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CONTRACT B7-5350/99/6141/MAR/C2

EVALUATION OF MANAGEMENT ROUTES FOR THE PALDISKI SARCOPHAGI

SECOND INTERMEDIATE REPORT

TASK 2 – DRAWING UP OF DISMANTLING STRATEGIES

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TA-215194 Ind. B

Version of : 03/07/01

A. SUMMARY (AND/OR MAIN CONCLUSIONS) :

Contract Number :	B7-5350/99/6141/MAR/C2
Title :	Evaluation of management routes for the Paldiski sarcophagi
Contractor :	TECHNICATOME – BNFL
Subcontractors :	AS ALARA Ltd – LI VNIPIET Institute – NUKEM Nuklear GmBH
Objectives and scope of this project :	Task 1 - Data Collection and analysis Task 2 - Drawing up of potential dismantling strategies Task 3 - Evaluation of dismantling strategies
Progress of work to date :	Completion of Task 2
Period covered :	January 2000 to February 2001
Objectives and scope of Task 2 :	Task 2 – Drawing up of dismantling strategies : – removal of the reactor compartments followed by their final disposal in a near-surface facility or in deep geological formation, – decontamination/cutting of the reactor primary systems into small pieces, and disposal of the resulting radioactive wastes, – melting of the metallic parts of the reactors with the view to volume reduction and/or recycling.

The aim of this contract is to identify feasible rational routes for the decommissioning of two nuclear power units of the former RF Navy Training Centre on the Pakri peninsula in Estonia. The main purpose of task 2 of this project is to draw up dismantling strategies in order to reach the third decommissioning level defined by IAEA, after a possible storage period in order to lower the remaining activity.

The applying regulations are the in progress Estonian regulations regarding radiation protection and radioactive waste management, and also the IAEA recommendations. An overview of the development of Estonian regulations regarding radiation protection and radioactive waste management is given this report.

On the basis of a comparison between :
- the description of the storage conditions of the decommissioned French nuclear reactors, and compartments,
- the data about the Paldiski Reactor Compartments collected in the course of the first task of the project,

an evaluation of the current storage conditions of the Paldiski reactor compartments is given, highlighting deficiencies as far as safety is concerned.

The conclusion of this evaluation is that the differences are significant regarding the risk of radioactivity dispersion. At Paldiski :
- The fire risk is not minimised.
- The provisions that have been made against humidity effects regarding primary circuit and reactor compartment corrosion risk are not sufficient.
- There's no way to check up the efficiency of radioactivity confinement provisions.
- Although support structures of the RCs were reinforced to resist against earthquake, it seems that no arrangements have been made against flood risk.
- Due to the presence of miscellaneous radioactive waste in the reactor compartments, radioactivity is not confined into the primary circuit.

According to European safety standards radioactivity remaining in RC must be totally enclosed in two successive confinement barriers. On the basis of the previous paragraphs, two suitable confinement barriers have been selected for the Paldiski reactors :
- it is impossible to consider Paldiski reactors primary circuits as reliable confinement barriers.
- RCs shell could be a reliable first confinement barrier for a few years, but at the present time, it's not possible to guarantee the reliability of this barrier for 50 years. However, we strongly think that the efficiency of this confinement barrier could be guaranteed for 50 years if a few complementary safety provisions are implemented.
- Sarcophagi could be the second confinement barrier, but it will be indispensable to implement complementary safety provisions

From the dismantling strategies that were devised in the beginning of the project, and from the above evaluation of the Paldiski Reactor compartments storage conditions, suitable dismantling options have been selected :

- **Strategy #1** : **Final disposal of the RCs as a whole**
  - Option #1 : in situ, in the sarcophagi, avoiding RC removal operations.
  - Option #2 : in a near surface disposal facility located on the Paldiski site.
- **Strategy #2:** **Full dismantling of the RCs**
  - Option #1 : Minimising cutting works in order to lower men exposure.
  - Option #2 : Decontamination and cutting components into small pieces in order to sort wastes, in view to reduce the resulting waste volume (using or not melting devices).

As waiting until the year 2050 could result in decrease up to one thousand times of gamma dose rates, and as men exposure is significant regarding the second decommissioning strategy, this strategy will be considered for operations performed after 50 years or after 100 years. The first decommissioning strategy will be considered for operations performed immediately or after 50 years.

