquids, there is a collection system inside the facility that prevents its release to the environment. The system has a retention tank that permits environmental monitoring before releasing the liquid to the environment.
- The facility has its own fire response equipment.

I–5. DESCRIPTION OF THE FACILITY, THE ACTIVITIES AND THE WASTE

I–5.1. Short description of the CSF

Facility name: Central Storage Facility (CSF) for institutional RW

Country: Slovenia

Status: in operation

Waste types: solid RW from medicine, industry, research and education activities

Waste streams:

- T1 (solid, compressible, combustible);
- T2 (solid, compressible, non-combustible);
- T3 (solid, non-compressible, combustible);
- T4 (solid, non-compressible, non-combustible);
- SRS (sealed radioactive sources);
- M (mixed waste).

External packing: 210 l drums, 210 l drums with inner concrete shielding, 320 l drums, original containers of SRS, polyethylene bags, polyethylene or metal containers.

Typical radionuclides: Co-60, Cs-137, Eu-152, Am-241, Ir-192, Kr-85, Sr-90, Am-241/Be, Eu-152/154, Ra-226, Pu-239, C-14, H-3, Se-75, U-238, Th-232.

Facility operations: accepting the waste packages, storing the waste packages, manipulation of the waste packages, monitoring, export (for treatment and conditioning or release).

Layout of the facility: Controlled area with 10 storage sections, separated with concrete walls.

Observation area: rooms for personnel, machinery room for ventilation system.

Capacity: 115 m^3 with the possibility to extend.

Systems: ventilation system with HEPA filters, fire protection system, drainage system, for potential liquid release collection, physical protection system.

Typical staffing: 6 persons.

I–5.2. Site conditions

The CSF is located inside the premises of the Research Reactor Centre of the IJS, north-east direction from Ljubljana (about 15 km). The research reactor is located adjacent (less than 50 m) to the CSF. The Sava River flows approximately 1 km south from the site and the Pšata River flows about 0.5 km to the north-east.

The perimeter of the site is enclosed by a simple fence. This fence is erected on the owner's property line and encloses the IJS buildings with the research reactor.

I–5.2.1. Demography

Table I–1 presents the number of inhabitants (as of 1 January 2013) at distances of 10 km in the surrounding areas adjacent to the CSF.

TABLE I–1. NUMBER OF INHABITANTS AT RADIAL DISTANCE FROM THE CSF

Radius [km]	Number of inhabitants
0.5	62
1	776
2	3633
3	7122
4	19379
5	42579
6	85227
7	140078
8	177450
9	221005
10	271780

Within a 0.5 km radius of the CSF, there are only 62 inhabitants who use the land around the facility mainly for farming. Nobody lives permanently inside the premises of the IJS, while around 150 people work there on a daily basis.

Most of the inhabitants of the urban areas are within a 6–7 km radius, as shown in Fig. I–1.

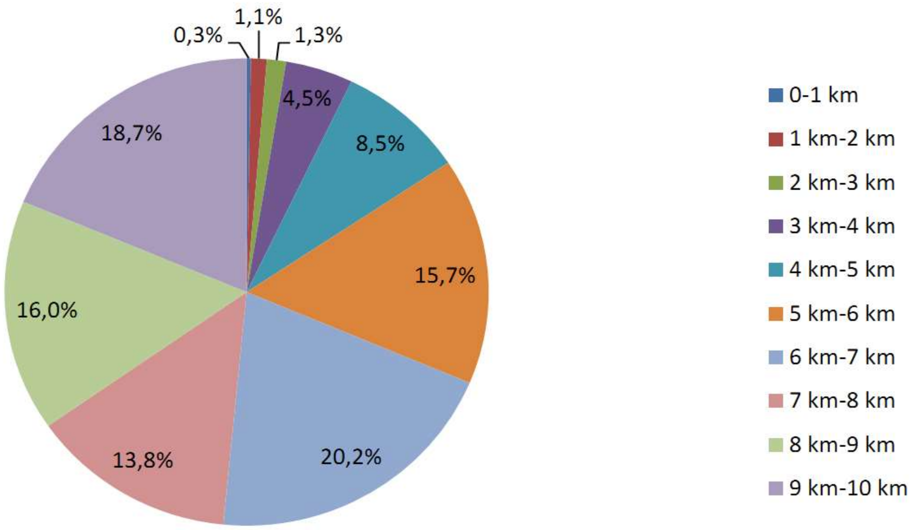

FIG. I–1. Percentage of inhabitants at radial distance from the CSF.

I–5.2.2. Meteorology

ARAO measures site meteorological conditions and collects data using the meteorological station located 250 m west from the CSF (see Fig. I–2). The site lies in an area with moderate continental climate, with low winter temperatures, frequent temperature inversion, relatively high summer temperatures, relatively high levels of precipitation and frequent fog.

FIG. I‒2. Meteorological station near the CSF.

Wind data have been collected since 2010 from the weather station adjacent to the site. As illustrated on the wind rose in Fig. I‒3, an evaluation of the characteristics of the wind shows that the most dominant winds are N and NW direction and E and SW direction with generally low wind speeds. The annual average wind speed is 1.3 m/s or approximately 4.7 km/h.

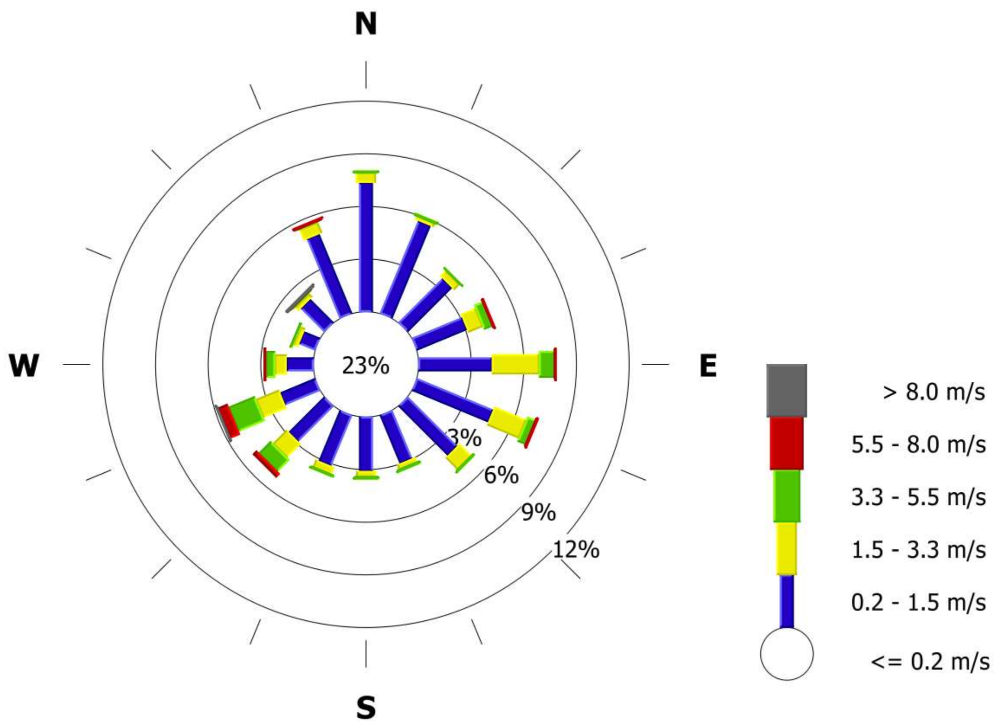

FIG. I‒3. Wind rose for the CSF site (from 1 January 2010 to 31 December 2012) on the height of 10 m.

Table I‒2 presents the number of days with strong wind (wind speed greater than 11 m/s) and the number of days with very strong wind (wind speed greater than 17 m/s).

TABLE I–2. THE NUMBER OF DAYS WITH STRONG AND VERY STRONG WIND IN THE CSF AREA

Year	Number of days with the strong wind speed > 11 m/s	Number of days with the very strong wind speed > 17 m/s
1981	13	0
1982	22	0
1983	41	1
1984	29	4
1985	34	0
1986	12	2
1987	19	1
1988	36	0
1989	36	3
1990	47	1
1991	57	1
1992	60	1
1993	64	1
1994	52	2
1995	46	3
1996	41	2
1997	45	1
1998	44	1
1999	49	0
2000	50	1
2001	59	2
2002	48	2
2003	57	4
2004	37	1
2005	44	1
2006	50	2
2007	55	2
2008	56	2
2009	49	0
2010	43	1

Temperature data indicates that the annual average temperature is 10.5°C. Table I–3 shows the average air temperatures, precipitations and average relative humidity in the CSF area.

TABLE I–3. AVERAGE AIR TEMPERATURES, PRECIPITATIONS AND AVERAGE RELATIVE HUMIDITY IN THE CSF AREA

Month	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec	Year
Average air temperature [°C]	-0.4	-0.3	6.7	11.1	14.9	19.3	21	20.7	16	9.6	6.8	0.4	10.5
Precipitation [mm]	37.7	42.2	40.8	60.1	90.3	89.3	108	94.5	202	153	92.9	101	1111
Average relative humidity [%]	86	81	75	72	75	78	76	72	76	87	91	92	80

I–5.2.3. Site geology and hydrology

The land where the CSF is constructed is located on flat terrain with a slope of about 1% (thus allows the drainage of rainwater) in the direction to the Sava River, with average elevation of 280 m above sea level.

From the map in Fig. I–4, it can be seen that the site of the CSF lies in an area that is not defined as a flooding area.

The CSF lies on the Sava River terrace at around 9 m above the Sava River, which flows 1 km away from the facility site. The area consists of quaternary conglomerate gravels and gravels with some layers of silt, clay gravels and clay.

In the area of the CSF, the thickness of the unsaturated zone is from 7.5 to 10.5 m. The average hydraulic conductivity of the upper 20 m layers is $5.6 \cdot 10^{-4}$ m/s.

The average temperature of ground water is around 11°C.

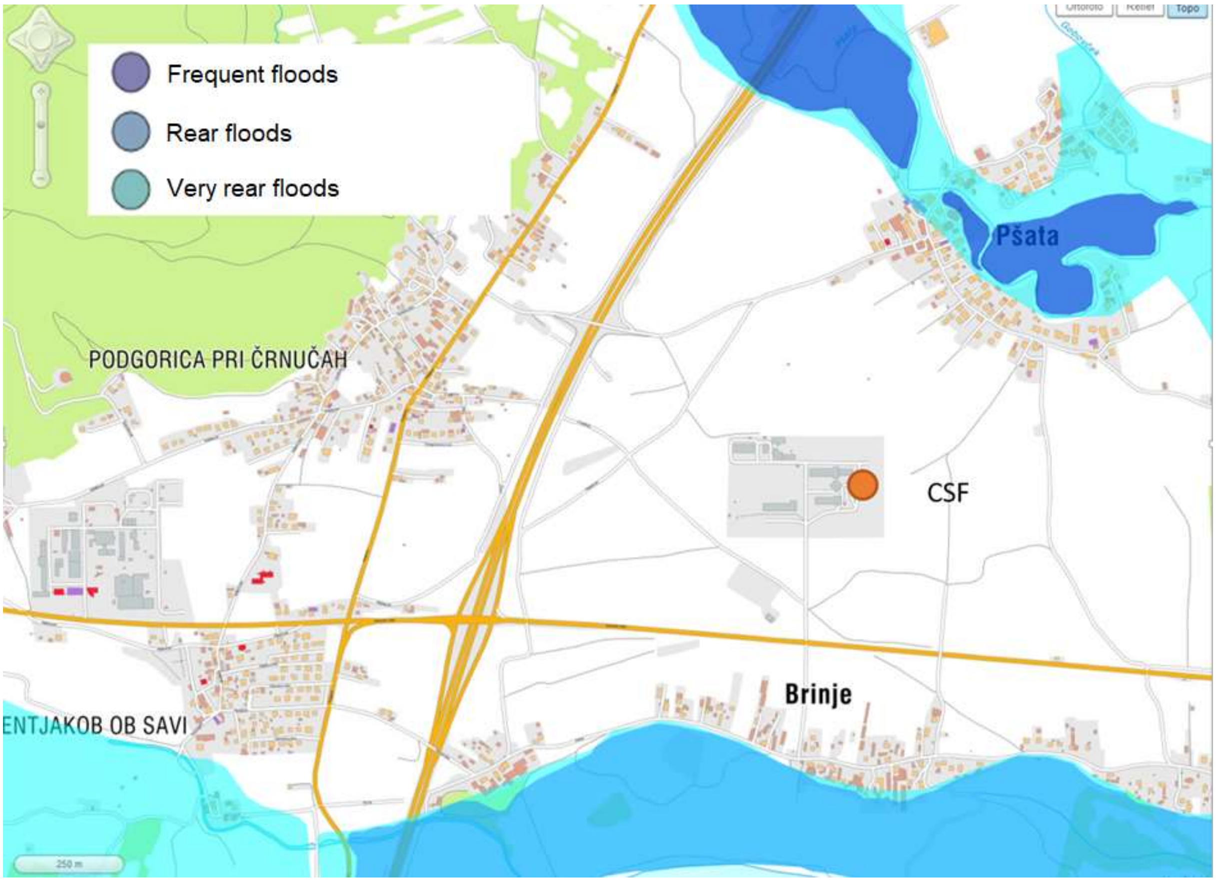

FIG. I–4. Illustration of the flood probability for the CSF [I–18]. In the labelling, 'Very rear floods' means floods with a return period of more than 50 years, 'Rear floods' means floods with a return period from 10 to 20 years, and 'Frequent floods' means floods with a return period from 2 to 5 years.

I–5.2.4. Site seismology

The area where CSF lies originates from older Pleistocene and is one of the earthquake-prone areas in Slovenia. In the last 20 years, two more significant earthquakes did occur:

- Bovec, April 1998, 80 NW km from Ljubljana, magnitude 5.8 with intensity 7–8 (EMS);
- Trebnje, August 1998, 30 SE km from Ljubljana, magnitude 4.3, with intensity 5–6 (EMS).

These earthquakes did not cause any damage to the CSF. The CSF design documentation shows that it was constructed as earthquake resistant (seismically safe).

I–5.3. Description of the facilities and activities

The CSF was constructed in 1986 for the purpose of storing LILW arising from the use of radioactive materials in medicine, industry and research activities in Slovenia. It lies inside the area of the Research Reactor Centre of the IJS in Brinje.

The CSF is a near-surface concrete building (seismic resistant) with the roof covered with a soil layer. The storage concept of the facility makes use of multiple barriers (reinforced concrete walls, cover, doors, additional shielding) to perform additional protection against radiation.

The building is subdivided by concrete walls into nine storage sections and an entrance area (see Figs I–5 and I–6). The ground plan of the facility is 10.6 m × 25.7 m with a height of 3.6 m. A small area is intended for a checkpoint between radiological controlled and supervised area and has a space for loading and unloading the waste and for internal transport. The storage section at the back end of the building is relatively deeper compared to the level of the other sections.

FIG. I–5. Entrance area of the CSF.

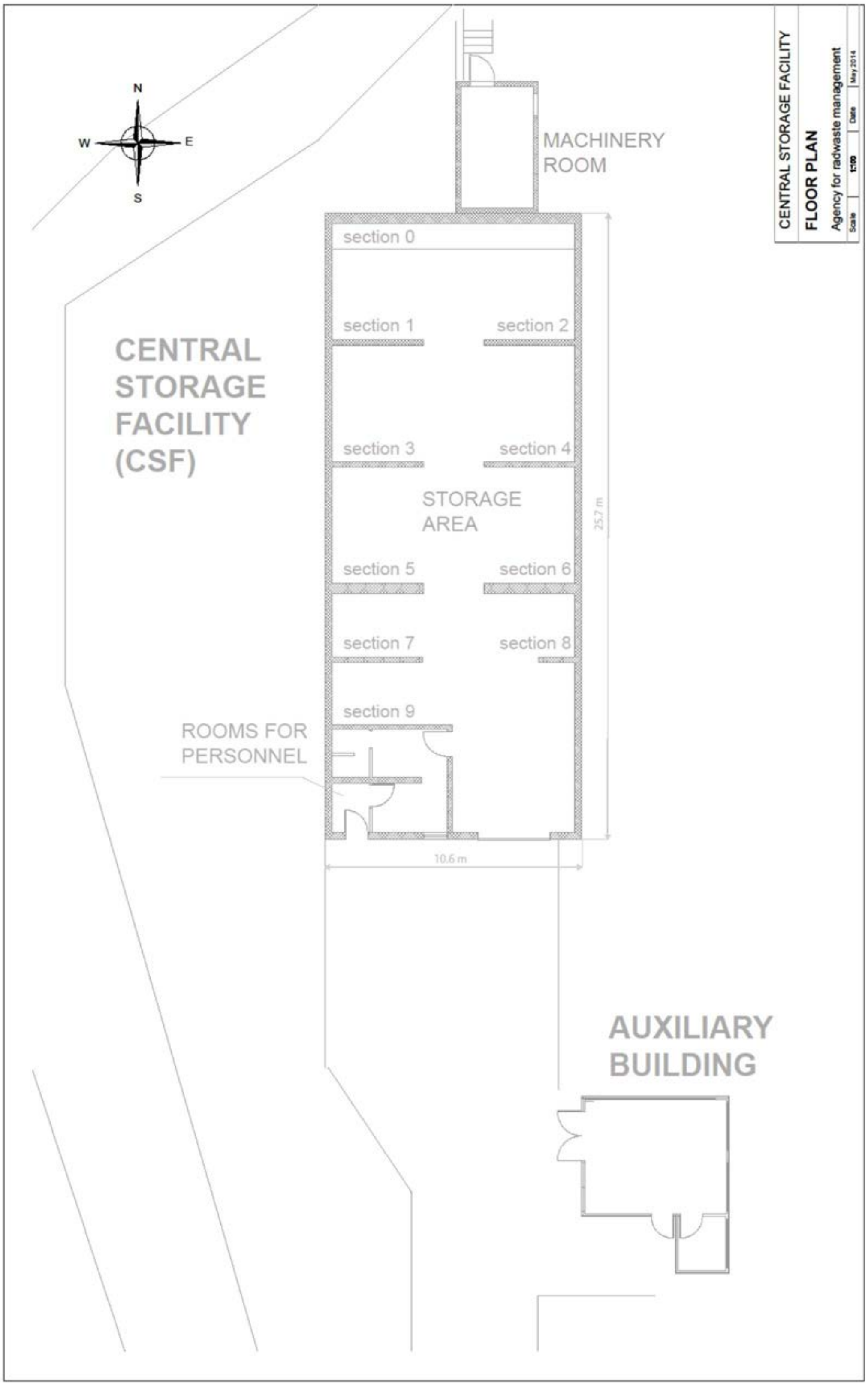

FIG. I–6. The CSF floor plan.

The facility is equipped with a ventilation system for reducing radon concentration and air contamination in the storage facility. To maintain relatively low and constant humidity, the CSF is equipped with an air-drying system. The water and sewage collecting system is designed as a closed system that retains all liquids from the storage facility in the sump. Liquids are only discharged after measurements of radioactive contamination are demonstrated to be below the prescribed regulatory limits. Collected liquids exceeding the limit are treated. The electricity supply system is used for lighting the storage facility and for powering the ventilation. The storage facility is physically and technically protected against fire, acts of violence, burglary, sabotage and similar events.

I–5.3.1. Facility operation

Operational activities within the CSF involve:

- Unloading of RW from the transport container;
- Control measurements;
- Packaging (take the package from transport overpack and put it in the storage drum);
- Storage.

The facility design is such that it makes these operational activities simple and easy to undertake in minimal time. Written operational procedures are prepared to ensure that the activities are carried out safely and in minimal time reasonably possible and to optimize safety and protection by ensuring that no individual dose constraints or limits are exceeded.

Operational radiation protection, maintenance and inspection procedures are formally documented and approved, an incident reporting system and emergency plans are prepared and approved. These procedures will be updated based on and justified by this Safety Case.

A record keeping system is in place for all operational activities, waste packages, DSRS and equipment. Stored waste in the CSF is clearly marked and labelled,

I–5.3.2. Operational radiation protection

The CSF is designated as a radiologically controlled area and people working in the facility are designated as occupationally exposed workers with the necessary training, dosimetry and medical control.

A radiation monitoring programme is implemented and covers routine monitoring of the facility and its environment, working environment, monitoring of specific operations such as treatment of waste, conditioning of DSRS and emplacement activities and some special monitoring that might be required from time to time. The programme makes provision to monitor external radiation levels and surface contamination [I–19].

I–5.4. Inventory of radioactive waste and waste acceptance criteria for the CSF

Relevant data associated with the waste inventory stored in the CSF are kept in a database that is managed by ARAO.

At the end of 2013, an RW inventory of 92.4 m^3, with total mass of 50 tons, was kept in the storage facility. The total activity of the waste was 3.2 TBq.

The storage facility currently contains 629 packages, which represents about 80% of its capacity (storage space), and consequently the operator takes actions for volume reduction. The total volume was reduced in the past few years due to several campaigns of DSRS repacking and exemption of emptied and cleared containers. There are approximately 50 receipts of waste from small producers (~2 m^3) per year. Figure I–7 indicates the radionuclides and total activities of the different waste types in storage at the CSF.

Type of RAO	Radionuklidies	Activity (Bq)
T1 (solid, compressible, burnable waste)	Ra-266, Co-60, Am-241, Cd-109, Ag-108, U-238, Co-57, Th-232, H-3	9,5E+08
T2 (solid, compressible, unburnable waste)	Ra-266, Co-60, Am-241, Cd-109, Ag-108, U-238, H-3, U-238, C-14	1,7E+10
T3 (solid, uncompressible, burnable waste)	Ra-266, Co-60, Th-232	1,2E+08
T4 (solid, uncompressible, unburnable)	Ra-266, Co-60, Cd-109,Cs-137, Ag-108, U-238, C-14, Th-232, Ba-133	1,6E+11
ZV0 (smoke detectors)	Am-241, Ra-266	5,7E+09
ZV1 (spent sealed sources: A≤3,7 [GBq])	Ra-266, Co-60, Am-241/Be, U-238, Th-232, Ni-63, Fe-55, Sr-90, Ru-106, H-3	4,4E+11
ZV2 (spent sealed sources: 3,7 [GBq]<A≤37 [GBq])	Ra-266, Eu-152, Co-60, Cs-137, Kr-85, Am-241/Be, Ba-133	1,4E+11
ZV3 (spent sealed sources: 37 [GBq]<A≤370 [GBq])	Eu-152, Am-241, Co-60, Ba-133, Am-241/be	1,7E+12
ZV4 (spent sealed sources: A>370 [GBq])	Co-60	6,3E+11
Total activity on 31. december 2010		3,1E+12

FIG. I–7. Waste forms stored in the CSF.

RW stored in the CSF include waste packed in drums, DSRS in original containers, plastic or metal boxes and plastic bags.

The drums contain mostly contaminated material such as paper, glass and plastic material with induced radioactivity caused by neutron exposure in the research reactor. DSRS are either stored in the original shielding containers or repacked in containers that are subsequently placed in the drums.

Since 2012, ARAO has carried out dismantling of ionizing smoke detectors as a regular activity. The volume reduction factor of this treatment is so significant that, despite the continuous receipt of new waste from waste producers at the facility, the volume of RW in the storage facility has actually been only very slightly increasing.

From 2012 to 2015, ARAO collected 600 litters of liquid waste from research activities in medicine. Prior to their acceptance to the storage facility, these liquid wastes were solidified. The annual increase is estimated to 3–6 m^3.

All waste accepted at the CSF is required to meet the CSF WAC [I–5]. The WAC defines the parameters that need to be met for acceptance at the CSF. These parameters are:

- Package record;
- Inner and outer packaging;
- Waste form (see Fig. I–7);
- Labelling;
- Dose rate;
- Contamination;
- State of matter;
- Corrosion resistance;
- Strength;
- Flammability;

- Explosiveness;
- Organic matter;
- Leachability;
- Chemical stability;
- Degradation effects due to radiation;
- Gas generation;
- Toxic substances;
- Poisonous substances;
- Free liquids;
- Chelating agents;
- Heat generation.

In 2010, the material composition of the inventory in storage in the CSF was assessed [I–20] and is presented in Table I–4. The total volume in Table I–4 is approximately 60 m^3. The remaining volume is designated as unknown (mainly sealed sources). The package type identified as 'D1' represents the standard 208 l drum.

TABLE I–4. MATERIAL COMPOSITION ESTIMATE OF CSF WASTE IN 2010

	Aluminium	Stainless steel	Steel	Undefined metals	Iron and cast iron	Depleted uranium	Cellulose	Sum
Mass (kg)	3100	1000	5000	3600	3900	50	1900	18550
Package type	D1	D1	D1	D1	D1	Shielded container	Mainly D1	
Packaged volume (m^3)	15	2.3	10	8	7.2	0.01	17.5	60

I–5.5. Interacting processes

The following processes interact with the development of the Safety Case:

- Interested parties;
- Independent review;
- The management system utilized to develop the Safety Case.

I–5.5.1. Involvement of interested parties

Relevant interested parties are engaged in the early stages of the development of the Safety Case to allow an understanding of the arguments included in the Safety Case. This includes the regulatory body responsible for nuclear safety, the environmental regulator and national governmental officials. Where relevant, public consultation is also undertaken.

Ensuring transparency

While managing RW in a professional and responsible way, ARAO constantly faces challenges related to social acceptance. Therefore, to accomplish its mission it is of utmost importance to keep in mind communication, education and cooperation with interested stakeholders, such as:

- Interested citizens;
- Residents of local communities where our activities are being performed or where the construction of facilities is planned;
- School-aged children and young people;
- Non-governmental organizations;

- The media.

ARAOs work in this area includes informing, raising awareness, monitoring public responses and opinions, and establishing a dialogue with key groups. These activities are carried out so as to increase knowledge about radiation and to raise awareness about the fact that professional and responsible RW management, as implemented by ARAO, contributes significantly to the quality of the environment and to sustainable development.

ARAO ensures transparent operation with the following activities:

- Informing and awareness raising via the ARAO website www.arao.si;
- Publishing in the media;
- Organizing events and presentations for various stakeholder groups;
- Participating in conferences, roundtables and other public events;
- Providing access to public and environmental information.

Relations with local communities
ARAO recognized the importance of communication in local and regional environments where its activities are carried out, i.e. in the Municipality of Dol pri Ljubljani and the Posavje Region. ARAO mostly communicates with communities through:

- The local media;
- Presentations and reports at meetings of municipal councils, and through mayors.

Doors Open Day
Every year, ARAO holds a 'Doors Open Day'. At a press conference, which takes place before the event itself, ARAOs work is being presented. Visitors are able to attend a lecture on the impact of ionizing radiation on living organisms. Using experiments, ARAO demonstrates that radioactivity is an omnipresent natural phenomenon. There is also a guided tour of the CSF.

Media relations
ARAO communicate with the media proactively and on a regular basis, providing prompt answers to any questions. In the past few years, the topics that the public and the media were most interested in were:

- Planning of the LILW disposal facility;
- Management of spent nuclear fuel;
- Compensation for limited use of space due to nuclear facilities;
- Carrying out the public service of RW management.

Monitoring stakeholder responses and measuring public opinion
ARAO followed stakeholder responses in the media and questions sent via the ARAO website. ARAO has not conducted a public opinion poll on RW management. However, ARAO has gathered some data relevant for RW management from other opinion polls.

I–5.5.2. Independent review

ARAO ensures that Safety Cases are subjected to independent review, as it is also obliged by national legislation. The independent reviewer (nominated by the SNSA) verifies the assumptions, models and assessment results. The positive opinion of the independent reviewer is required be the part of the license documentation delivered to the regulator.

I–5.5.3. Management system

The ARAO has an integrated management system in place that gives the required priority to safety. The ARAO integrated management system is based on IAEA Safety Standards Series No. GS-R-3[2], ISO 9001:2008 and ISO 14001:2004 requirements. Every year, internal audits and management reviews are conducted to ensure the suitability, adequacy and effectiveness of the implemented management system. In addition, external assessment and certification of the management system in accordance with ISO 9001:2008 and ISO 14001:2004 is also conducted every year.

Through a process approach, the ARAO continuously improves the effectiveness of its integrated management system to achieve company's goals and enhance nuclear safety. Based on ARAO's mission, vision and company policy, the main objectives are defined at ARAO official website (https://www.arao.si/index.php/en/arao/mission-and-vision-statement-objectives-and-safety-policies).

ARAO's management system [I–21] is compliant with international safety standards to assure that all safety related work is carried out at a high level of quality and only trained, qualified and competent persons will undertake work that is safety related, including the production of the safety assessment and Safety Case.

Safety is and will be assessed against international standards, and associated uncertainties will be identified, characterized and formally managed. Decisions to move from one step of the project to the next (e.g. operation, optimization, decommissioning) will only be made when compliance with international standards has been demonstrated.

Regulatory control over the facility design, construction and operation will be undertaken by the regulatory body independently from facility development and operational activities. Assurance of the independence will be demonstrated by the management system.

Elements of the management system that will interact with the Safety Case throughout its lifetime also include:

- Management system reviews;
- Internal auditing;
- External auditing as applicable.

I–6. SAFETY ASSESSMENT

I–6.1. Assessment context

The assessment is carried out to demonstrate safety of operations at the CSF. It provides assessments of radiation doses to workers and members of the public during normal operation of the facility and during accidents that could occur over the assumed lifetime of the facility[3], for comparison with relevant legal dose limits and constraints.

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[2] INTERNATIONAL ATOMIC ENERGY AGENCY, The Management System for Facilities and Activities, IAEA Safety Standards Series No. GS-R-3, IAEA, Vienna (2006). GS-R-3 has been superseded and replaced by the following publication: INTERNATIONAL ATOMIC ENERGY AGENCY, Leadership and Management for Safety, IAEA Safety Standards Series No. GSR Part 2, IAEA, Vienna (2016).

[3] In 2022, all the waste in the CSF is anticipated to be disposed of in a LILW disposal facility; afterwards, a decision will be made about future operation of the CSF.

An assessment of the impact of potential accidental events on the facility is performed in order to demonstrate that the design and safety features are sufficiently robust to withstand such events.

The assessment seeks to identify uncertainties and provide consideration to their importance and possible approaches to the management of those uncertainties considered to be important for safety.

A generally conservative approach is taken in respect of assumptions and the assessment.

The following radiological safety criteria have been derived from international safety standards and are used as a basis for evaluation of safety and protection:

- The dose limit for workers from all planned exposure situations is an effective dose of 20 mSv/a [I–22]. On the basis of past operation, storage facility refurbishment, measurements and dose assessment for workers in the CSF, ARAO proposed (in the safety report [I–23]) and the regulatory body approved an effective dose of 10 mSv/a as the dose limit for workers from all planned exposure situations. This criteria and its risk equivalent are not to be exceeded. A lower dose constraint might be established for radiation workers to ensure this limit is not breached.
- To comply with the public dose limit (1 mSv/a), a facility (considered as a single source) is so designed that the calculated dose or risk to the representative person who might be exposed as a result of the operation of the facility does not exceed a dose constraint of 0.1 mSv/a [I–23].

Given that the scope of activities at the CSF excludes the conditioning of spent sealed radiation sources, doses to extremities or to the lens of the eye are not considered.

I–6.1.1. End points for the assessment

The following end points for quantitative assessment will be considered:

- Radiation dose to workers performing the various normal RW management activities at the CSF. Doses received during the various activities are therefore accumulated for these workers.
- Radiation doses to workers and the public due to anticipated operational occurrences.

Doses calculated through the use of the different models and the SAFRAN tool [I–3] will be evaluated against the safety criteria.

I–6.1.2. Approaches to the assessment

Quantitative deterministic assessments of worker doses are performed that consider the full range of activities performed by various occupational groups of the CSF. Assessments are based on the following assumed tasks of personnel during normal operational activities performed at the CSF:

- Unloading of RW from the transport container: 2 workers, 50 activities per year, 10 minutes per activity.
- Control measurements: 2 workers, 50 activities per year, 10 minutes per activity.
- Packaging: 2 workers, 50 activities per year, 10 minutes per activity.
- Transfer of RW package to its storage location in the CSF.

- Monthly inspection and survey of the storage location and preventive maintenance (e.g. cleaning, change of light bulbs, control of fire protection system): 2 workers, 120 hours per year each.

In order to increase confidence in the results, hazards and initiating events relevant to the CSF were identified and assessed using two different methods. The first method that was used is referred to as the HAZOP method. The second method that was used was the SAFRAN tool [I–3], which implements the GSG-3 [I–1] and SADRWMS [I–2] methodologies.

Quantitative assessments are performed of the potential impact to workers and the public from possible occurrences as well as specific preventative and mitigating measures. Using a risk-based approach, design basis events and beyond design basis events are also considered for more detailed analysis as accident scenarios. The anticipated consequences associated with such events will be listed with comments and recommendations for further analyses and proposed preventative and mitigating measures. The SAFRAN tool (version 2.3.2.0) [I–3] was used, which forms the basis for the approach to the safety assessment.

A qualitative assessment is performed of the availability and level of implementation of an integrated management system to ensure a sustained level of safety with an emphasis on operational activities at the facility. The focus is on radiation protection, work procedures, quality assurance aspects (mainly record keeping and change management) and processes for the management of limits and conditions.

The results from the quantitative and qualitative assessment are compared to the proposed target and objectives established for the optimization of protection. No specific optimization is made in the case of doses below 1 mSv/a.

A qualitative assessment is performed for the non-radiological hazards of the facilities and the listing of specific control measures. Non-radiological hazards are listed and categorized in terms of their hazard potential. Comments and recommendations are made per hazard as applicable.

Uncertainties inherent to the assumptions made in the quantitative assessments and any other uncertainties identified during the safety assessment will be evaluated to determine their impact on safety. Uncertainties with a significant impact on safety will be listed with recommendation for its management.

I–6.2. Description of safety elements and functions

The main activity to be carried out in the facility is storage; other waste management operations are not considered. During normal operation, the key activities are related to:

- Unloading the waste drums/packages from the transport vehicle;
- Radiation and contamination monitoring of the transport vehicle and the waste drums/packages upon receipt;
- Transfer of the waste drums/packages into the storage building;
- Acceptance and placing of the waste drums/packages into their storage location within the storage building;
- Storage of the waste drums/packages for the entire storage period;
- Periodic inspection and radiological monitoring of the storage building and the waste drums/packages.

A simple process was defined in SAFRAN to describe the ongoing work in the CSF with the objective of defining activities and connecting them with rooms and waste components in SAFRAN (see Fig. I–8). The following activities were defined:

- Unloading of RW from the transport container;
- Control measurements;
- Packaging;
- Emplacement in Storage 0 up to Storage 9 (represent the storage in different compartments).

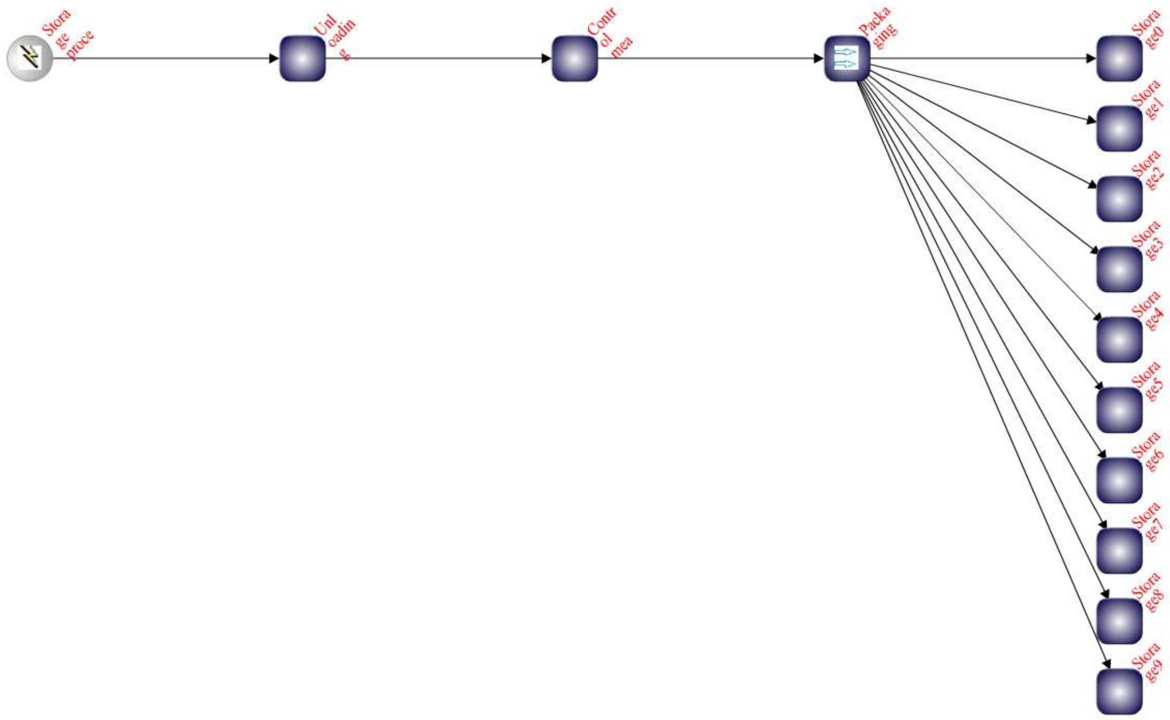

FIG. I–8. Representation of the CSF work processes in SAFRAN.

Taking into consideration Slovenian legislation and GSG-3 guidance, the safety functions and associated SSCs were identified for the CSF facility. All the SSCs were classified in two different classes considering Slovenian regulations and their importance for the safety of the facility. These classes are:

- SSCs important for safety;
- SSCs not important for safety.

The functions that were identified for the storage facility are:

- **Containment (C)** of the radioactivity – radionuclides stay during the operation of the facility limited inside the waste packages and storage facility respectively;
- **Shielding (S)** from the radiation that arises from the RW;
- **Protection (P)** of the waste – physical and technical security of the waste;
- **Supporting (Su)** function –supports the implementation of other safety functions.

Table I–5 lists all the SSCs with their safety classification and function(s) they are performing.

TABLE I–5. DEFINITION OF THE SSCS, THEIR SAFETY CLASSIFICATION AND FUNCTION(S)

SSC	Function *	Safety classification: Important for safety (Yes/No)
Storage building	C, S	Yes
Horizontal sewerage	C	Yes
Underground tank	C	Yes
Sewerage to collect precipitations	C	Yes
Ventilation system	C, Su	Yes
Electricity supply	Su	Yes
Safety lighting	Su	No
Telecommunication	Su	Yes
Lightning rods	Su	Yes
Active fire protection	C	Yes
Passive fire protection	Su	Yes
Physical and technical security	P	Yes
RW packages	C, S	Yes
Pallets	Su	No
Transportation system inside the storage facility	Su	No
Radiological monitoring	Su	No
Hydrological monitoring	Su	No
Meteorological monitoring	Su	No

* C = Containment, S = Shielding, P = Protection, Su = Supporting function

For all the SSCs that were classified as “important for safety”, the special procedures for maintenance and inspection were prepared.

I–6.3. Passive safety and defence in depth

I–6.3.1. Passive systems

Passive systems contributing to the safety of the facility and their operations were applied in three areas:

- Optimization of external exposure of workers and the public;
- Minimization of the potential impact to the environment and the public, due to operational occurrences and accidents;
- Prevention of unauthorized access to the facility.

The optimization of external exposure during the waste management operations is based upon:

- The shielding capacity of the external and internal wall structures due to their proper shielding characteristics;
- The labyrinth system used for the storage areas;
- The physical delineation of different areas, which contributes to optimization during the working activities.

Operational occurrences and accidents that might cause a release of radioactive material are evaluated (e.g. decontamination activities that might involve the use of liquids). For such events, the storage building is designed in way that the run off by gravity is allowed from any interior area and there is a residue collection and retention system that permits monitoring before release to the environment (see Figs I–9 and I–10). This passive system ensures the minimization of the potential impact into the environment and the public caused by operational occurrences and accidents.

FIG. I–9. System for collection and retention of liquids.

FIG. I–10. System for collection and retention of liquids, outer part.

A perimeter fence restricts access by unauthorized persons to the facility and its surroundings. Safety evaluation demonstrates that a representative at any location at any point within the perimeter, even under conservative conditions, does not exceed dose limits established for the public. In addition to the fence, other passive barriers, specifically the security system, limit the unintentional access of a non-authorized person (see Figs I–11 and I–12).

FIG. I–11. Video surveillance system monitor.

FIG. I–12. Security system commands.

To prevent the corrosion of the drums and other metallic material inside the CSF, a heating and moisture control system is installed and operated (see Figs I–13 and I–14).

FIG. I–13. Heating and moisture control system.

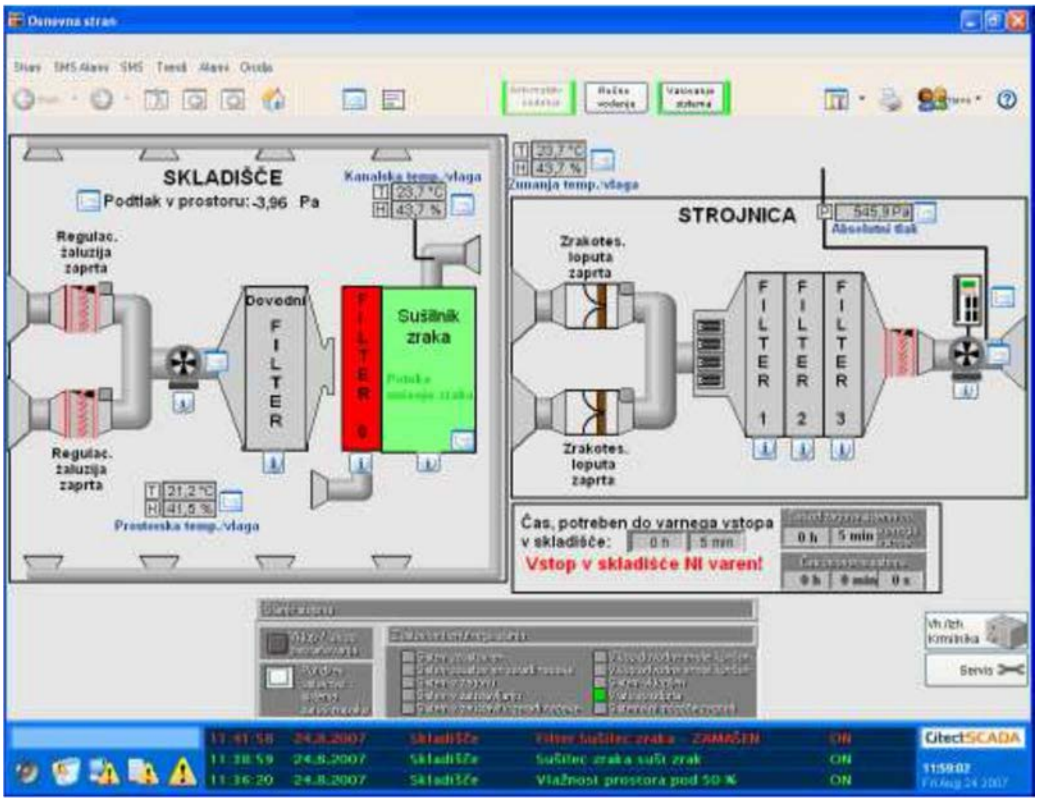

FIG. I–14. Application to operate the moisture control system.

Prevention of unauthorized access to the facility

The site of the storage building is located in a suburban area. Consequently, the establishment of an efficient physical protection system that prevents unauthorized access to the storage building it required. For this reason, the following barrier systems are considered:

- Site perimeter fence with a gate to control access;
- High integrity door to the personnel entrance;
- High integrity gate to the storage areas.

I–6.3.2. Defence in depth

Defence in depth principles were applied in three main areas:

- Storage of RW;
- Planned source and RW management operations;
- Minimization of the potential impact to the environment and the public, due to operational occurrences and accidents.

Storage of RW

Isolation of radioactive materials is one of the main objectives of storage. For this reason, the following physical and passive barriers were established:

- Facility perimeter fence;
- Storage building structure;
- Delineation of the storage areas with limited access to each (physical and technical protection);
- All the RW is packaged in proper containers (e.g. drums, overpacks) to meet the WAC (2 mSv/h contact dose).