1. <u>First decommissioning strategy option 1 : Final disposal of the RCs as a whole in situ, in the sarcophagi, avoiding RC removal operations</u>

The works to be carried out in order to transform the sarcophagi in small final disposal sites, in accordance with Estonian regulations and IAEA recommendationss are summarised below :

- Soil, geological, geotectonic and hydrological surveys on the possibility of permanent in-situ disposal
- Fitting out RCs with Fire extinguishing systems
- Fitting out RCs and sarcophagi with an air conditioning system in order to avoid confinement barriers corrosion,
- Filling the primary circuit with concrete, in order to stabilise waste and immobilise radionuclides
- Sarcophagi strengthening :
  - raft and earth works to ensure stability of the building, and to provide against flood,
  - works to reinforce sarcophagi superstructure,
  - waterproofing sarcophagi and improving containment capabilities,

Implementing a surveillance program in order to check the efficiency of the above measures. This surveillance program of Reactor Compartments and Sarcophagi should be in force for a minimal period of 300 years.

Regarding the problem related with the presence of small radioactive sources in RC#1, we think that as each source contains a rather little radioactivity of its own, it could be possible to extract them if necessary after 50 years, when the radiation dose rate in the reactor compartments is lower.

2. <u>First decommissioning strategy option 2 : Final disposal of the RCs as a whole in a near surface disposal facility located on the Paldiski site</u>

This decommissioning option rests on the following steps :

- Carrying out preliminary works in the RCs
- Erecting a new building on Paldiski site.
- Building a special heavy-duty route between the sarcophagi and the new building.
- Transferring the RCs as a whole from the sarcophagi to the new building.

The preliminary works to be carried out in the RCs consists in filling the primary circuit with concrete, in order to stabilise waste and to provide against corrosion risks, and to extract as much as possible most burnable materials (like rags, plastic, wood etc ..)

The new disposal building would have to be built as an extension (North side) of the building #301/302, to reduce the length of the heavy-duty way as much as possible. This building must be designed for both reactor compartments disposal, according to the IAEA requirements for the design of near surface repository for low and intermediate level waste.

After erecting a new disposal building, a heavy-duty route is to be constructed between the sarcophagi and the new building. This route could be designed for railway or wheel transfer.

Transferring the RCs in their entirety will be a difficult operation (each one weights about 1000 tons), that has been studied by VNIPIET institute. This work could be performed using hydraulic jacks to lift the reactor compartment and to lean it on a slipway trolley introduced in the free space under the reactor compartments. Reactor Compartment #2 would be transferred first to the disposal building. After that, Sarcophagus #2 would be completely dismantled, and the heavy-duty road would be prolonged towards Sarcophagus #1.

The Reactor Compartments transfer as described above is not fully consistent to the radioactive waste transport regulation due to their total enclosed activity, but as the waste packages are transported on tens of meters with no exit of the Paldiski site, we think that this option could be reasonable. This option is the one that was selected by TECHNICATOME to transfer in 1993 the reactor compartment of the first French nuclear submarine called Le Redoutable to the building dedicated to its temporary storage.

As for option 1, we suggest to postpone the extraction of small radioactive sources after a 50 years storage period, when the radiation dose rate in the reactor compartments is lower.

The surveillance program of Reactor Compartments and Disposal building should be in force for a minimal period of 300 years.

It appears that waiting fifty years before implementing this first decommissioning strategy brings no significant advantage, but implies additional cost to guarantee the reliability of the confinement barriers for the whole storage period. As a consequence, we recommend that this first decommissioning strategy would be implemented as soon as possible.

3. Second decommissioning strategy : Full dismantling of the RCs

The framework can be resumed as follows :

- Restoration of standardised storage conditions for the storage period :
  - Improvement of RCs resistance against corrosion.
  - Sarcophagi strengthening and improvement of sarcophagi confinement properties.
  - Improvement of sarcophagi resistance against flood.
  - Implementing the surveillance program.

- During the storage period, the following works should be performed :
  - Building of the Estonian radioactive waste surface storage site.
  - Transfer of the waste already stored in building #301/302 to this storage site.
  - Building of a “packaging workshop” in building #301/302.
  - Upgrading of the 50 tons crane lift of this building.

- Complete dismantling of the RCs into pieces to be transferred to the “packaging workshop” : usable techniques for operations are detailed, and schematic flowcharts of dismantling operations are provided.

- Packaging of the arising radioactive waste. Two different options will be considered for radioactive waste packaging:

  - ❑ Option # 1: Making special waste packages minimising cutting works.
    Special waste packages are prepared in the “packaging workshop” in accordance with the storage rules. These packages are made from whole NPU systems like reactor vessel, steam generator vessels, etc , minimising cutting works. The final volume of waste might be high, but the dismantling and packaging operations are simplified and men exposure minimised.

- Option # 2: Minimising the volume of definitive wastes.
  The aim of this packaging option is to minimise the final volume of waste, using techniques like decontamination, compaction, recycling by the mean of melting devices, etc. Most of the additional work is done in the “packaging workshop”.
  The study of this option considers the use of In–Situ decontamination. This method considers the use of further decontamination to re-categorise waste and melting to further reduce the volume of waste for disposal. Given the current state of information it could be possible to carry out In-Situ decontamination of the reactor coolant circuits to assist with their decommissioning. However this would involve the need to develop safety cases for its use along with the design of significant capital equipment. Lastly there would be the requirement to treat the liquid effluents resulting from the decontamination process used. The use of melting of the resulting waste as a means of achieving free release and or to reduce waste volumes and disposal costs has been discussed. Whether melting is a cost effective approach will depend on many factors and an economic evaluation will need to be carried out to establish the answer. Experience in other European countries gives some indication that it may be a cost-effective approach, though the whole picture needs to be taken into account.

– Release of the site after dismantling and decontamination of the sarcophagi.

We think that it’s possible to guarantee the reliability of the confinement barriers for 50 years provided that some complementary safety provisions be implemented as soon as possible. These provisions are briefly listed below (see § 6.2 for details) :
– Improvement of RC resistance against corrosion,
– Sarcophagi strengthening and improvement of sarcophagi confinement provisions,
– Improvement of sarcophagi resistance against flood,
– Implementation of a surveillance program.

But if it comes to extend the storage period up to 100 years or more, the works to be carried out would be similar to those described as First Strategy – Option 1 (Final disposal of the RC in their sarcopagi). These works are much heavier and much more expensive. Moreover, the gamma dose rate decrease will be much slower after 50 years, and the influence of the storage period might not be really significant either regarding waste volumes. For these reasons, we recommend that this second decommissioning strategy would be implemented after a storage period of 50 years.

The main advantages and drawbacks of each decommissioning option have been studied and synthesised in the following table.

The next task of this project (task 3), is aimed to evaluate these decommissioning options on the basis of the following criteria :
– cost estimation,
– radiological impact evaluation,
in view to recommend the best management route for the Paldiski sarcophagi.

Decommissioning Strategy	Advantages	Drawbacks
First Decommissioning Strategy Disposal of the RCs as a whole	– Except radioactive sources extraction, the works to be carried out would not imply high men exposure. – The risk of radioactivity release into the environment is reasonably low.	– Waste packages (RCs) are not consistent to the IAEA recommendations regarding waste management – Waste packages are not totally immobilized into the final disposal.
Decommissioning Option 1 In situ disposal in the sarcophagi	– This option does not require heavy dismantling works. – As a consequence, the global cost would be quite low. – No radioactive waste transport is required.	
Decommissioning Option 2 On-site near surface disposal at Paldiski	– Reactor compartments transfer to the disposal building could be considered as a radioactive waste transport, but this radioactive waste (the reactor compartments) remains on the Paldiski site.	– This decommissioning option requires heavy works to transfer the RCs into the disposal building. – As a consequence, the global cost would be much higher than for option #1. – The Reactor Compartments transfer is not fully consistent to the radioactive waste transport regulations due to their total enclosed activity, but as the waste packages are transported on tens of meters with no exit of the Paldiski site, this option seems to be reasonable.
Second Decommissioning Strategy Full Dismantling of the RCs	– Waste packages are consistent to the IAEA recommendations regarding waste management. – Waste package (parallelepipeds) will be totally immobilised into the final disposal. – Waste packages transport to the disposal site is fully consistent with Estonian regulations and IAEA recommendations regarding radioactive waste transport, except for RC#1 reactor vessel (but as explained in §6.6.1, RC#1 reactor vessel transport could be reasonably performed, providing eventually some complementary safety measures)	– This decommissioning strategy requires heavy works to dismantle the Reactor compartments. These works imply a very high labour, and a quite high total men exposure. – As a consequence, the global cost would be higher than for first strategy. – The risk of radioactivity release into the environment is reasonably low, but higher than for the first strategy.
Packaging Option 1 Disposal of big components as specific waste packages	– Total men exposure is lower than for packaging option #2	– The resulting waste volume is higher than for packaging option #2
Packaging Option 2 Minimising the waste volume by decontamination, recycling, etc ..	– The resulting waste volume is lower than for packaging option #1	– Total men exposure is higher than for packaging option #1