These isolation barriers also work as containment barriers for a potential radioactive material release.

Planned waste management operations

Storage of RW requires only limited operations. As explained previously, operations are mainly the reception of the RW, its placement in the storage area and routine inspection. During such operations, ample light is a factor important to ensuring safe performance (Fig. I–15). The existence of auxiliary autonomous equipment enables to perform the operations conceived in the event of a failure of the lighting system.

FIG. I–15. Light system in the CSF.

Because the CSF is partially an underground facility, there is a lot of radon exhalation from underground. To protect the workers and the visitors, a ventilation system is used before the facility is entered, to reduce the Rn-222 activity concentration below 200 Bq/m^3. To protect the public around the CSF, a filtration system is used (Fig. I–16).

FIG. I–16. Ventilation and filtration system.

Minimization of the potential impact to the environment and the public, due to operational occurrences and accidents

Barriers conceived to reduce the impact of events that cause the release of radioactive material in the order in which they function are:

- Segregation of storage and operation areas in such a way that the potential contamination is limited.
- The floor slab has a steel floated finish with an epoxy paint coating to provide a durable and easily decontaminated surface.
- The storage building is provided with an internal floor drain system to direct any internal liquid traces generated to a sump pit.
- The sump pit permits the monitoring of the radioactivity content and is only evaluated when compliance with established restrictions is necessary.
- The fire protection alarm system is installed to reduce the potential of fire.

I–6.4. Site characterization

The characterization of the site was based upon:

- Relevant information from meteorological data from the near weather stations;
- Information provided by the appropriate authority related to the facilities and activities in the surrounding vicinity of the site;
- Samples from drilled boreholes (Fig. I–17) and other performed tests;
- Visual observation and sampling in the field.

FIG. I–17. Borehole system around the CSF is equipped as piezometer.

Although the information provided above is adequate to perform the safety assessment, a conservative approach is adopted for the following aspects:

- Relevant characteristics of the regional physical geography, stratigraphy and lithography, as well as a geological structural history of the region;
- The current distribution of the population surrounding the facility, as well as that projected during its lifetime.

I–6.5. Operational safety aspects

Risks from operations not related to the management of CSF are not further considered. This is due to the following factors:

- The storage building is physically isolated from other activities;
- Facilities close to the storage building are not conducting activities that might impact the safety of the storage building.

I–6.6. Management system

The management system is designed and implemented based on IAEA Safety Standards Series No. GS-R-3[4]. The system defines clearly the responsibilities at all levels reflecting the management commitment to security and safety of the installation. Quality management is applied throughout the design, construction and operation of the CSF.

I–6.7. Waste management practices

The outcome of the qualitative assessment of the waste management practices, as implemented by the operating organization, is described in Table I–6.

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, The Management System for Facilities and Activities, IAEA Safety Standards Series No. GS-R-3, IAEA, Vienna (2006). GS-R-3 has been superseded and replaced by the following publication: INTERNATIONAL ATOMIC ENERGY AGENCY, Leadership and Management for Safety, IAEA Safety Standards Series No. GSR Part 2, IAEA, Vienna (2016).

TABLE I–6. IMPLEMENTED OUTCOMES OF THE QUALITATIVE ASSESSMENT OF THE WASTE MANAGEMENT PRACTICES

Item	Requirement	Compliance comments	Reference
1	Clearly defined responsibilities for waste management.	The legal framework of Slovenia specifies the responsibilities for the generation and management of RW. The construction and operation of the waste management facilities demonstrate intent and commitment.	IRPNSA [I–24]
2	Implementation of the principles of waste minimization and avoidance, namely, re-use or re-processing of waste, return to supplier, safe and secure storage and conditioning and final disposal of waste.	Principles defined in the legal framework and implemented in the case of RW to the point of conditioning. Final disposal for LILW is under final design.	Resolution on the 2006–2015 National Programme for Managing RW and Spent Nuclear Fuel [I–9]
3.	Hazards and the generation of secondary waste associated with all waste management operations (routine and ad hoc) are known, monitored, projected and managed by due management processes.	The treatment of RW from small producers is well planned and executed in the adjacent hot cell facility which is designed to mitigate exposure.	
4.	Interdependencies between the various steps of waste management are known and managed. WAC are defined, waste management activities and the outputs of such activities are aligned with set WAC.	Consignments stored RW are assessed in accordance with appropriate WAC at the generators facility and again at the CSF as part of collection and transport procedure. The operator will need to consider how the current actions and specification are aligned with future disposal options and associated WAC.	WAC for CSF [I–5]
6.	Conditioned RW will be stored in a dedicated storage area with passive safety features and adequate access control.	Only conditioned waste packages are transferred and emplaced in a dedicated long term storage facility.	Procedures and WAC

I–6.8. Development and justification of scenarios

I–6.8.1. Normal operation

The following normal operation scenarios for which worker doses are assessed are identified in Section I–6.1.2:

- Unloading of RW from the transport container;
- Control measurements;
- Packaging;
- Transfer of RW package to its storage location in the CSF;
- Monthly inspection and survey of the storage location and preventive maintenance (e.g. cleaning, change of light bulbs, control of fire protection system).

I–6.8.2. Identification and screening of hazards and initiating events

In order to increase confidence in the results, hazards and initiating events relevant to the CSF were identified and assessed using two different methods. The first method that was used is referred to as the HAZOP method. The second method that was used was the GSG-3 [I–1] and SADRWMS methodology, which is implemented in the SAFRAN tool.

Activities using the HAZOP method were carried out during a series of meetings by the ARAO multi-disciplinary team of experts who were familiar with the facility and also have experience in safety assessment, radiation protection, operation of the facility, civil engineering, environment protection, geology, hydrology, and meteorology. Initial work focused on the development of normal operational scenarios and then, using the HAZOP method, the hazards for the CSF were identified and screened on the basis of the risk posed by each hazard. Risk was defined as a function of consequence and likelihood, whereby the consequences that possibly result from the event were categorized into four classes:

- Class 1: Critical event – requires an immediate response; the SSCs do not take anymore their safety functions, the facility is damaged and releases of radioactive material in the environment can occur.
- Class 2: Serious event – without intervention some of the SSCs might be damaged and stop performing their safety functions. The facility is damaged but there is no release of radioactive material.
- Class 3: Event with medium importance – attention needs to be paid on this event; the safety functions are performed but without backup. The impact on the workers is very small but measurable.
- Class 4: Event with low importance – the event doesn't have an effect on SSCs, although it might cause inconveniences especially for workers. The impact is not measurable.

Likelihood (i.e. the probability that the event will occur) is divided into five classes:

- Likelihood 1: the event can occur every 5 years;
- Likelihood 2: the event can occur every 5 to 20 years;
- Likelihood 3: the event can occur every 20 to 50 years;
- Likelihood 4: the event can occur every 50 to 500 years;
- Likelihood 5: the event can occur every 500 years or more.

The risk is calculated as a function of the consequence and likelihood of the hazard scenario (Table I–7). Hazard scenarios with a risk between 1 and 9 are considered for quantitative analysis as accident scenarios within the safety assessment.

TABLE I–7. RISK AS A FUNCTION OF CONSEQUENCE AND LIKELIHOOD

Likelihood → Consequence ↓	1	2	3	4	5
1	**1**	**3**	**3**	4	5
2	**2**	**4**	**6**	8	10
3	**3**	**6**	**9**	12	15
4	4	8	12	16	20

As a result of the HAZOP assessment, the following scenarios were identified as requiring further analysis:

- Drop of the package or damaged package;
- Explosion at the CSF site – the scenario will be assessed through security assessment;
- Fire involving waste packages.

Theft of a RW package (hazard scenarios 13, 25, 26, and 27) was also identified as requiring further (quantitative) assessment, but will be assessed through a separate security assessment.

Table I–8 presents the scenarios that were identified as relevant to the CSF and assessed using the HAZOP method.

TABLE I–8. ASSESSMENT OF HAZARD SCENARIOS USING THE HAZOP METHOD

No.	Deviation	Cause	Consequences	Safety measures	C*	P*	Risk
1	Waste is not received in the CSF	Various causes	N/A				
2	Waste is received, but not accepted in the CSF	Waste properties, inadequate documentation, mistakes, superficiality	RW doesn't meet WAC, the discrepancy is removed.	Transparent procedures and instructions, education and training of employee, help producers with the preparation of packages for storage.	4	4	16
3	In addition to planned receipt, unplanned receipt also occurs	Lack of information, carelessness	On the CSF site, there is more waste than planned.	Extend working time or provide additional workers.	4	3	12
4	Package not in compliance with guidelines	Lack of information, carelessness	The package is received but not placed inside the CSF. A report is submitted to the regulatory body.	Producers are well informed.	4	3	12
5	Waste is not received in the CSF	Unfavourable weather conditions	Receipt is postponed – no influence on safety.	The weather forecast is checked.			
6	Waste is not received in the CSF	Power outage, CSF ventilation system doesn't work	Receipt is postponed – no influence on safety.	Power backup or natural ventilation of the facility.			
7	Waste is not received in the CSF	Earthquake or some other natural disaster	Receipt is postponed – no influence on safety.				
8	During receipt the unannounced visit of inspectors, or urgent receipt occurs	Coincidence, bad coordination	Receipt is postponed – no influence on safety.	Communication, planning.			
9	-	-	N/A	-	-	-	-
10	-	-	N/A	-	-	-	-

TABLE I–8. ASSESSMENT OF HAZARD SCENARIOS USING THE HAZOP METHOD (cont.)

No.	Deviation	Cause	Consequences	Safety measures	C*	P*	Risk
11	No transport	Forklift – truck doesn't work.	RW package is left outside the storage facility. Package is transported to the facility next day. There is no contamination.	Permanent check of transport equipment, regular services. Competent, educated and trained personnel. Agreement about fast service or the forklift – truck rental.	4	4	16
12	No transport	Hurrying, oversight	RW package is left outside the storage facility. Package is transported to the facility next day. There is no contamination.	More employees at the activity. Competent, educated and trained personnel. The platform in front of the facility is checked before personnel leave the facility site.	3	3	9
13	No transport	Hurrying, oversight	RW package is left outside the storage facility. Package can be stolen from the site.	More employees at the activity. Competent, educated and trained personnel. The platform in front of the facility is checked before personnel leave the facility site.	2	5	10
14	No transport	Hurrying, oversight	RW package is left outside the storage facility. Due to bad weather conditions (e.g. rain), there is a potential contamination of the surrounding.	More employees at the activity. Competent, educated and trained personnel. The platform in front of the facility is checked before personnel leave the facility site. The water from the platform is directed through a drainage system into the system for water collection and retention.	3	4	12
15	-	-	N/A	-	-	-	-
16	Transport is not successful	The package drops during the transport.	Contamination of the surrounding, there is also potential of external and internal radiation of workers and other people around the accident (institute workers, security guards, farmers).	Permanent check of the transport equipment, regular services. Competent, educated and trained personnel.	2	4	8

TABLE I–8. ASSESSMENT OF HAZARD SCENARIOS USING THE HAZOP METHOD (cont.)

No.	Deviation	Cause	Consequences	Safety measures	C*	P*	Risk
17	Something else occurs (beside transport).	Hurrying, carelessness.	Collision with a sharp object (doors, ventilation system, installations…). A package is damaged, potential of external and internal radiation of workers and other people around the accident exists.	Competent, educated and trained personnel. The transport complies with the guideline for internal transport.	2	4	8
18	Something else occurs (beside transport).	Hurrying, carelessness.	Collision with a sharp object (doors, ventilation system, installations…). The SSC important for the safety is damaged (ventilation system is damaged).	Competent, educated and trained personnel. The transport complies with the guideline for internal transport. During and after the transport, control of the facility is necessary. If this event happened, the responsible people are informed.	3	4	12
19	Something else occurs (beside transport).	Hurrying, carelessness.	Collision with a sharp object (doors, ventilation system, installations…). The SSC important for the safety is damaged (concrete construction is damaged).	Competent, educated and trained personnel. The transport complies with the guideline for internal transport. During and after the transport, control of the facility is necessary. If this event happened, the responsible people are informed.	4	4	16
20	Something else occurs (beside transport).	Hurrying, carelessness.	Collision with a sharp object (doors, ventilation system, installations…). SSCs important to safety (e.g. electrical installation) are damaged.	Competent, educated and trained personnel. The transport complies with the guideline for internal transport. During and after the transport, control of the facility is necessary. If this event happened, the responsible people are informed.	4	4	16
21	Transport occurs too early.	Hurrying, carelessness.	The transport of the package to the storage facility occurs before it is checked for compliance with the WAC for storage. Increased exposure of workers can occur.	Competent, educated and trained personnel. The transport complies with the guideline for internal transport. Double check before transport is necessary; if so, the responsible people are informed.	3	4	12

TABLE I–8. ASSESSMENT OF HAZARD SCENARIOS USING THE HAZOP METHOD (cont.)

No.	Deviation	Cause	Consequences	Safety measures	C*	P*	Risk
22	Transport occurs too early.	Hurrying, carelessness.	Transport of the package to the storage facility occurs before it is checked. Package is not packed correctly. Drop of the package can occur.	Competent, educated and trained personnel. The transport complies with the guideline for internal transport. Double check before transport is necessary; if so, the responsible people are informed.	3	3	9
23	Transport occurs too early.	Hurrying, carelessness.	Transport of the package to the storage facility occurs before it is checked. Package contains liquid waste. A spill of liquids can occur.	Competent, educated and trained personnel. The transport complies with the guideline for internal transport. Double check before transport is necessary; if so, the responsible people are informed.	3	4	12
24	Transport occurs too late.	Hurrying, carelessness.	Due to different reasons the package stays on the platform longer than is needed. In the meantime, rain and potential contamination of the surrounding can occur.	The water from the platform is directed through a drainage system into the system for water collection and retention. Competent, educated and trained personnel. The platform in front of the facility is checked before personnel leave the facility site.	3	4	12
25	There is no more storage space for the packages.	Storage facility is destroyed due to airplane crash or terrorist attack. The majority of SSCs don’t function anymore.	RW is scattered. The surrounding of the storage facility is contaminated.	The security of the facility is sufficient. The employees are educated, trained and competent.	1	5	5
26	The package is not stored.	A worker steals a package from the CSF.	Increased internal and external radiation to members of the public.	The security of the facility is sufficient. The employee is educated, trained and competent. The psychological health of the workers is checked.	2	4	8
27	The package is not stored.	A person steals a package from the CSF.	Increased internal and external radiation to members of the public.	The security of the facility is sufficient. The employees are educated, trained and competent.	2	5	10

TABLE I–8. ASSESSMENT OF HAZARD SCENARIOS USING THE HAZOP METHOD (cont.)

No.	Deviation	Cause	Consequences	Safety measures	C*	P*	Risk
28	The storage facility is damaged	A low magnitude earthquake.	Due to the earthquake some damage – cracks on the building can occur.	The facility design and construction is “seismically safe”. The waste is packed in the drums that still provide their function of containment. After such an event, immediate repair of the facility is needed.	3	4	12
29	The storage facility is damaged	A high magnitude earthquake.	Due to the earthquake the building is demolished. The building structure falls on the packages, and contamination of surroundings is possible.	After such an event, remediation of the waste and the site is needed.	2	5	10
30	The storage facility is damaged	Fire inside the storage facility.	Waste is burned as a result of the fire. Contamination of surroundings is possible.	Fire protection system. Work performed inside the building complies with fire safety guidelines and regulations.	2	4	8
31	The storage facility is damaged	Flood	Waste containers are damaged as a result of the flood. Transport of radionuclides in water is possible.	The CSF is constructed in area with very low probability of flooding.	3	5	15
32	The storage facility is damaged	Due to natural or artificial processes cracks in the concrete occur.	Water percolates into the CSF.	Drainage system inside the CSF. Some periodic inspection need to be done.	4	3	12
33	The storage facility is damaged	Due to natural or artificial processes corrosion of the drums and packages occur.	Waste containers are corroded as a result of moisture. Transport of radionuclides in water is possible.	Usage of suitable certified packaging. Regular inspection of the drums is prescribed [I–28].	4	3	12
34	-	-	N/A	-	-	-	-
35	-	-	N/A	-	-	-	-
36	-	-	N/A	-	-	-	-
37	-	-	N/A	-	-	-	-
38	-	-	N/A	-	-	-	-

TABLE I–8. ASSESSMENT OF HAZARD SCENARIOS USING THE HAZOP METHOD (cont.)

No.	Deviation	Cause	Consequences	Safety measures	C*	P*	Risk
39	CSF is abandoned	Different causes (natural disaster, social/political upheaval, etc.) result in the CSF is abandoned.	Natural degradation and disintegration of artificial natural barriers (SSCs) important for safety. Water can percolate into the facility and containers. Radionuclides are released to biosphere	Robustness of engineering barriers, multifunctional concept.	2	5	10
40	CSF is not maintained.	Financing of CSF maintenance is stopped.	Some of the SSCs stop performing their safety function (e.g. ventilation, air drying system, technical security). Faster degradation of the barriers due to corrosion, etc.	Permanent financing is ensured.	3	4	12
41	CSF is not maintained.	Financing of CSF maintenance is reduced.	Deterioration of the CSF SSCs is not immediately addressed.	Permanent and sufficient financing is ensured.	4	3	12
42	CSF is not monitored.	Financing of CSF monitoring is reduced.	Detection of CSF SSCs deterioration is not ensured, resulting in an increase in potential for accidents.	Permanent and sufficient financing is ensured.	4	3	12
43	-	-	N/A	-	-	-	-
44	-	-	N/A	-	-	-	-
45	Additional (unforeseen) activities on the CSF site	Careless or incompetent employee	Unnecessary external exposure or contamination can occur.	Adequate supervision on the activities in the CSF is ensured. The employees are educated, trained and competent.	3	4	12
46	Work starts earlier	Careless or incompetent employee	N/A	N/A			
47	Work starts late	Careless or incompetent employee	Late maintenance of the SSCs important for the safety. The SSCs don’t perform their functions well. Unnecessary external exposure.	Adequate supervision on the activities in the CSF is ensured. The employees are educated, trained and competent	4	4	16

TABLE I–8. ASSESSMENT OF HAZARD SCENARIOS USING THE HAZOP METHOD (cont.)

No.	Deviation	Cause	Consequences	Safety measures	C*	P*	Risk
48	Other activities in the storage facility start late.	Careless or incompetent employee	Late monitoring. Due to unknown conditions inside the facility, the radiological safety of the working place is assessed inappropriately, and unnecessary external exposure or contamination can occur.	Adequate supervision of the activities in the CSF is ensured. The employees are educated, trained and competent.	4	4	16

*C = Consequence, P – Probability

The second method used to perform the assessment involved using the SAFRAN tool to identify and screen hazards and initiating events, and identify and analysis relevant accident scenarios. Figure I–18 shows a screenshot of the postulated initiating events that were screened using the SAFRAN tool.

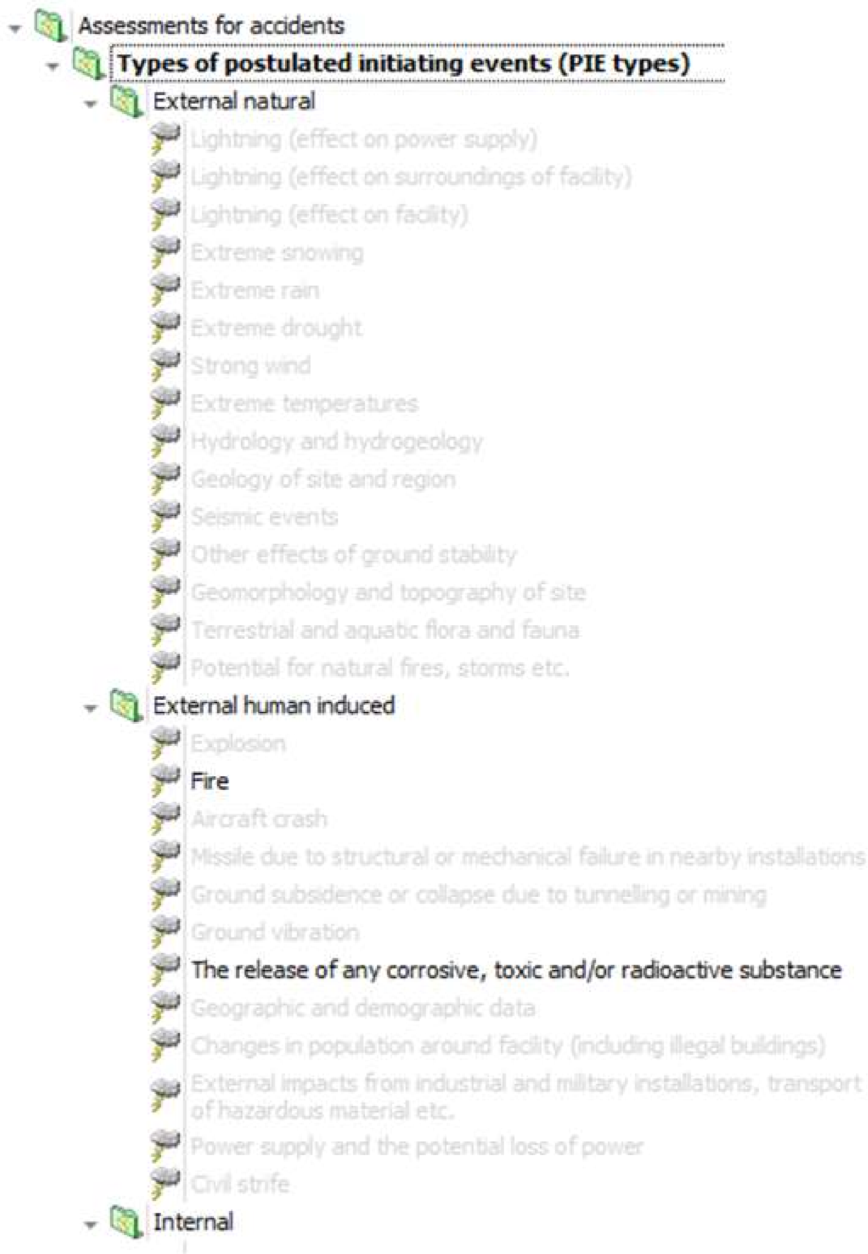

FIG. I–18. Screening of postulated initiating events in the SAFRAN tool.

The results of the screening of postulated initiating events are recorded in the SAFRAN file, along with justification of any scenarios considered not to be relevant. These results are summarized in Table I–9.

TABLE I–9. ASSESSMENT OF HAZARD SCENARIOS USING SAFRAN [I–3]

Name	Relevance	Relevance – justification (if not relevant)	Category
Lightning (effect on power supply)	Not relevant	The facility can perform its basic function (storage of RW) safely without electrical supply, all important systems have backup supply.	External natural
Lightning (effect on surroundings of facility)	Not relevant	Because the CSF is an underground facility, covered with a layer of soil, the lighting doesn't have an impact on the surroundings of the facility.	External natural
Lightning (effect on facility)	Not relevant	Because the CSF is an underground facility, covered with a layer of soil, the lighting doesn't have an impact on the facility.	External natural
Extreme snowing	Not relevant	Because the CSF is an underground facility, covered with a layer of soil, the extreme snowing doesn't have an impact on the facility.	External natural
Extreme rain	Not relevant	Because the CSF is an underground facility, covered with a layer of soil, the extreme rain doesn't have an impact on the facility and its safety. Also, the groundwater is 10 m below the surface, and above there is a very permeable layer that drains all the precipitations.	External natural
Extreme drought	Not relevant	Extreme draughts are not typical for the area of the facility.	External natural
Strong wind	Not relevant	Because the CSF is an underground facility, covered with a layer of soil, the strong wind doesn't have an impact on the facility and its safety.	External natural
Extreme temperatures	Not relevant	Because the CSF is an underground facility, covered with a layer of soil, the extreme temperatures don't have an impact on the facility and its safety.	External natural
Hydrology and hydrogeology	Not relevant	The storage facility lies on an area that is 10 m higher than the river that flows 1 km from the site, also groundwater is 10 m below the surface.	External natural
Geology of site and region	Not relevant	The geology is well known and is not expected to change in the near future.	External natural
Seismic events	Not relevant	The design and the construction of the facility are seismically safe.	External natural
Other effects of ground stability	Not relevant	There are no other effects of ground stability.	External natural
Geomorphology and topography of the site	Not relevant	Nothing from geomorphology or topography of the site can affect the safety of the facility.	External natural
Terrestrial and aquatic flora and fauna	Not relevant	The storage facility is well locked, and flora and fauna can't affect the interior of the facility.	External natural

TABLE I–9. ASSESSMENT OF HAZARD SCENARIOS USING SAFRAN [I–3] (cont.)

Name	Relevance	Relevance – justification (if not relevant)	Category
Potential for natural fires, storms, etc.	Not relevant	Due to the lightning rods system and because the surrounding of the facility is covered with grass, the possibility for natural fire is low.	External natural
Flooding	Not relevant	Because the facility site lies 1 km from the river and it is on the elevation that is 10 m higher than river no flooding is expected (see Fig. I-7).	External natural
Explosion	Not relevant	Different reasons (e.g. terrorist attack) could cause an explosion in the CSF. The scenario will be analysed through the security assessment.	External human induced
Fire	Relevant		External human induced
Aircraft crash	Not relevant	Same as for the explosion scenario – the scenario will be assessed through the security assessment.	External human induced
Missile due to structural or mechanical failure in nearby installations	Not relevant	A missile due the structural or mechanical failure from nearby installations is not foreseen.	External human induced
Ground subsidence or collapse due to tunnelling or mining	Not relevant	The CSF lies in a flat area where tunnelling is not foreseen and also the probability of the tunnel construction is low. There are no natural resources in the area suitable for mining and, therefore, no subsidence or collapse is anticipated to occur.	External human induced
Ground vibration	Not relevant	In the vicinity of the CSF, there is no source of ground vibration that can affect the the facility and its safety.	External human induced
The release of any corrosive, toxic and/or radioactive substance	Relevant	The PIE is included in the drop and the fire scenario.	External human induced
Geographic and demographic data	Not relevant	The impact of the facility is so low that change in geographic and demographic data can't affect the CSF.	External human induced
Changes in population around facility (including illegal buildings)	Not relevant	The impact of the facility is so low that change in geographic and demographic data can't affect the CSF.	External human induced
External impacts from industrial and military installations, transport routes of hazardous material, etc.	Not relevant	There is no such route in the vicinity of the CSF.	External human induced
Power supply and the potential loss of power	Not relevant	The power supply is not necessary for the CSF operation.	External human induced
Civil strife	Not relevant	The CSF and the waste emplaced therein (LILW) can't be seen important in the case of a civil strife.	External human induced

TABLE I–9. ASSESSMENT OF HAZARD SCENARIOS USING SAFRAN [I–3] (cont.)

Name	Relevance	Relevance – justification (if not relevant)	Category
The acceptance (inadvertent or otherwise) of incoming waste, waste containers, process chemicals, conditioning agents, etc., that do not meet the specifications (acceptance criteria) included in the design basis.	Not relevant	If this event occurs, the consequences will be negligible for workers and the environment and this “state” will last for a short period, as the waste package will be treated to meet the WAC for storage in the CSF.	Internal
The processing of waste that meets acceptance criteria but that is subsequently processed in an inappropriate way for the particular type of waste (either inadvertently or otherwise).	Not relevant	The CSF is not a processing facility.	Internal
A criticality event due to the inappropriate accumulation of fissile material, change of geometrical configuration, introduction of moderating material, removal of neutron absorbing material or various combinations of these.	Not relevant	The amount of fissile material stored in the CSF is limited so that a criticality event cannot occur. Parameters for ensuring subcriticality are controlled by engineered and/or administrative safety measures.	Internal
Explosion due to the evolution of explosive gas mixtures	Not relevant	The CSF is the operating facility that is monitored on daily basis and also the WAC are such that the waste producing gas are not acceptable. The major gas present in the CSF is radon, because the facility is partial under the surface and covered with soil layer. Owing to the presence of radon, the CSF has a ventilation system before the entrance into the facility. The occurrence of a mixture of explosive gases is not foreseen.	Internal
Spontaneous combustion	Not relevant	The WAC prevent that waste is stored that can be spontaneously combusted.	Internal
Local hot spots generated by malfunctions of SSCs	Relevant	The PIE is included in the fire scenario.	Internal
Sparks from machinery, equipment or electrical circuits	Relevant	The PIE is included in the fire scenario.	Internal

TABLE I–9. ASSESSMENT OF HAZARD SCENARIOS USING SAFRAN [I–3] (cont.)

Name	Relevance	Relevance – justification (if not relevant)	Category
Sparks from human activities such as welding or smoking	Relevant	The PIE is included in the fire scenario.	Internal
Explosions	Not relevant	Different reasons (e.g. a malicious act) could cause an explosion in the CSF. The scenario will be analysed through the security assessment.	Internal
Gross incompatibilities between the components of a process system and the materials introduced into the system.	Not relevant	The CSF is not a processing facility.	Internal
The degradation of process materials (chemicals, additives or binders) due to improper handling and storage.	Not relevant	The stored waste packages are inspected regularly, and if some degradation is noticed, remedial actions are required.	Internal
The failure to take account of the non-radiological hazards presented by the waste (physical, chemical or pathogenic).	Not relevant	The stored waste packages are inspected regularly, and if the failure is noticed, remedial actions are required.	Internal
The generation of a toxic atmosphere by chemical reactions due to the inappropriate mixing or contact of various reagents and materials.	Not relevant	All the stored waste meets WAC. The facility is inspected regularly, and procedures for non-compliant waste do exist. The risk for humans and the environment is very low.	Internal
Dropping waste packages or other loads due to mishandling or equipment failure, with consequences to the dropped waste package and possibly to other waste packages or to the SSCs of the facility.	Relevant	Included in the drop scenario.	Internal
Collisions of vehicles or suspended loads with the SSCs of the facility or with waste packages, waste containment vessels and pipes.	Relevant	Included in the drop scenario.	Internal
Failures of SSCs	Relevant	Included in the drop and the fire scenario.	Internal
The generation of missiles and flying debris due to the explosion of pressurized components or the gross failure of rotating equipment.	Not relevant	No such type of equipment that can generate missiles or flying debris is used in the CSF.	Internal

TABLE I–9. ASSESSMENT OF HAZARD SCENARIOS USING SAFRAN [I–3] (cont.)

Name	Relevance	Relevance – justification (if not relevant)	Category
The malfunctioning of heating or cooling equipment, leading to unintended temperature excursions in process systems or storage systems.	Not relevant	No heating or cooling equipment inside the CSF. Just the system to control the humidity inside the facility, but this system can't cause unintended temperature excursions.	Internal
The malfunctioning of process control equipment.	Not relevant	No such type of equipment is used in the CSF.	Internal
The malfunctioning of equipment that maintains the ambient conditions in the facility, such as the ventilation system or dewatering system.	Not relevant	The malfunction of such systems doesn't affect the safety of the facility directly. If there are some malfunctions, the facility need to be operated under the instructions prepared for such events. In case the ventilation system doesn't work, it is not allowed to enter the facility until the doors are opened and the facility is naturally ventilated.	Internal
The malfunctioning of monitoring or alarm systems so that an adverse condition goes unnoticed.	Not relevant	The monitoring and the ventilation system are regularly checked. The observed malfunctioning leads to immediate actions. If adverse conditions occur, their impact on workers and the environment is negligible.	Internal
Incorrect settings (errors or unauthorized changes) on monitors, alarms or control equipment.	Not relevant	The settings need to be checked regularly. Short incorrect settings have negligible impact on workers and the environment.	Internal
The failure to function when called upon of emergency equipment such as the fire suppression system, pressure relief valves and ducts.	Not relevant	In the CSF, there are no such active systems whose failure can cause problems with the operation of the facility.	Internal
The failure of the power supply, either the main system or various subsystems.	Not relevant	In the CSF, there are no such active systems whose failure can cause problems with the operation of the facility.	Internal
The malfunctioning of key equipment for handling waste, such as transfer cranes or conveyors.	Relevant	Included in the drop scenario.	Internal
The malfunctioning of SSCs that control releases to the environment, such as filters or valves.	Not relevant	In such case, the possible releases to the environment are so low that their impact on workers and the environment is negligible.	Internal

TABLE I–9. ASSESSMENT OF HAZARD SCENARIOS USING SAFRAN [I–3] (cont.)

Name	Relevance	Relevance – justification (if not relevant)	Category
The failure properly to inspect, test and maintain SSCs.	Not relevant	The safety functions of SSCs in the CSF are multiplied and malfunction of one doesn't affect the safety of the facility. All inspections, tests and maintenance are required to be planned in ahead and controlled during the performance. Such a failure has low probability to occur.	Internal
Incorrect operator action due to inaccurate or incomplete information.	Not relevant	All the actions of the operator are controlled or doubled, due to that this event has very low probability.	Internal
Incorrect operator action in spite of having accurate and complete information.	Not relevant	All the actions of the operator are controlled or doubled, due to that this event has very low probability.	Internal
Sabotage by employees.	Not relevant	This is part of the security assessment.	Internal
The failure of systems and components such as incinerator linings, compactor hydraulics or cutting machinery that poses the risk of significant additional radiation exposure of personnel called on to assist in effecting repairs or replacements.	Not relevant	In the CSF, the waste is just stored but not processed.	Internal
Encountering an unanticipated radiation source in decommissioning (e.g. different in nature or amount) and not recognizing immediately the changed circumstances.	Not relevant	Decommissioning is excluded from the scope of this safety case.	Internal
Removing or weakening a structure or component in decommissioning without realizing the possible effect on the structural competence of other structures and components.	Not relevant	Decommissioning is excluded from the scope of this safety case.	Internal
Traffic accident when transporting waste on-site to processing facility or from processing facility to storage building.	Not relevant	Partially included in the drop scenario. Traffic is not allowed on the transportation road.	Internal

The application of both methods resulted in the following scenarios being identified:

- Drop scenario;
- Fire scenario;
- Plane crash – explosion scenario – will be assessed through security case;
- RW package is stolen from the facility – will be assessed through security case.

The consequences of the first two scenarios above will be evaluated further in this safety assessment as accident scenarios.

Scenario 1 – Drop scenario

Considering the different types of RW stored in the CSF, six subscenarios (identified as P1 through P6) listed in Table I–10 were identified.

TABLE I–10. DROP SUBSCENARIOS P1 THROUGH P6

RW type	Subscenarios	Scenarios consequences
Solid RW	P1: Drop of package that contains dried resin P2: Drop of package that contains U-238	- External radiation - Inhalation - Ingestion - Contamination
DSRS	P3: Drop of package and release of high activity sealed source P4: Drop and subsequent release of unconditioned DSRS P5: Drop of package that contains conditioned DSRS P6: Drop and subsequent release of smoke detectors	- External radiation - Inhalation - Ingestion - Contamination

All of these scenarios (with the exception of scenarios P2 and P3) can occur in front of the CSF or inside the facility. Scenarios P2 and P3 were not assessed since, as subsequent or additional treatment of these packages is not foreseen, they are not transported outside the facility. The representatives of critical groups that can be exposed are:

- ARAO workers;
- IJS workers close to the CSF;
- Security guard performing routine inspections around the facility;
- Farmer behind the fence of the facility (60 m from the CSF).

Dried resins package (P 1)

This scenario involves a package containing dried resins that are packed in a 210 l drum; the resin is contaminated with Co-60, Cs-137, Eu-152 and Eu-154. The mass of the resin is 130 kg. Table I–11 presents the total radioactivity of each radionuclide in the container.

TABLE I–11. SCENARIO P1: RADIONUCLIDE ACTIVITIES

Radionuclide	Activity (Bq)
Co-60	3.8E+06
Cs-137	5.48E+07
Eu-152	3.0E+05
Eu-154	2.0E+05

Package that contains U-238 (P 2)

This package contains U-238 in powder form, with a total activity of 0.598 GBq.

Package with high radioactive sealed source (P 3)

This package contains DSRS with Cs-137 (total activity is $1.46 \cdot 10^3$ GBq), and it is assumed that, during the drop, the DSRS will be released from the drum. Table I–12 shows the dose rates at 1 m and 10 m distance from the drum.

TABLE I–12. SCENARIO P3: DOSE RATES

Radionuclide	Activity on 31.12.2012 [GBq]	Equivalent gamma factor [(mSv·m^2/h)/GBq]	Dose rate at 1 m [mSv/h]	Dose rate at 10 m [mSv/h]
Cs-137	1.46E+03	0.103	150	1.5

Package with unconditioned DSRS (P 4)

This package with DSRS contains Ir-192 with 90 kBq activity on 31 December 2011, and it is assumed that, during the transport, the source will be released from the shield. Table I–13 shows the dose rates for this package at 1, 10, 30, 40, 60 and 100 m distance from the package.

TABLE I–13. SCENARIO P4: DOSE RATES

Radionuclide	Activity on 31.12.2011 [kBq]	Equivalent gamma factor [(mSv·m^2/h)/GBq]	Dose rate at distance [mSv/h]					
			1 m	10 m	30 m	40 m	60 m	100 m
Ir-192	90	0.160	570	5.7	0.63	0.36	0.16	0.06

Note: All the dose rates in this table are assessed for the bare source – released from the shield.

Package with conditioned and repacked DSRS (P 5)

The properties of the package are summarized in Table I–14.

TABLE I–14. SCENARIO P5: PACKAGE PROPERTIES

Radionuclide	Activity on 31.12.2011 [kBq]	Equivalent gamma factor [(mSv·m^2/h)/GBq]	Dose rate at distance [mSv/h]					
			1 m	10 m	30 m	40 m	60 m	100 m
Co-60	460	0.370	170.2	1.70	0.19	0.11	0.05	0.02
Eu-152	12	0.202	2.424	0.02	2.7E-03	1.5E-03	1.1E-03	0.4E-03

Package with dismantled smoke detectors (P 6)

The properties of the package are summarized in Table I–15.

TABLE I–15. SCENARIO P6: PACKAGE PROPERTIES

Radionuclide	Activity on 31.12.2011 [kBq]	Equivalent gamma factor [(mSv·m²/h)/GBq]	Dose rate at distance [mSv/h] 1 m	10 m	30 m	40 m	60 m	100 m
Am-241	7	241 $8.48 \cdot 10^{-2}$	593.6 E-03	5.94 E-03	0.66 E-03	0.37 E-03	0.16 E-03	0.06 E-03

Scenario 2 – Fire scenario

Due to the use of the different electrical equipment inside the storage facility (e.g. moisture control system), a fire can occur. In the CSF, the following combustible materials can be found:

- Plastic baskets for disused personal health protection equipment;
- Combustible parts of electrical and engineering equipment;
- Combustible RW:
  - ZV0 (see Fig. I–7) – smoke detectors with plastic casing – prepared for dismantling;
  - T1 and T3 (see Fig. I–7) – paper, wood, fabric, cotton wool, polymers.
- Wooden benches.

It is assumed that the CSF is closed when a fire occurs and that all ventilation openings are not totally closed due to a failure of fire detection system. It is assumed that the fire lasts 90 minutes and that 10% of the activity of the burnable waste is released into the environment as a result of the fire. Representatives exposed to the radioactive smoke are:

- Guard;
- Workers of the IJS in the offices close to the CSF;
- Workers of the IJS working in a hot cell facility close to the CSF;
- IJS visitors;
- Farmer behind the fence;
- Inhabitants of the village nearby (1 year old child, 7–10 years old child, an adult – more than 18 years).

I–6.9. Formulation and implementation of assessment models

In order to perform the calculations for the safety assessment for the RW storage activities, certain measured and calculated data are used. In those instances where data is not available, certain assumptions are made based on experience performing similar types of activities elsewhere in the world. Further justification for the applied dose rate data is provided. The assumptions made are generally conservative.

I–6.10. Performance of calculations and analysis of the results

I–6.10.1. Radiological impact assessment for normal operation

To the extent possible, the assessment makes use of the data related to the characteristics of the facility, operations, and waste described in previous sections of this Annex.

As shown in Fig. I–19, the waste is segregated by radionuclides into storage compartments that are separated by half-walls; each storage compartment is characterized by unique dose rates.

FIG. I–19. Individual storage compartments in the CSF.

FIG. I–19. Individual storage compartments in the CSF.

For the purpose of performing the safety assessment using SAFRAN, the storage facility was represented as a facility with ten rooms (Fig. I–20), which allowed the consideration of the different dose rates in the individual compartments (Fig. I–21).

FIG. I–20. Rooms defined in SAFRAN.

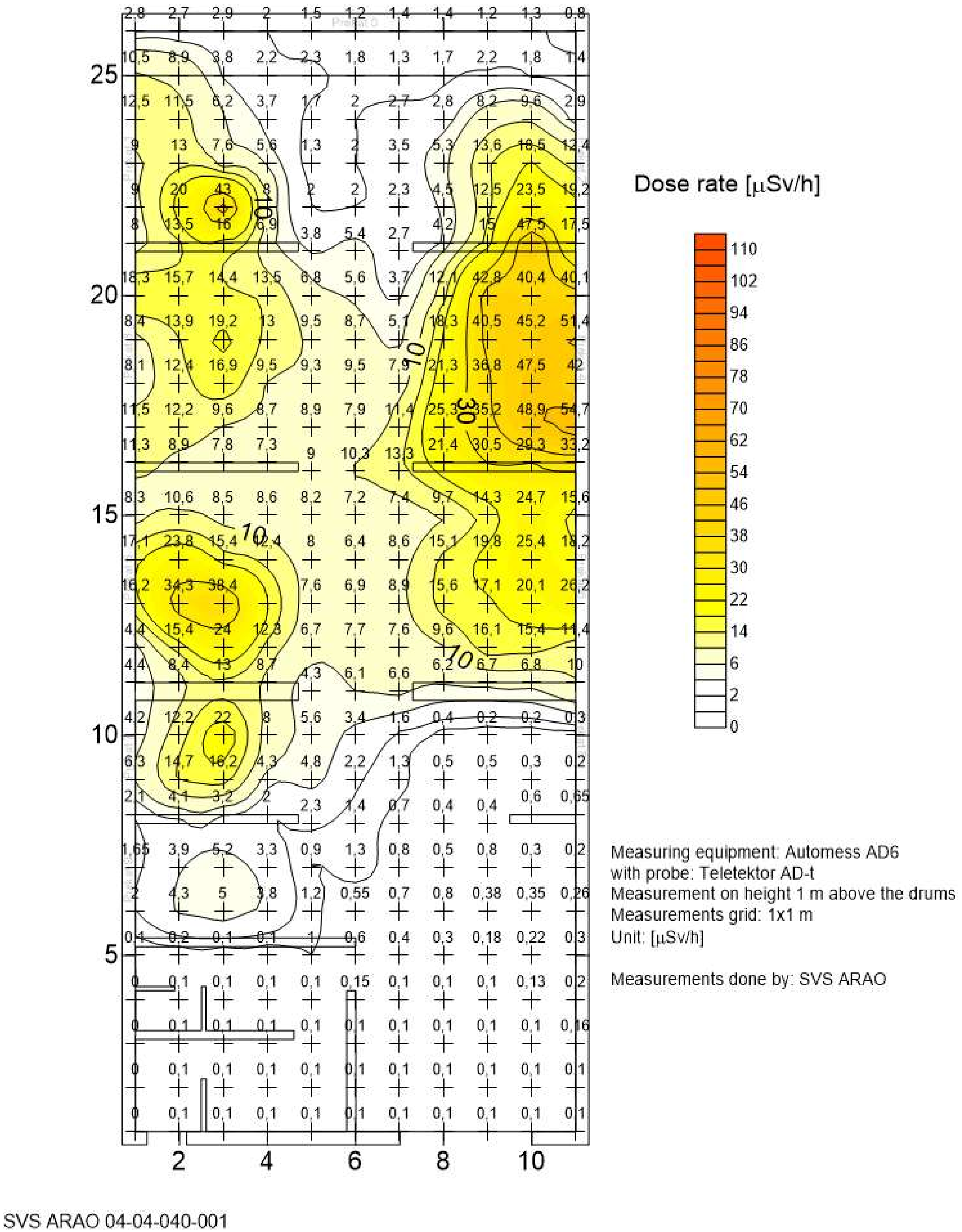

FIG. I–21. Dose rate inside the CSF.

Doses to the workers performing transfers of waste to the storage location are assessed using conservative values for dose rates in the individual compartments and time required to perform each operation. Table I–16 presents the activity duration and dose rates used to assess doses to the workers during normal operation.

TABLE I–16. ASSUMPTIONS USED TO ASSESS DOSES TO WORKER DURING NORMAL OPERATION

Operation	Duration of activity [min/a]	Dose rate [mSv/h] conservatively assessed
Unloading of RW	150	0.020
Control measurement	200	0.020
Packaging	150	0.020
Transfer to storage location:		
Compartment 0	100	0.009
Compartment 1	60	0.043
Compartment 2	200	0.047
Compartment 3	600	0.019
Compartment 4	100	0.047
Compartment 5	700	0.038
Compartment 6	600	0.025
Compartment 7	600	0.022
Compartment 8	2000	0.0006
Compartment 9	2240	0.004

Dose rates are assessed on the basis of regular measurements and the time on the basis of the experiences from previous years. It is also conservatively assumes that a single worker performs the work. Table I–17 presents doses for ARAO workers in the CSF that were assessed on the basis of the data listed in Table I–16.

TABLE I–17. DOSES TO WORKERS (NORMAL OPERATION)

Operation	Doses to workers [mSv/a]
Unloading of RW	0.050
Control measurement	0.067
Packaging	0.050
Transfer to storage location:	
Compartment 0	0.015
Compartment 1	0.043
Compartment 2	0.157
Compartment 3	0.190
Compartment 4	0.078
Compartment 5	0.443
Compartment 6	0.250
Compartment 7	0.220
Compartment 8	0.020
Compartment 9	0.149
Sum	1.732

As described in Section I–6.3.2, radon (and its progenies) is also present in the CSF. Due to this, the impact of radon on doses on workers is also assessed. The activity concentration of radon in the CSF after the ventilation system has been operational for an hour is 200 Bq/m^3. The assessment was performed in accordance with Slovenian regulation SV5 [I–25]. Due to the short lived nature of radon progenies and on the basis of the assumptions listed in Table I–16, the additional dose to the worker resulting from exposure to radon progenies is calculated to be 0.005 mSv/a, while the dose to the worker inside the facility from radon is assessed to be 0.0003 mSv/a.

The total dose for the worker during normal operation in the CSF is presented in Table I–18.

TABLE I–18. DOSE TO WORKER DURING NORMAL OPERATION (INCLUDING RADON)

Source	Dose [mSv/a]
External radiation	1.732
Short lived radon progenies	0.005
Radon	0.0003
Sum	1.7373

Table I–19 presents the total doses to the worker during normal operation, calculated using the SAFRAN tool.

TABLE I–19. DOSE TO WORKER DURING NORMAL OPERATION [I–3]

Impact	Dose [Sv/a]
Unloading from the transport container	5.00E-05
Control measurement	6.66E-05
Packaging	5.00E-05
Transfer to storage location	
Storage 0	1.50E-05
Storage 1	4.30E-05
Storage 2	1.57E-04
Storage 3	1.90E-04
Storage 4	7.85E-05
Storage 5	4.43E-04
Storage 6	2.50E-04
Storage 7	2.20E-04
Storage 8	2.00E-05
Storage 9	1.49E-04
Total	1.73E-03

The calculated doses for workers are below the dose limit of 10 mSv/a prescribed by Slovenian regulation but, due to the use of the conservative approach, are much higher than the actual (measured) doses. Actual (measured) annual doses for the ARAO workers in the storage facility are below 100 µSv/a.

I–6.10.2. Radiological impact assessment for accidents

The scenarios as defined above are assessed using simple calculations. Analyses of the results are presented in Section I–6.11.

Scenario 1 – Drop scenario

This scenario assesses the consequences from the drop of a waste package inside the facility. Because the scenario occurs inside the facility, it is assumed that, due to the design and the construction, ventilation (building is under negative pressure) and technical protection measures, only the ARAO workers are exposed. It is assumed that the worker needs 1 minute to move away from the accident and 10 minutes to protect the area.

Table I–20 presents the assumptions used to calculate the doses to the worker.

TABLE I–20. DROP SCENARIO, ASSUMPTIONS USED TO CALCULATE WORKER DOSES

Representative	Distance from source [m]	Time of exposure [min]	Breathing rate [m^3/h]
ARAO worker	1	1	1.5
	10	10	1.5

It is assumed that, as a result of the drop and the subsequent release, 0.1% of the total activity is released in the air at a height of 2 m.

The effective dose for the ARAO workers is calculated using MS Excel as follows:

$$E = \sum_i E_i = \sum_i \dot{E}_l \cdot t \qquad \text{(I–1)}$$

where

E is the total effective dose;

E_i is the effective dose due to external radiation, inhalation and ingestion;

$\dot{E}_l$ is the effective dose rate due to external radiation, inhalation and ingestion;

t is the duration of exposure.

It is also conservatively assumed that 10% of the inhaled radionuclides are additionally ingested.

Drop of spent dried resins container (P 1)

It is assumed that, as a result of the drop and the subsequent release, an area within a 5 m radius is contaminated. Table I–21 presents the dose rates calculated for selected distances using MicroShield v.6.04.

TABLE I–21. DROP SCENARIO P1, DOSE RATES AT SELECTED DISTANCES

Dose rate [μSv/h] at distance [m]					
1	10	30	40	60	100
0.038	0.010	0.001	<0.001	<0.001	<0.001

Assuming that 0.1% of the activity is released into the air with a volume of 160 m^3, the assessed activity concentrations of the radionuclides in the air are:

- 24 Bq/m^3 for Co-60;
- 342 Bq/m^3 for Cs-137;
- 2 Bq/m^3 for Eu-152;
- 0.1 Bq/m^3 for Eu-154.

Table I–22 presents the results of calculations assessing doses to the worker from external and internal radiation using MS Excel.

TABLE I–22. DROP SCENARIO P1, TOTAL DOSE TO WORKER FROM EXTERNAL AND INTERNAL RADIATION

Representative	Assessed dose due to external radiation [μSv]	Assessed dose due to internal radiation [μSv]	Total dose [μSv]
ARAO worker	< 0.01	0.37	0.37

The SAFRAN tool was used to calculate doses to the ARAO worker resulting from two external exposure scenarios, assuming an exposure time of 1 min and a distance of 1 m and an exposure time of 10 min at distance of 10 m. Because the data for radionuclide Eu-152 was not included in the SAFRAN database at the time of this assessment, only the activity for Eu-154 was considered in the calculation. The results are presented in Table I–23.

TABLE I–23. DROP SCENARIO P1, EXTERNAL EXPOSURE OF WORKER [I–3]

Exposure conditions	Nuclide	Activity [Bq]	Dose rate [Sv/h]	Dose [Sv]
Exposure time = 1 min, distance = 1 m	Cs-137	3.80E+06	4.17E-08	7.10E-10
	Co-60	5.48E+07	2.43E-06	4.14E-08
	Eu-154	3.20E+05	7.27E-09	1.24E-10
	Total		2.48E-06	4.22E-08
Exposure time = 10 min, distance = 10 m	Cs-137	3.8E+06	2.86E-09	5.06E-10
	Co-60	5.48E+07	1.67E-07	2.95E-08
	Eu-154	3.20E+05	4.98E-10	8.82E-11
	Total		1.70E-07	3.01E-08

Dose due to inhalation was also assessed using the SAFRAN tool. Table I–24 presents the input parameters and results of the calculations.

TABLE I–24. DROP SCENARIO P1, INHALATION DOSE TO THE WORKER [I–3]

Nuclide	Activity [Bq]	Airborne release factor	Room volume [m^3]	Dispersion factor [h/m^3]	Protection factor	Dose [Sv]
Eu-154	3.20E+05	1.00E-03	200	1.18E-03	0	5.67E-08
Co-60	5.48E+07	1.00E-03	200	1.18E-03	0	1.89E-06
Cs-137	3.8E+06	1.00E-03	200	1.18E-03	0	9.06E-08
Total						2.04E-06

Package that contains U-238 (P 2)

Table I–25 presents the results of calculations assessing doses to the worker from external and internal radiation using MS Excel.

TABLE I–25. DROP SCENARIO P2, TOTAL DOSE TO THE WORKER

Representative	Assessed dose due to external radiation [μSv]	Assessed dose due to internal radiation [μSv]	Total dose [μSv]
ARAO worker	< 0.01	3.1E+03	3.1E+03

Table I–26 presents the results of calculations using SAFRAN to determine:

- External exposure for a worker who spent 1 min on a distance of 1 m.
- External exposure for a worker who spent 10 min on a distance of 10 m.

TABLE I–26. DROP SCENARIO P2, EXTERNAL EXPOSURE OF WORKER [I–3]

Exposure conditions	Nuclide	Activity [Bq]	Dose rate [Sv/h]	Dose [Sv]
Exposure time = 1 min, distance = 1 m	U-238	5.98E+08	7.39E-10	1.26E-11
Exposure time = 10 min, distance = 10 m	U-238	5.98E+08	8.27E-12	1.46E-12

Table I–27 presents the inhalation dose calculated with SAFRAN together with the input parameters that were used to perform the calculations.

TABLE I–27. DROP SCENARIO P2, INHALATION DOSE TO THE WORKER (1 MIN AT 1 M DISTANCE)

Nuclide	Activity [Bq]	Airborne release factor	Release inside [Bq]	Room volume [m^3]	Dispersion factor [h/m^3]	Protection factor	Dose [Sv]
U-238	5.98E+08	1.00E-03	5.98E+05	200	1.18E-03	0	1.55E-02

Package with high radioactive sealed source (P 3)

Table I–28 presents the results of calculations assessing doses to the worker from external and internal radiation using MS Excel.

TABLE I–28. DROP SCENARIO P3, TOTAL DOSE TO THE WORKER

Representative	Dose due to external exposure [μSv]	Dose due to internal exposure [μSv]	Total dose [μSv]
ARAO worker	2.75E+03	–	2.75E+03

Table I–29 presents the results of calculations using SAFRAN to determine:

- External exposure for a worker who spent 1 min on a distance of 1 m.
- External exposure for a worker who spent 10 min on a distance of 10 m.

It is assumed that inhalation is not possible due to the properties of the package and the radioactive sealed source, precluding contamination of the room and of the air.

TABLE I–29. DROP SCENARIO P3, EXTERNAL EXPOSURE OF WORKER [I–3]

Exposure conditions	Nuclide	Activity [Bq]	Dose rate [Sv/h]	Dose [Sv]
Exposure time = 1 min, distance = 1 m	Cs-137	1.46E+12	1.10E-01	1.94E-03
Exposure time = 10 min, distance = 10 m	Cs-137	1.46E+12	1.23E-03	2.18E-04

Package with unconditioned DSRS (P 4)

Table I–30 presents the results of calculations assessing doses to the worker from external and internal radiation using MS Excel.

TABLE I–30. DROP SCENARIO P4, TOTAL DOSE TO THE WORKER

Representative	Assessed dose due to external radiation [μSv]	Assessed dose due to internal radiation [μSv]	Total dose [μSv]
ARAO worker	10.45E+3	–	10.45E+3

In the SAFRAN common database, the data for Ir-192 is not yet included. Although it is possible for the assessor to introduce new data or change existing data for individual radionuclides into the SAFRAN database, it was decided not to include this calculation in the SAFRAN calculation, due to the lack of availability of the necessary parameters at the time of the preparation of this report.

Package with conditioned and repacked DSRS (P 5)

Table I–31 presents the results of calculations assessing doses to the worker from external and internal radiation using MS Excel. In this case, dose assessment calculations involving Eu-152 is performed in accordance with Safety Reports Series No. 19 [I–26].

TABLE I–31. DROP SCENARIO P5, TOTAL DOSE TO THE WORKER

Representative	Assessed dose due to external radiation [μSv]	Assessed dose due to internal radiation [μSv]	Total dose [μSv]
ARAO worker	316	–	316

Package with dismantled smoke detectors (P 6)

Table I–32 presents the results of calculations assessing doses to the worker from external and internal radiation using MS Excel.

TABLE I–32. DROP SCENARIO P6, DOSE TO THE WORKER

Representative	Dose from external radiation [μSv]	Dose from internal radiation [μSv]	Total dose [μSv]
ARAO worker	10.88	0.24	11.12

Table I–33 presents the results from SAFRAN calculations for external exposure:

- External exposure for a worker that spent 1 min at a distance of 1 m;
- External exposure for a worker that spent 10 min at a distance of 10 m.

TABLE I–33. DROP SCENARIO P6, EXTERNAL EXPOSURE OF WORKER [I–3]

Exposure conditions	Nuclide	Activity [Bq]	Dose rate [Sv/h]	Dose [Sv]
Exposure time = 1 min, distance = 1 m	Am-241	7E+09	1.99E-05	3.52E-07
Exposure time = 10 min, distance = 10 m	Am-241	7E+09	2.23E-07	3.94E-08

Table I–34 presents the inhalation dose calculated with SAFRAN, together with the input parameters that were used to perform the calculations.

TABLE I–34. DROP SCENARIO P6, INHALATION DOSE TO THE WORKERS [I–3]

Nuclide	Activity [Bq]	Airborne release factor	Release inside [Bq]	Room volume [m^3]	Dispersion factor [h/m^3]	Protection factor	Dose [Sv]
Am-241	7E+09	4.00E-09	28	1200	1.18E-03	0	3.87E-06

In the case of a drop in front of the CSF, it is assumed that the ARAO worker needs 1 minute to move away from the accident and 10 min to protect the area. The security guard is assumed to be at the time of the drop near the CSF and needs 2 min to move away and an additional 15 min at a distance of 30 m to inform personnel about the accident. It is assumed that the representatives remain in the plume of contaminated air for 5 min. Due to the movement of the air, it is assumed that this plume will expand quickly and that the particles will not reach the farmer located 60 m from the accident. It is also assumed that 0.1% of the total activity is released in the air and that the plume has a volume of 1000 m^3.

Table I–35 presents the distances, exposure times and breathing rates used to calculate doses to the representatives as a result of the drop scenario.

TABLE I–35. ASSUMPTIONS USED IN THE DROP SCENARIO

Representative	Distance from the source [m]	Exposure time [min]	Breathing rate [m^3/h]
ARAO worker	1	1	1.5
	10	10	1.5
Security guard	10	2	1.5
	30	15	1.5
IJS worker close to CSF	40	30	1.5
Farmer behind the fence	60	60	1.5

Drop of spent dried resins container (P 1)

Table I–36 presents the results of the assessment of doses to the workers and the public from external and internal radiation as a result of the drop scenario, using MS Excel.

TABLE I–36. DROP SCENARIO P1, DOSES TO THE WORKER AND THE PUBLIC

Representative	Dose [μSv]		
	External radiation	Internal radiation	Total
ARAO worker	<0.01	0.29	0.29
Security guard	<0.01	0.29	0.29
IJS worker close to CSF	<0.01	0.29	0.29
Farmer behind the fence	<0.01	-	<0.1

Table I–37 presents the external radiation to the ARAO worker resulting from exposure outside the facility, using SAFRAN.

TABLE I–37. DROP SCENARIO P1, EXTERNAL EXPOSURE OF WORKER [I–3]

Exposure conditions	Nuclide	Activity [Bq]	Distance [cm]	Dose rate [Sv/h]	Exposure time [h]	Dose [Sv]
Exposure time = 1 min, distance = 1 m	Cs-137	3.8E+06	100	2.86E-07	1.77E-02	5.06E-09
	Co-60	5.48E+07	100	1.67E-05	1.77E-02	2.95E-07
	Eu-154	3.20E+05	100	4.98E-08	1.77E-02	8.82E-10
	Total			1.70E-05		3.01E-07
Exposure time = 10 min, distance = 10 m	Cs-137	3.8E+06	1000	2.86E-09	1.77E-01	5.06E-10
	Co-60	5.48E+07	1000	1.67E-07	1.77E-01	2.95E-08
	Eu-154	3.20E+05	1000	4.98E-10	1.77E-01	8.82E-11
	Total			1.70E-07		3.01E-08

Table I–38 presents the input parameters and results for the inhalation dose assessed with the SAFRAN tool.

TABLE I–38. DROP SCENARIO P1, INHALATION DOSE TO THE WORKER [I–3]

Nuclide	Activity [Bq]	Airborne release factor	Release outside [Bq]	Dose conversion factor [Sv/Bq]	Dose [Sv]
Cs-137	3.80E+06	1.00E-03	3800	2.03E-13	7.71E-10
Co-60	5.48E+07	1.00E-03	54 800	5.27E-13	2.89E-08
Eu-154	3.20E+05	1.00E-03	320	2.67E-13	8.54E-11
Total					2.97E-08

The results for the inhalation are assumed to be the same for all representatives (ARAO worker, security guard, IJS worker close to the CSF, farmer) who are located within a 60 m radius of the drop. It is assumed that all particles settle on the ground within the 60 m radius. Because the farmer is located at a distance greater than 60 m from the accident, it is assumed that he does not receive any inhalation dose as a result of the drop.

It is assumed that the security guard is performing a regular inspection when the drop occurs, and that he spends 2 min at a distance of 10 m from the source to identify what is occurring. Table I–39 presents the results of using SAFRAN to assess external exposure.

TABLE I–39. DROP SCENARIO P1, EXTERNAL EXPOSURE TO SECURITY GUARD (2 MIN AT 10 M DISTANCE) [I–3]

Nuclide	Activity [Bq]	Distance [cm]	Dose rate [Sv/h]	Exposure time [h]	Dose [Sv]
Cs-137	3.8E+06	1000	2.86E-09	3.30E-02	9.43E-11
Co-60	5.48E+07	1000	1.67E-07	3.30E-02	5.50E-09
Eu-154	3.20E+05	1000	4.98E-10	3.30E-02	1.64E-11
Total			1.70E-07		5.61E-09

After 2 min, the security guard moves to a distance of 30 m from the drop and it takes 15 min to inform responsible persons about this accident, after which time he moves away from the exposure zone of the accident. Table I–40 presents the results using SAFRAN to assess the external exposure for this 15 min period.

TABLE I–40. DROP SCENARIO P1, EXTERNAL EXPOSURE TO SECURITY GUARD (15 MIN AT 30 M DISTANCE) [I–3]

Nuclide	Activity [Bq]	Distance [cm]	Dose rate [Sv/h]	Exposure time [h]	Dose [Sv]
Cs-137	3.8E+06	3000	3.51E-10	2.50E-01	8.78E-11
Co-60	5.48E+07	3000	2.05E-08	2.50E-01	5.12E-09
Eu-154	3.20E+05	3000	6.12E-11	2.50E-01	1.53E-11
Total			2.09E-08		5.22E-09

It is assumed that the IJS worker is in his office (located at a distance of 40 m from the accident) during the accident and that he needs 30 min to move away from the site. Table I–41 presents the results using SAFRAN to assess the external exposure for this event.

TABLE I–41. DROP SCENARIO P1, EXTERNAL EXPOSURE TO IJS WORKER [I–3]

Nuclide	Activity [Bq]	Distance [cm]	Dose rate [Sv/h]	Exposure time [h]	Dose [Sv]
Cs-137	3.8E+06	4000	1.99E-10	5.00E-01	9.93E-11
Co-60	5.48E+07	4000	1.16E-08	5.00E-01	5.79E-09
Eu-154	3.20E+05	4000	3.46E-11	5.00E-01	1.73E-11
Total			1.18E-08		5.91E-09

Table I–42 presents the results from using SAFRAN to calculate doses to the farmer who works 60 min at a distance of 60 m away from the accident.

TABLE I–42. DROP SCENARIO P1, EXTERNAL EXPOSURE TO FARMER [I–3]

Nuclide	Activity [Bq]	Distance [cm]	Dose rate [Sv/h]	Exposure time [h]	Dose [Sv]
Cs-137	3.8E+06	6000	8.87E-11	1	8.87E-11
Co-60	5.48E+07	6000	5.17E-09	1	5.17E-09
Eu-154	3.20E+05	6000	1.54E-11	1	1.54E-11
Total			5.27E-09		5.27E-09

Package with unconditioned DSRS (P 4)

Due to the properties of the DSRS packaged while in storage at the CSF, contamination of the air as well as internal radiation and inhalation are not considered. For this subscenario, the calculation with SAFRAN is not possible at the moment, because Ir-192 is not yet included in the SAFRAN database.

Table I–43 presents the dose due to external and internal radiation to each representative, calculated using MS Excel.

TABLE I–43. DROP SCENARIO P4, EXTERNAL AND INTERNAL DOSES

Representative	Dose due to external radiation [µSv]	Dose due to internal radiation [µSv]	Total dose [µSv]
ARAO worker	10.45E+03	-	10.45E+03
Security guard	350	-	350
IJS worker close to CSF	180	-	180
Farmer behind the fence	160	-	160

Package with conditioned and repacked DSRS (P 5)

The dose assessment for Eu-152 for this scenario is performed in accordance with Safety Reports Series No. 19 [I–26]. Table I–44 presents the assessed dose due to external and internal radiation to each representative.

TABLE I–44. DROP SCENARIO P5, EXTERNAL AND INTERNAL DOSES [I–26]

Representative	Assessed dose due to external radiation [µSv]	Assessed dose due to internal radiation [µSv]	Total dose [µSv]
ARAO worker	316	-	316
Security guard	104.8	-	104.8
IJS worker close to CSF	55.8	-	55.8
Farmer behind the fence	51.1	-	51.1

Package with dismantled smoke detectors (P 6)

Table I–45 presents the assessed dose (calculated using MS Excel) due to external and internal radiation to each representative.

TABLE I–45. DROP SCENARIO P5, EXTERNAL AND INTERNAL DOSES

Representative	Assessed dose due to external radiation [µSv]	Assessed dose due to internal radiation [µSv]	Total dose [µSv]
ARAO worker	10.88	0.24	11.12
Security guard	0.36	1.20	1.56
IJS worker close to CSF	0.19	1.20	1.39
Farmer behind the fence	0.16	-	0.16

Table I–46 presents the distances, exposure times and dose rates used to calculate doses using SAFRAN due to external exposure to the Am-241 source (activity of 7×10^9 Bq) to the different representatives.

TABLE I–46. DROP SCENARIO P6, EXTERNAL EXPOSURES FROM AM-241 [I–3]

Representative	Exposure conditions	Nuclide	Dose rate [Sv/h]	Dose [Sv]
ARAO worker	Exposure time = 1 min, distance = 1 m	Am-241	2.91E-06	5.15E-08
	Exposure time = 10 min, distance = 10 m	Am-241	1.99E-07	3.52E-08
Security Guard	Exposure time = 2 min, distance = 10 m	Am-241	1.99E-07	6.57E-09
	Exposure time = 15 min, distance = 30 m	Am-241	2.45E-08	6.11E-09
IJS worker	Exposure time = 30 min, distance = 40 m	Am-241	1.38E-08	6.92E-09
Farmer	Exposure time = 60 min, distance = 60 m	Am-241	6.18E-09	6.18E-09

In order to calculate the dose due to inhalation to the ARAO worker, security guard, and IJS worker close to CSF (all of whom are located within a radius of 60 m of the drop), it is assumed that all particles settle on the ground within the 60 m radius. Because the farmer is located at a distance greater than 60 m from the accident, it is assumed that he does not receive any dose due to inhalation. Table I–47 presents the results of the assessment performed using SAFRAN.

TABLE I–47. DROP SCENARIO P1, INHALATION DOSE TO THE WORKERS [I–3]

Nuclide	Activity [Bq]	Airborne release factor	Release outside [Bq]	Dose conversion factor [Sv/Bq]	Dose [Sv]
Am-241	7E+09	1.70E-08	119	2.09E-11	2.49E-09

Scenario 2 – Fire scenario:

This scenario assumes a fire in the CSF caused by an electrical fault in the facility that expands to other combustible materials inside the facility. Table I–48 presents the total quantities and activities of combustible waste stored in the CSF.

TABLE I–48. COMBUSTIBLE WASTE STORED IN THE CSF

Type of RW	Volume [m^3]	Activity [GBq]
T1 – solid, compressible, combustible	18.1	0.95
T3 – solid, non-compressible, combustible	3.35	0.12
ZV0 – smoke detectors	7.55	6.61
Total	29.1	7.68

To analyse the fire scenario, the following conservative assumptions are adopted:

- There are no ARAO workers at the site at the time of fire.
- The doors of the facility are closed when the fire occurs.
- The fire protection system is activated, and the security guard checks the situation, as required.
- The fire hatches don't close completely due to technical problems.
- The duration of the fire is 90 min (includes initiation of the fire, situation check by the security guard, summoning of the firefighting service, conduct of the firefighting, extinguishing of the fire, and cessation of smoke from the fire).
- Smoke contaminated with radioactive particles escapes from the facility through the partially closed fire hatches as well as through the damaged ventilation system (it is also assumed the filters are damaged as a result of the fire), resulting in 10% of the total activity released inside the facility escapes into the atmosphere.

Table I–49 presents the representatives and parameters used to assess the doses to the representatives (including both workers and the public).

TABLE I–49. REPRESENTATIVES AND PARAMETERS USED IN FIRE SCENARIO

Representative	Distance from the source [m]	Exposure time [min]	Breathing rate [m^3/h]
Security guard	10	2	1.5
	40	15	1.5
IJS Worker in offices close to CSF	40	25	0.54

TABLE I–49. REPRESENTATIVES AND PARAMETERS USED IN THE FIRE SCENARIO (cont.)

Representative	Distance from the source [m]	Exposure time [min]	Breathing rate [m^3/h]
IJS worker in hot cell facility adjacent to CSF and IJS visitors	40	25	0.54
Farmer behind the fence	60	30	1.5
Inhabitants of nearby village (child 1 y)	500	90	0.35
Inhabitants of nearby village (child 7–10 y)	500	90	1.12
Inhabitants of nearby village (adult)	500	90	1.5

The methodology described in IAEA-TECDOC-1162 [I–27] (procedure E5a) was used to assess the impact of the fire on the representatives presented above. Table I–50 presents the radionuclides and the associated fire release fractions considered by the model. Table I–51 presents the results of the assessment.

TABLE I–50. RADIONUCLIDES AND FIRE RELEASE FRACTIONS CONSIDERED IN THE FIRE SCENARIO

Radionuclide	Fire release fraction
Cs-137	0.01
Eu-152	0.01
Am-241	0.001
Co-60	0.001
Pu-239	0.001
Ra-226	0.001
U-238	0.001

TABLE I–51. FIRE SCENARIO DOSES TO REPRESENTATIVES [I–27]

Representative	Distance from the source [m]	Dose due to inhalation [μSv]		Dose due to ingestion [μSv]	
		Sunny	Cloudy	Sunny	Cloudy
Security guard	10	64.7	524	9.63E-3	7.8E-2
	40	64.7	524	9.63E-3	7.8E -2
IJS Worker in offices close to CSF	40	11.9	99.6	1.78E-3	1.48E-2
IJS Worker in hot cell facility adjacent to CSF and IJS visitors	40	33.5	279	4.98E-3	4.15E-2
Farmer behind the fence	60	20.1	1.5	2.99E-3	2.9E-2
Inhabitant of nearby village (child 1 y)	500	Less than adult	Less than adult	Less than adult	Less than adult
Inhabitant of nearby village (child 7–10 y)	500	Less than adult	Less than adult	Less than adult	Less than adult
Inhabitant of nearby village (adult)	500	0.5	10	7.47E-5	1.49E-3

Doses received as a result of exposure to the fire scenario were also assessed using the SAFRAN tool. However, as Eu-152 is not included in the common SAFRAN database, it was not considered in the calculation. Table I–52 presents the doses to the public; doses to the workers are assumed to be the same as for the public.

TABLE I–52. FIRE SCENARIO DOSES TO REPRESENTATIVES [I–3]

Nuclide	Release inside [Bq]	Filtration efficiency	Release outside [Bq]	Dose conversion factor [Sv/Bq]	Dose [Sv]
U-238	2.17E+05	9.00E-01	2.17E+04	2.05E-12	4.45E-08
Ra-226	6.00E+04	9.00E-01	6.00E+03	9.51E-12	5.71E-08
Pu-239	4.31E+04	9.00E-01	4.31E+03	2.50E-11	1.08E-07
Am-241	6.64E+06	9.00E-01	6.64E+05	2.09E-11	1.39E-05
Cs-137	3.16E+06	9.00E-01	3.16E+05	2.03E-13	6.41E-08
Co-60	1.16E+05	9.00E-01	1.16E+04	5.27E-13	6.11E-09
Total					1.42E-05

I–6.10.3. Interdependencies

Activities performed in neighboring facilities, such as the IJS, might have an impact on the safety of the CSF. The development of activities that could introduce additional or new hazards to the CSF, such as the production and handling of explosives or the performance of activities that could pose a fire risk, might impact the safety of the CSF. Discussions will be carried out with the regulatory authority to address this issue.

I–6.10.4. Management of uncertainties

While performing the safety assessment, the most significant source of uncertainty that could potentially impact the assessed safety of the facility and activities at the CSF concerns the level of confidence in the characterization of the RW inventory. It is important that any insufficiencies in the characterization of the current inventory of RW at the CSF could not negatively impact the result of the safety assessment. In order to address this uncertainty, a conservative approach was taken in the selection of conservative but realistic inventory data and assumptions. Furthermore, the 'screening' method was used to evaluate the impact of important input data and assumptions on occupational and public exposures.

Ageing of the existing facility is also considered to be another important source of uncertainty, and to address this periodic safety reviews are performed every ten years. The results from the periodic safety reviews are used to revise the safety case and extend the operating license.

I–6.11. Analysis of assessment results

The following subsections compare the results obtained during the quantitative and qualitative assessments for both normal operation and accident conditions against the proposed targets and objectives set for the optimization of protection.

Normal operation

Table I–53 presents the results of the safety assessment for normal operation obtained using classical calculations as well as those obtained using SAFRAN. All of the doses, calculated using conservative assumptions, are (by a factor of 10) below the dose limit of 10 mSv/a prescribed by the Slovenian regulation, and are well below the actual (measured) annual doses to the ARAO worker (below 100 μSv/a). The results obtained using classical calculations versus the SAFRAN tool are the same.

TABLE I–53. COMPARISON OF DOSES TO THE WORKER DURING NORMAL OPERATION

Operation	Using classical calculations [mSv/a]	Using SAFRAN [mSv/a]
Unloading of RW	0.050	0.050
Control measurement	0.067	0.067
Packaging	0.050	0.050
Transfer to storage location:		
Compartment 0	0.015	0.015
Compartment 1	0.043	0.043
Compartment 2	0.157	0.157
Compartment 3	0.190	0.190
Compartment 4	0.078	0.079
Compartment 5	0.443	0.443
Compartment 6	0.250	0.250
Compartment 7	0.220	0.220
Compartment 8	0.020	0.020
Compartment 9	0.149	0.149
Total	1.732	1.732

Accident scenarios

Drop scenario inside the facility

Table I–54 summarizes the results of the dose assessment using classical calculations and using SAFRAN for the accident scenarios involving the drop of a RW container inside the facility. Doses that are less than 0.01 µSv are presented as “< 0.01”. As Table I–54 shows, the results are well below the prescribed dose limit of 20 mSv for the ARAO worker. In most cases, the SAFRAN tool resulted in more conservative (i.e. higher) doses to the worker.

TABLE I–54. COMPARISON OF DOSES TO THE WORKER RESULTING FROM CONTAINER DROP ACCIDENTS INSIDE THE FACILITY

Scenario, Dose representative	Classical calculations			SAFRAN tool		
	External radiation [μSv]	Inhalation [μSv]	Total dose [μSv]	External radiation [μSv]	Inhalation [μSv]	Total dose [μSv]
P1, package containing dried resins						
ARAO worker	< 0.01	0.37	0.37	<0.01	2.04	2.04
P 2, package containing U-238						
ARAO worker	< 0.01	3.1E+03	3.1E+03	<0.01	15.5E+03	15.5E+03
P 3, package containing high activity Cs-137 sealed source						
ARAO worker	2.750E+03	-	2.75E+03	2.158	-	2.16E+03
P 4, package containing unconditioned Ir-192 source						
ARAO worker	10.45E+03	-	10.45E+03	Note 1		
P 5, package containing conditioned Co-60 and Eu-152 sources						
ARAO worker	316	-	316	Note 1		
P 6, package containing dismantled Am-241 smoke detectors						
ARAO worker	10.88	0.24	11.12	0.39	3.87	4.26

Note:

1. The assessment with SAFRAN was not performed (see Section I–6.8.2).

Drop scenario outside the facility

Table I–55 summarizes the results of the dose assessment using classical calculations and using SAFRAN for the accident scenarios involving the drop of a RW container outside the facility. As Table I–55 shows, the doses to all representatives are well below the prescribed dose limits of 20 mSv for the ARAO worker and 1 mSv for non-ARAO workers and the public.

TABLE I–55. COMPARISON OF DOSES TO THE WORKER FOR CONTAINER DROP ACCIDENTS OUTSIDE THE FACILITY

Representative	Classical calculations			SAFRAN tool		
	Ext. radiation [μSv]	Inhalation [μSv]	Total dose [μSv]	Ext. radiation [μSv]	Inhalation [μSv]	Total dose [μSv]
	P1, package containing dried resins					
ARAO worker	<0.01	0.29	0.29	0.33	0.03	0.36
Security guard	<0.01	0.29	0.29	0.01	0.03	0.04
IJS worker close to CSF	<0.01	0.29	0.29	0.01	0.03	0.04
Farmer behind the fence	<0.01	-	<0.1	0.01	0.03	0.04
	P 2, package containing U-238					
P 2	Note 1					
	P 3, package containing high activity Cs-137 sealed source					
P 3	Note 1					
	P 4, package containing unconditioned Ir-192 source					
ARAO worker	10.45E03	-	10.45E03	Note 2		
Security guard	350	-	350	Note 2		
IJS worker close to CSF	180	-	180	Note 2		
Farmer behind the fence	160	-	160	Note 2		
	P 5, package containing conditioned Co-60 and Eu-152 sources					
ARAO worker	316	-	316	Note 2		
Security guard	104.8	-	104.8	Note 2		
IJS worker close to CSF	55.8	-	55.8	Note 2		
Farmer behind the fence	51.1	-	51.1	Note 2		
	P 6, package containing dismantled Am-241 smoke detectors					
ARAO worker	10.88	0.24	11.12	0.09	<0.01	0.09
Security guard	0.36	1.20	1.56	0.01	<0.01	0.01
IJS worker close to CSF	0.19	1.20	1.39	0.01	<0.01	0.01
Farmer behind the fence	0.16	-	0.16	0.01	<0.01	0.01

Notes:

1. The assessment was not performed (see Section I–6.8.2).
2. The assessment with SAFRAN was not performed (see Section I–6.8.2).
3.

Fire scenario

Table I–56 summarizes the results of the dose assessment using classical calculations and using SAFRAN for the accident scenarios involving a fire. As Table I–55 shows, all of the results are well below the dose limit of 1 mSv for all representatives. While the doses calculated using SAFRAN are less conservative than classical assessments for the more exposed representatives and are more conservative for less exposed representatives, the results are still considered comparable.

TABLE I–56. COMPARISON OF DOSES TO THE WORKER FOR FIRE ACCIDENTS

Representative	CLASSICAL					SAFRAN
	Limit [μSv]	Dose – inhalation – sunny [μSv]	Dose – inhalation – cloudy [μSv]	Dose – ingestion – sunny [μSv]	Dose – ingestion – cloudy [μSv]	Dose [μSv]
Security guard	1E03	64.7	524	<0.01	0.08	14.2
	1E03	64.7	524	<0.01	0.08	14.2
Worker of IJS in the offices close to CSF	1E03	11.9	99.6	<0.01	0.01	14.2
Worker of IJS working in hot cell facility near to CSF and IJS visitors	1E03	33.5	279	<0.01	0.04	14.2
Farmer behind the fence	1E03	20.1	1.5	<0.01	0.03	14.2
Inhabitants of the village nearby (child 1 y)	1E03	Less than adult	Less than adult	Less than adult	Less than adult	14.2
Inhabitants of the village nearby (child 7–10 y)	1E03	Less than adult	Less than adult	Less than adult	Less than adult	14.2
Inhabitants of the village nearby (adult)	1E03	0.5	10	<0.01	<0.01	14.2

Assessment of Uncertainties

Table I–57 presents the main sources of uncertainties that might impact the quantitative assessment of doses to the workers and the public at the CSF, along with recommendations for their management.

TABLE I–57. ASSESSMENT OF UNCERTAINTIES

Item	Uncertainty	Recommendations for management of uncertainty
1	Uncertainty regarding the dose rate information used in the safety assessment. This issue was resolved by the use of conservative data.	Perform monitoring performed to verify the dose rate assumptions to the extent possible to update exposure scenarios and data.
2	The data used for accidental scenarios assessment were taken from literature.	Increase efforts to get as much realistic data as possible.
3	There is uncertainty regarding the radiological characterization of the waste stored in the facility.	Increase efforts to improve the data and knowledge about the waste radiological characteristics.
4	Uncertainty regarding the data about the mixed waste (toxic waste) exists in characterization of the waste.	Increase efforts in: • Identification of toxic substances • Characterization of stored waste regarding toxic components

I–7. ITERATION AND OPTIMIZATION

The evaluation of the design of the CSF and the safety assessment have been undertaken with the best available data and applying a qualitative approach based on expert judgement.

Nevertheless, aspects such as those mentioned below could impose the necessity for iteration in the safety assessment process:

- New data about the site might become available.
- The building design or the supported systems might need to be modified.
- The features of the security system, to be defined in the near future, might interfere with the safety measures proposed.
- Due to the development of the knowledge and technology, new good practices or systems will be developed.

An appropriate management system for controlling and registering the information plays an important role in this iterative process.

I–8. IDENTIFICATION OF SAFETY MEASURES

The assessment undertaken indicates that if the storage facility is operated and maintained according to the provisions set out in this Safety Case, it will comply with international safety standards and meet the relevant dose limitation criteria with respect to workers and members of the public. The assessment has been carried out using conservative assumptions and straightforward methodology. No particular consideration has been given to the sensitivity of the assumptions used as the dose assessment made only a few simple assumptions, such as the distance of the building from the fence, the residence time of the members of the public and occupancy time of workers in or near the storage building.

Radiation shielding is provided by the walls and internal structures, source containers, and packaging. These are simple, passive, engineered components with a strong level of robustness and reliability. The isolation of the RW is provided by the building structure and the site boundary fence, which as stated earlier are simple, passive, engineered components with a similarly strong level of robustness and reliability. Control of access to the building is achieved by the boundary fence, gate and the access doors to the building.

Limits are placed on the acceptance of RW for storage to ensure that all conditions of storage are met; operating procedures will be in place to ensure compliance with limits.

Inspection and maintenance programmes are in place and a management system providing for trained personnel, formalized procedures, records, reports and an assurance regime over all aspects important to safety and security is also established.

The building is of a strong monolithic design, which ensure its ability to withstand the impact of severe winds, storms and any possible ground erosion and slippage.

This Safety Case presents the second assessment of the safety of the CSF and will be updated and revised in the future.

The tabulated list of safety functions and safety features can be found in Table I–5.

I–9. LIMITS, CONTROLS AND CONDITIONS

Based on the safety assessment, the following facility operational limits, controls, and conditions are derived:

- On the basis of the Safety Case results, the existing WAC are confirmed;
- When WAC for the future disposal facility are developed, the inventory for compliance with the prescribed WAC for disposal is checked and, if necessary, the WAC for storage are revised;
- Based on this Safety Case, a limitation on the total activity of the facility is not foreseen, because the capacity of the facility (in terms of volume) is limited. Therefore, the maximum volume of waste that can be stored in the facility needs to be better derived and specified.

I–10. CONCLUSIONS

I–10.1. Comparison with safety criteria

The results of the quantitative safety assessment as reflected above are well within the national and international safety criteria for workers and the public. During the preparation of the Safety Case, a very conservative approach was used (the maximum inventory value was typically taken into account) and all the assessment results are well below the prescribed limits.

The Safety Case for the RW storage facility and storage operations, defined above, is supported subject to a formal plan and schedule to address the identified unresolved issues as covered above.

The key findings and conclusions for the safety of operations within the CSF are as follows:

- **Strategy:** The CSF facility and its associated operations to store RW are in line with the national policy and strategy.
- **Facility design and engineering:** The CSF is a robust facility with features that indicate that safety and security have been considered.
- **Facility operation:** The safety assessment indicates that the facility can be operated well within the current safety criteria identified in this document. Uncertainties exist mainly regarding characterization of RW and assumptions regarding data used for assessment. As described above, management of such uncertainties requires continued action on the verification of assumptions and scientific data. Some facility specific limits and conditions have also been recommended in order to mitigate some uncertainties.
- **Optimization of protection:** The margin for optimization of protection associated with the RW activities is limited in view of the relative low consequences and conservatism of assumptions made. Some facility design and procedural changes could, however, be considered for further optimization of protection. An operational optimization of protection programme that is based on activity specific radiation protection surveillance, personal dosimetry results and scheduled optimization review sessions is recommended.
- **Waste management practice:** Good waste management practice is generally evident from the intent of the legal framework, organizational arrangements and defined responsibilities for establishing the CSF and its operations. The interdependencies amongst the various waste management steps seem to be considered up to now. The alignment between conditioning, storage and disposal will need to be considered. Recommendations regarding unresolved issues are given in Section I–10.2 below.

- **Integrated management system:** Practically the whole management systems and procedures have been implemented, further development of the Management system is required in the sense of the optimization.
- **Uncertainties:** The identified uncertainties are neither of such a nature nor extent that the associated detriment in confidence in the Safety Case would result in the recommendation of drastic measures. As stated in Section I–6.11, uncertainties are managed by implementation of specific facility limits and conditions, use of WAC and implementation of monitoring and inspections.

I–10.2. Identification of issues requiring clarification

Table I–58 presents aspects that were identified during the assessment as requiring further clarification, along with the commensurate proposed management recommendations and actions.

TABLE I–58. ASPECTS REQUIRING FURTHER CLARIFICATION

Item	Aspects requiring clarification	Recommendation/Action
1. Implemented waste management practice		
1.1	Waste acceptance criteria.	Check that WAC will be in compliance with the WAC for RW disposal.
1.2	Interdependencies related to disposal	Develop and implement a national waste management plan and strategy to make provision for and commitments to long term actions including disposal.
2. Implementation of a new waste management practice		
2.1	Possibility to temporarily store unconditioned liquid waste.	Perform the related safety assessment for this new activity.
3. Interfaces between safety and security		
3.1	Interfaces between safety and security	Develop a security assessment for the CSF considering the site features. Consider the possible need for modification of the safety case.
4. Interdependencies between CSF and other facilities		
4.1	Interdependencies between CSF and other facilities	Consider the possible impact on safety.

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ANNEX II
ILLUSTRATIVE SAFETY CASE AND SAFETY ASSESSMENT FOR THE RETRIEVAL OF WASTE FROM A HISTORICAL RADON-TYPE FACILITY IN THE RUSSIAN FEDERATION

II–1. INTRODUCTION

This Annex presents an illustrative example of a safety case for radioactive waste (RW) retrieval operations from a typical historical RADON-type waste storage facility. Specifically, the safety case only considers retrieval of radioactive waste from Vault 1 of the RADON-type facility. This illustrative safety case follows the guidance provided in IAEA Safety Standards Series No. GSG-3, The Safety Case and Safety Assessment for the Predisposal Management of Radioactive Waste [II–1], and illustrates the assessment of normal operational and accident dose scenarios using the methodology developed in the framework of the IAEA International Project on Safety Assessment Driving Radioactive Waste Management Solutions (SADRWMS project) [II–2]. Doses arising from normal operation and accident conditions were determined using the Safety Assessment Framework (SAFRAN) software tool (version 2.3.2.7) [II–3] to demonstrate application of the assessment methodology.