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B. ATTRIBUTES :

Nature of Doc : **PRO**	Manual (P1) **DD**	Process (U1) : **DEMANTELEMENT**
Confidentiality : **NP**	WBS (P2) **PAL**	Free (U2) :
Activity :	Work Package **SDPTP200**	Free (U3) :
Related works : **PAL**	NTA (P4)	Key Words(U4): **CAD**
Tech. Value :		
File reference :		
Record duration : **L**	Record ref **PAL**	Origin : **DE/SMCI/AI**
Company	**Project**	**Issuing Unit**

C. INPUT DOCUMENTS USED TO ELABORATE PRESENT DOCUMENT:

Guide	Issuer	MAIDOC Reference	Rev	Date	External Reference	Title
<1>	*EUROPEAN COMMISSION DG XI*	*EXT-11488*		*17/05/99*	B7-5350/99/6141/MAR/C2	Evaluation of management routes for the Paldiski sarcophagi – Study contract

D. MODE OF DISTRIBUTION :

External distribution :

- ☒ *covering letter*
- ☐ *dispatch note*
- ☐ *standard distribution*

Internal distribution :

- ☐ *standard distribution*
- ☐ *see underneath*
- ☐ *chrono issuing*
- ☐ *copy for Archives (for L or M duration) and issuing Units*

E. DESCRIPTION OF REVISIONS (AND/OR HISTORY OF DOCUMENT) :

List of writers, verifiers, approving of present document

Rev	Writer	Verifier	Approving
A	BNFL : W. ELDER TA : M. ROMESTAN TA : L. ANTONEL	BNFL : J. MILLER TA : B. ROBIN	TA : L. ANTONEL
B	TA : L. ANTONEL	TA : B. ROBIN	TA : L. ANTONEL

Indicate main changes made to document :

Rev A : First issue.
Rev B : Update taking into account the comments made at the occasion of the third progress meeting that took place in Paris, 7-8 June 2001.

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Draft BNFL Engineering Report F0298 C JUNE 2000

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List of abbreviations

ACP	Auxiliary Coolant Pump
AF	Activity Filter
CACP	Cooler of Auxiliary Coolant Pump
CDB	Central Design Bureau
CMCP	Cooler of Main Coolant Pump
CPS	reactor Control Protection System
ERPC	Estonian Regulation and Protection Center
IWP	Iron-Water Protection
LWSB	Liquid Waste Storage Building
LWTF	Liquid Waste Treatment Facility
MCP	Main Coolant Pump
MTB	Main Technological Building
MWP	Metal-Water Protection
NPP	Nuclear Power Plant
NPU	Nuclear Powered Unit
NS/M	Nuclear Submarine
NSSS	Nuclear Steam Supply System
PIERG	Paldiski International Expert Reference Group
SG	Steam Generator
SPP	Steam Producing Plant
RC	Reactor Compartment

Note :
The plans and drawings included in appendices were reduced in order to be printed on paper of standard format. All the digital files are available on the attached CD-ROM. When drawings were implemented with AUTOCAD, the AUTOCAD files are also available on the attached CD-ROM.

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1 GENERAL

1.1 BACKGROUND

The necessity to develop the submarine fleet in Russia required constructing a special training base for a preliminary training of submarine crews made most realistic in conditions. To this purpose two prototypes of nuclear power units (NPU), close analogous of NPU installed at nuclear submarines (NS/M) were constructed and commissioned in the sixties on the Navy training centre's base located on the Pakri peninsula in the town of Paldiski (Estonia).

According to an agreement between Government of the Russian Federation and Government of Estonian Republic of July 30, 1994, for transfer of this training facility of Navy Training Centre located in Pakri Peninsula (town of Paldiski), with laid-up nuclear reactors and nuclear waste storage facilities to the ownership of the Estonian Republic, nuclear fuel was discharged from the reactors and transported to Russia while the reactors themselves were prepared for prolonged storage.

A number of uncertainties remain concerning the way the Russians carried out this enclosure process and consequently in predicting dismantling operations.

The other site buildings consist mainly of a liquid waste treatment plant, a decontamination plant, liquid and waste stores, a radiochemical laboratory, each of them being in poor shape. Important work is being done through PIERG (Paldiski International Expert Reference Group) members to clean up the site and to start building new waste packaging and storage facilities.

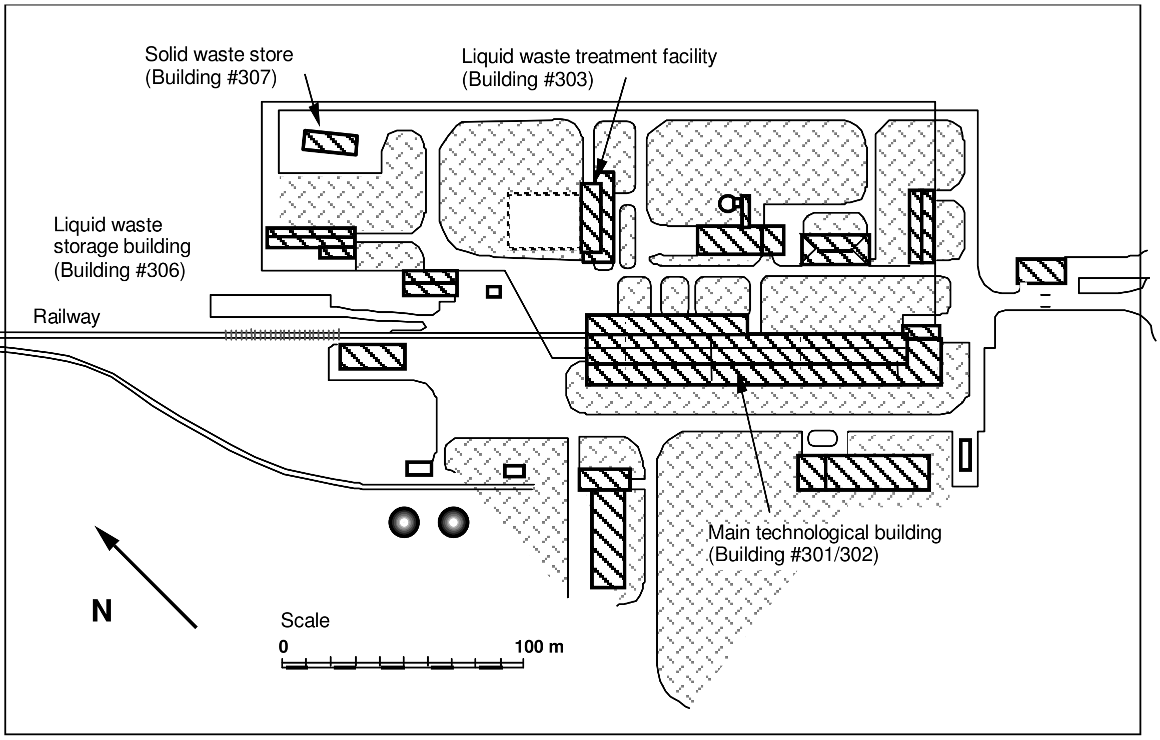

Figure 1 - Paldiski site plan

1.2 PRESENTATION OF THE PROJECT

The main purpose of this contract is to identify feasible rational routes of dismantling and complete removal of radioactive components from two nuclear power units of the former RF Navy Training Centre on the Pakri peninsula in Estonia.

This study will be made up of three different tasks:

- Task 1 : Data collection analysis
- Task 2 : Drawing up of potential Dismantling strategies
- Task 3 : Evaluation of Dismantling options

This report is the second intermediate report of the study: it presents the results of the second task of the project.

1.3 INVOLVED PARTIES

The two companies TECHNICATOME and BNFL have joined to perform this study, with the main aim to define the best dismantling route for the Paldiski sarcophagi, regarding cost and radiological impact on the environment, in accordance with IAEA safety recommendations and Estonian radioactive waste management regulations. TECHNICATOME has been the leader of the contract.

Two subcontractors have been involved in this study :

- The Russian design institute VNIPIET, which designed the complex of buildings and premises in which the power stands were located and, later on, the general concept of NPU decommissioning and sarcophagi design,
- The Estonian Waste Management Agency AS ALARA, which is responsible for the site,

Several Russian design and development organisations were involved in designing the training power stands including :

- the engineering bureau CDB ME «Rubin», which was involved in preparation of stands 346A and 346B for prolonged storage,
- the research and development institute RDIPE, which designed the 346A unit,
- the engineering machine-building bureau OKBM, which designed the 346B unit.