The RADON-type facility comprises four concrete storage vaults filled with approximately 185 m^3 of miscellaneous low and intermediate level solid wastes and two empty liquid waste storage tanks. The scope of the activities will result in retrieval of all wastes, repackaging and consignment of the items to another processing and storage facility elsewhere off site. This safety case considers a simple and flexible methodology for RW retrieval activities that has been developed using a combination of manual and semi-remote techniques. Retrieval activities are anticipated to take approximately 100 working days.

The safety case demonstrates compliance with national and international dose limits and constraints. Results of the safety assessment indicate a maximum dose to the worker of 6.2 mSv under normal operation and 7.0 mSv under accident conditions. The maximum dose to a member of the public under accident conditions is 0.8 μSv, with negligible public dose under normal operation.

II–1.1. Characteristics of Radon-type facilities

In the late 1950s, the Soviet Union created a chain of specialized sites to conduct the collection and disposal of RW generated outside of the nuclear fuel cycle. These sites eventually became known as the RADON network, or RADON system.

RW storage facilities (RWSF) of “RADON type” were built in the 1960s in various regions of the Soviet Union, as well as in a number of Eastern European countries (former USSR block countries). These facilities were constructed according to a standard design, with specific modifications to address local conditions of the storage locations and predicted volumes of RW.

The RADON-type facilities took their name from the RADON system that was established in the former Soviet Union for collecting, transportation, processing and near surface disposal of low and intermediate level institutional RW including disused sealed radioactive sources (DSRS).

There were 35 RADON-type facilities, in most cases located close to regional scientific centres and covering all territories of the former Soviet Union. Near surface RADON-type facilities were sited, designed, constructed and operated as disposal facilities with a typical design based on the understanding of safety and the level of knowledge of that time.

In most cases, a RADON-type facility is a trench in clayey rock, approximately 4 m depth, in which rectangular vaults were constructed using reinforced concrete and covered with concrete slabs equipped with loading hatches. Some of these were subdivided into several sections using wooden or concrete walls, effectively created independent vaults. The typical initial design of a RADON-type facility includes three or four disposal vaults of 200 m^3 each and one or two 200 m^3 underground tank of stainless steel for temporary storage of liquid RW.

During the 1960s–1980s, RW management technologies were limited to the placement of RW into disposal facilities, either packaged into various overpacks or in bulk heaps. The design solutions of the time did not include provisions for subsequent retrieval. RW was disposed of without treatment or conditioning and with very simple waste acceptance criteria (WAC), sometimes without any criteria.

Initial characteristics of the waste and packages might have changed over time due to a variety of degradation mechanisms, such as corrosion, biodegradation, chemical reactions and radioactive decay. Therefore, the original waste package documentation cannot be relied upon to describe completely the current status of the waste and waste packages.

RW stored in the RADON-type facilities are commensurate with the categories of very low level wastes, low level wastes and intermediate level wastes. They are characterized by a wide variety of radionuclide compositions and forms. The storage or disposal vaults were filled with RW via hatches in the top of the vaults or in the case of large-sized packages these were loaded into the vaults by removing covers (concrete finger slabs).

Initially, the RADON-type facilities were considered disposal facilities for RW. However, in compliance with the current regulations, many of these facilities fail to meet actual safety requirements and, thus, are only considered as interim storage facilities. Therefore, it is necessary to retrieve the RW and transfer it to other appropriate facilities in order to facilitate decommissioning of the legacy RADON-type facilities.

II–1.2. Historical background

Before a decision was made on waste retrieval from the historical RADON-type facility, two options were considered within the framework of historical RW management strategy.

The first option envisaged removal of all RW from the storage facility and remediation of the site. The activities include retrieval of the waste, loading into containers (with or without prior conditioning), transportation, transfer of the containers in a new storage facility, and dismantling of the old facility structures. This option would provide for a complete remediation of any negative impact of the historical RW facility on the environment and the public, which is particularly important for sites with residential communities in the vicinity, and create new engineered barriers around the retrieved RW made of modern materials and compliant with the latest requirements for safety, reliability and longevity. Disadvantages of this option are as follows: long implementation time and high costs, increased risk of accidents, and the need to put in place extensive measures for radiation protection of personnel and the environment.

The second option envisaged entombment of RW in the existing storage vaults. After upgrading the storage facility, a monitoring system would be established, including the drilling and arrangement of a network of wells for periodic collection and radiometric examination of water samples. Monitoring would remain in place until the facility was decommissioned, after which the RW would no longer pose a danger to the environment and human health.

There were a number of reasons why waste emplaced in the historical RADON-type facility required retrieval and reconditioning. These included:

- Recognition of a real or potential problem that could lead to negative safety, environmental and radiological impacts, specifically through human intrusion scenarios (e.g. leaching of contaminants into groundwater or impending structural failure of the facility);
- Risks associated with RW that is not properly stored increasing over time as retrieval is delayed;
- Lack of RW inventory data (e.g. significant uncertainty over the quantity of long lived radionuclides in a facility designed for short lived waste);
- Degradation of RW packages and facility structures in a way that might compromise the current or future safety of the facility;
- Implementation of a conditioning programme for RW stored in an unconditioned state;
- A desire to consolidate several smaller facilities into a larger facility;
- As a precursor to the decommissioning of the historic facility;
- Changes in national regulations.

II–1.3. Legal and regulatory framework

Although the details in the national legislation might differ, the safety case typically comprises the following elements:

- A demonstration of the required level of safety of the facility;
- A demonstration of the protection of the environment both in the short and long term perspective;
- An assurance that the generation of secondary RW in the facility is kept to the minimum practicable;
- A demonstration that account is taken of interdependencies among all steps in RW management;
- An assurance that any processing of RW will be compatible with the anticipated type and duration of the storage and the need for retrievability of the RW from storage;
- The cost estimates of the waste management facilities and the liability of the operator with regard to the management of RW in the long term;
- An assurance that account is taken of anticipated waste arisings, accountability of waste, disposal options and safety considerations;
- An assurance of acceptance or tolerance of the facility by the public;
- An assurance of adequate physical protection.

The retrieval and subsequent management of RW was performed in conformity with the national strategy for management of RW and with national legislation.

II–1.3.1. National federal laws (Russian Federation)

The **Federal Law No. 170-FZ of 21 November 1995 “On Atomic Energy Use”** [II–4] is the fundamental document regulating relationships in the field of the use of atomic energy, which is aimed at environmental protection, protection of health and life of people by the use of atomic energy and determines the legal basis for regulation of safety.

This law (Article 44) stipulates that the State policy in the management of nuclear materials, radioactive substances and RW provide for an integrated solution of issues related to normative regulation of their production, generation, use, physical protection, collection, registration and accounting, transportation, storage and disposal.

Articles 45–48 of the law stipulate that, during the transportation, storage and reprocessing of nuclear materials (including spent nuclear fuel) and RW, as well as by the disposal of RW, the reliable protection of workers of objects of the use of atomic energy, population and environment are ensured against radiation impact and radioactive contamination impermissible in accordance with norms and rules valid in the field of use of atomic energy and legislation of the Russian Federation in the field of environmental protection.

The **Federal Law No. 3-FZ of 9 January 1996 "On Radiation Safety of Population"** [II–5] defines the legislative basis to ensure the radiation safety of the population in order to protect its health. The law establishes main principles to ensure radiation safety, main hygienic normative standards (permissible dose limits) of exposure to irradiation in the territory of the Russian Federation arising from the use of ionizing radiation sources.

The following terms are used in the framework of the Federal Law:

- 'Public radiation safety' (hereinafter referred to as radiation safety) is the condition of protection of the current and the future generations of people against the harmful effect of ionizing radiation on their health.
- 'Ionizing radiation' is the radiation resulting from radioactive decay, nuclear transformations, and deceleration of charge particles in the substance, and that generates ions with different charges when interacting with the media.
- 'Natural radiation background' is the dose of ionizing radiation generated by the cosmic radiation and radiation from natural radionuclides contained in ground, water, air, and other biosphere elements, food and human body.
- 'Technologically modified natural radiation background' is natural radiation background changed as a result of human activity.
- 'Effective dose' is the value of action of the ionizing radiation used as a measure of risk of the long term effects of exposure of a human body and its parts considering their radiosensitivity.
- 'Control area' is the territory around a source of ionizing radiation where the level of public exposure under normal operation conditions of the source can exceed the dose limit for public. Both temporary and permanent residence are prohibited in the control area; the restricted regime of economic activities is introduced in the control area, and radiation monitoring is conducted there.
- 'Supervised area' is the territory beyond the boundaries of the control area where the radiation monitoring is performed.
- 'Employee' is a physical person who directly works with sources of ionizing radiation on permanent or temporary terms.
- 'Radiation accident' is the loss of control over a source of ionizing radiation due to equipment malfunctioning, erroneous personnel actions, natural disasters or other causes which could have led or led to irradiation of people beyond the established limits or to radioactive contamination of environment.

Article 3 of the Law establishes principles of radiation safety assurance. The main concepts of radiation safety assurance are:

- The normalizing principle is non-exceedance of the allowable public individual exposure doses from all the sources of ionizing radiation.
- The principle of justification is inhibition of all types of activity on the use of sources of ionizing radiation unless positive results for the human and society achieved by these

activities exceed the risk of possible harm caused by radiation in addition to the natural background.
- The principle of optimization is maintenance of individual radiation doses and the number of exposed individuals due to use of any ionizing radiation source at as low as reasonably achievable level considering economic and social factors.

In case of a radiation accident, the public radiation safety assurance system adheres to the following principles:

- Suggested measures for mitigation of radiation accident consequences have more advantages than disadvantages.
- Types and scope of activity on mitigation of radiation accident consequences are implemented in a way ensuring maximum advantages from the decrease of ionizing radiation dose, without the harm inflicted by this activity.

Article 9 of the Law establishes the following hygienic standards (permissible dose limits) of exposure in the territory of the Russian Federation due to use of ionizing radiation sources:

- The average annual effective dose for public is 0.001 Sv or effective dose for the life span (70 years) is 0.07 Sv; in some years, large effective dose values are allowed provided that the average annual effective dose calculated for five successive years does not exceed 0.001 Sv.
- The average annual effective dose for the personnel is equal to 0.02 Sv, or effective dose for the period of professional life (50 years) is 1 Sv; the annual effective dose of 0.05 Sv is allowed provided that the average annual effective dose calculated for five successive years does not exceed 0.02 Sv.
- In case of radiation accidents, exposure greater than the prescribed basic hygienic standards (allowable dose limits) is allowed during a certain period of time and within the limits specified in sanitary codes and regulations.

According to the Russian legislation (Article 12), public associations have the right for public control of compliance with the requirements of the codes, standards, and regulations in the field of radiation safety assurance.

According to Article 14 “Requirements to Radiation Safety Assurance in Handling Sources of Ionizing Radiation”, organizations are required to do the following while handling sources of ionizing radiation:

- Follow the rules of the given Federal Law, other federal laws and regulatory legal acts of the Russian Federation, and other laws and regulatory legal acts of the Constituent Entities of the Russian Federation, codes, rules and standards in the field of radiation safety assurance;
- Plan and implement measures aimed at radiation safety assurance;
- Assure radiation safety of new (upgraded) products, materials, substances, technological processes and production which are the sources of ionizing radiation, for human health;
- Systematically conduct production control of radiation situation at workplaces, in premises, sites, control and supervised areas, and of release and discharge of radioactive substances;
- Control and account individual exposure doses of employees;

- Provide radiation safety assurance training and qualification of managers and specialists performing activities, production control specialists and other individuals who permanently or temporarily work with ionizing radiation sources;
- Arrange of preliminary (pre-employment) and periodical medical examinations of employees (personnel);
- Regularly inform employees (personnel) on ionizing radiation levels at their workplaces and on amount of their individual exposure doses;
- Timely inform the federal executive bodies which are authorized for state regulation and supervision in the field of radiation safety assurance, executive authorities of the Constituent Entities of the Russian Federation, about emergencies, deviations from the process regulations jeopardizing radiation safety assurance;
- Implement conclusions, decrees, prescriptions of authorities of the authorized executive bodies which exercise state regulation and supervision in the field of radiation safety assurance;
- Ensure enforcement of rights of the citizens in the field of radiation safety assurance.

The **Federal Law No. 190-FZ of 11 July 2011 "On Radioactive Waste Management"** [II–6] regulates the relationships occurred by the management of accumulated and being formed RW, stipulates principles of functioning and structure of unified State system for RW management, establishes organizational and legal basis for RW management.

According to this law, the unified state system for management of RW is created, the main aim of which is to organize and ensure the safe and economically effective RW management, including disposal.

Article 20 of the law stipulates the creation of the national operator for RW management – organization, defined by the Government of the Russian Federation according to the proposal of State authority, which administer RW management, to plan, organize and conduct activities of RW management, including their long term storage and disposal. For financial provision of the RW management activities, it is envisaged by the law to use the special reserve fund, which is created on the basis of regular payments by the producers of RW.

The main structure of Federal Law No. 190-FZ is as follows:

- Basic terms definition;
- RW classification:
    - 'Special' RW and 'Retrievable' (disposable) RW
    - Classes of retrievable RW (based on disposal option).
- For special (non-retrievable) RW:
    - Emplacement site
    - Site for conservation.
- Deep well injection of liquid low level waste and intermediate level waste on operating sites.

Initial RW registration is provisioned in Article 23 of the Federal Law "On Radioactive Waste Management" and has been implemented during the first phase of development of the unified state system of RW management, lasting from 15 January 2013 till 31 December 2014. The surveys have been carried out in all facilities having RW that are subject to the initial registration.

Based on the survey results special commissions will draw up the acts of initial registration, on the basis of which proposals will be developed to classify RW storage facility into a particular category of RW storage facilities.

The Russian State Corporation "Rosatom" has confirmed the schedule for survey and initial registration of the storage facilities (sites) for RW generated before 15 July 2011. Initial RW registration will provide the necessary information to form a registry of RW and the inventory of the RW storage facilities (sites). From a legal point of view, the initial RW registration will allow:

- To assign the status of "accumulated waste" to the waste generated before the Federal Law "On Radioactive Waste Management" has entered into force;
- To define the categories and types of storage facilities and of the accumulated RW according to the new Russian legal framework (retrievable and special waste, the temporary and long term storage, sites of emplacement and sites of conservation of special waste).

The type of RW and of the category of RW storage facility are crucial for planning of treatment of previously accumulated waste in accordance with the requirements of Article 24 of the Federal Law No. 190-FZ "On Radioactive Waste Management". Within the context of this Federal Law, RW is grouped into:

- Retrievable RW – RW for which radiological and other risks, as well as cost of its removal from storage facilities and subsequent management, including disposal, do not exceed risks and costs of its in-situ disposal;
- Special (non-retrievable) RW – RW for which radiological and other risks, as well as cost of its removal from storage facility and subsequent management, including disposal, exceed risks and costs of its in-situ disposal.

Retrievable RW is categorized based on the following characteristics:

- Half-life of radionuclides present in RW: long lived RW, short lived RW;
- Specific activity: high level waste, intermediate level waste, low level waste, very low level waste;
- Aggregate state: liquid RW, solid RW, gaseous RW;
- Nuclear material inventory: RW containing nuclear material, RW not containing nuclear material;
- Spent sealed radiation sources;
- RW resulting from mining and processing of uranium ore;
- RW produced as a result of non-nuclear mining and processing activities with mineral and organic raw materials with high content of natural radionuclides.

According to Article 24 of the Law, accumulated RW which is classified as retrievable RW is to be retrieved, processed, conditioned and disposed.

II–1.3.2. Federal norms and rules of nuclear and radiation safety

Federal norms and rules valid in the field of use of atomic energy are elaborated on the basis of normative legal Acts of the Russian Federation, the Convention on Nuclear Safety [II–7], as well as the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management [II–8], taking into account the recommendations of the international organizations acting in the field of use of atomic energy, in whose work the Russian Federation takes part.

Safe management of RW is regulated by a series of Federal norms and rules, elaborated taking into account the recommendations of the International Commission on Radiological Protection and the Organization for Economic Co-operation and Development, as well as various IAEA safety standards in the field of RW management (including IAEA Safety Series No. 111-F, The Principles of Radioactive Management, IAEA Safety Standards Series No. WS-G-1.2, Management of Radioactive Waste from the Mining and Milling of Ores, No. WS-R-1, Near Surface Disposal of Radioactive Waste, and others[5]).

"Radioactive Waste Management Safety. General Provisions" (NP-058-04) [II–9] establishes the aims and the principles to ensure safe RW management, as well as the general requirements to ensure safety.

Norms and rules NP-019-2000 [II–10], **NP-020-2000** [II–11], **NP-021-2000** [II–12] establish requirements to ensure safety during the collection, reprocessing, storage and conditioning of liquid, solid and gaseous RW, correspondingly, at nuclear facilities, radiation sources, storage facilities of nuclear materials and radioactive substances, and RW storage facilities.

"Radioactive Waste Disposal. Principles, Criteria and Basic Safety Requirements" (NP-055-04) [II–13] establishes principles, criteria and main safety requirements for the near-surface RW disposal and for the RW disposal in deep geological formations.

"Near-surface Disposal of Radioactive Wastes. Safety requirements" (NP-069-06) [II–14] develops and concretizes the requirements from Federal norms and rules NP-058-04 and NP-055-04 regarding safety achievement for the near-surface disposal of RW.

"Safety Regulations for Transport of Radioactive Material (NP-053-04)" [II–15] establish main technical and organizational requirements for systems transportation of nuclear materials, radioactive substances and RW, including spent nuclear fuel, aimed to ensure safety during the storage and transportation of nuclear materials, radioactive substances and RW at the objects of use of atomic energy.

"Radiation Safety Standards" (NRB-99/2009) [II–16]. In accordance with the requirements of Radiation Safety Standards in Russian Federation NRB-99/2009, the annual dose limit for radiation workers is 20 mSv/a and for the general public 1 mSv/a.

"Basic Sanitary Rules of Radiation Safety Assurance" (OSPORB-99/2010) [II–17]. According to the rules, the annual effective dose to the critical population group from RW management activities and facilities is not allowed to exceed 0.1 mSv.

II–1.3.3. Regulatory guidelines

Administrative Procedures for the Public Service of Licensing Activities in the Field of Atomic Energy Use to be Provided by the Federal Environmental, Industrial and Nuclear Supervision Service [II–18]

[5] Note that, even though the aforementioned IAEA safety standards have been superseded, they are still considered in the legal framework of the Russian Federation.

The regulatory body of the Russian Federation – the Federal Environmental, Industrial and Nuclear Supervision Service of Russia (ROSTECHNADZOR) – provides guidance on the detailed contents of documents to be submitted to the regulatory body in support of the application for authorization of RW processing and storage facilities and the ways of obtaining the required information.

The necessary set of documents for justifying radiation safety of operation of the radiation source, storage facility of radioactive substances, RW storage, the management of radioactive substances, as well as the use of radioactive substances for research and development works, includes the following:

- Safety analysis report of the nuclear facility for its operation or in the exercise of the declared activity;
- Operational regulations for the nuclear facility;
- Radiation safety instruction(s);
- Reference of personnel training and assessment of knowledge of radiation safety codes and standards, personnel appraisal, briefing and working authorizations for radiation-hazardous operations;
- Instructions on prevention and mitigation of accidents and fires, and their mitigation;
- Criteria for decision making in the case of a radiation accident (might be incorporated into instructions for accident and fire prevention and mitigation);
- Action plan to protect personnel and the population against radiation accident and its consequences;
- Description of the structure and composition of radiation safety service;
- Reference of the documents that define the procedure for radiation-hazardous operations, including process regulations and instructions, operational instructions, and maintenance and repair instructions (documents to be provided to ROSTECHNADZOR at request);
- List of the documents that specify requirements on safety of the nuclear facility and the declared type of activity (federal codes and standards, safety guides, regulations of ROSTECHNADZOR, national standards, organization in-house standards and documents of the license applicant). Information on documentation completeness in the applicant organization and the system of their accounting and amending;
- Certificate of accounting and control of radioactive substances and RW;
- Certificate of assurance of physical protection;
- Description of the existing quality management system of the applicant organization in the exercise of the declared activity;
- Quality assurance programme for the declared activity;
- Information on organizations that render engineering and technical support of the declared activity and engage in works and provide services in the field of atomic energy use in the exercise of that activity, listing the scope of works (services);
- Reference of certificates for the applied equipment, devices and technologies for radiation sources, storing facilities of radioactive substances and RW storages;
- Procedure instruction for road traffic collisions (to be provided only for a license for management of radioactive substances and/or waste in transport).

II–2. CONTEXT OF THE SAFETY CASE

II–2.1. Purpose of the safety case

This illustrative Safety Case takes into consideration IAEA Safety Standards Series Nos GSR Part 5, Predisposal Management of Radioactive Waste [II–19], GSG-3, The Safety Case and

Safety Assessment for the Predisposal Management of Radioactive Waste [II–1], and GSR Part 4 (Rev. 1), Safety Assessment for Facilities and Activities [II–20]. Safety criteria are taken from the Russian regulatory framework [II–16, II–17] and IAEA Safety Standards Series No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [II–21].

According to the licensing processes in the Russian Federation, the operating organization is required to attach to the license application a safety analysis report that includes justifications for the selected site, covering safety-related issues, giving general description of the nuclear facility and its safety impact on the environment and population, and containing preliminary safety and physical protection analysis taken as required by regulations in force.

In order to follow international practice and requirements, the philosophy of Safety Case has been implemented for justification of waste retrieval from a historical RADON-type facility. The principal purposes of this Safety Case are:

— To demonstrate safety of the RADON-type facility during historical waste retrieval;
— To justify continued operations and identify areas for decommissioning of the facility.

The iterative development of the Safety Case throughout the lifetime of the facility resulted in the following achievements:

— The systematic collection, analysis and interpretation of the necessary scientific and technical data;
— The development of plans for operation;
— Iterative studies for design optimization, operation and safety assessment with progressively improving data and comments from technical and regulatory reviews.

The following specific aspects will be addressed in this Safety Case:

— Demonstration of safety of the RADON-type facility;
— Demonstration of safety of various RW management activities conducted by the operating organization;
— Optimization of the respective waste management activities;
— Management systems implemented in support and to ensure the safety of the respective waste management activities;
— Definition of limits, controls and conditions that will be applicable to the facilities and the respective activities;
— Input to the improvement of existing radiation protection programmes and procedures for the conduct of activities.

II–2.2. Scope of the safety case

The full scope of the Safety Case includes the retrieval of solid RW from the historical RADON-type facility as a precursor to its decommissioning, historical waste packaging and preparation of waste packages for further transportation to an existing authorized waste storage site.

The process of waste retrieval from each vault is divided into the following phases:

— Phase I. Unloading of large-sized RW packages available for gripping and retrieval;
— Phase II. Unloading of small-sized RW packages;

— Phase III. Accomplishment of unloading of large-sized RW packages released from under the debris;
— Phase IV. Collection and packaging of spillages.

For illustrative purposes only, the waste retrieval from Vault 1 is considered in this Safety Case. Waste retrieval operations from Vault 1 are anticipated to take approximately 100 working days.

An assessment of the non-radiological hazards of the facility and activities and associated identification of specific control measures is outside of the scope of this safety case.

To take into account the variation of dose rates from different packages, operations performed during Phase I have been divided into four subphases:

— Subphase I1. Retrieval of waste containers К4 – К7, Б9;
— Subphase I2. Retrieval of waste containers К3, Б3 – Б5, Б8;
— Subphase I3. Retrieval of waste containers К1, К2, К8, К9, Б2;
— Subphase I4. Retrieval of waste containers Б6, Б7.

Table II–1 provides a detailed list of these activities.

TABLE II–1. OPERATIONS PERFORMED DURING RETRIEVAL FROM VAULT 1

Activity	Area
Phase I1, Initial monitoring of package in the vault	Area A
Phase I1, Entering the vault for slinging	Area A
Phase I1, Slinging package and leaving the vault	Area A
Phase I1, Lifting package to the height of 10 cm	Area A
Phase I1, Entering the vault and additional slinging (if necessary)	Area A
Phase I2, Initial monitoring of package in the vault	Area A
Phase I2, Entering the vault for slinging	Area A
Phase I2, Slinging package and leaving the vault	Area A
Phase I2, Lifting package to the height of 10 cm	Area A
Phase I2, Entering the vault and additional slinging (if necessary)	Area A
Phase I3, Initial monitoring of package in the vault	Area A
Phase I3, Entering the vault for slinging	Area A
Phase I3, Slinging package and leaving the vault	Area A
Phase I3, Lifting package to the height of 10 cm	Area A
Phase I3, Entering the vault and additional slinging (if necessary)	Area A
Phase I4, Initial monitoring of package in the vault	Area A
Phase I4, Entering the vault for slinging	Area A
Phase I4, Slinging package and leaving the vault	Area A
Phase I4, Lifting package to the height of 10 cm	Area A
Phase I4, Entering the vault and additional slinging (if necessary)	Area A
Phase II, Entering the vault and slinging DSRS	Area A
Phase II, Checking of slinging (lifting to the height of 10 cm) and leaving the vault	Area A
Phase III, Initial monitoring of package in the vault	Area A

TABLE II–1. OPERATIONS PERFORMED DURING RETRIEVAL FROM VAULT 1 (cont.)

Activity	Area
Phase III, Entering the vault for slinging	Area A
Phase III, Slinging package and leaving the vault	Area A
Phase III, Lifting package to the height of 10 cm	Area A
Phase III, Entering the vault and additional slinging (if necessary)	Area A
Phase IV, Cleaning the vault and retrieval of debris (not finished)	Area A
Phase I1, Parking of transport container in area B	Area B
Phase I1, Loading of waste package into transport container	Area B
Phase I1, Checking dose rate and contamination	Area B
Phase I1, Removal of slinging from waste package	Area B
Phase I1, Removal of loaded transport container from area B	Area B
Phase I2, Parking of transport container in area B	Area B
Phase I2, Loading of waste package into transport container	Area B
Phase I2, Checking dose rate and contamination	Area B
Phase I2, Removal of slinging from waste package	Area B
Phase I2, Removal of loaded transport container from area B	Area B
Phase I3, Parking of transport container in area B	Area B
Phase I3, Loading of waste package into transport container	Area B
Phase I3, Checking dose rate and contamination	Area B
Phase I3, Removal of slinging from waste package	Area B
Phase I3, Removal of loaded transport container from area B	Area B
Phase I4, Parking of transport container in area B	Area B
Phase I4, Loading of waste package into transport container	Area B
Phase I4, Checking dose rate and contamination	Area B
Phase I4, Removal of slinging from waste package	Area B
Phase I4, Removal of loaded transport container from area B	Area B
Phase III, Parking of transport container in area B	Area B
Phase III, Loading of waste package into transport container	Area B
Phase III, Checking dose rate and contamination	Area B
Phase III, Removal of slinging from waste package	Area B
Phase III, Removal of loaded transport container from area B	Area B
Parking of transport container in area C	Area C
Placing DSRS on the decontamination platform	Area C
Checking dose rate from “BGI” DSRS, working distance 10cm	Area C
Checking dose rate from other DSRS, working distance 10cm	Area C
Transfer “BGI” DSRS to transport platform	Area C
Fixing “BGI” DSRS on transport platform	Area C
Loading transport platform to transport container	Area C
Handling “BGI” DSRS at distance of 100 cm	Area C
Handling other DSRS at distance of 100 cm	Area C
Loading damaged DSRS into shielded container	Area C

TABLE II–1.OPERATIONS PERFORMED DURING RETRIEVAL FROM VAULT 1 (cont.)

Activity	Area
Transfer shielded container to storage location	Area C
Personnel stay in area D	Area D
Personnel stay in area E during phase I	Area E
Personnel stay in area E during phases II, III and IV	Area E
Phase I1, Truck loading with transport container	Area F
Phase I2, Truck loading with transport container	Area F
Phase I3, Truck loading with transport container	Area F
Phase I4, Truck loading with transport container	Area F
Phase II, Truck loading with transport container	Area F
Phase III, Truck loading with transport container	Area F
Phase IV, Truck loading with transport container	Area F

II–2.3. Demonstration of safety

This section describes the approach to demonstration of safety; specifically, the applicable safety objectives, safety principles, and regulatory requirements. Taking cognizance of the scope of the Safety Case and the application of the graded approach as described in Section 2.4, the safety of the waste retrieval activities will be evaluated and demonstrated as described below.

Approach to basic engineering analysis

A combined qualitative and quantitative assessment will form the basis of the basic engineering analysis, which will cover the following aspects:

- Basic site characteristics and credible external events considered in the design of the hangar facility;
- Quality assurance considered in the design, construction and commissioning of the new facility;
- Application of national construction codes and standards;
- Inspection and maintenance plans;
- Formal processes for the evaluation, approval and implementation of modifications;
- Safety and security aspects.

The following specific assessments will be performed:

- For normal operations, quantitative deterministic assessments of worker dose resulting from the range of activities performed by workers, including determination of the allowed working hours in Supervised and Controlled areas;
- For anticipated operational occurrences, quantitative deterministic assessments of occupational and public doses as applicable;
- For all other credible accident scenarios, a quantitative and qualitative assessment of the impact of other occurrences with identification of specific preventative and mitigating measures.

Approach to safety assessment

The radiological assessment will be based upon a realistic and conservative approach taking into consideration measured data from surveys.

Uncertainties inherent to the assumptions made in the quantitative assessments or any other uncertainties identified during the safety assessment will be evaluated to determine their impact on safety. Uncertainties with a significant impact on safety will be listed with recommendations for their management.

A qualitative assessment will be performed of the availability and level of implementation of an integrated management system to ensure a sustained level of safety. This assessment focuses on radiation protection, work procedures, quality assurance aspects and processes for the management of operating limits and conditions.

II–2.4. Graded approach

A graded approach is applied to define the extent and depth of this safety case by the use of qualitative assessment of hazards and deterministic analysis of doses to potential receptors (i.e. workers and public). This takes into consideration the relative safety significance and complexity of operations and the maturity of the operating organization.

The steps to undertake waste retrieval are comparable to decommissioning activities and are, in fact, a relatively straightforward precursor to decommissioning itself. Therefore, elements of the graded approach applied to decommissioning, as emphasized in IAEA Safety Standards Series No. WS-G-5.2, Safety Assessment for the Decommissioning of Facilities Using Radioactive Material [II–22], will be applied herein, including the following factors:

- — The RADON-type facility is a relatively small, stand-alone facility with no surrounding nuclear installations;
- — The radioactive inventory of the facility is relatively low;
- — Low complexity of the waste retrieval operations;
- — Good quality characterization data for the stored wastes;
- — Simplicity of engineering safety measures provided for the retrieval operations.

Since waste retrieval activities necessitate a significant number of manual operations, a detailed breakdown of the steps (divided into phases and subphases) will enable a comprehensive dose assessment to be undertaken for normal operations.

II–2.5. Strategy for safety

This section describes the strategy for safety, including the approach that was taken in the facility design and all the respective waste retrieval activities to comply with the regulatory requirements and to ensure that good engineering practice has been adopted and that safety and protection are optimized.

In view of the scope of the Safety Case, the following strategies for demonstrating safety are adopted:

- — Safety principles – all the safety principles defined by IAEA requirements and resumed in Federal Laws of the Russian Federation are met.
- — Step by step approach is used with principle that the facility and the activities performed in the facility can adapt to good new findings and practice.

- Defence in depth – Care is taken to ensure that multiple safety layers are established. This principle is considered to ensure that no important safety argument is based on a single level of protection.
- Passive safety – The use of passive safety systems wherever possible.
- Shielding – Ensuring that doses to workers and the public are as low as possible. This also includes the optimization of shielding usage during all waste management activities is considered.
- Optimized waste management procedures − Clear roles and responsibilities; trained and competent staff; use of radiation surveys and dose monitoring to inform procedures; use of remote or semi-remote equipment to undertake higher dose rate activities.

Overall approach to safety of the facility

The first step for implementing the retrieval process is to set up a controlled area around the retrieval site. In this Safety Case, a construction of temporary structure (hangar) is considered. The purpose of this structure is to limit access to the area during waste retrieval and to control the potential spread of contamination that might be created by disturbing the waste. The structure also protects the work area and workers from sun, rain and wind.

A simple robust design has been adopted for the construction of the hangar and associated waste retrieval equipment so as to make operations within the facility simple and easy to undertake. The facility design and construction provide defence in depth and is designed to rely predominantly on passive safety features. No single design feature will be relied on for the overall safety of the facility.

All activities in connection with the retrieval of old waste will be carried out in full conformity with radiation protection quality and safety requirements as defined by national legislation.

II–3. DESCRIPTION OF THE SITE, THE RADON-TYPE FACILITY AND THE WASTE

According to GSG-3 [II–1], the safety case includes the following elements: a full description of the structures, systems and components (SSCs) of the facility and their importance for safety; the quantity and characteristics of the waste to be handled at the facility; the range of conditions under which the facility might operate; the hazards to which the facility might be exposed; and the required performance criteria.

II–3.1. Description of the site

II–3.1.1. General description of the site

The site that accommodates the historical RADON-type facility is levelled and partially laid with asphalt. There are some oversized boulders on the site, their size reaching 2.2 m across. Absolute elevations range from 220 to 229 m in accordance with the Baltic Height System.

Topsoil in the RWSF location area is not fertile. The surrounding territory is not used for agricultural purposes. Ground water is not used as a fresh water source for domestic purposes.

The RWSF location area is not characterized by any notable level of human-induced radioactive contamination. The largest contribution to the gamma-radiation dose rate is introduced by natural radioactive elements (uranium, thorium and potassium), which are contained in the rocks. Prevailing radiation levels in the whole territory of the region are rather low. Radiation levels over boggy terrains are especially low.

The annual radiation dose exposure to natural background radiation for the population of the RWSF location area does not exceed 1 mSv. Thus, the background radiological situation allows for monitoring of the facility with a good sensitivity.

Climatic conditions of the RWSF site create no significant impediments for its operation.

The territory is seismically calm. Some phenomena in the rocks can lead to an earthquake of magnitude 6. The systems important to safety, for example transportation and ventilation systems, are designed to withstand an earthquake of such magnitude.

Groundwater in the territory adjacent to the RWSF occurs at depths between 0.5 and 1.3 m. Groundwater is unconfined. The expected water table rise is to 0.0 m. The RWSF was constructed on top of a human-made soil fill over 1.5 m high; therefore, its flooding is not expected. Still, rain and melt water ingress inside the storage is taking place. Ingressing water flow is rather slow, about 30 m^3 per year. During the 2000s, water was pumped out and purified twice. No radioactive effluents from the storage facility were detected during monitoring.

The geological section of the site is represented by (from top to bottom): fill-up soil with boulders, pebbles, gravel, up to 1 m thick; glacial sediments (sandy loam and sand with boulders, pebbles, gravel) – up to 1.8 m; crumbling rocks. Rocks are characterized by a rather high perviousness. The filtration factor is 0.04 m/day for glacial sediments, and up to 30 m/day for rocks.

The location of the RWSF is shown in Fig. II–1.

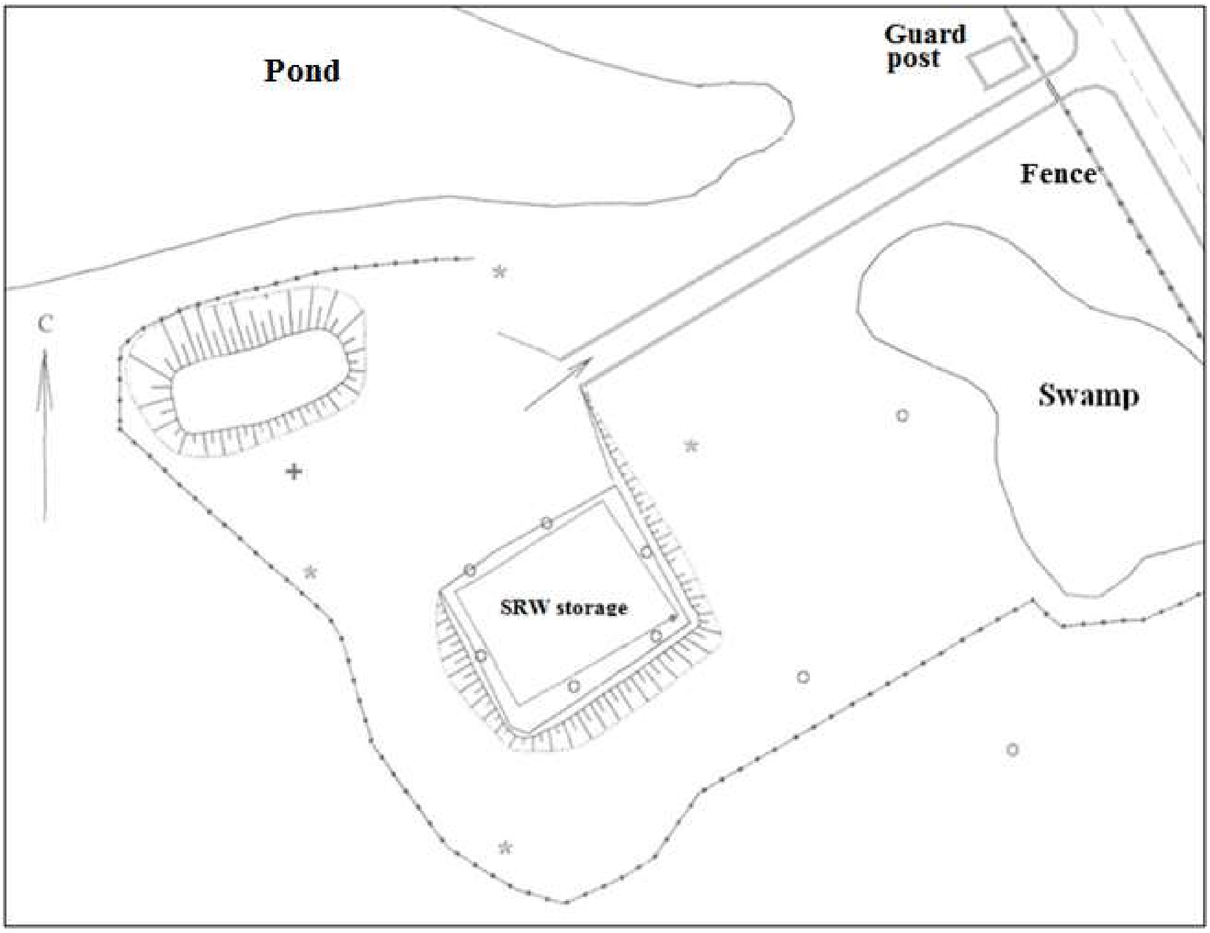


FIG. II–1. Illustration of historical RADON-type facility.

II–3.1.2. Topography

The RWSF location area occupies the north-eastern edge of the crystalline shield formed by Precambrian and upper Palaeozoic rocks. The surface represents a system of uplands and depressions smoothed over under the action of the glacier. The northern coast is steep and split by deep and narrow bays. The eastern and southern coasts are lowland.

The relief of the RWSF location area is relatively smooth due to the position of the dome-shaped top of moraine rocks. A small water body lies 17 m to the north-west of the site. This pond was formed by atmospheric precipitation after road construction in the territory of the RWSF, and has no organized water discharge. A boggy area was formed on the other side of the road; its larger part lies outside the site.

Some areas of the RWSF are covered with low shrubs and trees. The vegetation period is 80 to 130 days.

The RWSF site is levelled and partially laid with asphalt. There are separate standing boulders at the site; their dimensions reach 2.2 m across. Absolute elevations range from 220 to 229 m in accordance with the Baltic Height System.

II–3.1.3. Population and demography

The largest city is located in the centre of the RWSF location region with its population of approximately 325 000 people. At the beginning of 2005, the population of the whole region was 872 000 people, with urban population constituting 798 000 people and rural population 74 300. Populated areas occupy 0.4% of the territory of the region, and agricultural lands 0.2%.

The average population density for the entire RWSF location area is 6 people per km^2. There are no urban communities near the RWSF, and the rural population density is 0.5 people per km^2.

Industrial development of the RWSF location area began at the end of the 1920s and was connected with the exploitation of mineral deposits.

The major contributors to population employment are industry (27%), social security, education, culture, science (21.25%), trade, sales, procurement (16.4%), transportation and communication (10.1%), and construction (4.5%).

The distance from the RWSF to the nearest community is 10 km. A large open water body (lake) is located 1000 m from the RWSF, and the nearest river 10 km from the RWSF.

In the RWSF location area – within a radius exceeding the radius of the controlled area – there are no gas/oil trunk lines, industrial facilities or plants, warehouses, or water reservoirs.

II–3.1.4. Meteorology

Winds from the seas bring humid air that contributes to cloudiness and precipitation. Days are overcast 198 days on average in a year, with fully clear days averaging only 16 days a year.

The annual precipitation is 488 mm, including 166 mm during the cold season, and 322 mm during the warm season. Average seasonal precipitation is as follows: spring – 14%, summer – 40%, autumn – 29%, winter – 17%. The maximum daily precipitation is 58 mm. Precipitation falls almost 150 days a year, mostly in summer.

As summer is short and cool, only a small portion of precipitation turns into vapour, and the remaining portion is discharged; which is why the surrounding land is rich in rivers, lakes and swamps. The rivers cut into solid crystalline rocks; therefore, the area features many rapids and waterfalls. Most rivers flow out of or through the lakes, which regulate river discharge. Several large lakes are located in the RWSF location area. The number of small lakes in the area is over twenty thousand.

Typical snowfall begins in late September. However, stable snow cover is formed only by the second week of November. Snow cover is uneven, and depends mainly on relief and prevailing winds. Snow depth varies between 25 cm and 75 cm. Snow melts at the end of May or in early June. The snow cover is deepest in late March. Duration of snow cover is 200 days. The number of snowstorm days ranges from 23 to 111; snowstorms usually begin in October and end in May.

In winter, the average air temperature in the RWSF area is similar to central areas of the European part of the country. The average annual air temperature in the area is plus 0.2°C. The temperature changes frequently: thaws can occur in any winter month, and light frosts can occur in summer due to collision of cold air masses with warm air currents.

Winter lasts for five months (from November to March). The average air temperature of the coldest month of the year (January) is −8°C on the seaside, and −13°C in the internal areas. Considerable air temperature fluctuations are possible: from −50°C up to 4–10°C. Frosty days with an average daily temperature below −20°C, −25°C, or −30°C are relatively seldom. The number of days with persistent frost varies between 140 and 160.

The average air temperature of the warmest month of the year (July) is 12 to 14°C. In summer, the number of hot days (temperatures over 20°C) is about 16 to 27 in the central areas of the region. The first light frost can occur as early as August, and the last of the light frosts can occur in late May and in June. The duration of the frost-free period varies between 50 and 100 days. The depth of frost penetration in clay and loamy soils is 130 cm. The RWSF location area is outside the permafrost region, and only in some places the ground temperature is below zero all the year round. The annual average temperature of the ground is 1−3°C, depending on the cover and protection.

Wind conditions in the RWSF location area are highly diverse. Winds of southern directions prevail in the RWSF location area in winter; winds of northern directions prevail in summer. Prevailing winds in winter are south and south-west; in summer winds of northern directions prevail. Winds in transitional seasons are less stable, though the winds of southern direction prevail.

Erosive processes are practically non-existent in the territory of the RWSF location area.

Foggy fumes appear over the non-freezing gulf in the cold season. Sometimes they are very thick and reduce the visibility down to 2–5 m. In summer, fog is often observed on the seaside, when the winds are onshore.

Because of significant cloudiness, the average annual insolation is just slightly over half of the potential insolation for the given latitude. Only during the mainly clear months – March and April – the average annual insolation reaches 2/3 of the potential insolation. The solar spectrum contains UV radiation only from April to September.

II–3.1.5. Site geology and hydrology

Hydrogeological conditions within the survey depth are characterized by one water-bearing system, which is confined to unbroken glacial sediments and crumbling rock.

Groundwater in loose thickness of quaternary sediments is opened by two pits (bore pits No. 1688 and No. 1690), which are located in the south-eastern part of the surveyed site (outside the territory of the RWSF), at the depth of 0.5 and 1.3 m, respectively.

When building the RWSF, the surface was levelled using fill-up soil (1.5 m and higher). Ground water within the site is opened by two drilled bores in rock at a depth of 2.2–4.9 m. Groundwater is unconfined.

Engineering and geological holes (open stoppings and separate bore pits) were used as monitoring wells to measure the groundwater level. The results are shown in Table II–2.

TABLE II–2. RESULTS OF OBSERVATIONS OF GROUND WATER DEPTH AND LEVEL

Well number	Distance from the ground surface and absolute elevations are indicated for every measurement (in meters)						
	Date observed						
	28.11.02	03.12.02	04.12.02	09.12.02	23.12.02	04.01.03	23.01.03
1680	–	5.1 220.92	4.0 222.02	3.2 222.82	2.2 223.82	2.2 223.82	2.2 223.82
1681	–	–	2.2 224.13	2.2 224.13	2.2 224.13	2.2 224.13	2.2 224.13
1682	–	2.4 224.14	2.4 224.14	2.4 224.14	2.4 224.14	2.4 224.14	2.4 224.14
1683	–	–	5.0 221.48	4.9 221.58	4.5 221.98	4.4 222.08	4.4 222.08
1684	5.1 223.3	4.4 224.00	4.4 224.00	4.4 224.00	4.3 224.10	4.3 224.10	4.3 224.10
1685	no	no	no	no	no	no	4.9 223.83
1694	–	–	–	–	4.6 224.03	4.5 224.13	4.4 224.23

When drawing the groundwater contour, the data for well No. 1683 was mapped out as it could not be unambiguously interpreted, and at the present time it is impossible to confirm and revise the hydrogeological situation in its location area, since well No. 1683 was destroyed in 2007 in the course of preliminary construction works connected with the impending reconstruction of the RWSF.

On the whole, groundwater flow follows the general direction of lower elevations. The prevailing direction of groundwater flow from the area of the RWSF location is south-east. Within the local fill-up plateau which accommodates the RWSF, groundwater stays practically at the same elevation (224.1 m). An average slope of the aquifer to the southeast can be determined from comparison of the data for well No. 1685 near the northwest corner of the storage and bore pit No. 1688, which is located approximately 115 m to the southeast. The difference in water levels for these holes is 1.4 m, and the hydrodynamic gradient is estimated to be 0.012.

During spring floods and high levels of precipitation, flooding can occur up to the depth of 1.5 m in the storage territory and reaching land surface outside its boundaries; seasonal perched water in the lows can also occur.

Basic characteristics of the underground water are:

- Aquifer number (top to bottom): 1;
- Underground water type and character: free aquifer;
- Aquifer depth of occurrence:
  - During the survey period: 0.5–4.9;
  - Expected maximum: 0.0;
  - Expected minimum: 4.9.
- Water-bearing rocks: unbroken glacial sediments and crumbling rocks;
- Aquifuge: untapped;
- Aquifer boundary conditions:
  - Feed: infiltration of precipitation;
  - Discharge (drainage): outside the site.

The following negative characteristics of natural environment in the RWSF location area were identified:

- High groundwater level, which occur in immediate proximity to the bottom of the RWSF;
- A water body (pond) is located near the RWSF;
- Poor thickness of quaternary and moraine sediments;
- The old storage is located in crumbling rocks.

II–3.1.6. Site seismology

The geological processes prevailing in the territory of the RWSF location area are diverse in genesis and intensity of manifestation. On the whole, the territory is seismically calm. However, the RWSF location area is characterized by highly irregular distribution of stresses. Abnormally high stresses in crystalline rocks cause inrush, bounce and rock bursts in mine works. This is connected with block faulting along northwest faults that can lead to earthquakes of magnitude 6.

II–3.1.7. Radiological conditions in the vicinity of the site

A survey was carried out in 2002–2003 in order to develop a RWSF reconstruction design. Six boreholes were drilled and 32 pits developed at a distance from 0.5–20 m from the storage wall. Water and soil samples were studied in the state sanitary and epidemiological control centre of the region. Soil and ground samples were extracted from 0–5 m boreholes near the storage site.

The results demonstrated the following:

- The near field concentration of Cs-137 is insignificant, at the level of 0–2.7 Bq/kg (measurement error over 100%), and there is no clear dependence of activity on the sample depth.

- Low concentration of Cs-137 in the samples extracted near the storage – both at a depth and on the surface, where earthwork was underway and therefore the ground was perturbed – also counts in favour of the conclusion that Cs-137 with a concentration of 100–180 Bq/kg in the samples of the unperturbed ground surface farther from the storage is a product of global fallouts.

— ^{40}K, ^{226}Ra, ^{232}Th nuclides are present in concentrations of natural origin.

The specific activities of Cs-137 and Sr-90 in the soils near the RWSF correspond to the average values that are characteristic of the area.

Tritium samples were taken in the lakes to the south-west and north-east of the storage facility. In the water bodies, no high content of tritium was detected. No cases of beta activity exceeding background values were found in the water bodies inside the controlled area.

The radiation dose rate at the RWSF ceiling ranges from 0.3 to 2.0 μSv/h; over the hatches up to 100 μSv/h; the rest of the storage site from 0.07 to 0.25 μSv/h (corresponding to the background values in the locality).

Annual average individual doses of the RWSF personnel ranged from 0.37 to 1.56 mSv in the period from 2002 to 2008; the maximum values for some employees ranged from 0.63 mSv in 2008 to 1.9 mSv in 2005.

II–3.2. Description of the historical RADON-type facility

The RWSF design includes four buried vaults with the capacity of 200 m^3 each, intended for storage of solid RW (SRW). RWSF as seen prior to waste retrieval is presented in Figs II–2 and II–3. Details on RW storage vaults are presented in Table II–3.

TABLE II–3. DESIGN OF RW STORAGE VAULTS

RADON-type facility	Number of vaults	Brief description
SRW storage vaults: — Length: 15 m; — Width: 5 m; — Depth: 3 m; — Storage capacity: 200 m^3.	4	Reinforced concrete buried vaults (design TP-4891)


FIG. II–2. RADON-type facility as seen prior to and after the construction of the hangar.

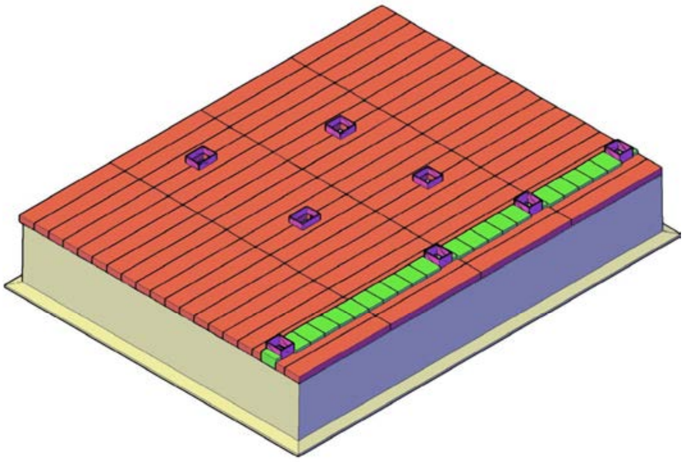


FIG. II–3. Ceilings of RADON-type vaults.

II–3.3. Description of the historical waste in the RADON-type facility

At most of the Radon-type facilities, there are a few common problems relating to waste inventory records. One specific problem relates to the uncertainty associated with insufficient information on the waste inventory, e.g. the waste is commonly labeled as ‘mixed fission products’ or simply ‘RW’.

The absence and inadequacy of waste inventory records makes the issue of restoring records and transferring them to a modern records management system a rather complex and cumbersome task, if at all practical. In cases, when data restoration and verification using administrative methods are not feasible, or available records are not adequate, the actual retrieval of the waste or waste packages might need to be undertaken to generate relevant data on the waste inventory.

II–3.3.1. Available records on historical waste

The major sources of data on the RW being in the vaults of the historical RADON-type facility are two volumes of the “SRW Vaults Loading Register”. The first of these was kept from December 1961 to December 1976 and the second one was kept from January 1977 to June 1993. A review of the entries in the registers has shown the following.

Vault 1: Most of contents were emplaced in October 1976 when 20 containers with ion exchange resin of total capacity 190 m^3 were loaded. Later, additional amounts were loaded occasionally in the period from May 1977 to July 1987. According to the register records, the amount of the waste loaded during this time is about 11 m^3, which makes a total of 201 m^3 together with the initially loaded quantity. This value exceeds the tank capacity value. The cause of this discrepancy is likely to be an incorrect estimation of the initial load amount.

Vault 2: The vault was loaded fairly regularly from December 1961 to April 1991. An entry with respect to the total SRW amount loaded in the last year was made in the register practically every year (there are no records for the years 1970, 1971 and 1984 to 1986). Summing up all of the above amounts gives a total of ~251 m^3 of SRW loaded into the vaults, which exceeds the available tank capacity by a considerable amount.

Vault 3: The vault was loaded fairly regularly from June 1991 to June 1993. Summing up all of the above amounts gives a total of ~26 m^3 of SRW loaded into the tank.

Vault 4: There are no records on the loading of this vault indicating that it is empty.

The RW in storage in the vaults consist mostly of DSRS and are composed of gamma relays, level gages, thickness gages, gamma-ray flaw defectors, medical sources and radiation standards.

Apart from the above RW, the tanks contain wastes that were packed in paper and plastic bags (overalls, personal protective equipment, laboratory utensils, tools, rags).

After summing up the radionuclide activities based on the loading register records and taking into account the gamma source decay process, it can be concluded that the activity in the SRW storage tanks is as follows: 9.6×10^{13} Bq of Cs-137, 1.7×10^{12} Bq of Sr-90, 1.3×10^{13} Bq of Ra-226, and 1.3×10^{12} Bq of Pu-239. These estimates are approximately the same as the data given in the 2006 SRW inventory taking statement. The containers with ion exchange resin in tank 1 are attributed with an activity of approximately 3.7×10^{14} Bq, whilethe loading register entry of 19 October 1976 specifies an activity of 4.4×10^{12} Bq for all containers. The reason for the discrepancy is a 1993 letter of the shipping company, which stated that the total activity of the ion exchangers loaded into the tank was estimated to be 3.7×10^{14} Bq (as of 1993). The letter only identifies ten containers and not twenty as noted in the registers.

II–3.3.2. Radiation and visual survey results

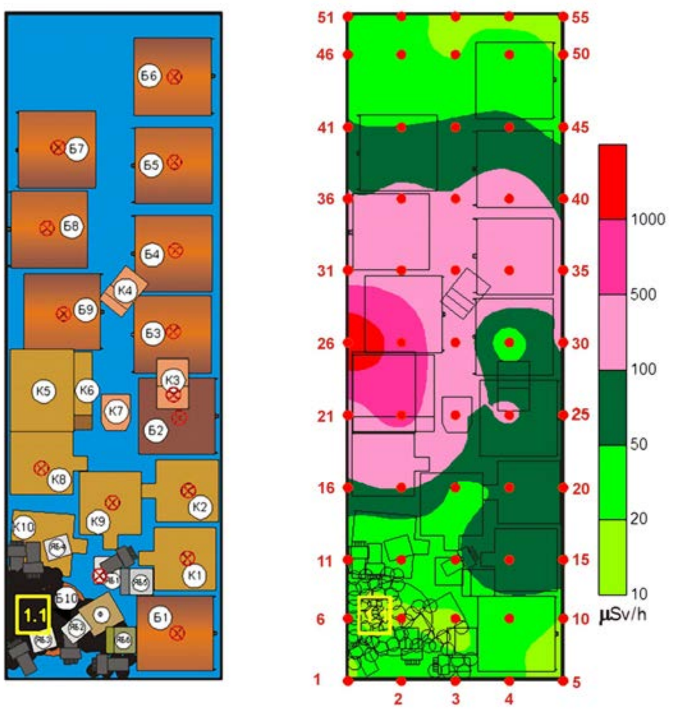
Vault 1:

The following items were found in Vault 1:

- 10 cylindrical metal containers of 1.7 m in diameter, 1.75 m height and volume ~4 m^3 each. On the layout in Fig. II–4 (a), they are shown as Б1–Б10.
- 7 metal containers with lateral dimensions of 1.7 m × 1.8 m × 1.4 m and a volume of ~3.6 m^3 each. In Fig. II–4 (a), they are shown as К1–К2, К5–К6, К8–К10.
- 2 metal containers with the dimensions of 0.7 m × 0.7 m × 0.95 m and a volume of 0.68 m^3 each. In Fig. II–4 (a), they are shown as К4 and К7.
- 1 metal container with the dimensions of 0.65 m × 0.65 m × 0.8 m and a volume of 0.47 m^3. In Fig. II–4 (a), it is shown as К3.
- 6 wood cases embedded in concrete from the inside with a volume of up to 0.2 m^3 each. In Fig. II–4 (a), they are shown as ЯБ1–ЯБ6.
- 1 wood case with air filters of a volume of up to 0.2 m^3. In Fig. II–4 (a), it is shown as Ф.
- About 40 blocks of gamma-ray sources of BGI E-1M type representing steel ball-shaped containers and containing a gamma-ray source inside, with the diameter of ~30 cm and a volume of not more than 0.015 m^3. In Fig. II–4 (a), they are shown as dark balls.
- Construction debris and rubbish that occurred as a result of cracking and destruction of RW packages (hereinafter referred to as "spillages").

The total amount of SRW in Vault 1 makes up less than 70 m^3. The majority of SRW containers (Б1–Б10 cylindrical metal containers and К1–К10 metal containers) was loaded into the vault when the upper coating slabs were removed and spread relatively evenly inside the vault. Other objects (blocks of gamma-ray sources, cases embedded in concrete) were loaded through the designed loading hatch, thus they are located directly under the hatch. The wood encasement of the cases embedded in concrete has been burst and partially peeled off over the period of storage. Data was obtained from the storage radiation survey and measuring exposure dose rates (EDRs) on the surfaces of containers piled inside Vault 1.

The RW layout and EDR distribution at the ceiling level inside Vault 1 are shown in Fig. II–4.



(a) (b)

FIG. II–4. (a) Layout of Vault 1 and (b) EDR distribution at the ceiling level inside the vault.

EDR values in measurement points for certain containers are presented in Table II–4.

TABLE II–4. GAMMA-RAY EDR FROM CERTAIN PACKAGES IN VAULT 1

	EDR from certain objects, µSv/h				
Packages	Centre	North	South	West	East
Б1	15	50	7	30	20
Б2	200	120	70	160	150
Б3	70	1000	120	56	18
Б4	500	600	1000	**2000**	**2000**

TABLE II–4. GAMMA-RAY EDR FROM CERTAIN PACKAGES IN VAULT 1 (cont.)

	EDR from certain objects, μSv/h				
Package	Centre	North	South	West	East
Б5	40	8	600	50	21
Б6	19	2	10	12	13
Б7	60	50	24	50	50
Б8	70	24	70	12	600
Б9	340	70	700	65	150
Б10	N/A*	500	N/A	N/A	500
К1	450	60	80	230	120
К2	30	64	60	20	N/A
К3	55	N/A	60	190	30
К4	100	320	100	55	350
К5	700	1300	350	620	550
К6	700	80	220	N/A	140
К7	150	100	200	170	N/A
К8	600	140	25	150	220
К9	30	65	40	40	16
К10	15	85	320	5	60

* N/A means no access

Details on radionuclides determining radiation from packages in Vault 1 are presented in Table II–5.

TABLE II–5. RADIONUCLIDES DETERMINING RADIATION FROM PACKAGES IN VAULT 1

Package	Main radiation
Б2, Б3, Б4, Б5, Б8, Б9 К5, К6, К8, К9	Radiation of Cs^{137}
К1	Radiation of Cs^{137} + radiation with an energy of more than 700 keV
Б1, Б6, К2	Radiation of Cs^{137} + radiation of Eu^{152} on the container surface
Б7, К3, К10, Яб1, Яб2, Яб3	Scattered radiation of Cs^{137}

Based on the measurement results, the major radiation source in the packages is Cs-137. In addition, Eu-152 was detected on the surface of some packages.

The gamma EDR was measured at the ceiling level inside the tank using an MKS-14ETs gamma dosimeter. The data obtained will help to assess the radiation situation over the tank after the concrete ceiling is removed.

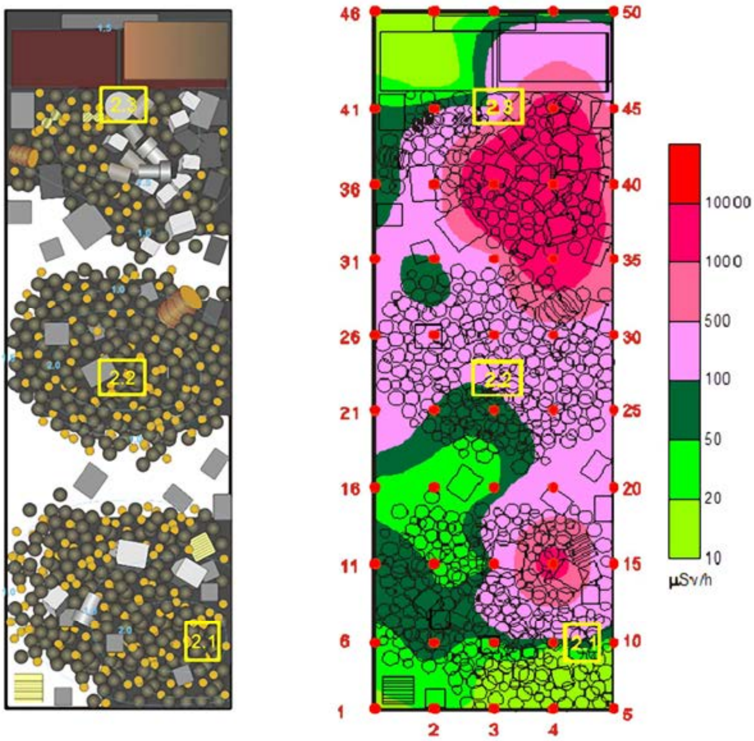
Vault 2:

Up to 70% of all SRW in the tank are the BGI E-1M and BGI-75 gamma source units. These units are steel spherical containers with a gamma source inside with a diameter of ~30 cm (see Fig. II–5) and a capacity of about 0.015 m^3.

FIG. II–5. Illustration of a block of gamma-ray source (E-1M type).

Figure II–6 provides the RW layout and EDR distribution at the ceiling level inside Vault 2. The total volume of SRW in tank 2 is about 100 m^3. In addition, the tank contains:

— Not less than 5 concrete-grouted metal drums of 0.1 to 0.2 m^3;
— Not less than 30 concreted wooden cases of 0.2 m^3 each;
— Several radioisotope level gauges (ur-8 and others), radioisotope smoke detectors (rid-1, rid-5), containers for beta sources (tsr-m), bags with pipes, and other small objects.



(a) (b)

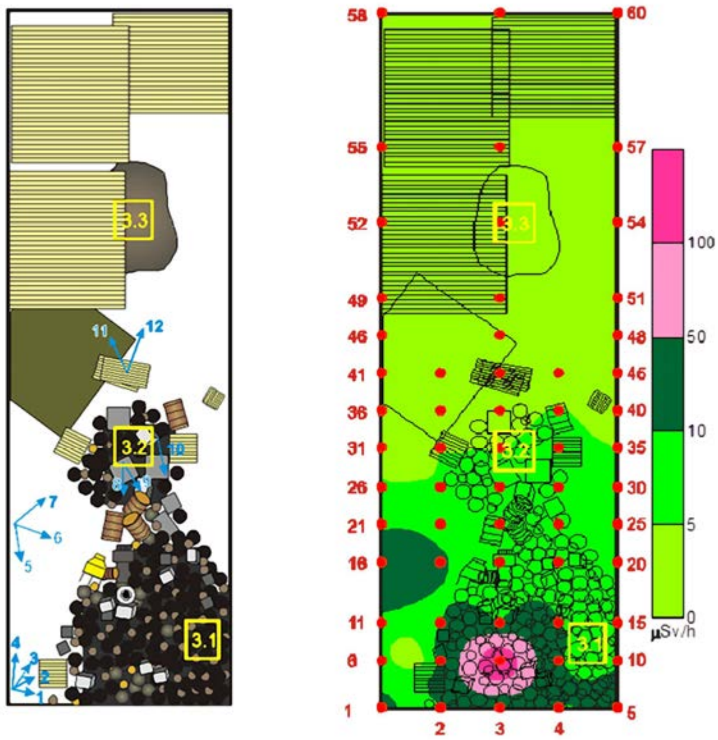
FIG. II–6. (a) Layout of Vault 2 and (b) EDR distribution at the ceiling level inside the vault.

The gamma EDR at the ceiling level was measured inside the vault using an MKS-14ETs gamma dosimeter. The major gamma emitter in Vault 2 is Cs-137.

Vault 3:

The BGI E1M and BGI-75 gamma source blocks account for about 50% of all SRW in the vault. Some 35–40% are concrete blocks in a wooden formwork, i.e. these are just concreted packages and concrete-grouted metal drums. The capacity of the packages is 0.2–0.3 m^3. The number of the packages is not less than 15. The number of the drums is not less than 4. The capacity of the drums is up to 0.1 m^3. Most of the formwork has rotted and fragment when touched. The metal drums are rather strong. The concrete blocks are damaged in part. The remaining 10–15% of the SRW include other objects, namely fire alarms, metal boxes with sources and metal containers of different shape. The total amount of SRW in Vault 3 is not more than 15 m^3.

The RW layout and EDR distribution at the ceiling level inside Vault 3 are shown in Fig. II–7.



(a) (b)

FIG. II–7. (a) Layout of Vault 3 and (b) EDR distribution at the ceiling level inside the vault.

The gamma EDR was measured at the tank ceiling level using an MKS-14ETs gamma dosimeter. The EDR measurement points, illustrated in red in Fig. II–7, also gives a graphic representation of the gamma EDR distribution at the tank ceiling level. The major gamma emitter in Vault 3 is Cs-137.

Vault 4:

Reviews of the available data resulted in the conclusion that RW has never been loaded into Vault 4. The measurements showed that the gamma EDR in the vault did not exceed 0.5 μSv/h and the beta-particle flux from the walls did not exceed the lowest instrument measurement limit, i.e. 20 particles/($cm^2 \cdot$min).

Dose burden during examination

Personal monitoring of the specialists was done using DKG-AT2503 dosimeters that recorded the equivalent gamma dose value at the breast level.

The preliminary examination of the SRW storage vaults involved four specialists. The collective dose for the work period was 140 person-μSv. The maximum personal dose was 58 μSv.

The main examination stage involved five specialists. The collective dose for the work period was 1760 person-μSv. The maximum personal dose was 700 μSv.

The total collective dose for all measurements was 1.9 person-mSv. The maximum personal dose was 0.74 mSv.

Evaluation of air volumetric activity

According to measurements of air samples, the activity concentration of Rn-222 in the air inside the storage vault is up to 22 kBq/m^3.

Summary of RW examination

The following results of examining the RADON-type storage facility have been obtained:

- According to the loading registers, the total SRW activity in the storage tanks is as follows: Cs-137 – 9.6×10^{13} Bq, Sr-90 – 1.7×10^{12} Bq, Ra-226 – 1.3×10^{13} Bq, and Pu-239 – 1.3×10^{12} Bq.
- Most of the tank ceiling outside has a gamma EDR in the range from 0.2 to 1 μSv/h. The EDR over the hatches beneath which there are SRW heaps (hatches 1.1, 2.1, 2.2, 2.3, 3.1 and 3.2) is 2.5–8.6 μSv/h. The exclusion is hatch 2.3 with the EDR over this being 61 μSv/h. The said value is determined not by the concreted object stuck therein (a washing machine drum) but by radiation from the tank. The EDR might grow threefold to fourfold if this object is withdrawn.
- The total amount of the SRW in the tanks is about 185 m^3, of which:
  - 70 m^3 are in tank 1;
  - Some 100 m^3 are in tank 2;
  - Some 15 m^3 are in tank 3 (tank 4 is empty and contains water).
- Plans and 3-D models of the tanks and the SRW therein have been developed. The tanks have the following dimensions: length 14.75 m, width 4.75 m, height 2.95 m.
- The gamma EDR in tank 1 is determined by radiation from metal containers (up to 2 mSv/h on the surface). On the tank ceiling, the maximum EDR value is about 1 mSv/h, with the EDR not exceeding 100 μSv/h on the two thirds of the ceiling area.
- The maximum gamma EDR value in tank 2 is 32 mSv/h on the surface of the SRW heap under hatch 2.1 and 111 mSv/h on the surface of the heap under hatch 2.3. The radiation sources are practically point ones and are covered with other objects. These might be Cs-137 sources that fell out of the gamma source units during dropping.
- The said sources create the maximum EDR of 2.7 mSv/h at the ceiling level and an EDR of over 1000 μSv/h on 60% of the ceiling area.
- At the ceiling level near hatch 3.1 in tank 3, the maximum EDR value is 170 μSv/h. This is determined by radiation from the gamma source unit without a plug on the SRW heap surface. The second such gamma source unit without a plug was also found in the heap under hatch 3.2 but radiation from this is directed towards the tank wall. The EDR

at the outlet of the openings in such units without plugs is 36 mSv/h. It can be suggested that the plugs and the sources could fall out of the gamma source units when these were dropped into the tanks.
— The metal containers in tank 1 have a layer of radioactive corrosion products and deposits. These containers are classified as low level SRW in terms of contamination level.
— Tank 2 contains superficially contaminated objects that have long been beneath the water. The oily deposit layer on the objects is classified as low level waste.
— No such deposit has been found in tanks 3 and 4.
— The thickness of the deposit layer in tank 2 is about 1 cm. No sludge has been found on the tank bottoms but it can be assumed that there is a layer of such deposit on the bottoms of tanks 1 and 2.

II–3.4. Interacting processes

The following processes interact with the development of the Safety Case:

— Involvement of interested parties;
— Independent review;
— The management system utilized to develop the Safety Case.

II–3.4.1. Involvement of interested parties

Waste retrieval activities involve a number of organizations such as the owner of the waste, the consignors, the provider of the storage or disposal facility, the regulatory body.

Relevant interested parties are engaged in the early stages of the development of the Safety Case to allow an understanding of the arguments included in the Safety Case. This includes the regulatory body responsible for nuclear safety, the environmental regulator, and national governmental officials.

In compliance with the Environmental Impact Assessment Provision of the State Committee of Russia for Environmental Protection (order No. 372 dated 16 May 2000) [II–23], the operating organization is responsible for organizing public hearings in order to collect and take into consideration recommendations and suggestions from the public. In general, these public hearings are only conducted in relation to construction of new facilities.

Under normal circumstances, following completion of the Safety Case report, including incorporation of independent review comments, the Safety Case will be submitted to the regulatory body for approval and issue of a license to operate the facility in accordance with the national regulatory requirements. However, since this is an illustrative safety case, this will not be done in this instance.

II–3.4.2. Independent review

Under normal circumstances, the operating organization will ensure that the safety case has been subjected to independent review in line with the requirements of GSG-3 [II–1], through the use of independent personnel to check and verify the assumptions, models and assessment results. The output and response to the independent review would be summarized with a reference or provided in an appendix to the safety case.

Although independent reviews of this illustrative safety case were not performed, the safety case was reviewed at technical meetings and consultancy meetings held in the scope of the CRAFT project. Comments resulting from these reviews were discussed and addressed.

II–3.4.3. Management system

The management system for the programme of waste retrieval activities incorporates the individual management systems of a series of operators carrying out successive steps in the retrieval, transportation, handling, storage and disposal of waste.

In developing the processes for waste retrieval activities, care was taken:

- To ensure the continuity of control of the waste and waste management activities;
- To maintain linkages and relationships between organizations if more than one organization is involved;
- To allow for the potentially long duration of the waste management activities.

Management of retrieval of historical RW requires special attention, specific preparation and appropriate implementation. Initiation of retrieval activities introduces many challenges associated with the selection of appropriate techniques, instrumentation, protective equipment and WAC.

Description of the operating organization's management system

The operating organization of the RADON-type facility performs centralized collection, segregation, transportation, conditioning, and interim storage of low and medium level RW throughout the country.

The management system of the operating organization is based on the principles of ISO 9004:2009 [II–24] and GS-R-3[6]. The management system of the operating organization integrates safety, health, environmental, security, quality and economic elements. Safety is the fundamental principle upon which the management system is based.

The integrated management system of the operating organization defines that regulators and stakeholders play a significant role in defining requirements as inputs. It addresses customer satisfaction by requiring the enhancement of interested party satisfaction, as long as safety is not compromised, in the activities and interactions of the organization.

The management system of the operating organization exists in the form of complete operation documentation, job descriptions, radiation and fire safety instructions, emergency preparedness measures. The documentation is being updated on a regular basis.

The documentation of the operating organization's management system includes:

- The policy statements of the operating organization;
- A description of the management system;
- A description of the structure of the organization;

[6] INTERNATIONAL ATOMIC ENERGY AGENCY, The Management System for Facilities and Activities, IAEA Safety Standards Series No. GS-R-3, IAEA, Vienna (2006). GS-R-3 has been superseded and replaced by the following publication: INTERNATIONAL ATOMIC ENERGY AGENCY, Leadership and Management for Safety, IAEA Safety Standards Series No. GSR Part 2, IAEA, Vienna (2016).

— A description of the functional responsibilities, accountabilities, levels of authority and interactions of those managing, performing and assessing work;
— A description of the processes and supporting information that explain how work is to be prepared, reviewed, carried out, recorded, assessed and improved;
— Detailed work control documents (e.g. instructions, checklists, process control cards and forms).

Organizational structure of the operating organization

The management structure of the operating organization is based on the following principles:

— One person is appointed as the head of the RADON-type facility in which is vested complete responsibility and accountability for the operation and safety of the facility.
— Sufficient numbers of staff are appointed to cover the range of waste retrieval tasks.
— The tasks of safety and quality are independent of operational responsibilities. The person(s) appointed to manage those tasks has (have) a direct reporting line to the head of the facility.
— Persons appointed to key tasks are suitably qualified and experienced. If either qualifications or experience are lacking, then opportunity to remedy the deficiency is made available prior to taking up the post.
— The management structure endures even when the facility is quiescent. The continuation of the management structure, with its defined accountabilities and responsibilities, will demonstrate a prudent approach to the management of radioactive materials.

The staff includes experienced health physicists, drivers of specialized vehicle, and workers of the RADON-type facility.

The activities of the operating organization are authorized by the license of ROSTECHNADZOR, Sanitary, Epidemiological and Environmental Authorities.

Senior managers of the operating organization hold the licenses of ROSTECHNADZOR for the right to conduct activities in the field of RW management, accounting, control and physical protection of radioactive substances and waste.

For the period of waste retrieval operations, a team of eight specialists will be set up. Experienced specialists of other organizations can be invited for the works.

The operating organization has an established procedure for personnel selection, training and work authorization.

Senior management

The operating organization is managed by the Director.

Project manager

The project manager is responsible for providing technical leadership, advice and guidance to customers and the company's business on RW throughout the waste lifetime.

Middle management

Middle management of the operating organization includes:

— Supervisor (operations manager);
— Quality manager.

Operational staff

Operational staff includes:

— Health physicist;
— RW processer (Slinger);
— Hoist operator (Handler);
— RW accounting staff (Check person);
— Decontaminator.

Other staff who are not directly involved in the retrieval activities (and not further considered in this document) include drivers of special vehicles and security guards.

Planning of waste retrieval

Management of waste retrieval projects requires careful planning procedures. The RADON-type facility contains a variety of waste with a wide range of radiological and physical properties. Therefore, more than one retrieval technique is required. In addition, the waste retrieval plan for the RADON-type waste retrieval project recognizes that the initial characteristics of the waste and packages might have changed over time due to a variety of degradation mechanisms, such as corrosion, biodegradation, chemical reactions, and radioactive decay.

The waste retrieval plan for the RADON-type waste retrieval project identifies:

— The overall plan for waste retrieval and management of the RADON-type facility;
— The waste data and characterization required to select and/or support retrieval processes;
— Where retrieval and processing actions fit into the overall remediation sequence;
— The further waste characterization required to define potential downstream processes;
— The final waste product to be produced for interim storage and/or disposal;
— The operators and managers responsible for each set of actions;
— The interfaces with other functional activities;
— The project schedule and budget;
— Cooperation and interface with the regulatory authority;
— The change control process for incorporating and approving changes in the plan that might occur over the project life.

The safety and environmental protection factors considered in the waste retrieval plan include:

— A risk assessment of occupational exposure to ionizing radiation;
— Utilization of adequate and acceptable practical means and available technology to minimize the impact on the environment and protect workers and the general public;
— Utilization of the as low as reasonably achievable (ALARA) concept;
— Minimization of disruption of adjacent areas, including waste storage or disposal areas not subject to the remediation (e.g. the effects of inadvertent removal of shielding from adjacent areas).

These factors are important for the safe execution of the waste retrieval project and might lead to decisions regarding the selection of remotely operated technologies versus hands-on practices.

An important input to the planning process is the information from the initial characterization on dose rates and contamination levels necessary to ensure that the work can be accomplished without undue exposure of the staff and spread of contamination to the environment.

The steps of the waste retrieval process are shown in Fig. II–8.

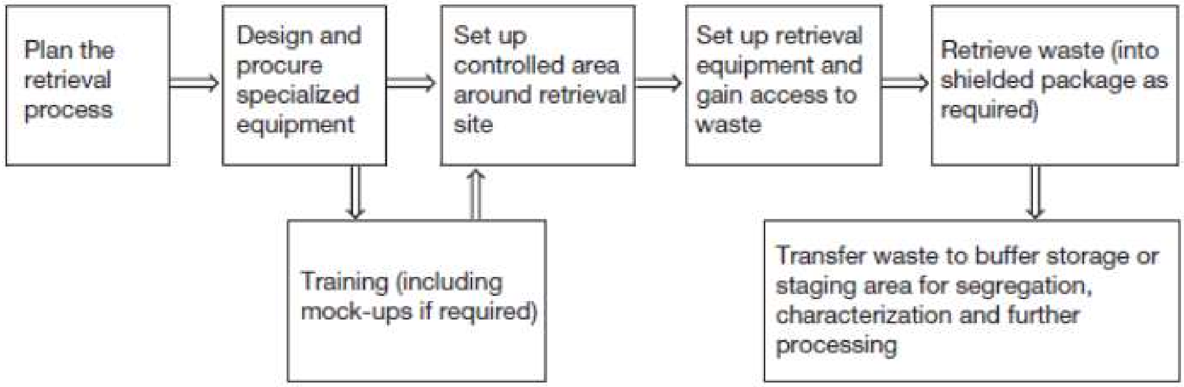


FIG. II–8. Steps of the waste retrieval process.

Distribution of responsibilities

RESPONSIBILITIES OF THE OPERATING ORGANIZATION

The prime responsibility for the safety of waste retrieval from the RADON-type facility rests with the operating organization.

The responsibilities of the operating organization are:

- To ensure that the generation of secondary RW is kept to the minimum practicable;
- To establish and implement a suitable waste retrieval programme with an appropriate management system to ensure compliance with conditions of authorization;
- To ensure that RW is managed by providing appropriate collection, segregation, characterization, classification, packaging, storage and transportation arrangements, including timely transfer between waste management steps;
- To ensure that equipment is available to perform waste retrieval operations safely;
- To ensure that suitable staff are adequately trained and have operational procedures available to perform their duties safely;
- To maintain an awareness of practices in waste management and to ensure the feedback of relevant operating experience;
- To conduct safety assessment(s) of the waste retrieval project;
- To establish and keep records of information on the generation, packaging and storage of RW, including the maintenance of an up to date inventory of RW;
- To ensure the monitoring, recording and reporting to the regulatory body of discharges in sufficient detail and accuracy to demonstrate compliance with any discharge authorization;
- To report promptly to the regulatory body any discharges or releases exceeding the authorized amounts;

— To provide to the regulatory body an inventory of RW held, discharges made and radioactive material removed from the RADON-type facility, at such intervals, in such a form and containing such details as required by the regulatory body;
— To assess the integrity of the waste control measures and facilities to ensure that they are fault tolerant;
— To establish contingency plans and emergency procedures;
— To notify the regulatory body of events and accidents;
— To provide any other information on RW as required by the regulatory body.

The fulfilment of these responsibilities is achieved through the operating organization's management system, together with selection and provision of appropriate equipment for retrieval and storage of RW.

RESPONSIBILITIES OF THE MIDDLE MANAGEMENT

The management system of the operating organization defines clearly the responsibilities at all levels, reflecting the management commitment to security and safety of the facility. Management of the operating organization communicates to individuals at all levels the need to adopt these values and behavioural expectations as well as to comply with the requirements of the management system.

The senior management of the operating organization has developed values and behavioural expectations for the organization to support the implementation of the management system. Quality and safety policies are also developed by the senior management of the operating organization.

The senior management of the operating organization is ultimately responsible for the management system and ensures that it is established, implemented, assessed and continually improved.

The senior management is responsible for the organization and implementation of waste retrieval, its planning, selecting and using the available technologies and resources.

Supervisor

A supervisor with practical experience with the handling of radioactive material and quality control is responsible for supervising day-to-day operations.

The supervisor is responsible for coordination of actions among work participants, decision making based on the results of radiation monitoring, and performing supervisory control over works.

The duties and responsibilities of the supervisor include:

— Receiving, storing and conditioning all RW in accordance with quality arrangements;
— Segregating RW based on its characteristics;
— Maintenance of records of receipt, storage and conditioning of RW for the appropriate period of time;
— Management of the operational staff.

Quality manager

The quality manager is an experienced staff trained to university degree level in radiochemistry or radiation physics, with experience in RW management. The duties and responsibilities of the quality manager include:

- Implementation, management and maintenance of the defined quality management system;
- Audit, internal and external, of the workings of the quality management system;
- Ensuring that non-compliances and corrective actions are followed-up promptly and to their proper and logical conclusion.

RESPONSIBILITIES OF THE OPERATIONAL STAFF

The list of operations assigned to every job position is presented in Table II–6.

TABLE II–6. RESPONSIBILITIES OF THE OPERATIONAL STAFF

No.	Job position	Works/operations to be performed
1	Health physicist	The health physicist is experienced in radiological protection procedures and regulations. The duties and responsibilities of the radiation protection supervisor include: — Establishment of the necessary monitoring regime; — Receiving and assessing the results from the dosimetry service; — Maintenance of the dosimetry records for the appropriate period of time; — Taking necessary action on the basis of the radiation records and dosimetry. Radiation monitoring and radiation survey of as follows: — Vaults with RW and neighboring space; — Work areas; — RW packages; — Transport containers in the course of and upon their charging with SRW; — Supervision over works performed by the RW processer.
2	RW processer (Slinger)	Slinging of gamma-ray source blocks, small packages and SRW containers inside the vault. Applying of slings and handling group to lift the damaged RW packages. Plugging of the open collimator of the gamma-ray source block. Fixing of the gamma-ray source block on the transport platform. Collection of spillages into polyethylene sacks, loading them onto the pallet. Loading of SRW packages into the transport container. Loading of transport containers onto a special vehicle.
3	Hoist operator (Handler)	Parking in the work areas of as follows: — Pallet for lifting of cargoes; — Transport platform for gamma-ray source blocks; — Decontamination pallet to perform EDR measurements for the items; — Transport containers. Placement of items on the transport platform. Lifting and transportation of SRW items from the vault.
4	RW accounting staff	Accounting and comparison with the recorded data. Installation of tamper indicating devices.
5	Decontaminator	Decontamination of contaminated surfaces

Management of resources

PROVISION OF RESOURCES

Senior management determines the amount of resources necessary and provides the resources to conduct the activities of the operating organization and to establish, implement, assess and continually improve the management system.

The information and knowledge of the operating organization is also managed as a resource.

HUMAN RESOURCES

The senior management of the operating organization determines the competence requirements for individuals at all levels and provides training or take other actions to achieve the required level of competence.

The senior management ensures that individuals are competent to perform their assigned work and that they understand the consequences for safety of their activities.

The staff has received appropriate education and training, and has acquired suitable skills, knowledge and experience to ensure their competence. The staff is aware of the relevance and importance of their activities and of how their activities contribute to safety in the achievement of the operating organization's objectives.

Organization of personnel selection and training

The internal documents of the operating organization establish that only persons aged 18 and older with the required skills, subjected to preliminary medical examination, introductory and primary trainings on the site, and trained for safe methods of operation and having the appropriate certificates, are admitted to operations with ionizing radiation sources.

In addition to the introductory and primary training, a periodical instruction is provided twice a year. When preparing for non-typical radiation hazardous works, unscheduled training is provided. Results of the trainings are documented in the log-books.

The programme for personnel training related to the radiation safety is enforced. The programme includes such sections as the ionizing radiation sources; measurement units; biological radiation effect; radiation source work (practical part); regulatory and technical; documentation. The training course is designed for 40 hours.

Examination of knowledge is carried out annually by a committee appointed by the order of the operating organization. Results of the examination are recorded in the protocols. In addition to the radiation safety issues, the committee examines whether the personnel is familiar with the rules of transportation of RW, fire safety regulations, and the rules of handling of the portable electrical equipment.

The examination of personnel knowledge is performed in the presence of a representative of the inspection department of radiation hazardous facilities.

Developing and implementation of processes

To function effectively, an organization has to manage numerous linked activities. An activity or set of activities using resources, and managed in order to enable the transformation of inputs into outputs, can be considered as a process.

Operational processes of the operating organization with a focus on waste retrieval from RADON-type facility include:

- Receipt of vehicles carrying empty waste packages;
- Off-loading of empty waste packages from the transport vehicle;
- Transfer of the waste packages into the facility;
- Unloading of large-sized RW packages;
- Unloading of gamma-ray source blocks;

— Accomplishment of unloading of large-sized RW packages released from under the debris;
— Collection and packaging of spillages;
— Radiation and contamination monitoring of the wastes;
— Loading of waste packages with retrieved waste;
— Acceptance and placing of the waste packages onto a special vehicle;
— Maintenance of storage records;
— Periodic inspection and radiological monitoring of the storage building and the waste packages;
— Maintenance of the RADON-type facility and all associated equipment;
— Ensuring physical protection of the RADON-type facility;
— Radiation safety of workers, e.g. possible monitoring of operating and maintenance staff on exit from the RADON-type facility.

The development of each process ensures that the following are achieved:

— Process requirements, such as applicable regulatory, statutory, legal, safety, health, environmental, security, quality and economic requirements, are specified and addressed.
— Hazards and risks are identified, together with any necessary mitigatory actions.
— Interactions with interfacing processes are identified.
— Process inputs are identified.
— The process flow is described.
— Process outputs (products) are identified.
— Process measurement criteria are established.

The following generic processes are developed in the management system of the operating organization:

— Control of documents;
— Control of products;
— Control of records;
— Purchasing;
— Communication.

Quality assurance programme

The quality assurance programme for waste retrieval from the historical RADON-type facility was developed by the operating organization in accordance with the regulatory requirements for quality assurance programmes NP-090-11 [II–25]. The programme establishes a procedure for sharing the responsibilities between top managers and chief specialists of the operating organization.

The programme is structured to cover the following elements: metrological support, supply management, equipment control, document control, control of non-conformances, personnel training, corrective measures, and inspection. These elements characterize the methods of fulfilment of respective activities and emphasize the most important aspects of individual procedures.

Periodic inspection and testing is defined for main and auxiliary equipment. Inspection results are registered. Manufacturing documents (certificates, passports, forms) are made available for all instruments and equipment. External organizations performing work are required to be licensed for the respective activities.

Newly arriving equipment is registered through relevant documentation. The quality assurance programme also includes periodic calibration of instruments. In the case that a calibration error is found, the results of respective measurements are cancelled and a decision is taken on further actions with regard to restoring the calibration or performing repairs or exceptional verification of the instruments.

II–4. SAFETY ASSESSMENT

II–4.1. Assessment context

The context for the assessment involves the following key aspects: the purpose of the assessment, the philosophy underlying the assessment, the regulatory framework, and the end points and time frame for the assessment.