All these organisations were involved in this project as VNIPIET's subcontractors.

1.4 TASK 2 : DRAWING UP OF DISMANTLING STRATEGIES

The main purpose of task 2 of this project is to draw up dismantling strategies in order to reach the third decommissioning level defined by IAEA (as far as possible), after a possible storage period in order to lower the remaining activity. These three decommissioning levels are the followings :

- level 1 : the reactor is defueled, circuits are drained and confinement barriers are maintained,
- level 2 : the reactor compartment is taken out of the submarine, size reduced to a minimum, and air-tightness of the compartment maximised, simplifying reactor monitoring.
- level 3 : all radioactive components have been removed – radiation monitoring and inspection is no longer required.

Three main dismantling options will be considered :

- removal of the reactor compartments from the sarcophagi followed by their final disposal in a near-surface facility or in deep geological formation according to the waste acceptance criteria defined by the Estonian authorities for radioactive waste disposal,

- decontamination/cutting of the reactor primary systems into small pieces, in order to facilitate further handling and disposal of the resulting radioactive wastes,

- melting of the metallic parts of the reactors with the view to volume reduction and/or recycling, provided that criteria for conditional or unconditional recycling of the resulting ingots are met.

For each dismantling option, the dismantling sequences and dismantling process are detailed, the volumes of the different waste types and effluents generated are estimated, and the required equipment and buildings is defined, according to the Estonian waste management regulations and to the IAEA recommendations.

Each strategy is to be optimized in terms of cost and workforce dose taking into account a storage period of 10, 50 or even 100 years before starting operations.

On the basis of this report, we will be able to evaluate in task 3 the different dismantling options considering cost and radiological impact criteria, in order to select the most convenient strategy.

2 MAIN BOUNDARY CONDITIONS FOR THIS STUDY

Currently an interim radioactive waste storage is located in the building #302, north of the sarcophagi #2 with reactor 346B, and a waste treatment and packaging facilities in the eastern annex of building #302. This interim waste storage is in operation since 1997.

No time limitation has been decided for the use of the interim storage, but the containers will have to be transferred sooner or later to a final disposal site.

Another project is to be launched to define the criterion in order to choose a suitable site and to decide what type of repository would be constructed. Six possible locations have been identified.

These possible locations were chosen because of suitable geological layers. They are all located along Estonia northern cost. Paldiski is one of these alternatives, but for geological reasons, Pakri peninsula might not be the best place to set up a final disposal site.

Anyway, the repository must be available before starting sarcophagi and reactor compartments dismantling operations. In the other case, the cost of waste management would be much higher, because we would have firstly to store the waste packages in the interim storage, and then to transfer them later to the repository.

So we will assume that the future final repository in Estonia for the decommissioning waste will be available before starting dismantling works.

3 ESTONIAN LEGISLATIVE ACTS IN RADIOACTIVE WASTE MANAGEMENT

At present a fast development of national radiation protection and radioactive waste management infrastructures is in progress. The objective of this development is to enact internationally accepted principles, criteria, provisions and their implementation procedures for an efficient regulation of radiation safety issues, including practices involving radioactive waste management and decommissioning. An overview of this development is given below (from reference <4>).

3.1 RADIATION ACT

In 1997 the principal legal instrument of the radiation protection infrastructure, the Radiation Act, was passed by the Parliament and its amendment was accepted in 1998 [4.1]. The purpose of the Radiation Act is to protect people and the environment against the harmful effects of radiation. The Act bases on the concepts, terms and limits laid down by the International Basic Safety Standards (BSS) [4.23] and by the Directive 96/29/EURATOM [4.24]. Both above standards have been extensively used as models in drafting the Radiation Act. The basic principles of these international documents, e.g. justification of practices, optimisation of protection and safety (ALARA), limitation of individual doses, adoption of justified and optimised interventions, authorisation of radiation practices, the primary responsibility of a undertaking (licensee), an equal basis for consideration of occupational exposures caused by artificial or natural sources, etc., are explicitly formulated as provisions of the Act. The exemption criteria as well as the dose limits are also fully adopted.

The Act specifies also a governmental regulatory authority, Estonian Radiation Protection Centre (ERPC), and its responsibilities. ERPC is authorised to issue licences for radiation practices, including for decommissioning of nuclear facilities, to provide additional licence conditions and guidance to the licensees and to execute surveillance.

The Act defines radioactive materials and components produced in the process of decommissioning of nuclear facilities or installations as radioactive waste, and lays down general radiation safety provisions for management, transport, export and import of these wastes.
The Act consists of 36 paragraphs in 6 chapters. A brief review of the items covered in the Act is given below :

– Ch. 1 General Provisions (purpose and scope of the Act, terms and principles for acceptable practices and interventions, authorisation of the ERPC for the enforcement of the requirements).

- Ch. 2 Requirements to the Practices (licence requirements and exemptions, obligations of the licensees, cancellation of licences).
- Ch. 3 Exposures (dose limits for occupational and public exposure, requirements for natural exposure, obligations of workers, dose register, age limit, medical surveillance of workers, assessment of public exposures, requirements for medical exposure and emergency exposure). Dose limit values as the basic regulatory safety criteria are explicitly included in §13 and §19 of the Act (see Table 1).

Table1. Effective and equivalent dose limits for occupational and public exposure

Doses	Occupational exposure (mSv/a)	Training of apprenties (mSv/a)	Public exposure (mSv/a)
Effective dose			
5 year average	**20**	**6**	**1**
any single year	**50**		**5**
Equivalent dose			
lens of the eye	**150**	**50**	**15**
skin and extremities	**500**	**150**	**50**

- Ch. 4 Radiation Sources (safety of sources, installation and repair of sources, type approval, transportation of radioactive substances and wastes).

- Ch. 5 Radioactive Wastes (management principles and requirements for, transfer and restriction, export requirements, import prohibition for disposal).

- Ch. 6 Final Provisions (supervision, implementation of the Act, sanctions for legal persons for breach of the provisions).

During the last two years more than 10 regulations of the Government and those of the Ministers under the Act (specifying its requirements) have been accepted [4.2-4.12]. A number of provisions concerning some general aspects of radioactive waste management and decommissioning, incl. non-radiological aspects, are also addressed in the other enforced legal acts or international conventions, e.g., in Environmental Monitoring Act, Emergency Situation Act, Occupational Safety Act, etc [see, e.g., 4.13-4.19].

Two regulations on guidelines for implementing environmental impact assessment are also applicable for non-radiological environmental aspects of decommissioning in Estonia [4.20,4.21]. These regulations also address the important role of local authorities in the process of environmental impact assessment.

Nevertheless, a number of aspects concerning waste management and decommissioning are not yet sufficiently regulated. Rules for the release of materials, buildings and land for reuse, clear classification criteria of radioactive wastes, rules for waste treatment, storage or disposal, are only a few examples of these non-existing regulations. Decommissioning projects as such are not mentioned as a licensable activity in the Radiation Act.

The gaps in national legislation have considerably hampered the practical decommissioning activities at the Paldiski and the Sillamäe sites. In these cases the operator has applied international good-practice procedures and recommendations. In a few occasions, a case-by-case approach has been applied by the Government or by the Ministers.

At present, there is no specific program document or a conceptual action plan declaring the Estonian policy in radioactive waste management, incl. disposal and decommissioning of former nuclear or nuclear cycle facilities. Nevertheless, the preparation and adoption of the national radiation safety action plan is a requirement in the existing legislation (§4 of the Radiation Act). Hopefully a draft of this action plan will be completed and presented to the Government for adoption by the end of 2000. The authorisation of the document should specify necessary general principles and measures to ensure radiation safety, including also the identification for options of radioactive waste disposal, decommissioning as well as the requirements for their achievement and the relevant funding policies.

Chapter 5 of the Act establishes generic provisions and general framework for radioactive waste management. In the following, the English translation of the Act prepared by the Estonian Translation and Legislative Support Centre, Tallinn, is used. Paragraph 28 (Definition of radioactive waste) defines radioactive waste as: "..

1) substances and objects which contain or are contaminated with radioactive substances in which the quantity of the radioactive substances is larger than the limits established in § 6 of this Act and which are not intended to be used in the future;
2) radioactive substances or radiation generators containing radioactive substances the owner of which cannot be established;
3) radioactive substances or materials contaminated with radioactive substances produced in nuclear installations or radioactive components of decommissioned nuclear installations".