The assessment is carried out to demonstrate safety of retrieval operations at the historical RADON-type facility. It provides assessments of radiation doses to workers and members of the public during normal operation of the facility and during accidents that could occur over the assumed lifetime of the facility, for comparison with dose limits and constraints. For the purposes of this Safety Case, the total duration of activities to empty Vault 1 of the historical facility is 100 working days.

An assessment is performed of the impact of potential accidental events on the facility with a view to demonstrate that the design and safety features are sufficiently robust to withstand such events. The assessment seeks to identify uncertainties and provide some consideration to their importance and possible approaches to the management of those uncertainties considered to be important for safety. A generally conservative approach is taken with respect to assumptions and the assessment.

It is necessary to reiterate that the safety assessment provided in this Section is not a fully comprehensive and complete assessment. This Safety Case is an illustrative example of how the methodology presented in GSG-3 [II−1] can be applied to waste retrieval from a historical RADON-type facility.

A detailed assessment of the dose arising from normal operations of waste retrieval from Vault 1 of the facility is modeled using the SAFRAN tool (version 2.3.2.7) [II−3]. Selected accident scenarios are modeled to demonstrate the application of the SADRWMS methodology [II−2] and the SAFRAN tool [II−3].

II–4.1.1. *End points for the assessment*

The following end points for quantitative assessment will be considered:

- Radiation dose to workers performing the various normal RW management activities at the historical RADON-type facility;
- Radiation doses to workers and the public due to anticipated operational occurrences.

Doses are evaluated against the safety criteria through use of different models and the SAFRAN tool.

Doses will be evaluated against the safety criteria and will also be compared with annual dose limits for occupationally exposed persons recommended by national and international standards. The assessments will cover activities taking place over a period of less than one year, currently estimated to be 100 working days.

In accordance with the requirements of NRB-99/2009 [II–16], the annual dose limit for radiation workers is established to be 20 mSv/a and for the general public to be 1 mSv/a.

In order to maintain individual dose rates at ALARA level taking account of economic and social factors, control levels of radiation exposure are established by the operating organization for the waste retrieval from the historical RADON-type facility. Accordingly, a dose constraint of 10 mSv/a is established for workers, which corresponds to half of the dose limit. The public dose constraint is set at 0.1 mSv/a, in accordance with [II–17].

II–4.1.2. *Approaches to the assessment*

For normal operation, quantitative deterministic assessments of worker doses due to the range of activities by various occupational groups have been performed.

A qualitative assessment will be performed on the implemented waste management practices. In the approach to waste management, the following elements will be considered as contributing to safety:

- Clearly defined responsibilities for waste management;
- Implementation of the principles of waste minimization and avoidance, namely, re-use or re-processing of waste, return to supplier, safe and secure storage and conditioning and final disposal of waste;
- Hazards of the generation of secondary waste associated with all waste management operations (routine and ad hoc) are known, monitored, projected and managed by due management processes;
- Interdependencies between the various steps of waste management are known and managed.

A qualitative assessment of the availability, level of implementation of an integrated management system to ensure a sustained level of safety during the operational phase of the facility will be performed. This assessment will focus on radiation protection, work procedures, quality assurance aspects (mainly record keeping and change management) and processes for the management of limits and conditions.

Uncertainties inherent to the assumptions made in the quantitative assessments or any other uncertainties identified during the safety assessment will be evaluated to determine their impact on safety. Uncertainties with a significant impact on safety will be listed with recommendations for their management.

II–4.2. Description of the predisposal waste management facility and the waste

II–4.2.1. Description of the waste

A 3-dimensional illustration of the SRW storage facility model is presented in Fig. II–9.

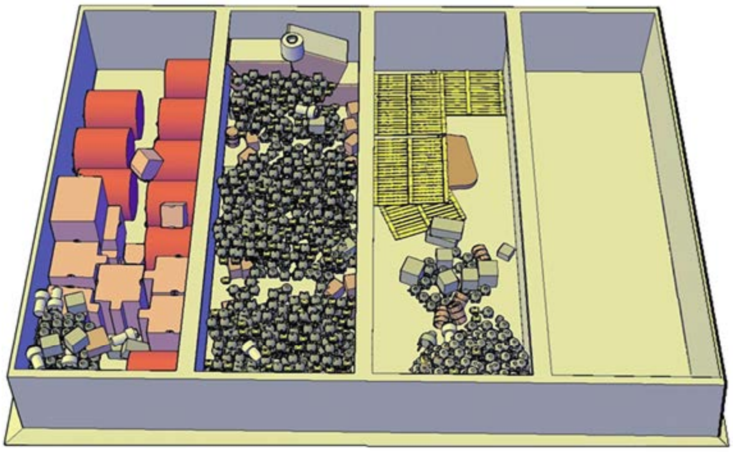


FIG. II–9. Illustration of the contents of the four SRW storage vaults.

The following waste types were selected to create a model for safety assessment of operations related to RW retrieval from storage:

1. Blocks of gamma-ray sources of E-1M type (Fig. II–5);
2. Large-sized metal containers (Fig. II–10);
3. Spillages.

FIG. II–10. Large-sized metal containers.

II–4.2.2. Description of the predisposal waste management facility

Hangar building

The steel framework, cladding and roof have been designed to support all superimposed structural loads including all applicable live and dead loads. The equipment, materials and items were selected by the designer with account taken of the climatic conditions in the area of construction and environmental factors. Specific external effects of human-induced and natural origin were not considered.

Civil structures of the storage are designed for standard loads and stresses and with no account taken of the loads from specific external effects (e.g. accidental aircraft crash, seismicity at the level of the maximum design basis earthquake, explosion, tornado).

The building is rectangular in plan with axial dimensions 24 m × 30 m (Figs II–11 to II–14). The building has rolled-strip roofing with downpipes as a water drain. The roofing material includes (from bottom to top): vapor seal (IzoPlastHPP); lightning protection grid; heat insulation (Izover, Finland); sandy concrete cover for making inclinations; insulation material (two layers, IzoPlastHPP and IzoPlast EKP).

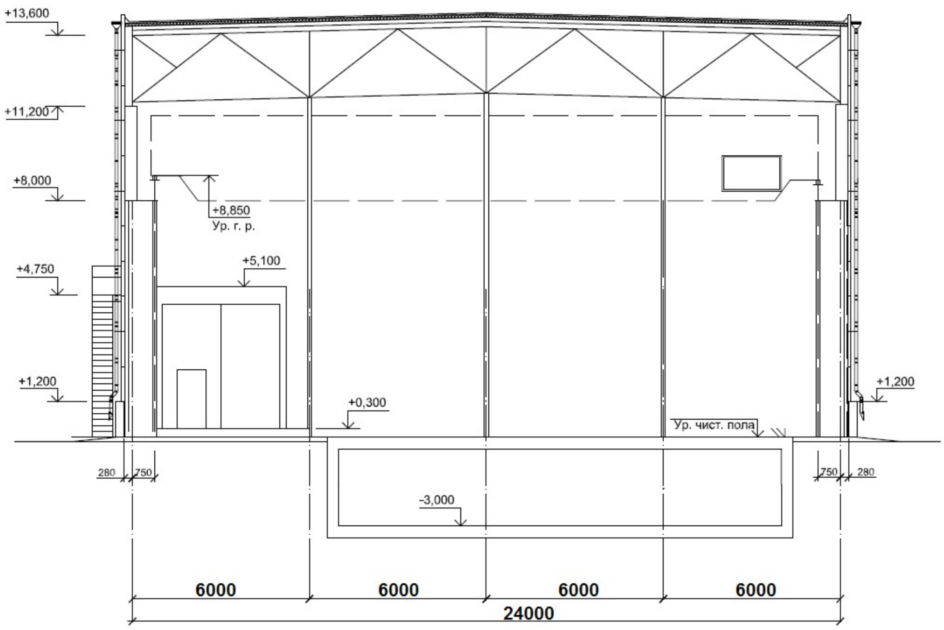


FIG. II–11. Schematic representation of the building (front view).

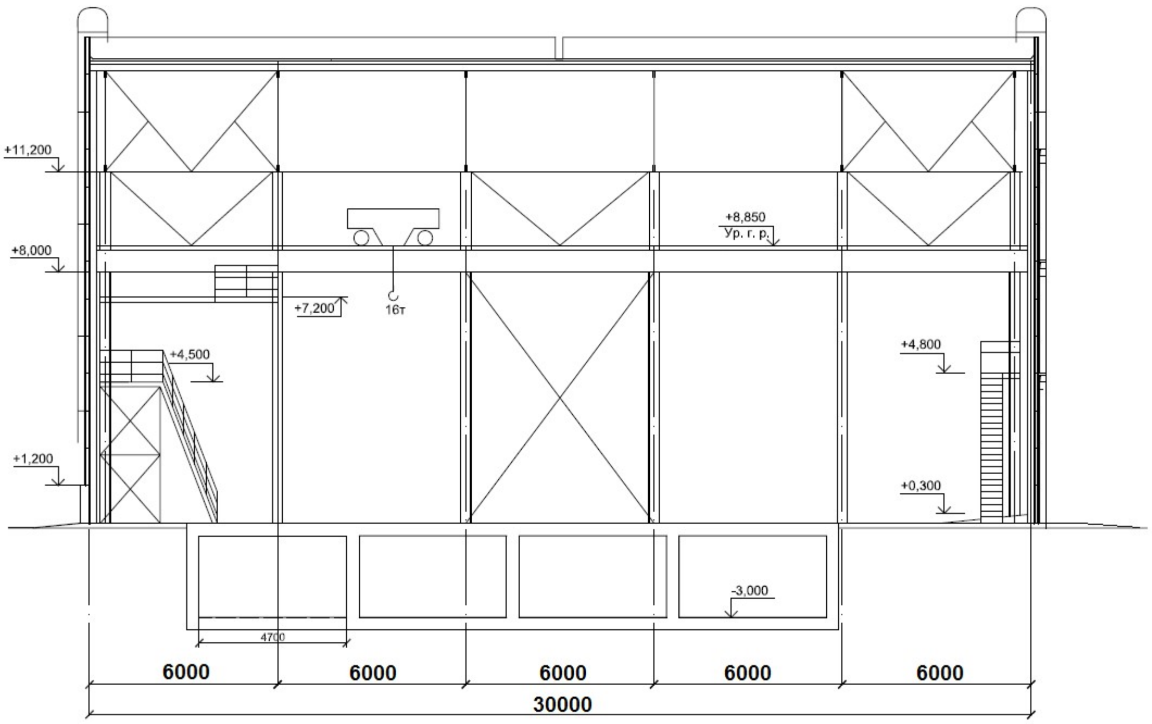


FIG. II‒12. Schematic representation of the building (side view).

FIG. II‒13. Schematic representation of the vaults (plan view).

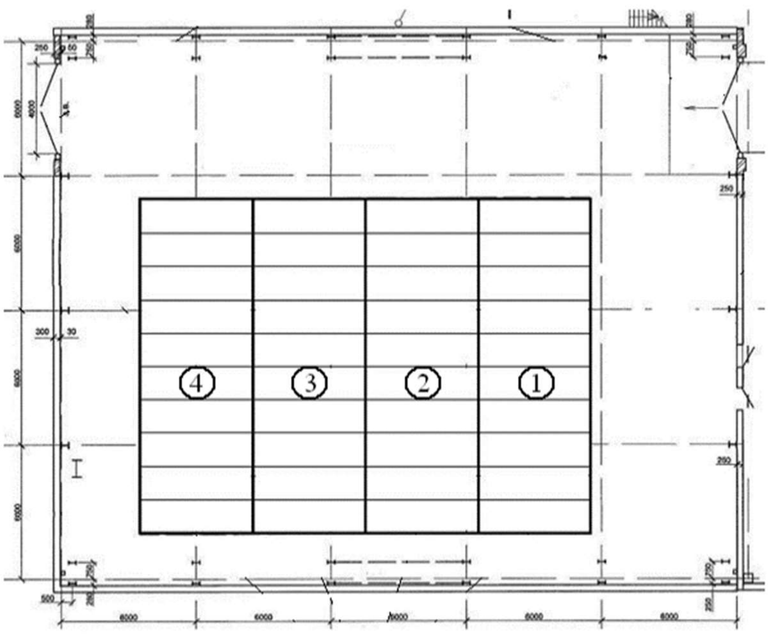


FIG. II–14. View of the SRW storage hangar.

The load-bearing frame is all-metal using standard steel structures. It is provided with an electric supported overrunning crane with a lifting capacity of 16 tons and a span of 22.5 m. Columns of the building frame are series 1.424-4, issue 1, ceiling trusses (L=24 m) are provided with parallel chords in accordance with series 1.460.2-10/88.1. Crane girders (L=6 m) are series 1.426.2-7, issue 3.

The floor slab is able to support the concentrated point loads of the waste containers and live loads of vehicles/equipment used to load the packages.

The RWSF building site is located on top of the hill. The “zero” elevation of the building (top of the old storage facility) is 1.5 m higher than the elevations of the adjacent territory and corresponds to the absolute elevation of 229.0 m in accordance with the Baltic Height System. Surface water discharge is designed with an outlet to a water course to the south of the building. A concrete flume with a section of 0.4 m × 0.4 m is designed along the wall of the building, to be combined with a blind area. The existing circular drainage with a leak control well will be kept after the reconstruction, since it fits within the limits of the RWSF building.

The flooring in the assembling hall, which will be arranged upon completion of the RW retrieval operations, is inclined towards the channel with sumps located along the perimeter of the building, in order to collect condensate from the walls and floor of the room, and also various kinds of leaks. Water is removed to the liquid RW reservoirs or special sewerage tanks. The concrete floor in the assembling hall has a damp proof membrane installed.

Shielding

The facility building structure does not provide any significant shielding from radiation as it is a steel-framed structure with a profiled cladding system. There is therefore no safety function in the building walls that will limit exposures outside the building. Instead, localized shielding will be used within the facility when waste retrieval operations are in progress to provide sufficient dose attenuation factors that achieve dose rate objectives.

Figure II–15 identifies the following areas that are established within the facility during waste retrieval operations, noting that these are dynamic areas that will change as waste retrieval operations progressing from Vault 4 through to Vault 1. The waste retrieval process starts from Vault 3, and areas C1 and C2 will always be above empty vaults.

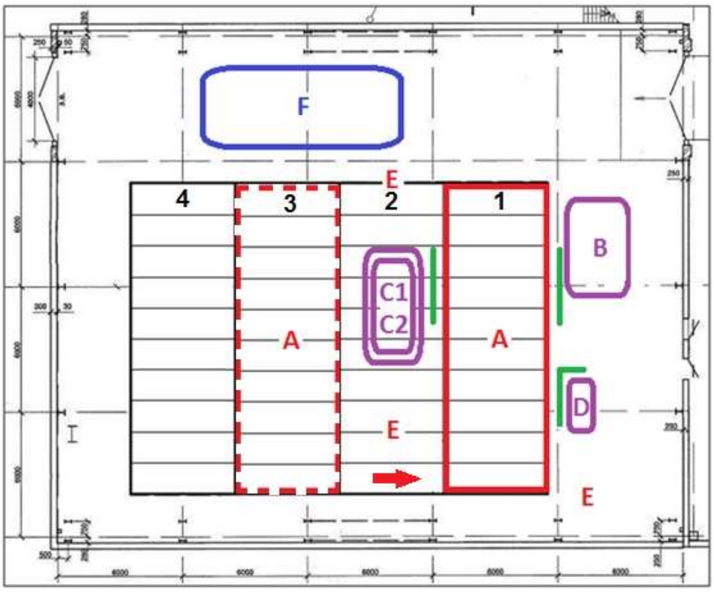


FIG. II‒15. Location of work areas during RW retrieval from Vault 1.

A description of work performed in different areas is presented in Table II‒7.

TABLE II‒7. DESCRIPTION OF WORKS PERFORMED IN DIFFERENT AREAS

Area	Description of work
A	RW storage vault
B	Area for performance of radiometric measurements, decontamination of large-sized containers with RW and their loading into the transport container
C	Area for conduct of radiation monitoring of gamma-ray source blocks and other small-sized wastes, as well as spillages, for their decontamination, placement of gamma-ray source blocks onto transport crates and loading into the transport container
D	Radiation protection area
E	Area for non-permanent attendance beyond the limits of the assigned work areas
F	Area for loading of filled containers onto the transport vehicle

Note: Two symbols ‒ C1 and C2 ‒ are used for Area C due to differences in conditions of personnel dose rate calculation at performance of different operations in this area.

Area E is the area for short -term attendance of RW processer (slinger) and health physicist, and for primary attendance of supervisor and hoist operator. Considering that the amount of waste in the area will be reduced over the course of retrieval activities, and taking into account the complexity of movements in this area and variability in the EDR field, the EDR is assumed to be the average EDR value along the long side of the vault neighboring Area B (where operations are performed during this phase) over the entire period during which operations are performed in Phase I.

For other phases, the approximate EDR for Area E is assumed to be half the minimum EDR value of the vault. This value is based on the EDR value calculated for the upper slab level resulting from spillage on the bottom of the vault.

The EDR associated with Area F depends on the work phase. In order to determine the EDR in Area F in Phases I and III, it is taken into account that three large-sized RW packages can be located in one UKTN-24000 transport container. The transport container slinging is carried out by means of eye-rings located in the upper part. It conservatively assumed that slinging of the container is performed by means of the slinging platform and requires participation of the RW processer only in certain cases, i.e. when an abnormal option of slinging occurs. The probability of an abnormal option is associated with the expected EDR, based on the radiation characteristics of packages that might be located in the transport container. Values of external radiation dose rates in work areas are presented in Table II–8.

TABLE II–8. VALUES OF EXTERNAL RADIATION DOSE RATES IN WORK AREAS

Area	External radiation dose rates in work areas, μSv/h			
	Phase I	Phase II	Phase III	Phase IV
A	Table II–12	50	70	14
B	Table II–12	–	70	–
C1/C2	–	0.073/0.32	–	14
D	1.8	1.8	1.8	1.8
E	~60	7	7	7
F	~60	7	7	7

Access and physical security

A security service is provided with responsibilities for security and protection of the RWSF. Compensatory measures are in place against failures of any component of the engineered facilities providing physical protection. Approved procedures are in place for arrangement and maintenance of the RWSF physical protection system. Contracts are in place with special organizations that provide armed guards.

Measures are in place to ensure compliance with requirements for the RWSF physical protection system. There is an authorization-based system in place for access of staff (personnel), business travelers, visitors and vehicles to the secured rooms, buildings and territory, where work with radioactive sources, substances and waste takes place.

Activities of the security service are aimed at the following:

- Protection of the RWSF against unauthorized actions;
- Access control with regard to radioactive sources and substances within the RWSF;
- Operation of engineered facilities of the physical protection system of the RWSF;
- Timely detection of unauthorized actions and proper response;
- Suppression of unauthorized actions.

Fire protection

Fire protection measures are based on the following design concepts.

Classification of fire safety and classification of explosive and fire-hazardous premises of the storage, based on [II–26], are as follows:

— Fire resistance rating of the building – II (the second of five, where the fifth is the least fire-resistant buildings); this means that the fire resistance limit of load-bearing structural elements is 90 min, external non-load-bearing walls – 15 min, intermediate floors – 45 min, other signs of ultimate states.
— Class of design fire safety – C0 (less hazardous, in total four classes – C0 to C3); this means that the fire safety class of all structural units is K0 (fire-proof); structural units include: columns, girders, trusses, exterior walls, interior walls, partitions, walls of the stairwells and fire barriers; others.
— Functional fire hazard – Class F5.1 (by method of use – "production buildings and structures, laboratory rooms, workshops"; depending on the class, the requirements to arrangement of fire rescue paths, stairs, and stairwells are regulated).

All the structures of the storage facility are made of non-combustible materials, and wastes are stored in non-combustible reinforced-concrete containers.

The fire-fighting measures include:

- Access to fire-fighting vehicles is provided from three sides of the RWSF.
- The access way, via the RWSF transport access, is finished with a turning circle with sizes of 12 m × 15 m for turning of vehicles.
- Water will be taken from the pond that is located at 70 m distance from the building. For these purposes, a retaining wall will be made of concrete building blocks at 2.0 m height, and a parking lot of 12 m × 12 m for two fire-fighting vehicles. Water consumption for external firefighting is assumed in accordance with the regulatory requirements based on a fire area of at least 150 ha and equal to 15 l/s.
- For entrance of fire-fighting vehicles, provisions are made for the gates in the existing chain link fence, and the access way of 4 m width with pitch-grouted macadam.

Ventilation

The hangar building is provided with a ventilation system that was installed to provide nominal air changes.

A specific local extract ventilation system will be installed via one of the access hatches in the vault where waste retrieval operations are being undertaken. A containment tent, measuring approximately 2.8 m × 1.1 m × 2.3 m, will be installed around the hatch. The extract rate will be 12 860 m^3/h.

The ventilation system is designed for removal and coarse purification of the air from RW storage compartments. The system is connected to the general building ventilation system by means of a flexible air pipe with a diameter of 900 mm.

The air removed by local exhaust from the compartments of the storage goes through the following stages of filtration:

- Prefilter (efficiency 75%);
- High efficiency aerosol filter FU-350/F-23 (efficiency 90%).

The rate of air extracted from the working compartment (i.e. the compartment from which the RW are being removed) is calculated assuming that the area of the opening is 6 m^2, and that the air speed in the opening is 0.3 m/s.

Electrical power and lighting

Electrical power is provided for lighting, use of small power tools and detection/warning equipment. All installations and equipment are of high quality and comply with national standards. Good levels of lighting are provided throughout the facility and additional local lighting will be installed as required to support waste retrieval operations.

II–4.2.3. Operational safety measures

Operational radiation protection

The facility is designated as a radiologically controlled area and people working in the facility are designated as occupationally exposed persons with the necessary training, dosimetry and medical control.

The radiation protection programme is implemented and covers routine monitoring of the facility and its environment, monitoring of specific operations such as waste retrieval activities and any special monitoring that might be required from time to time. The programme makes provision to monitor external radiation levels and surface contamination.

In areas where the dose rate might vary during the process of retrieval, the health physicist performs continuous monitoring of dose rates as well as performing regular surveys. All persons working in such environments are required to carry electronic dosimeters that alarm at pre-set levels. This method provides an immediate warning to the individual if he or she enters a high dose rate area or if the work activity results in a sudden increase in the dose rate.

Contamination control

Radiation protection controls at and adjacent to the retrieval area includes personnel contamination monitors, portable radiation instruments and personal dosimetry and appropriate personnel protective gear. If there is a potential for internal contamination, whole body counting or bioassay might be appropriate.

Any equipment or waste package will be removed from the control area only after checking the surface for loose radioactive contamination by swipe tests. A lock system and administrative procedures are established to ensure positive control over material and equipment movement.

In higher dose rate situations, such as with intermediate level waste, the use of supplemental portable shielding and remote handling techniques is required; worker scheduling and rotation has also to be considered.

Radiation monitoring prior to transportation

Gamma radiation dose rate measurement is made during acceptance of radiation cargo from the external surface of each radiation package at a distance of one meter.

During the loading of radiation packages into the special vehicle, the radiation supervisor checks that the dose rates at any point of the external vehicle body surface at a distance of 1 m and in the driver's cabin do not exceed the acceptable levels (0.1 mSv/h and 0.012 mGy/h, respectively). Packages with categories II and III waste and spent ionizing radiation sources are loaded in the last turn into the back part of the vehicle body.

In order to ensure that packages and containers are reliably fixed in the vehicle body, the vehicle body is equipped with the tools for fastening radiation packages, as well as preventing the RW packages and containers from tilting along its transportation route.

Prior to RW shipping, the representative of the specialized plant has to make sure that the RW package is reliable in order to prevent contamination of external environment with radioactive substances.

II–4.2.4. Passive safety and defence in depth

Passive systems contributing to the safety of the facility and their operations are applied in three areas:

- The optimization of external exposure to workers and the public;
- The limitation of facility impact during accidents leading to radioactive contamination;
- The limitation of public exposure due to non-authorized access to the facility.

The optimization of external exposure during the waste retrieval operations is based upon:

- Use of localized shielding to minimize worker dose based on real-time radiation surveys in addition to preliminary calculations and assessments.
- Dose limitation is achieved by adherence to the rules established by radiation safety instructions, radiation control of practices in accordance radiation safety instructions and techniques, using a wide range of engineering and organizational measures.
- Removal of high dose waste components early to reduce background dose rates.
- Zoning of the areas:
  - Area of possible contamination (RWSF inside the fence);
  - Controlled area (500 m radius with the centre in the storage location);
  - Monitored area (within 1 km radius).
- Mechanization of work where necessary; remote control of mechanisms.
- Using basic personal protection equipment in accordance with the class of work to be carried out, and additional personal protection equipment depending on the kind and conditions of work.
- Training and testing personnel knowledge of radiation safety regulations and standards.
- The principle of optimization (ALARA) is implemented by limiting the number of exposed persons, rational organization of the industrial process, keeping individual doses to personnel as low as reasonably achievable. The documents of the operating organization establish control levels of radiation parameters for all categories of exposed persons. The values of control levels are established lower than the values of principal dose limits and derived controlled parameters. The outcomes of commissioning operations might require revision of the internal radiation safety documents of the operating organization, and adjustment of the controlled levels, if necessary.

Defence in depth principles were applied primarily to plan waste management operations as follows.

— Radiation protection of personnel is ensured by using room zoning, ventilation systems, radiation protection structures, and continuous radiation control, as well as by using necessary administrative measures.
— Shielding structures to be used include:
  ▪ Radiation protection wall of lead bricks; standard lead bricks will be used; thickness – 50 mm, height – 2 m.
  ▪ Vertical radiation-protection steel structure; represents a steel plate with eyes for slings; the size is 3 m × 5 m, thickness – 50 mm, weight – 6 tons.
  ▪ Horizontal radiation-protection steel structure is used for covering the working opening of the storage compartment for 2/5 or 4/5 of its area in the process of RW retrieval, and also for complete covering of the opening at the end of the shift; represents a 3 m × 6 m steel plate with eyes for slings; thickness – 50 mm, weight – 7.2 tons.
  ▪ Mobile radiation-protection structures; the material of these screens consists of a bismuth-based metal polymer; the size is 3 m × 1.7 m, thickness – 40 mm, weight – 16 tons; a 10-fold reduction of Co-60 radiation will need a 19-mm layer of BIECOM material; these items are used for shielding container NZK in the process of its filling with the wastes retrieved from the old storage.
— Use of active ventilation systems and containment tents to reduce the impact of airborne activity on both workers and the public. This includes a localized ventilation extract as well as the building ventilation system.
— Use of personal protective equipment to prevent worker internal doses, such as respirators, powered air-fed respirators and/or air-fed pressurized suits as required by the conditions.
— Use of area gamma monitors to warn of abnormal dose rates during waste retrieval operations in addition to localized health physics surveys.
— Use of personnel dosimetry (thermoluminescent dosimeters).

II–4.2.5. Engineering systems ensuring safety

Taking into consideration Russian legislation and also para. 4.53 of GSG-3, the safety functions and associated SSCs were identified for the historical RADON-type facility. All the SSCs were classified in four different classes, taking into account Russian regulations and their importance for the safety of the facility.

According to national federal norms and rules NP-016-05 [II–27], systems and components are, with respect to their safety importance, divided into:

— Systems and components important to safety;
— Other systems and components not related to safety.

Systems and components are divided by their functions into:

— Systems and components of normal operation (N);
— Safety systems and components.

Safety systems and components are divided into four safety classes:

— Protection systems (P);
— Localizing systems (L);
— Support systems (S);
— Control systems (C).

These four safety classes are given a ranking that reflects the safety significance of their components.

Safety Class 1 includes the components whose failures can become initiating events of beyond design basis accidents leading to exposure of workers and/or the public, and the release of radioactive substances to the environment which exceeds the limits established for design basis accidents.

Safety Class 2 includes components whose failures can initiate events leading to design basis accidents.

Safety Class 3 includes:

— Components of systems important to safety not attributed to Safety Classes 1 and 2;
— Components that contain radioactive and/or toxic substances, which ingress into the premises and/or the environment in the event of failures of such components might lead to an excess of levels established in accordance with the regulatory documents;
— Components that perform monitoring functions of radiation protection of the employees (personnel) and population.

Safety Class 4 includes:

— Components of normal operation which do not affect safety and are not attributed to Safety Classes 1, 2 or 3;
— Components used for accident management, which are not included in Safety Classes 1, 2 or 3.

In Table II–9, all SSCs are listed with their safety classification and function(s) they are performing.

TABLE II–9. SAFETY CLASSIFICATION SAFETY FUNCTION(S) OF SSCs

SSC	Safety function	Safety class of components
Hangar structures	N	2
System of physical barriers	P	3
RW packages	N	2
Radiation control system	N	3
Transportation system inside the RWSF	S	2
Industrial television	N	3
Automatic fire alarm system	N	3
Communication system	N	3
Ventilation system	N	3
Electricity supply system	N	3

Engineering aspects that ensure safety during normal operation are:

— The engineering characteristics of the building. Information expressed in the building design document show the engineering features. Its design ensures structural stability under extreme environmental conditions.
— The lighting system is adequate and permits the performance of operations in a safe manner.
— Each delineated area has sufficient physical space to ensure a minimal probability of accident occurrence during waste management operations.
— Local ventilation systems minimize the spread of any contamination and its impact on workers, the public and the environment.

For anticipated operational occurrences and accident conditions, the engineering systems ensuring safety are:

— Floor and wall finishes allow easy decontamination;
— The facility has its own fire response equipment;
— Local ventilation systems minimize the spread of any contamination and its impact on workers, the public and the environment.

Basic and robust engineering systems have been selected to provide high reliability. There are no complex control systems and interacting engineering processes.

II–4.3. Description of the waste retrieval activities

II–4.3.1. Methods used for waste retrieval

For retrieval of unconditioned low level waste, simple industrial equipment might be used, such as backhoes, remotely controlled clamshell diggers, forklift trucks, small mobile cranes and similar equipment. For retrieval of intermediate level waste, more sophisticated, remotely operated or shielded equipment might be required. This includes robotic arms, shielded transfer casks, long reach cranes, remote grappling devices and similar equipment.

Standard industrial equipment often can be used, but sometimes custom designed devices are needed for a specific job; for example, large volumes of soil, sand and gravel that are sometimes used for backfilling of waste repositories can be removed using conventional digging equipment, or (if it is loosely packed) with vacuum equipment. All removed soil, sand and gravel might be contaminated and, thus, needs to be monitored. In cases in which it can be shown to be only very slightly contaminated, much of this material can be cleared for conditional or free release from regulatory control, depending on the national regulations. Alternatively, the material could be reused for backfill or in the construction of other waste disposal facilities.

Table II–10 identifies the equipment that would be employed to retrieve the various wastes.

TABLE II−10. EQUIPMENT EMPLOYED DEPENDING ON WASTE CATEGORY

Waste category	Equipment employed	Comments
Loose low level waste, low dose rate	Manual removal, clamshell bucket, small crane	Some initial characterization and segregation might be performed at the retrieval site (e.g. have several receiving containers, properly identified by colour coding or numbering, available to sort the waste at source); waste is typically placed in a container suitable for transfer to a buffer storage or staging area for further segregation, characterization, treatment, etc.
Waste in intact containers	Crane, forklift truck	Depending on the condition of the original container, it might be placed into a secondary container or overpack for transfer to a buffer storage or staging area.
Higher dose rate waste	Remotely operated crane (required capacity depends on the size and weight of the shielded package), custom designed robotics, remote grapple, shielded casks	Waste is usually retrieved remotely and placed immediately into a shielded container or cask for transfer to a buffer storage or staging area; retrieval of higher dose rate waste typically requires a high degree of planning in order to avoid radiation exposure of the workers.
Waste that was previously subject to in-situ conditioning	Cutting equipment such as diamond saws or jackhammers to remove the waste from the conditioning matrix or to cut the monolith into pieces that can be handled, crane	Approach and equipment selected to minimize the risk of cutting through waste objects such as spent sealed radiation sources or through containers of unconditioned, mobile waste such as ion exchange resins and sludge; depending on the dose rate, remote operated equipment might be required; if concrete was used as the conditioning matrix, the dose rate might rapidly increase as the concrete (shielding) is removed; a high level of loose or airborne contamination might result from breaking up the matrix or by cutting through a discrete waste item during matrix cutting and removal.

II–4.3.2. Preliminary work – Hangar building

Preliminary work is the first stage of waste retrieval. A building structure (hangar) is constructed above the existing SRW storage to prevent dispersion of radionuclides by wind during this stage. A sanitary inspection room is provided for the personnel who are engaged in waste retrieval works (Fig. II−16).

FIG. II–16. Sanitary inspection room.

In accordance with Russian safety rules NRB-99/2009 [II–16], the effective dose limit for personnel is set to 20 mSv/a. Personnel working time is 1700 hours a year. The permissible dose rate on the site under these conditions is 12 µSv/h, which is considerably lower above the vaults. The staff who are engaged in construction of the hangar can work without limitation of time (36 hours a week).

The hangar accommodates all equipment and systems, which are necessary for retrieval operations. It also accommodates ventilation and water supply systems.

Atmospheric emission of air from rooms of the second zone and the first zones is carried out via gas-cleaning systems to prevent contamination of the environment.

II–4.3.3. Work procedure for waste retrieval from the vault

Overview of work performance

The process of RW retrieval from the storage vault consists of the following sequence of operations at the storage site:

- — Placement of the package containing RW (hereinafter also referred to as the item) on the load-lifting mechanism;
- — Lifting and withdrawal of the package from the vault;
- — Characterization of the package;
- — In some cases, placement of packages onto the transport crate;
- — Loading of the RW package or the transport crate into the transport container;
- — Placement of the packed transport container onto the vehicle.

Wastes are transported to the centralized waste management facility in accordance with transport requirements and are packaged in accordance with the WAC of the receiving facility.

SRW is received and transported in radiation packages of different transport categories (I, II and III) in an amount not exceeding 10 packages in one vehicle. The equivalent radiation dose rate is not allowed to exceed 2 mSv/h at any point of the external surface of the special vehicle, and 0.1 mSv/h at a distance of 1 m from this surface.

The following types of containers are envisaged to be loaded with retrieved RW:

— Waste of relatively small size is loaded in KTO-800 solid waste containers shown in Fig. II–17. Overall dimensions of the KTO-800 container are (in mm): length – 1266; width – 1120; height – 865. Weight of the KTO-800 container is 240 kg. Transport category – III.

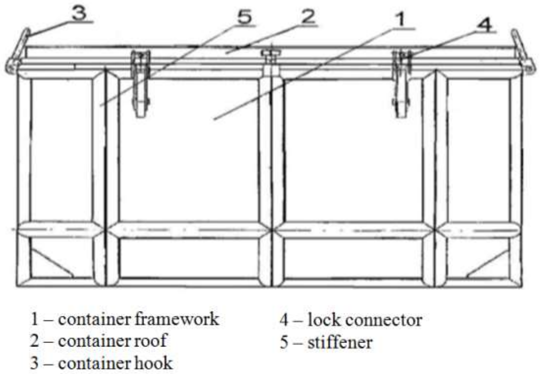


FIG. II–17. Structural design of the KTO-800 container.

— Sources with partial or complete loss of shielding are placed into a UKT1B-100 container shown in Fig. II–18. Overall dimensions of the internal shielded container are as follows (in mm): diameter – 248, height – 415. Shielded container weight is 180 kg. Overall dimensions of the UKT1B-100 transport packaging set is (in mm): diameter – 640, height – 730. Weight of the UKT1B-100 transport packaging set is 350 kg. Transport category – III.

FIG. II–18. UKT1B-100 container.

— Large-sized low-level RW is placed into UKTN-24000 containers, shown in Fig. II–19. Transport category – III.

FIG. II–19. UKTN-24000 container.

Fig. II–20 shows the special vehicles that are used for transportation of RW packages.

FIG. II–20. Waste transportation vehicle.

Lifting and handling of RW packages inside the hangar are performed by means of a bridge crane with the load-lifting capacity of 16 tons (see Fig. II–21).

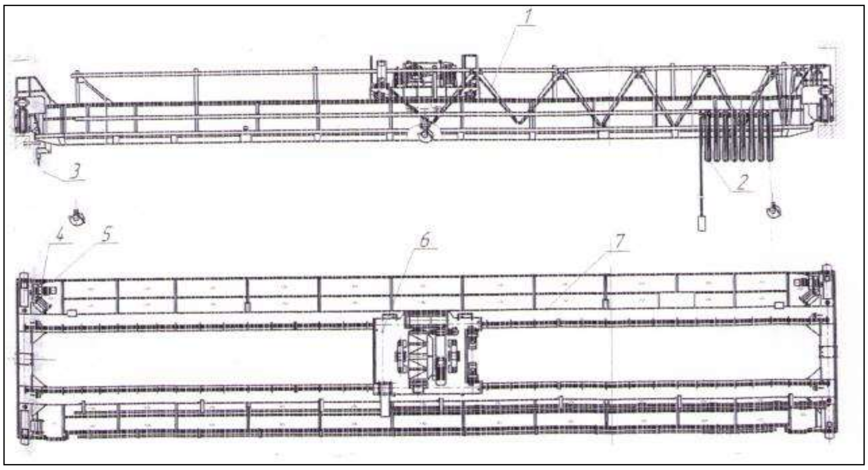


FIG. II–21. Bridge crane.

Operations to be performed

The section contains the description of work phases and operations for each phase, specifying the job positions involved in the operations. The work is divided into four phases, as shown in Table II–11.

TABLE II–11. DESCRIPTION OF WORK PHASES

Phase	Description
I	Unloading of large-sized RW packages
II	Unloading of small-sized RW packages (gamma-ray source blocks)
III	Unloading of large-sized RW packages released from under the debris
IV	Collection and packaging of spillages

Activities performed during each of these phases are represented in SAFRAN in Fig. II–22. A detailed list of activities performed during each work phase is provided in Table II–1.

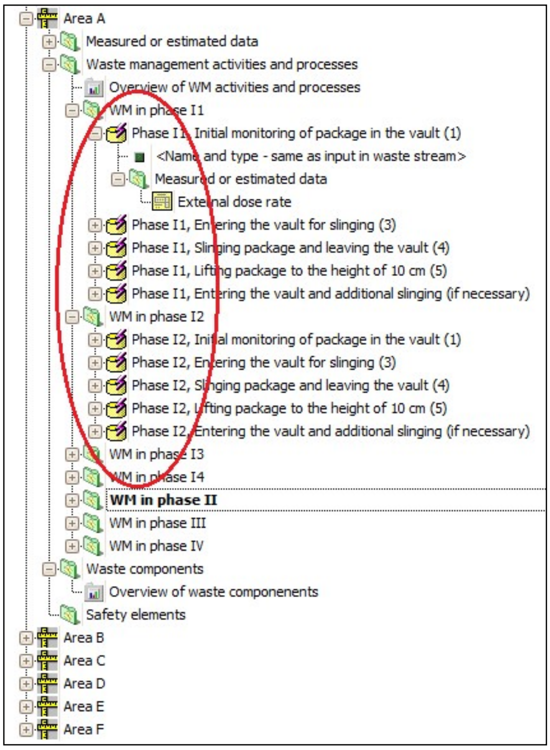


FIG. II–22. The structure of grouping of works, performed at activity phases.

II–4.4. Development and justification of scenarios

The term ‘scenario’ is defined as a postulated or assumed set of events and conditions that can lead to human exposure and/or environmental contamination. Scenarios that are selected for safety assessment strongly influence the assessment results and their relevance and credibility. Development and justification of scenarios was made using systematic approach to identify and screen hazards taking into account the description of presented above facility and activities.

To identify potential hazards, the list of PIEs from Annex I of GSG-3 [II–1] was considered and analysed, taking into account waste retrieval and supporting activities, the inventory, activity, physical conditions and location of the waste packages and working procedures predefined by the retrieval project. Because of the limited time of considered waste retrieval operations (no erosion, landslides etc., no loose of control within 100 days), predefined summer season (no snow, extreme freezing etc.), very long distance from the seacoast (no tsunamis), thin layer of clay and loamy soil (no sand storm), average precipitation and temperature regime for this area and other specific features, a significant number of PIEs were screened out as not relevant, and only the remaining PIEs were assessed in terms of possible consequences and screening in terms of hazard (see Section II–4.4.2).

Following screening, the consequences of the relevant PIEs were evaluated in this safety assessment as either anticipated operational occurrences or accident scenarios.

Doses under normal operations and postulated accident conditions were modelled using SAFRAN [II–3] to demonstrate application of the assessment methodology.

II–4.4.1. Normal operation scenario of waste retrieval

Prior to the beginning of RW retrieval from Vault 1, the following sequence of preliminary works are performed:

— Lighting equipment is installed.
— A segment of the intermediate slabs is removed.
— Preliminary radiation survey is performed.
— Ladders for getting inside the vault are installed.
— Radiation protection structures are mounted.

The process of waste retrieval from Vault 1 is divided into four phases described below.

Phase I: Unloading of large-sized RW packages available for gripping and retrieval

The sequence and duration of operations during RW retrieval in Phase I is presented in Table II–12. The duration of each operation is specified as a potential range or an average value.

TABLE II–12. SEQUENCE OF WORKS AND DURATION OF OPERATIONS ON RW RETRIEVAL FROM VAULT 1 (PHASE I)

Operation		Personnel attended areas				Operation duration, min	
Phase I. Unloading of large-sized RW packages available for gripping and retrieval		Supervisor	Slinger	Hoist man	Health physicist	min	max
1	Action for health physicist together with the RW processer: get inside Vault 1 and check visually the large-sized RW packages and exit the vault thereafter.	E	A	E	A	10	25
2	Park UKTN-24000 container on the ground near the exit from Vault 1.	E	B	B	D	15	30
3	Action for RW processer: get into Vault 1 and come to the container.	E	A	D	D	0.5	2
4	Sling the large-sized RW package with the help of load-handling accessories (or sling around in case the eye rings are damaged) and exit the vault.	E	A	D	D	2 (10)	8 (20)
5	Action for hoist operator: lift the large-sized RW up to the height of about 100 mm and ascertain in reliability of the catching accessory and slinging.	E	D	E	D	1	3
6	Action for hoist operator: load the item into UKTN-24000 container.	E	B	E	D	10	20
7	Action for health physicist: measure gamma and neutron radiation EDR on the container outer surfaces and take swabs to check surface contamination.	E	D	D	B	2	5
8	Action for RW processer: remove the slings.	E	B	B	D	1	5
9	Fulfill Operations 3–8 to achieve the permissible weight for load-lifting capacity of a 16 tons bridge crane. Thereafter transport the filled UKTN-24000 container to the place of interim storage.	E	B	E	D	15	30
10	Perform Operations 2–9 to accomplish unloading of large-sized RW that is not underneath the debris from Vault 1.	–	–	–	–		
11	Action for RW processer in response to a command from the supervisor: sling the container; reload it by means of the bridge crane onto the previously prepared place for storage of packed RW.	E	D	E	O	10	25

Note: it is assumed that operations are performed in response to supervisor commands.

When loading the RW that has been withdrawn from the vault to RW transport containers, the RW accounting staff marks each item by means of labelling it with an individual number. For the purposes of this illustrative Safety Case, actions of the accounting staff and the decontaminator are not included into Table II–12.

The EDR values in Vault 1 (Area A) vary in the process of RW retrieval. To take account of this variation, it is assumed that large-sized containers are characterized by highest EDR values according to Table II–4 and Fig. II–4 and are retrieved first.

Accordingly, Phase I was divided into four subphases. EDR values in Area A obtained on the basis of maximum values of EDR predicted for these subphases of Phase I are given in Table

II‒13. Table II-13 also provides the averaged EDR values of containers for the mentioned subphases obtained on the basis of Table II‒4.

TABLE II‒13. DESCRIPTION OF SUBPHASES OF PHASE I AND MAXIMUM VALUES OF PREDICTED EDR AT SUBPHASES OF PHASE I

	SUBPHASE I1	SUBPHASE I2	SUBPHASE I3	SUBPHASE I4
Containers to be retrieved	К4 – К7, Б9	К3, Б3 – Б5, Б8	К1, К2, К8, К9, Б2	Б6, Б7
Number of cycles of operations	5	5	5	2
EDR_A, μSv/h	1000	220	150	100
EDR_B, μSv/h	320	370	160	30

Phase II: Unloading of small-sized RW packages

The sequence of work performed and the duration of operations on RW retrieval through Phase II is presented in Table II‒14.

TABLE II–14. SEQUENCE OF WORKS AND DURATION OF OPERATIONS ON RW RETRIEVAL FROM VAULT 1 (PHASE II)

Operation		Personnel attended areas				Operation duration, min	
Phase II. Unloading of small-sized RW packages		Supervisor	Slinger	Hoist man	Health physicist	min	max
1	Action for RW processer: park the container and the transport pallet in the work area.	D	C1	C1	D	2	5
2	Action for RW processer: get inside Vault 1. Sling a certain gamma-ray source block to the hook of the bridge crane with the operating eye ring (if there is no eye-bolt, perform slinging around).	E	A	E	D	2	5
3	Action for hoist operator: lift the gamma-ray source block up to the height of about 100 mm. Action for RW processer: ascertain in reliability of the eye-bolt and slinging and exit the vault.	E	A	E	D	0.5	0.5
4	If any damaged gamma-ray source blocks are detected at unloading of RW layers, take operations on their removal from the work area as a priority.	See Table II–15					
5	Action for hoist operator: lift the gamma-ray source block above the intermediate slab level up to the height of about 500 mm and load it onto the decontamination pallet.	E	C1	E	C1	1	3
6	Action for health physicist: measure EDR of the gamma-ray source block and based on the results of measurements verify the integrity of the package protection properties. In case of damaged protection – refer to Action 10.	C1	C1	C1	C2	2	4
7	Action for hoist operator: load the gamma-ray source block onto the transport crate.	C1	C1	C1	D	1	2
8	Action for RW processer: fix the gamma-ray source block on the transport crate (6 pieces).	C1	C2	C1	D	2	5
9	Action for hoist operator: load the transport crate into the certified KTO-800 container.	C1	C2	C1	D	1	3
10	In case of damaged protection the gamma-ray source block to be placed into a separate barrel-type container, which is further loaded into KTO-800 transport container.	C1	C2	C1	D	2	5
11	Action for RW processor and hoist operator upon loading of KTO-800 container: reload the container to the place specified for interim storage.	E	E	C1	D	5	10
12	Repeat Operations 1–11 till unloading of all small-sized RW packages from Vault 1.	–	–	–	–		

When performing work in Area C under normal operation mode, it is assumed that the RW processer and the health physicist perform their actions at a distance of 0.1 m from a gamma-ray source block, while the supervisor and the hoist operator do not approach to the item closer than for 1 m.

Expert-calculated estimates of the EDR from E-1M gamma-ray source blocks are given in Table II–15.

TABLE II–15. EDR AT DISTANCE FROM THE SOURCE

Ionizing radiation source	EDR at the calculation point, μSv/h		
	on the surface	10 cm from the source	100 cm from the source
E-1M gamma-ray source block	0.436	0.157	0.073

EDR values given in Table II–15 for a single source are used to predict the EDR during the course of operations performed to fix the gamma-ray source block on the transportation crate. Since a crate is envisaged for up to 6 gamma-ray source blocks, the conservative EDR value will be determined considering radiation emitted by 6 gamma-ray source blocks. However, it is required to take account of the sequence of loading the gamma-ray source blocks onto the crate and the difference in distances between each of them and the operator. In this regard, the average EDR value during loading of gamma-ray source blocks onto the crate is approximately twice that of the EDR value from a single source.

The sequence of work performed during unloading of damaged small-sized RW packages from Vault 1 is provided in Table II–16.

TABLE II–16. SEQUENCE OF WORK PERFORMANCE AT UNLOADING OF DAMAGED SMALL-SIZED RW PACKAGES FROM VAULT 1

Operation		Personnel attended areas				Operation duration, min	
Unloading of damaged gamma-ray source blocks		Supervisor	Slinger	Hoist man	Health physicist	min	max
1	Action for health physicist: visual and radiometric survey of damaged gamma-ray source blocks.	E	D	D	E	2	5
2	Action for RW processer: if the eye-bolt of the damaged gamma-ray source block is available, grip the gamma-ray source block from the intermediate slabs by means of the process hook. Engage the second end of the process hook equipped with the eye-ring with the hook of the bridge crane. Lift the damaged gamma-ray source block up to the height of not more than 100 mm. Reload it to the side of SRW debris and load it to the vault bottom near the wall. Keep the collimator's opening directed to the wall.	E	E	E	E	5	10
3	In case if the damaged gamma-ray source block is unavailable for handling from the intermediate slabs, it is required to determine clearly the direction of the radiation ray.	E	D	D	E	2	5

TABLE II-16. SEQUENCE OF WORK PERFORMANCE AT UNLOADING OF DAMAGED SMALL-SIZED RW PACKAGES FROM VAULT 1 (cont.)

Operation		Personnel attended areas				Operation duration, min	
Unloading of damaged gamma-ray source blocks		Supervisor	Slinger	Hoist man	Health physicist	min	max
3.1	Action for RW processer: get into the vault from the side of the lowest EDR value, sling the gamma-ray source block (by the process hook), engage the sling fitting with the crane hook. Exit the vault.	E	A	E	D	0.5	4
3.2	Action for hoist operator: lift the damaged gamma-ray source block up to the height of not more than 100 mm. Reload it to the side of SRW debris and load it to the vault bottom near the wall. Keep the collimator's opening directed to the wall.	E	A	E	E	1	5
4	If performance of Operations 2–3 makes risk of personnel exposure above the established level, use the remotely controlled machine. In response to the supervisor command, the machine lowers down to the bottom of the vault.	E	D	E	E	10	20
4.1	By means of a gripper lift the damaged gamma-ray source block and reload it to the side from the SRW debris to the vault wall with its opening directed to the wall.	D	D	D	D	10	30
5	Action for RW processer: get into the vault and install the protection plug onto the collimator.	E	A	D	E	1	3
6	If it is impossible to install the plug, the damaged gamma-ray source block is to be totally relocated to the shielded container that is preliminarily placed onto the bottom of the vault.	E	A	E	E	1	3
7	Lift the gamma-ray source block or the shielded container with the gamma-ray source block to the work area.	E	C	E	E	1	3
8	Perform necessary surveys for the gamma-ray source block.	C	D	D	C	2	10
9	If the collimator of the gamma-ray source block is closed, fulfill Operations 7–9, 11, 12 of Phase II. If the plug for collimator is impossible to be installed or there is damage to the protection of the gamma-ray source block, fulfill Operations 10–12 of Phase II.	–	–	–	–		
10	Repeat Operations 1–9 at handling of the next damaged gamma-ray source block.	–	–	–	–		

Note: All operations are required to be performed in response to supervisor commands.

When loading the RW withdrawn from the vault to RW transport containers, the accounting staff marks each item by means of labelling it with an individual number. Actions of the accounting staff and the decontaminator are not included into Table II–16.

Phase III: Unloading large-sized RW packages released from under the debris

The sequence of work performed and the duration of operations on RW retrieval through Phase III is presented in Table II–17.

TABLE II–17. SEQUENCE OF WORKS AND DURATION OF OPERATIONS ON RW RETRIEVAL FROM VAULT 1 (PHASE III)

Operation		Personnel attended areas				Operation duration, min	
Phase III. Accomplishment of unloading of large-sized RW packages released from under the debris		Supervisor	Slinger	Hoist man	Health physicist	min	max
1	Perform sequentially Operations 1–9, 11 of Phase I.	–	–	–	–		

Phase IV: Collection and packaging of spillages

The sequence of work performed and the duration of operations on RW retrieval through Phase IV is presented in Table II–18.

TABLE II–18. SEQUENCE OF WORKS AND DURATION OF OPERATIONS ON RW RETRIEVAL FROM VAULT 1 (PHASE IV)

Operation		Personnel attended areas				Operation duration, min	
Phase IV. Collection and packaging of spillages		Supervisor	Slinger	Hoist man	Health physicist	min	max
1	Action for RW processer and hoist operator: park KTO-800 container in the work area.	E	C	C	C	2	5
2	Action for RW processer: localize the spillages into the shape-forming packages by using the shovel and under supervision of Health physicist. Placement of shape-forming packages onto the lifting pallet.	E	A	D	A	2	5
3	Action for hoist operator when the pallet is loaded: move the pallet to the work area. Action for health physicist: measure EDR on the surface of packages.	E	A	E	C	1	3
4	Action for RW processer: load the shape-forming packages into KTO-800 container manually.	C	C	D	D	5	10
5	Perform Operations 1–4 till complete unloading of RW spillages from Vault 1.	–	–	–	–		
6	Action for hoist operator upon radiation survey: reload the filled containers by means of a bridge crane onto the place specified for interim storage of packed RW.	E	E	E	D	5	10

In order to estimate EDR of external radiation at the phase of collection of spillages and due to unavailability of measurements, the minimum EDR value measured in the vault charged with wastes and given in Figure II-4 are used as the upper limit. However, it is possible to obtain the estimate value by means of calculation. For this purpose the following assumptions are used:

— Spillages contain low level RWs, which contain Cs-137 and Ra-226.
— Spillages are evenly spread within the area of vault basement shaping a layer of about 1 cm, which corresponds to an approximate weight of 1200 kg.

EDRs are also estimated for one sack, containing spillages and weighing 30 kg, and sacks located on the pallet.

II–4.4.2. Hazard identification and screening

For the purpose of this illustrative Safety Case, only hazards related to radiation exposures were considered; namely, external exposure from radiation sources and internal exposure due to inhalation of radionuclides.

A systematic approach was taken to the identification of hazards, and the following steps were applied to identify normal and accident scenarios that could lead to the exposure of workers and members of the public:

— Identification of hazards and initiating events;
— Hazard screening;
— Identification of scenarios.

Identification of hazards and initiating events

The following PIEs were considered in the hazard identification process for waste retrieval activities:

(a) External initiating events:
    — Natural events such as adverse meteorological conditions (e.g. wind, snow, rain, ice, temperature, flooding, lightning), earthquakes or biological intrusion;
    — Human-made events such as accidental aircraft crashes (with or without subsequent fires), explosions, fires, loss of electric power or other services, and human intrusion (mainly in cases where the facility is in a state of deferred dismantling).
(b) Internal initiating events at the facility or on the site, such as fire, explosion, structural collapse, leakage or spillage, failure of ventilation, dropping of heavy loads and failure of protective measures (e.g. failure of shielding or of personal protective equipment).
(c) Human induced initiating events, such as operator errors and violations, and misidentifications leading to the performance of incompatible activities.

The analysis of evolution of the initiating events was carried out using the event tree analysis technique, which is briefly introduced in IAEA-TECDOC-1494 [II–28].

Hazard screening

Hazards lacking the potential to cause harmful consequences for workers, the public and the environment to an extent that is not in compliance with relevant safety requirements or criteria, as well as hazards that could not be realized in view of the scope of the waste retrieval activities being assessed, were screened out from the subsequent hazard analysis.

The following potential exposure pathways, through which the identified hazards could cause harmful consequences for workers, were considered in the screening process:

(a) External exposure due to direct radiation from gamma emitting radionuclides from radioactive material (e.g. sealed sources, RW packages);
(b) Internal exposure due to inhalation or ingestion from airborne releases (e.g. aerosols and particulates from spillages), or in fires (for the purpose of this illustrative safety case, fire scenarios are not further considered);
(c) A combination of radiological contamination and physical injuries (e.g. the contamination of wounds).

The results of the screening are recorded in the SAFRAN file along with justification of any scenarios that were not considered to be relevant. These results are summarized in Table II−19.

TABLE II−19. RESULTS OF QUALITATIVE HAZARD SCREENING

Name	Relevance	Relevance – justification (if not relevant)	Category
Lightning (effect on power supply)	Not relevant	The facility can operate (i.e. perform its basic function – storage of waste and its retrieval) safely without electrical supply, all important systems have backup supply.	External natural
Lightning (effect on surroundings of facility)	Not relevant	The RADON-type facility is an underground facility – covered with concrete pavement, the lighting doesn't have an impact on the surrounding of the facility.	External natural
Lightning (effect on facility)	Not relevant	The RADON-type facility is an underground facility – covered with concrete pavement and subsequently with hangar structures, the lighting doesn't have an impact on the facility.	External natural
Extreme snowing	Not relevant	Waste retrieval activities are planned for the summer season.	External natural
Extreme rain	Not relevant	Hangar structures are designed with account taken of extreme rain.	External natural
Extreme drought	Not relevant	Extreme draughts are not typical for the area of the facility.	External natural
Strong wind	Not relevant	Hangar structures are designed with account taken of strong wind.	External natural
Extreme temperatures	Not relevant	Extreme temperatures are not typical for the area of the facility.	External natural

TABLE II–19. RESULTS OF QUALITATIVE HAZARD SCREENING (cont.)

Name	Relevance	Relevance – justification (if not relevant)	Category
Hydrology and hydrogeology	Not relevant	Waste retrieval activities are planned for the summer season when groundwater is far below the surface.	External natural
Geology of site and region	Not relevant	Geology is well known, and is not expected to change in the near future.	External natural
Seismic events	Not relevant	The design and the construction of the facility are seismically safe.	External natural
Other effects of ground stability	Not relevant	There are no other effects of ground stability.	External natural
Geomorphology and topography of the site	Not relevant	Nothing from geomorphology or topography of the site can affect the safety of the facility.	External natural
Terrestrial and aquatic flora and fauna	Not relevant	The RADON-type facility is closed well, and flora and fauna can't affect the processes inside the facility.	External natural
Potential for natural fires, storms, etc.	Not relevant	Within a radius exceeding the radius of the controlled area of the facility, there are no gas/oil pipelines, industrial facilities, warehouses, water reservoirs, etc.	External natural
Flooding	Not relevant	In the territory adjacent to the facility, ground water is opened at a depth from 0.5 to 1.3 m. Groundwater is unconfined. Expected water table rise is to 0.0 m. The RADON-type facility was constructed in a human-made soil fill over 1.5 m high, therefore, its flooding is unrealistic, especially in the summer season.	External natural
Accidental aircraft crashes; Nearby military activities	Not relevant	There are no military objects and/or military activities nearby the facility. Taking into account that there are no regular airlines routes in the vicinity of facility, only light firefighting and traffic monitoring vehicles could, in principle, crash in the vicinity of the facility with very low probability. Potential consequences of these events have to be evaluated.	External human factors

II–4.4.3. Identification of scenarios

As a result of this being an illustrative safety case, although a hazard identification and screening assessment has been undertaken, only a limited number of anticipated operational occurrences and accident scenarios have been selected for detailed hazard analysis to demonstrate the use of SAFRAN and its implementation of the SADRWMS methodology for accident scenario modelling.

Accident scenario: Radiation source dropped from gamma-ray source block in the work area

In the course of work with gamma-ray source blocks in the RW vault, an accidental situation related to opening of the collimator's opening or to the source drop-out from the block is likely to occur. As a result, the EDR increases abruptly. In the first case, the direction of radiation exposure will be narrow; in the latter, the direction will cover a wide area.

Prior to commencement of works, a thorough visual inspection of the gamma-ray source block is performed to detect open collimator's openings (if any), to establish the direction of a "beam" and to close the openings thereafter. In the course of lifting up of gamma-ray source blocks, there can occur a gamma-ray source block with the open collimator's opening, which was initially covered by the uplifted objects. In case of visual detection of such gamma-ray source block its radiometric survey are conducted with further decision making on closure of the opening with the operating/standby plug or on placing of this gamma-ray source block to a concrete container.

Such an event is considered as unlikely to happen. However, it cannot be excluded that the initially shielded radiation source will open or will drop out from the shielded container in the course of works. In case of detection of such sources, the following actions are to be performed immediately:

- Workers leave the SRW vault.
- Radiometric survey is carried out by the health physicist.
- Based on the survey results, the decision on loading of the dropped-out source to the UKT-100 container is taken by the supervisor.
- Additional technical tools, such as manipulators or remote grippers, are used by the decontaminator.

Accident scenario: Loss of ventilation due to power supply failure

In the event of a failure of the ventilation system, it is assumed that Rn-222 begins to accumulate, and that personnel exposure increases due to inhalation. However, natural convection is assumed to limit the accumulation of Rn-222 in the room.

Accident scenario: Dropping of a RW container from height (malfunction of lifting equipment)

Dropping waste packages or other loads due to mishandling or equipment failure, resulting in damage to the dropped waste package and possibly to other waste packages or to the SSCs of the facility, is considered a viable accident scenario.

In such an event, according to work and emergency procedures and rules, workers are to be immediately evacuated from the place of accident, and the immediate area will be subject to visual inspection and radiation survey.

II–4.5. Formulation of models and identification of data needs

II–4.5.1. Estimation of personnel exposure under normal operation

Dose rate estimates are based on the assumption that personnel perform their duties in accordance with technical specifications, work and emergency procedures. It is assumed that, at any single time, only one operation prescribed in the specifications can be performed, i.e. there can be no overlapping of operations. Therefore, personnel involved in an operation are assumed not to take part in other work activities and to stay in the protection zone during breaks from work.

In order to calculate the duration of work activities and associated doses to persons involved in the work, the following equations are used:

Duration of work of "i" job position during "n" phase:

$$T_{i,n} = \sum_j T_{i,n,j} \tag{II–1}$$

Dose rates of "i" job position during "n" phase:

$$D_{i,n} = \frac{1}{60} \sum_j W_{i,n,j} \cdot T_{i,n,j} \qquad T_{i,n} = \sum_j T_{i,n,} \tag{II–2}$$

Annual dose rate of "i" job position:

$$D_i = \sum_n D_{i,n} \qquad D_{i,n} = \frac{1}{60} \sum_j W_{i,n,j} \cdot T_{i,n,} \tag{II–3}$$

Where:

i is the job position in compliance with the number, provided in Table II–8;

n is the phase number;

j is the operation number;

$T_{i,n,j}$ is the duration of work performed by "i" job position (min);

$W_{i,n,j}$ is the EDR in the work area of "i" job position in the relevant operation (μSv/h).

Flowcharts of work performance in each of the phases are shown in Fig. II–23 (Phases I and III), Fig. II–24 (Phase II), and Fig. II–25 (Phase IV).

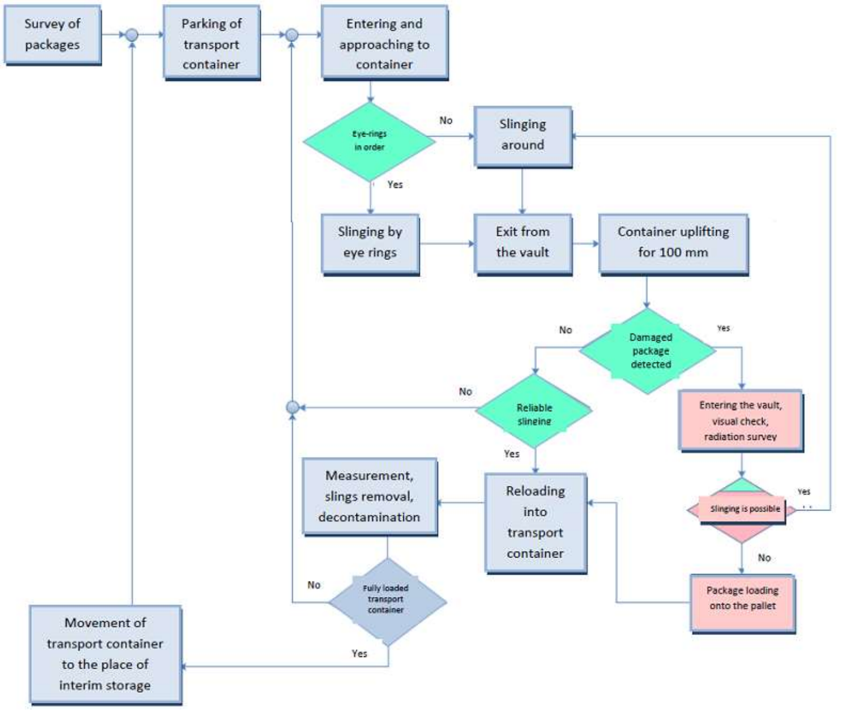


FIG. II–23. Flowchart for works performed at Phases I and III.

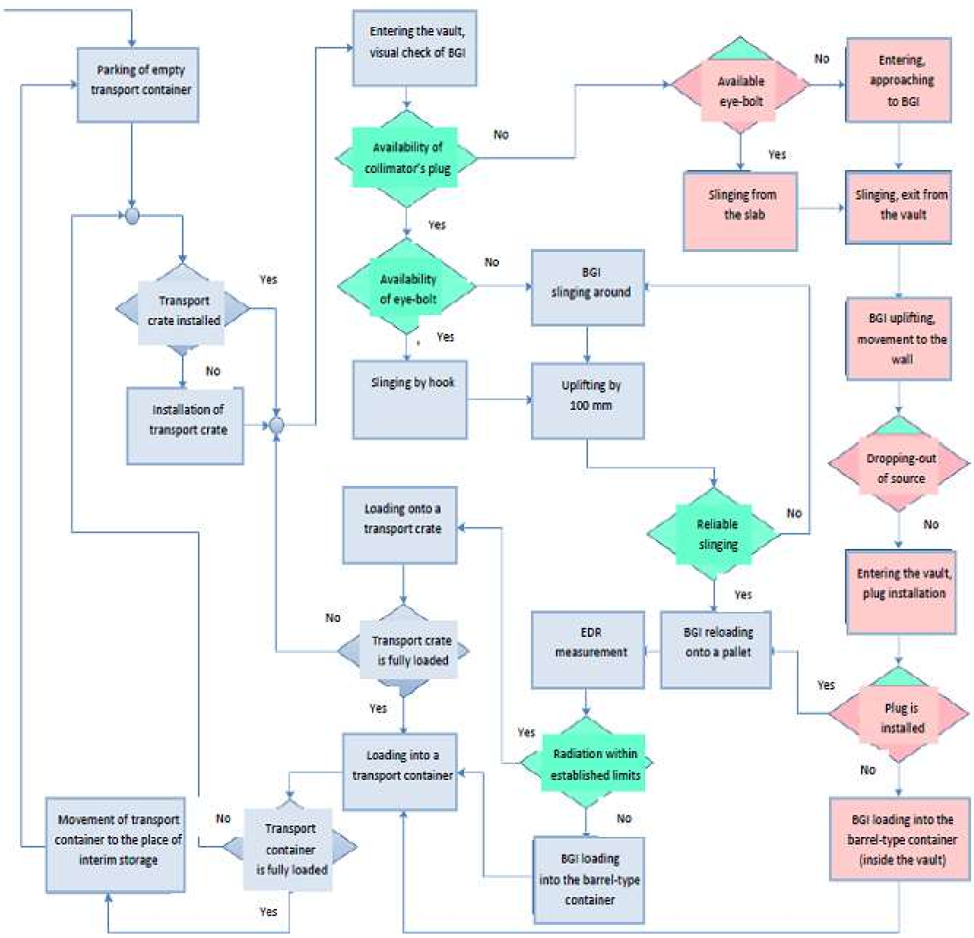


FIG. II–24. Flowchart for works performed at Phase II.

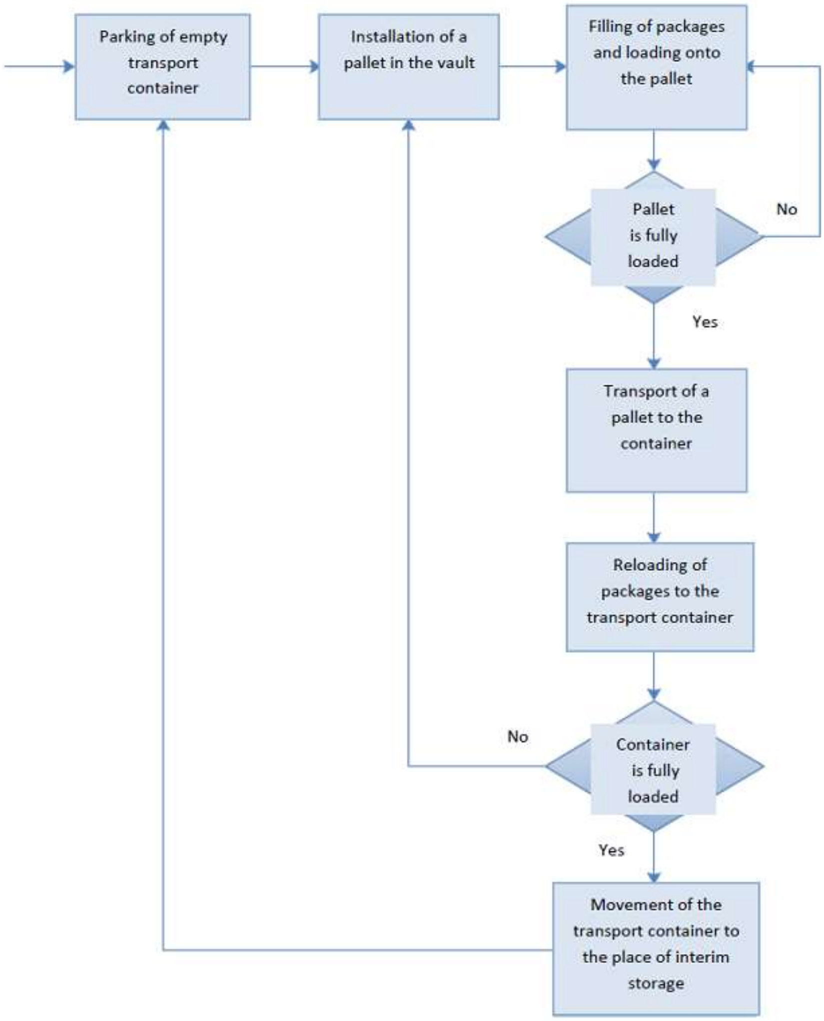


FIG. II–25. Flowchart for works performed at Phase IV.

II–4.5.2. Assessment of doses using the SAFRAN tool

The SAFRAN tool (version 2.3.2.7) [II–3] was used to perform safety assessment in this document. The SAFRAN tool was developed within the International Project on Safety Assessment Driving Radioactive Waste Management Solutions (SADRWMS project) (2004–2010) [II–2] to implement methodologies for safety assessment. The main objective of the SAFRAN tool is to assist operators, regulators and technical support specialists in performing systematic and structured safety assessments of predisposal RW management facilities in compliance with national regulations, relevant international safety standards, and good international practice. The software tool was designed to be generic and to cover all kind of predisposal waste management activities, including management of DSRS. The SAFRAN tool aids the user in:

- Describing predisposal RW management activities in a systematic way;
- Conducting the safety assessment with clear documentation of the methodology, assumptions, input data and models;
- Establishing a traceable and transparent record of the safety basis for decisions on proposed waste management solutions;
- Demonstrating clear consideration of and compliance with national and international safety standards and recommendations.

The SAFRAN tool includes the following databases:

- Radionuclide half-lives;
- Clearance levels;
- Gamma constants – dose rates at 1 m from a point source;
- Screening dose rates for normal and accidental situations;
- Screening release rates for normal operation;
- Screening releases for accidental situations;
- Release fractions;
- Dispersion factors;
- Dose conversion factors for normal operation and accidental situations.

The SAFRAN tool also supports a number of features, both in terms of usability and applicability. This can greatly assist in the elaboration of the safety case. These include:

- Help pages to guide the users in filling the various forms within SAFRAN;
- Comment boxes that allow both users and reviewers to record comments, and thus provide a means of dialogue;
- Link to documents and other electronic material (e.g. pictures, maps) for uploading as part of the safety assessment;
- Exporting data (including results of hazard screening, assessment calculations) to MS Excel, PDF or other formats.

Within the SAFRAN tool, a 'model' site is defined which the user performs a safety assessment of; the model consists of a user-specified configuration of facilities, rooms, areas, and processes that describe the RW operations and the area where these operations are performed. Once the model is created, the user can then specify parameters that will be used to perform the safety assessment calculations; these parameters include specific nuclides, concentration of radionuclides in the air, external EDR, etc. The user then defines the RW that is processed within the model site, assigning properties to the RW as it is processed in the model (including RW properties that might be affected during processing, such as activity concentration or volume). The user then defines the scenarios that will be considered, including normal operations and accident conditions. Doses for each scenario are calculated summarily for the areas, assuming that all people who are involved in the activity are located in one area and are in similar conditions of radiation exposure and under the same duration of exposure.

The development of the safety assessment in the SAFRAN tool starts with the description of the facility in which the activity is performed (see Fig. II–26). In this illustrative safety case, the model consists of a single unit of the facility, "Hangar", consisting of a single compartment, "Waste retrieval room". Within this compartment, six areas are distinguished (see Fig. II–26).

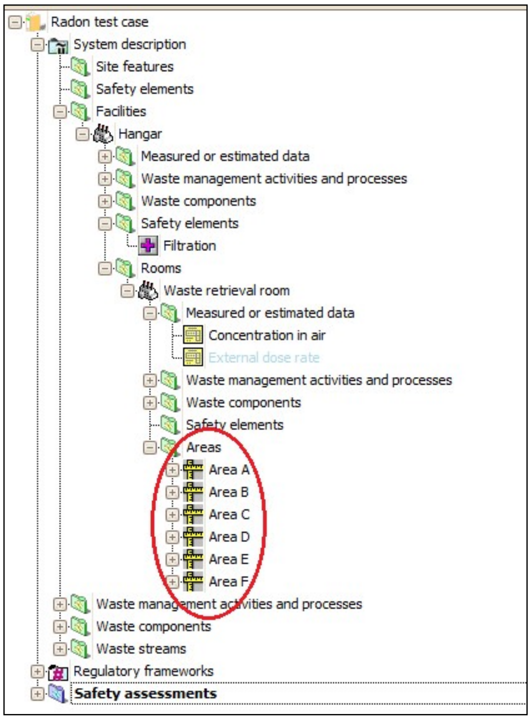


FIG. II–26. Designation of areas in the illustrative safety case.

For each of the areas, the types of activities to be performed are assigned. The activities are grouped according to work phases. Where values are set up for external EDRs at work performance phases, the following phases are specified:

— Waste management in phase I1;
— Waste management in phase I2;
— Waste management in phase I3;
— Waste management in phase I4;
— Waste management in phase II;
— Waste management in phase III;
— Waste management in phase IV.

The structure reflecting the grouping of works is illustrated in Fig. II–27.

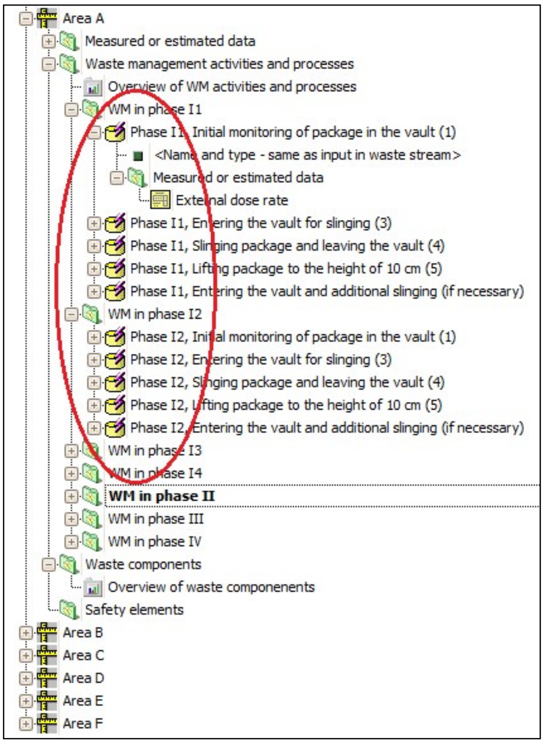


FIG. II–27. Structure of grouping of work activities performed during each phase.

In addition to showing the division of work activities into phases, Fig. II–27 shows the contents of work phases: "waste management in phase I1" and "waste management in phase I2". These phases are composed of similar sets of operations that are performed under different values of external EDRs. External EDR values are established for each of the operations.

The general list of operations to be performed (listed in Table II–1) is presented in the folder "Overview of waste management activities and processes" and is illustrated in Fig. II–28.

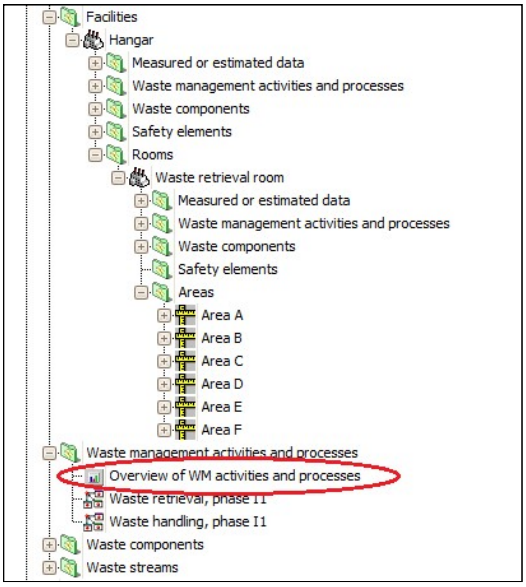


FIG. II–28. Folder "Overview of waste management activities and processes".

The folder “Incoming waste” is used to describe the RW components (Fig. II–29) which are to be retrieved during the mentioned subphases.

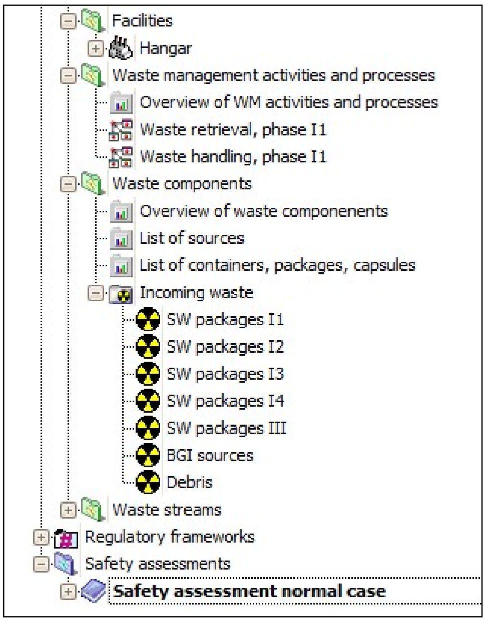


FIG. II–29. Description of RW.

As an example, Fig. II–30 illustrates the properties of “Solid waste packages I1”. In the case of this safety assessment, the content of radioactive substances is not entered in this component of the SAFRAN tool, since this data is not used to estimate EDRs of personnel involved in the work.

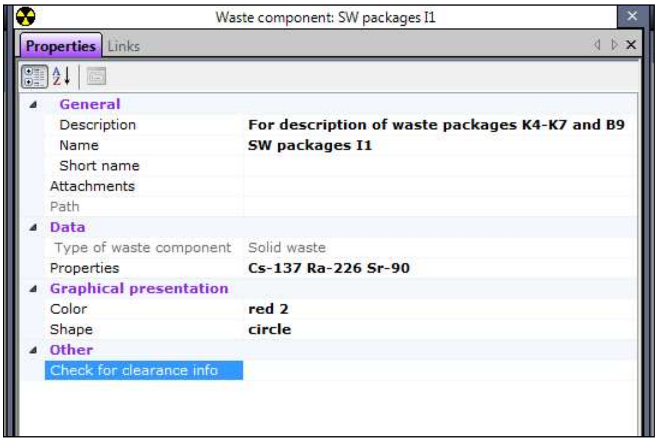


FIG. II–30. Description of “Solid waste packages I1”.

II–4.5.3. Assessment of doses during normal operation

Safety assessment is carried out for normal operation modes of RW retrieval. Receptors are defined by their job descriptions under “Common endpoints” (Fig. II–31).

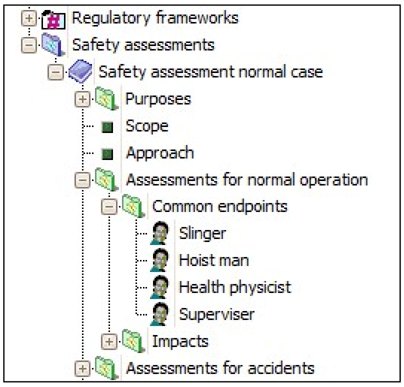


FIG. II–31. Introduction into the model of job positions performing works.

The establishment of the structure of impacts (exposures) during work in Phases I through III is shown in Fig. II–32. Phase I is divided into subphases I1 to I4, each of which assigned to work areas. Inside the structure of these areas, impacts are assigned for each of the activities defined.

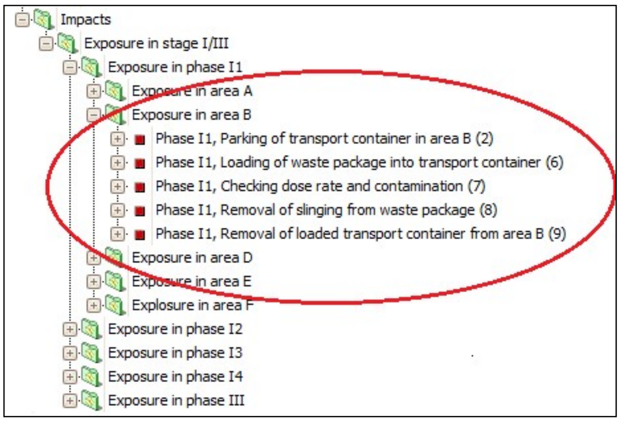


FIG. II–32. Establishment of the structure of impacts.

It is assumed that, inside the hangar, the air mixes evenly and, therefore, the concentration of radionuclides in the air is equivalent in all of the areas and is assigned a common value – the value established for the "Waste retrieval room". During normal operation (with an operational ventilation system), it is assumed that one radionuclide that considerably affects personnel in the room is available in the air of the room (Rn-222). Based on calculated estimates, the activity concentration of Rn-222 under operating ventilation is assumed to be 262 Bq/m^3.

A filtration coefficient of 0.9 for the ventilation system is assumed for both the "Waste retrieval room" and the "Hangar" (which in this case is not realistic).

EDRs were entered for each area and for each subphase of work. In order to calculate doses, an exposure time was also entered. For this purpose, a table was created, where the data concerning work performance time intervals are generalized, with account taken of the number of repeated operations. Table II–20 presents the minimum and maximum values of the predicted duration of work under normal operation for Phases I and III.

TABLE II–20. PREDICTED DURATION OF WORK PERFORMED UNDER NORMAL OPERATION MODE

No.	Area	Personnel job position	Operation	Duration, hour									
				Phase I1		Phase I2		Phase I3		Phase I4		Phase III	
				min	max	min	max	min	max	min	max	min	max
1	A	RW Processer	Entering the vault, visual inspection of large-sized RW package	0.17	0.42	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
2			Entering the vault, approaching to the container	0.04	1.67	0.04	1.67	0.04	1.67	0.02	0.67	0.03	1.00
3			Slinging of a large-sized package	0.17	0.67	0.17	0.67	0.17	0.67	0.07	0.27	0.10	0.40
4		Health physicist	Entering the vault, visual inspection of large-sized RW packages	0.17	0.42	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
5	B	RW Processer	Parking of transport container for large-sized RW packages	0.50	1.00	0.50	1.00	0.25	0.50	0.25	0.50	0.25	0.50
6			Loading of large-sized package into the transport container	0.83	1.67	0.83	1.67	0.83	1.67	0.33	0.67	0.50	1.00
7			Removal of slings from the large-sized package	0.08	0.42	0.08	0.42	0.08	0.42	0.03	0.17	0.05	0.25
8			Slinging of the transport container upon its loading	0.33	0.83	0.33	0.83	0.17	0.42	0.17	0.42	0.17	0.42
9		Hoist operator	Parking of transport container for large-sized RW packages	0.50	1.00	0.50	1.00	0.25	0.50	0.25	0.50	0.25	0.50
10			Removal of slings from the large-sized package	0.08	0.42	0.08	0.42	0.08	0.42	0.03	0.17	0.05	0.25
11		Health physicist	Survey of the large-sized package	0.17	0.42	0.17	0.42	0.17	0.42	0.07	0.17	0.10	0.25
12	D	RW Processer	Staying in area D	0.42	0.67	0.42	0.67	0.42	0.67	0.17	0.27	0.25	0.40
13		Hoist operator	-----”-----	0.38	1.25	0.38	1.25	0.38	1.25	0.15	0.50	0.23	0.75
14		Health physicist	-----”-----	2.71	6.00	2.71	6.00	2.04	4.58	1.22	2.68	1.49	3.32
15	E	Supervisor	Work in area E	3.04	6.83	2.88	6.42	2.21	5.00	1.28	2.85	1.59	3.57
16		Hoist operator	-----”-----	2.25	4.83	2.25	4.83	2.08	4.42	0.93	2.02	1.32	2.82
17	F	RW Processer	Reloading of transport container onto the platform of the special vehicle	0.50	1.00	0.50	1.00	0.25	0.50	0.25	0.50	0.25	0.50

The data provided in Table II–20 for each exposure and each operation are entered into the calculation model (circled in Fig. II–33) labelled “Min dose” and “Max dose”. Data are also entered under “Use in cumulative results” to specify whether the result related to this component will be accounted in the summary estimates.

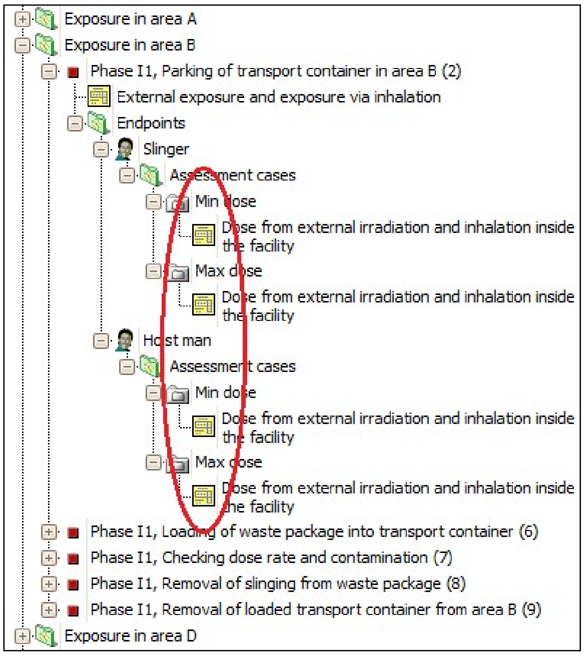


FIG. II–33. Components of the calculation model, wherein the values of the predicted minimum and maximum durations of works performance are to be entered.

At this point all the input data necessary to calculate personnel dose rates during waste retrieval under normal operation mode is complete. Calculated EDR values are reflected in the form of a table “Comparison of doses inside” in the folder “Analysis” (Fig. II–34). The data are also available in the SAFRAN tool in the form of bar charts. The results of subphase I1 are provided in Section 4.6.

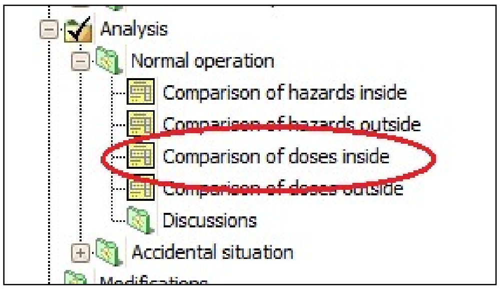


FIG. II–34. Bookmarks “Comparison of doses inside” inside the folder “Analysis” containing the results of personnel dose rate calculation.

II–4.5.4. Radiation source dropped from gamma-ray source block in work area

The methodology for modelling an accidental situation is demonstrated by the example of works performed at Phase II, related to retrieval of gamma-ray source blocks. This has been done with the use of the SAFRAN tool.

The following initiating events selected from the list provided in GSG-3 [II–1] and illustrated in Fig. II–35 are considered:

- External human induced:
  - Power supply and the potential loss of power.

— Internal:
  - Dropping waste packages or other loads due to mishandling or equipment failure, with consequences to the dropped waste package and possibly to other waste packages or to the SSCs of the facility;
  - The malfunctioning of equipment that maintains the ambient conditions in the facility, such as the ventilation system or dewatering system;
  - The failure of the power supply, either the main system or various subsystems;
  - The malfunctioning of key equipment for handling waste, such as transfer cranes or conveyors;
  - Incorrect operator action in spite of having accurate and complete information.

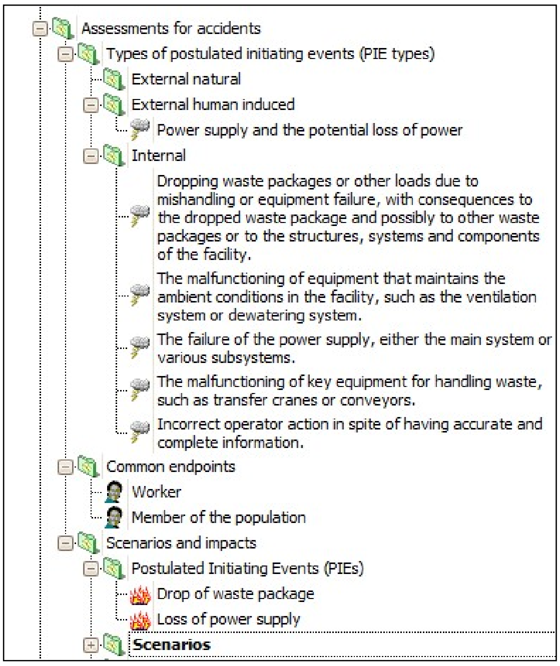


FIG. II–35. Initiating events for accidental scenarios.

The scenarios selected to determine the potential consequences of the considered PIEs (impacts) are illustrated in Fig. II–35:

— Drop of waste package;
— Loss of power supply.

Workers and members of the public are collectively addressed as end points for the scenarios.

The SAFRAN representation of the scenarios is depicted in Fig. II–36.

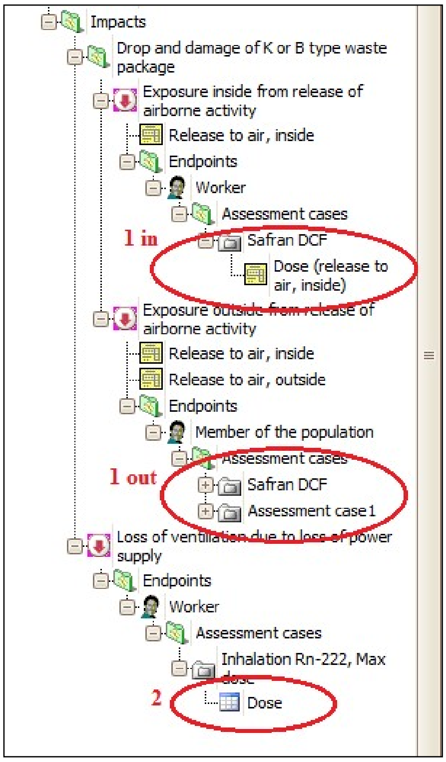


FIG. II–36. SAFRAN representation of the scenarios.

II–4.5.5. Drop of waste package

Under the first accidental scenario, it is assumed that the accident leads to loss of leak tightness of a metal package containing 2×10^{13} Bq of Cs-137 and 2×10^{13} Bq of Sr-90, ingress of the radionuclides into air of the room and escape of the radionuclides away from the hangar, taking into account a filtration efficiency of 0.9.

In order to calculate the propagation of radionuclides in the room air, the transfer model (described in Annex I of IAEA-TECDOC-1777 [II–2]) (folder “1 in” in Fig. II–36) was used. The SAFRAN tool provides the possibility to select model parameters from a predefined set of values.

The following values were used for this accident scenario:

— Room volume – 3000 m^3;
— Distance from the place of accident – 2 m;
— Exposure duration – 10 min;
— Airborne release fraction – 10^{-6}.

The parameters and results obtained are illustrated in Fig. II–37.

Waste component	Nuclide	Activity (...	ARF	Release i...	Room volu...	Distance (m)	Exposure tim...	Dispers...	.	Dose
SW packages I1 - [Phas...	Cs-137	2E+13	1.00E-006	2E+07	3000	2	10	3.56E-004	0	1.44E-004
SW packages I1 - [Phas...	Ra-226	0	1.00E-006	0	3000	2	10	3.56E-004	0	0
SW packages I1 - [Phas...	Sr-90	2E+13	1.00E-006	2E+07	3000	2	10	3.56E-004	0	6.41E-004

FIG. II–37. The window for selection of parameters and results obtained with regard to exposure inside the room at the first accidental scenario.

II–4.5.6. Loss of ventilation

Under the second accidental scenario (element “2” in II–36), the consequences of ventilation malfunction are considered, due to which the room starts accumulating Rn-222, and personnel exposure increases due to inhalation. In the case of unavailability of forced ventilation and permanent ingress of Rn-222 into a room, its concentration value is influenced by two processes:

- Air outlet from the room due to natural convection;
- Radioactive decay (half-life 3.823 days).

II–4.6. Performance of calculations and evaluation of results

II–4.6.1. Assessment of doses during normal operation (direct external exposure)

The following procedure is applied to assess doses resulting from direct external exposure. Within the SAFRAN tool, the user selects the affected waste components and identifies the number of packages and total inventory for each radionuclide in the hazard screening step. Doses are calculated using the following equation:

$$\text{Dose (i,k)} = \text{DoseRate (i,k)} * \text{time (k)} \qquad \text{(II–4)}$$

Where:

i is radionuclide;

k is a particular waste component;

DoseRate(i,k) is the dose rate from a single component;

time(k) is the duration of exposure.

Calculation of external EDRs from a single waste component can be performed using the SAFRAN models for external exposure calculations. There are models for several geometries (cube, cylinder, sphere, point source, and disc). For complex situations, it is necessary to use other tools. The SAFRAN models allow for performing calculations either with or without consideration of shielding. Common parameters necessary for the models are inventory (Bq) and distance (cm). The SAFRAN tool automatically collects and sends the values for these parameters to the models.

Once doses resulting from each radionuclide have been evaluated, the total dose from all radionuclides in the affected waste component can be calculated as follows:

$$\text{Dose (k)} = \text{SUM (Dose (i, k))} \qquad \text{(II–5)}$$

EDRs calculated for normal operation in Phase I are presented in Table II–21.

TABLE II–21. RESULTS OF EDR CALCULATIONS FOR PHASE I

Personnel	Doses to personnel at different subphases of phase I, Sv/a					Total dose (Phase I), Sv/a
	Area A, B				Area D, E, F	
	I1	I2	I3	I4		
Slinger	3.1E-3	1.7E-3	6.5E-4	9.8E-5	1.9E-4	5.8E-3
Hoist operator	4.5E-4	5.3E-4	1.5E-4	2.1E-5	9.9E-4	2.1E-3
Health physicist	5.5E-4	1.66E-4	6.8E-5	5.2E-6	3.7E-5	8.3E-4
Supervisor	–	–	–	–	1.3E-3	1.3E-3
Check person (RW accounting staff)	1.3E-4	1.5E-4	6.8E-5	5.2E-6	2.1E-4	5.6E-4
Decontaminator	1.3E-4	1.5E-4	6.8E-5	5.2E-6	3.8E-5	3.9E-4

Table II–22 presents values of cumulative doses calculated for the full duration of waste retrieval operations.

TABLE II–22. DOSES TO PERSONNEL AT DIFFERENT PHASES OF WASTE RETRIEVAL

Personnel	Doses to personnel at different phases of waste retrieval, Sv/a				Total dose, mSv/a
	Phase I	Phase II	Phase III	Phase IV	
Slinger	5.8E-3	1.9E-4	1.9E-4	7.7E-5	6.2
Hoist operator	2.1E-3	7.4E-5	7.6E-5	2.4E-5	2.3
Health physicist	8.3E-4	5.7E-5	2.6E-5	5.7E-5	0.97
Supervisor	1.3E-3	8.9E-5	2.7E-5	4.9E-5	1.5
Check person (RW accounting staff)	5.6E-4	5.2E-5	2.8E-5	3.1E-5	0.67
Decontaminator	3.9E-4	3.3E-5	2.1E-5	5.1E-5	0.49

Graphical illustrations of doses to the different groups of workers for the period of retrieval operations are presented in Figs II–38 to II–43.

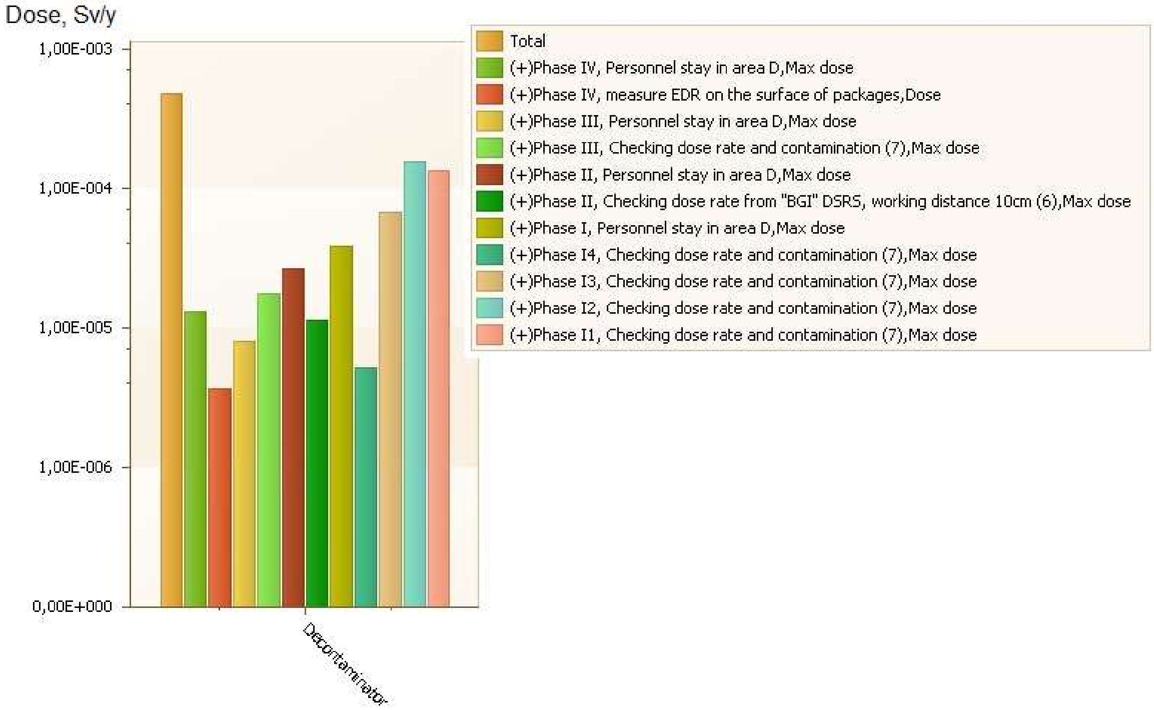


FIG. II–38. Doses to the decontaminator during waste retrieval operations.

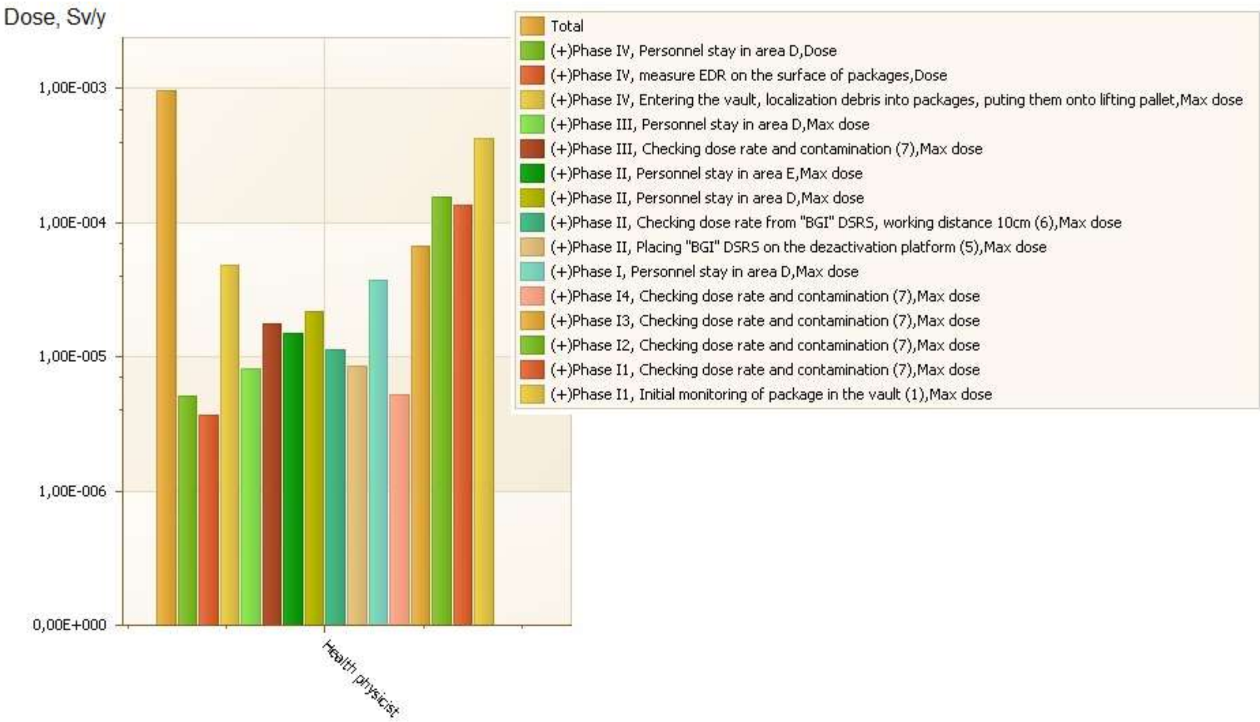


FIG. II–39. Doses to the health physicist during waste retrieval operations.

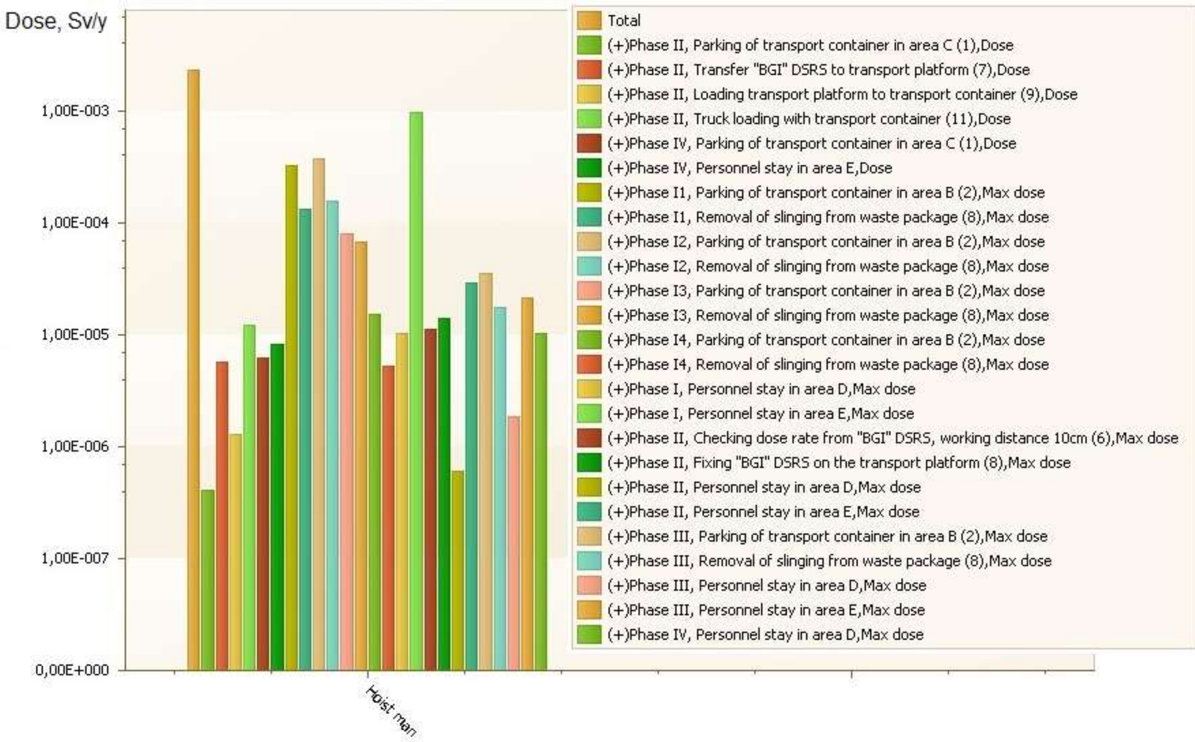


FIG. II–40. Doses to the hoist operator during waste retrieval operations.

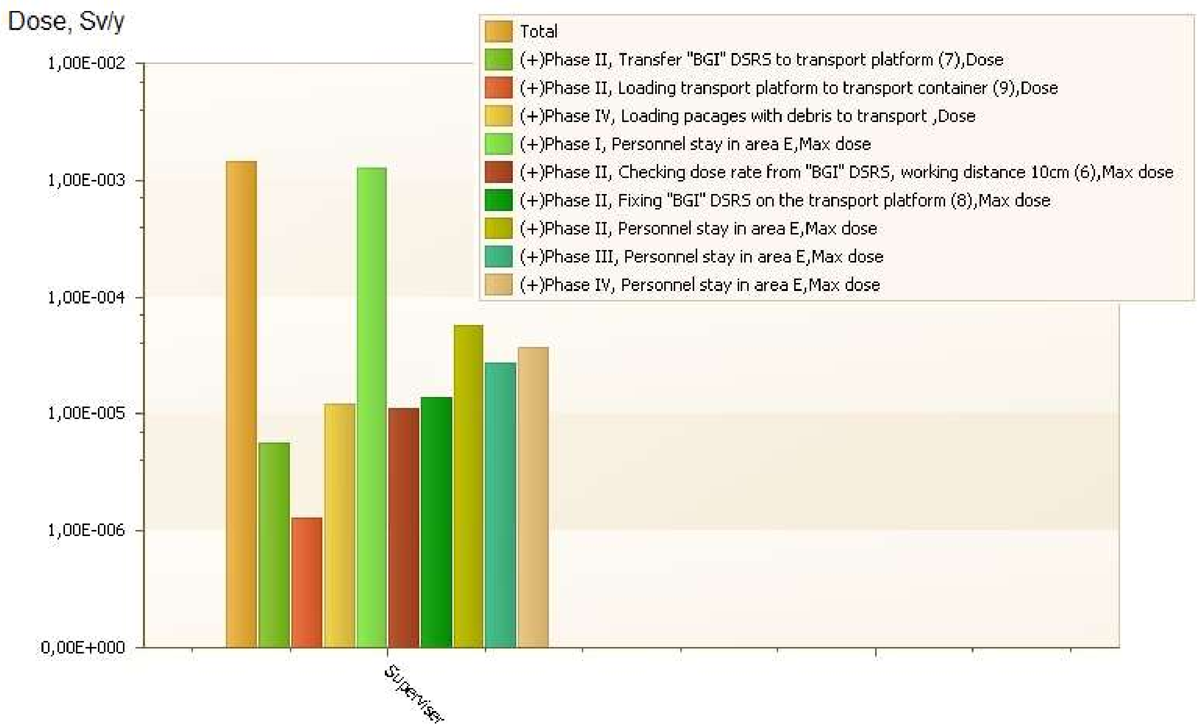


FIG. II–41. Doses to the supervisor during waste retrieval operations.

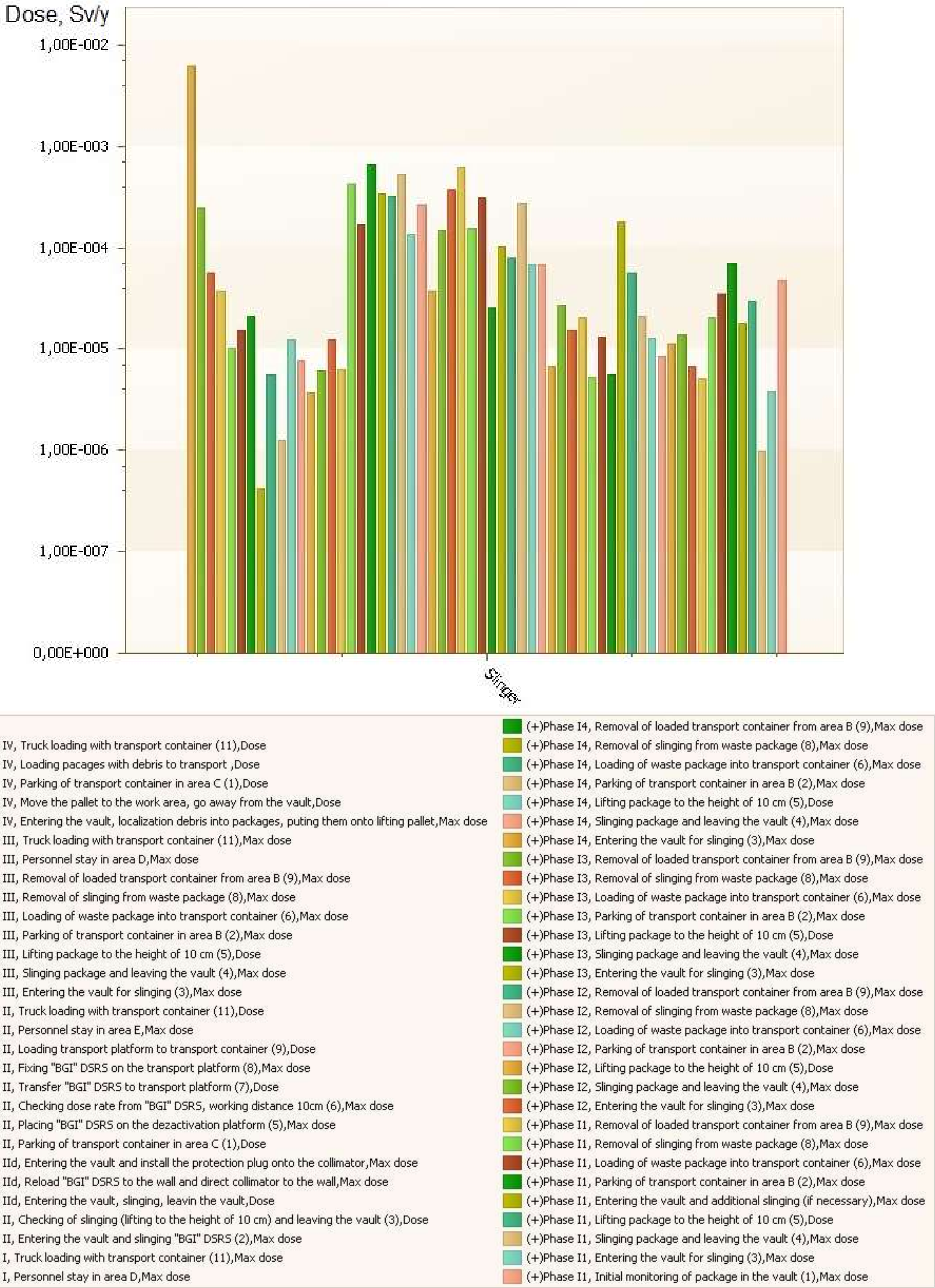


FIG. II–42. Doses to the slinger during waste retrieval operations.

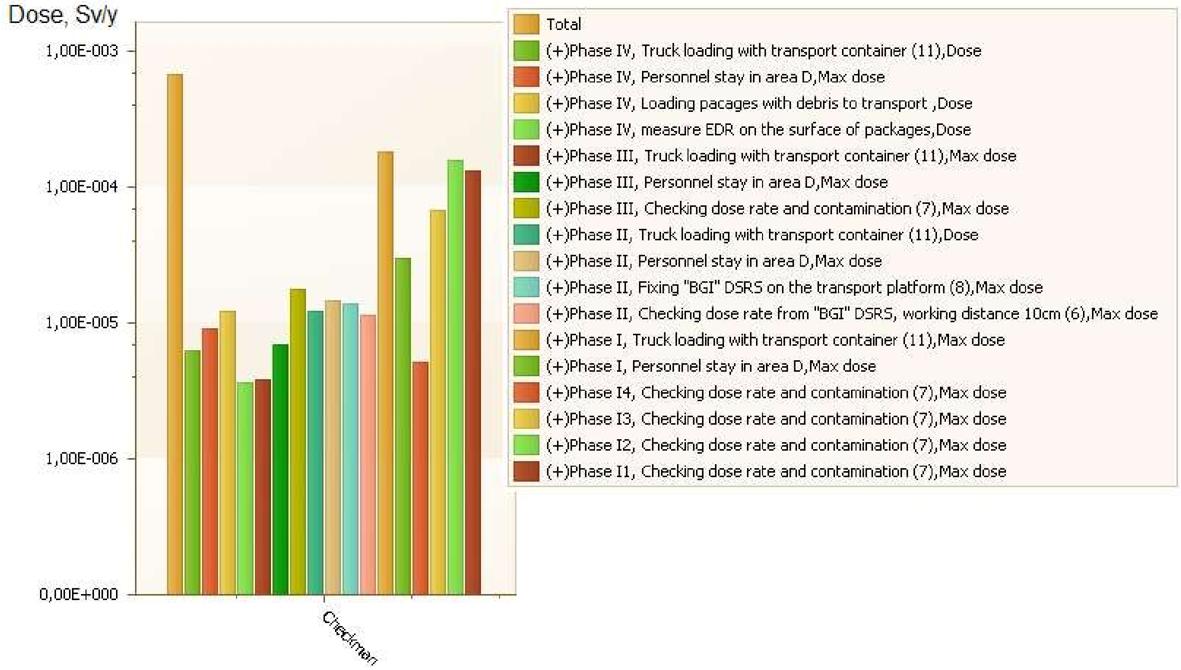


FIG. II–43. Doses to the check person during waste retrieval operations.

II–4.6.2. Accident scenarios

Radiation source dropped from gamma-ray source block in the work area

In case of such an accident a gamma survey is carried out by the health physicist. Tools such as remote grippers are used by the decontaminator to load the dropped-out source to UKT-100 container. It is expected that the gamma survey takes conservatively 1 minute and loading of the dropped-out source to a special container takes 5 minutes. The length of the remote gripper is 1 m.

Expert-calculated estimates of the EDR resulting from exposure to an unprotected E-1M gamma-ray source block are given in Table II–23.

TABLE II–23. EDR DEPENDING ON THE DISTANCE FROM THE SOURCE

Radiation source	EDR at calculation point, mSv/h	
	10 cm from the source	100 cm from the source
DSRS	$7.48 \cdot 10^2$	7.45

The results of dose assessment for this anticipated operational occurrence are presented in Table II–24.

TABLE II–24. DOSES TO PERSONNEL FROM DROPPED-OUT SOURCE

Personnel	Doses to personnel, mSv
Health physicist	0.12
Decontaminator	0.37

Loss of ventilation

The dose assessment was carried out using the SAFCALC module of the SAFRAN tool (model AAir_Worker_Inside.eco). Assuming a one hour period and an equilibrium value of Rn-222 concentration in the room under long term loss of ventilation, the calculated dose to the worker is 0.393 mSv (Fig. II–44).

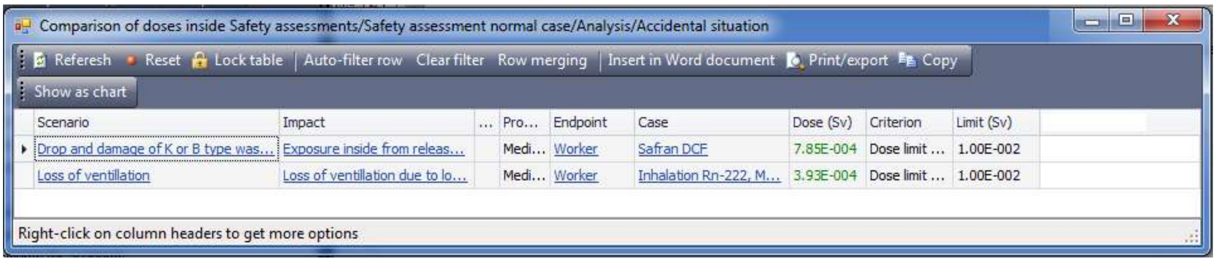


FIG. II–44. Doses to personnel for accident scenarios.

Drop of a waste package

In order to calculate EDRs outside the building (element “1 out” in Fig. II–35), the Gaussian model for atmospheric dispersion for different diffusion categories is used. Calculations are performed using the SAFRAN module SAFCALC.

Parameters necessary to perform the calculations include:

— The radionuclide composition, in this case Cs-137 and Sr-90;
— “Accident released activity” (ingress of radionuclides in the air outside the building in the content of inhaled fraction): 212 Bq (106 Bq each for Cs-137 and Sr-90);
— “Building height” and “Building width”: 8 m and 30 m, respectively;
— “Exposure location” (distance from the emission source to the place of exposure) – assumed to be 500 m.

Radiation doses are calculated taking into account the following exposure pathways:

— External exposure from the radioactive plume;
— Internal exposure due to inhalation;
— External exposure from contaminated ground.

Consideration of the paths of exposure excludes intake of radionuclides into the human body through swallowing of contaminated food and water, since at a distance of some kilometers from the facility the usage of open sources for portable water supply and the production of agricultural products are excluded. Calculation results for radiation doses obtained for all the considered exposure pathways and for different diffusion categories at the site boundary (located at a distance of 500 m from the emission source) are provided in Table II–25.

TABLE II–25. INDIVIDUAL DOSE FOR THE PUBLIC AT A DISTANCE OF 500 M FROM THE EMISSION SOURCE

Diffusion categories (Atmospheric stability conditions)	Dose (μSv)
Class A	0.175
Class B	0.329
Class C	0.350
Class D	0.434
Class E	0.775
Class F	0.368

Atmospheric stability classes (a method of categorizing the stability of a region of the atmosphere in terms of the horizontal surface wind, the amount of solar radiation, and the fractional cloud cover) are as follows:

— Class A – extremely unstable conditions;
— Class B – moderately unstable conditions;
— Class C – slightly unstable conditions;
— Class D – neutral conditions (applicable to heavily overcast day or nighttime conditions);
— Class E – slightly stable conditions;
— Class F – moderately stable conditions.

For Class E, the individual exposure dose for the public reaches its maximum (see Table II–25) and is approximately 0.8 μSv.

II–4.7. Analysis of assessment results

In summary, based on the available evidence and safety analysis, the conclusion of this safety case, which is not a fully comprehensive and complete assessment, is that the waste retrieval operations can be safely undertaken and provide a solution to the hazards currently posed by the interim storage of wastes at the historical RADON-type facility. The key findings and conclusions for the safety of operations within the RADON-type facility are as follows.

II–4.7.1. Comparison with safety criteria

The results of the quantitative safety assessment for the retrieval of the waste from Vault 1 as reflected above are well within the national and international safety criteria for workers and the public.

The assessed dose for workers for normal operation is 6.2 mSv, in comparison to the dose constraint of 10 mSv/a.

The assessed dose to workers for the accident scenario is 7.0 mSv.

The assessed dose to the public for the accident scenario is 0.8 μSv, in comparison to the dose constraint of 0.1 mSv/a.

II–4.7.2. Use of the SAFRAN tool

A detailed assessment of doses arising from normal waste retrieval operations from Vault 1 of the facility is modelled using the SAFRAN tool (version 2.3.2.7) [II–3]. Selected accident scenarios are modelled using the SAFRAN tool to demonstrate application of the assessment methodology.

The given calculated example of safety assessment of the activity on RW retrieval from the RADON-type storage facilities has demonstrated the useful application of the SAFRAN tool for this purpose.

The general sequence of work performed using the SAFRAN tool included the following steps:

- Description of the facilities;
- Creation of the area structure, where works are to be performed, and parameters of exposure in work areas;
- Description of operations performed in the course of the activity;
- Establishment of the control levels of exposure, according to the national regulations;
- Description of the regulatory framework for normal and accidental situations;
- Input of personnel job positions who are assigned to perform aforesaid operations;
- Establishment of a list of impacts relevant to performed operations, and setting up time parameters for operations performance;
- Analysis of the results for the primary mode and abnormal operation mode;
- Establishment of accidental mode scenarios and calculation of the relevant EDRs by means of applying the SAFRAN tool's SAFCALC module.

Application of the SAFRAN tool allows processing of the input data, creation of the demonstrative safety assessment structure and analysis of the alternative options for personnel response actions, occurring in the course of implementation of the concerned activity – under normal operation mode, abnormal operation mode and accidents.

Certain inconveniences in application of the SAFRAN tool occur due to a lack of the possibility for simple mathematical processing of data directly inside the programme body. Therefore, in order to perform the analysis for abnormal operation modes there is the need to create separate files or additional structures inside the programme, which leads to considerable expansion of the calculation model.

II–5. MANAGEMENT OF UNCERTAINTIES

In general, historical RW are those that are generated without a complete traceable characterization programme or quality management system in place. This introduces potential uncertainty in the contents and condition of the stored wastes.

The characteristics of historical wastes in the RADON-type facility comprised the following issues:

- Unconditioned or partially treated waste;
- Poor or no information and/or traceability;
- Cannot conclusively identify the originating process and/or location;
- Mixed waste streams;
- Incomplete history;
- Incomplete or improper characterization and/or processing of the waste;
- The quality management system did not cover the whole lifetime at the time of waste generation.

During the safety assessment, some uncertainties have been identified that might impact the safety of the facility and waste retrieval activities. Sources of uncertainty were categorized as:

- Data and/or parameter uncertainties;
- Scenario uncertainties.

In order that uncertainties associated with poor characterization of the current inventory of RW do not unduly influence the results of the safety assessment, a conservative safety evaluation has been carried out. Conservative but reasonable values were used as input data and the "screening" method was used to evaluate the behaviour of the main indicators of the impact in occupational and public exposure.

Specific uncertainties identified for this Safety Case and the possible approaches to their management are described below:

- Uncertainty over the exact inventory and condition of the wastes stored in the vaults has been minimized through a thorough intrusive characterization survey. However, uncertainties remain throughout the retrieval operations.
    - A flexible retrieval methodology has been adopted to enable innovation within the bounds of the safety case. This makes use of a combination of manual and remote or semi-remote operations as the situation requires.
    - The safety case has taken a realistic and conservative approach to give flexibility during waste retrieval operations.
    - Conservative estimates of time taken to deal with each package. This allows for unexpected situations, where waste packages are in poor condition and retrieval operations are extended.
    - A variety of waste packages have been identified for use to cater for emergent situations, such as historically degraded or damaged waste packages.
    - Final characterization of the waste for consignment will be carried out as wastes are retrieved.
- Uncertainty of the radiation levels within the vaults has been minimized through surveys. However, not all areas were accessible and there might be some self-shielding by the waste items themselves.
    - Real time dose monitoring will be undertaken, with advice from the health physics advisor and the supervisor.
    - Local shielding will be available to enable operators to retreat from high dose items if required.
- Concentrations of activity in air are not known for retrieval operations.
    - Air sampling has been undertaken to establish a quiescent baseline and to enable conservative modelling in the safety assessment.
    - A local ventilation system has been designed to reduce airborne activity levels during retrieval operations. The performance parameters of the system have been overspecified to ensure flexibility to changing conditions.

In order to minimize internal dose uptake, in addition to the use of the ventilation system, operators will wear personal protective equipment.

II–5.1. Data uncertainty

Data uncertainties have been treated by obtaining data on:

— The radiation situation and radioactive contamination of the storage facility;
— The quantity and range of RW in the tanks (vaults) and their respective radiation characteristics.

The tasks of the examination were to:

— Collect and review the available initial data;
— Take photographs and videos of the storage tanks;
— Update the parameters and range of the tank contents;
— Create plans and 3-D models of the tanks, including their respective contents;
— Perform dose rate, radiometric and spectrometric measurements in the tanks;
— Identify the tank contents being the major sources of radiation and determine their spectral characteristics;
— Sample the tank contents and further measure their radioactivity;
— Process and review the measurement results.

The activities were conducted in accordance with the work programme in two stages:

— Preliminary examination which included collection of the available “Radon” data on the states and filling levels of the SRW storage tanks, measurement of hatches and monitoring of the radiation situation near these, and determination of the accessibility of the vaults through the existing hatches;
— Main examination.

The summary of waste examination of the RADON-type facility is presented in Section II–3.3.2.

II–5.2. Scenario uncertainty

A scenario is a postulated or assumed set of conditions and/or events. They are most commonly used in analysis or assessment to represent possible future conditions and/or events to be modelled, such as possible accidents at a nuclear facility, or the possible future evolution of a disposal facility and its surroundings.

This safety assessment of waste retrieval is concerned with the impact of the waste on workers over the period of retrieval operations.

Incomplete knowledge about the current status of historical waste packages is a major source of uncertainty in this safety assessment. For example, some of the gamma-ray source blocks in the RADON-type facility have lost their collimator’s plugs and eyebolts. Handling of such blocks can lead to an accident situation with dropping-out of the gamma-ray source from the block and exposure of workers.

Scenario uncertainties have been treated by assessing doses to workers through different scenarios described in Section II–4.3.3.

II–6. ITERATION AND OPTIMIZATION

The evaluations of the waste retrieval techniques and the safety assessment have been undertaken with the best available data and applying expert judgement.

Nevertheless, aspects such as the following could result in the need for iteration of the safety case:

- New data about the waste becomes available as waste retrieval operations progress.
- The unforeseen issues might be identified during waste retrieval, such as performance issues with remote equipment.

Specific actions to optimize dose protection can be summarized as follows:

- Use of ventilation systems to minimize airborne activity, resulting in lower doses to workers and the public;
- Providing shielded areas for workers to reduce doses during retrieval operations;
- Removal of wastes with high radiation doses from the vaults first, to reduce general background doses;
- Use of remote or semi-remote tools to assist in retrieval activities;
- Real-time dose monitoring to assist operators and supervisors in decision making;
- Use of appropriately shielded waste containers to receive the waste;
- Timeliness (i.e. adherence to schedules) in carrying out waste retrieval operations.

II–7. IDENTIFICATION OF SAFETY MEASURES

The assessment undertaken indicates that if the retrieval operations are implemented according to the provisions set out in this safety case, it will comply with the required national safety standards and meet the relevant dose limitation criteria with respect to workers and members of the public. The assessment has been carried out using conservative assumptions and straightforward methodology.

The hangar structures itself does not provide any significant shielding. However, localized shielding will be in place to reduce doses to workers.

Inspection and maintenance programmes will be in place for the installed and portable equipment and a management system providing for trained personnel, formalized procedures, records, reports and an assurance regime over all aspects important to safety and security will also be established.

II–8. LIMITS, CONTROLS AND CONDITIONS

According to national regulations and guidance documents, a set of limiting conditions and controls will be implemented to ensure safety of operations, as follows:

- The current assessment assumes that no more than 185 m^3 of solid RW are to be retrieved from Vaults 1 to 4.
- Waste retrieval operations are only undertaken with the use of a local ventilation system operating at the vault, to ensure that airborne activity concentrations are minimized.
- The maximum dose rate assumed in the SAFRAN model for manual intrusive work is 1 mSv/h at 1 m distance from the source. Items with dose rates higher than this would require use of additional localized shielding or semi-remote retrieval operations to ensure that total doses are optimized.

All waste will be packaged and recorded in compliance with the waste acceptance criteria of the receiving waste management facility.

II–9. CONCLUSIONS

As stated earlier in this annex, a fully comprehensive safety assessment has not been performed in this illustrative version of the safety case. However, the results of the quantitative safety assessment for the retrieval of the waste from Vault 1 as reflected above are well within the national and international safety criteria for workers and the public. Therefore, based on the available evidence and safety analysis, the conclusion of this safety case is that the waste retrieval operations can be safely undertaken and provide a solution to the hazards currently posed by the historical wastes emplaced in the RADON-type facility.

The assessed dose to workers during normal operation is 6.2 mSv, in comparison to the dose constraint of 10 mSv/a.

The assessed dose to workers as a result of the accident scenario is 7.0 mSv.

The assessed dose to the public as a result of the accident scenario is 0.8 μSv, in comparison to the dose constraint of 0.1 mSv/a.

The results of the quantitative safety assessment are well within the national and international safety criteria for workers and the public. The safety case for the operations is supported by a formal plan to address identified unresolved issues.

Evaluation of uncertainties has been undertaken and each area of uncertainty has been managed appropriately. A list of key unresolved issues has been identified and planned steps to resolve them have been cited.

The key findings and conclusions for the safety of waste retrieval operations are as follows:

— **Strategy:** The facility and its associated activities to retrieve, package and dispatch the waste is in line with the national policy and strategy.
— **Facility design and engineering:** A simple approach has been taken to design and engineer a waste retrieval facility, with a balance of engineered and operational safety measures appropriate to the hazards. All engineering features are anticipated to perform their operational and safety functions adequately. A key item of equipment is the local ventilation system, which will maintain low levels of airborne activity within the facility.
— **Facility operation:** A detailed series of activities has been described and assessed for waste retrieval operations. Implementation of these operations is expected to result in safe retrieval of the wastes.
— **Optimization of protection:** A series of optimization measures have been identified in Section 6, which will result in doses that are as low as reasonably practicable to workers and the public.
— **Waste management practice:** Good waste management practice is generally evident from the intent of the legal framework, organizational arrangements and defined responsibilities.
— **Integrated management system:** Although some management systems and procedures have been implemented, further development of the management system is required. Management of unresolved issues as covered above addresses recommendations regarding the development of an integrated management system.
— **Uncertainties:** Uncertainties have been identified and mitigating actions put in place, either in the assessment itself or in the subsequent waste retrieval activities.

PLANS FOR ADDRESSING UNRESOLVED ISSUES

The safety case indicates some information gaps that need to be addressed before it will be regarded as a document that can be submitted to the regulatory authority for review and approval.

The identified aspects requiring further clarification with commensurate management recommendations and actions are described in Table II–26.

TABLE II–26. ASPECTS REQUIRING FURTHER CLARIFICATION

Item	Aspects requiring clarification	Recommendation/Action
1. Legal and regulatory framework		
1.1	None identified.	
2. Basic engineering analysis		
2.1	None identified.	
3. Optimization of protection		
3.1	Optimization for exposures related to normal operation.	Development and implementation of a formal operational optimization programme where actual doses are measured, and specific reduction strategies are considered and implemented during waste retrieval activities.
3.2	Identification of hazards, hazard screening and full hazard assessment for anticipated operational occurrences and accidents has not been carried out in this illustrative safety case. Nominal scenarios have been assessed as examples only.	If this safety case is to be used in a real application, full safety assessment needs to be completed.
4. Non-radiological hazards		
4.1	Comprehensive assessment of non-radiological hazards has not been carried out.	Plan, schedule and conduct a comprehensive non-radiological hazard assessment.
5. Implemented waste management practice		
5.1	None identified.	
6. Integrated management system		
6.1	Detailed supporting information on the Management system is not currently referenced.	Provide details of the management system and safety culture in the safety case.
7. Management of uncertainties		
7.1	None identified.	
8. Facility specific limits and conditions		
8.1	None identified.	

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LIST OF ABBREVIATIONS

ALARA	as low as reasonably achievable
ARAO	Slovenian Agency for Radioactive Waste Management
CRAFT	Complementary Safety Reports: Development and Application to Waste Management Facilities
CSF	central storage facility
DSRS	disused sealed radioactive sources
EDR	exposure dose rate
HAZOP	hazard and operability study
IAEA	International Atomic Energy Agency
IJS	Institute Jožef Stefan
LILW	low and intermediate level radioactive waste
PIE	postulated initiating event
ROSTECHNADZOR	Federal Environmental, Industrial and Nuclear Supervision Service of Russia
RW	radioactive waste
RWSF	radioactive waste storage facility
SADRWMS	International Project on Safety Assessment Driving Radioactive Waste Management Solutions
SAFRAN	Methodology and Safety Assessment Framework Tool
SNSA	Slovenian Nuclear Safety Administration
SRW	solid radioactive waste
SSCs	structures, systems and components
WAC	waste acceptance criteria
WASSC	Waste Safety Standards Committee

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Zakaria, E.	Egyptian Atomic Energy Authority, Egypt
Zeroual, S.	Centre National de Radioprotection, Morocco

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