The qualities of both unofficial and official translations of the Act into English are not fully satisfactory and, in several cases, even misleading. E.g., in the following, *"the handling of radioactive waste"* has been replaced by "*radioactive waste management*".
In § 29 of the Act, general requirements for *radioactive waste management* are established. Among others the following general provisions are established: "..

1) In radioactive waste management, a person holding a radiation practice licence is responsible for radiation safety.
2) If a person holding a radiation practice licence fails to comply with the requirements for radioactive waste management, the requirements shall be fulfilled at the expense of the person by way of public procurement.
3) If the owner of radioactive waste is not known or it is impossible to establish the person who is responsible for the production of radioactive waste, the waste is managed at the expense of the State.
4) If acquisition or taking possession of a radiation source or radioactive waste is contrary to this Act, the radiation source or radioactive waste shall be transferred to an agency engaged in radioactive waste management.
5) The State administers the disposal of radioactive waste through permanent emplacement thereof in special repositories".

The same paragraph authorises the Minister of the Environment to establish the procedure for radioactive waste management. According to this requirement, the corresponding regulation was issued in 1998 [4.2].

The next paragraph (§ 30) establishes restrictions on transfer and receipt of radioactive waste and authorises the Minister of the Environment to issue the procedure, pursuant to which radioactive waste is transferred (sub-paragraph 1): "..

1) The recipient of radioactive waste must hold a radiation practice license for practice involving radioactive waste management.
2) The recipient of radioactive waste is required to prove the legality of possession of the radioactive waste at the request of a state agency with corresponding authority.
3) The state takes possession of radioactive substances or radioactive waste without charge if they have been acquired unlawfully, or if they are used in a manner dangerous to human health or the environment".

In the case of radioactive waste transferred into the possession of the state, the Minister of the Environment shall decide on their further management. Sub-paragraph 1 of § 31 (Exportation of radioactive waste) of the Act states that "a permit is required for the exportation of radioactive waste …" and authorises the Government of the Republic by a regulation to establish the issuing authority and the procedure for issue of such permits. The next sub-paragraph introduces the provisions of the EU legislation on export prohibitions into: "..

1) regions located to the south of the sixty degrees of south latitude;
2) the states which have not joined the European Union but have entered into a corresponding agreement with the European Union;
3) states, the laws of which prohibit the importation of radioactive waste;
4) states, within which there is reason to believe that safe handling of such waste is not possible".

The Act (§ 32) also prohibits the importation of radioactive waste into Estonia for their disposal.

3.2 REGULATIONS

3.2.1 Radioactive waste management

Up to now, no categories of radioactive waste, clearance levels and release limits have been established in Estonia. According to the Radiation Act and the presently accepted regulations, the established exemption levels coincide with those of clearance, actually - of unconditional clearance. The regulation on monitoring and assessment of population doses [4.3] establishes the well-known international generic clearance levels of 1 *manSv* (annual collective dose) and 10 μSv (annual individual dose) as well as the procedure for a case-by-case clearance basing on analytical surveys and assessments. It is considered, however, that the procedure should undergo a significant improvement for the clearance of large quantities of materials produced during decommissioning operations. At present a draft regulation on release criteria, on clearance levels, on the classification of waste, on the waste acceptance criteria as well as on the order of their implementation, is under preparation. The draft regulation will include the basic international and the recent EC recommendations in the field of radioactive waste management and decommissioning.

An overview of the contents of the draft Regulation of the Minister of Environment "The order of management, registration and transfer of radioactive waste arising in medicine, industry, and research and as a result of nuclear activities and the levels of specific activity in their management" is presented hereafter. Hopefully it would replace an existing radioactive waste management regulation [4.2] in 2000.

The draft consists of two chapters :

1. General provisons,
2. Management, registration and transfer of radioactive waste.

In the first chapter, for the first time in Estonia, the categories of radioactive waste are introduced (See Table 2).

Table 2. Categories of radioactive waste

Category	Description	Storage requirements
Cleared waste	**Activities less than exemption or clearance levels**	**No restriction imposed after clearance**
Low activity waste	**$A < 10$ MBq, $t_{1/2} < 100$ days**	**Decay in storage**
Short-lived LILW	**$t_{1/2} > 30$ years; beta and gamma; alpha $C < 4$ kBq/g per waste package (averaged over all packages $C < 0.4$ kBq/g)**	**Interim or final repository**
Long-lived LILW	**$t_{1/2} > 30$ years; $C > C$(short-lived LILW)**	**Interim or final repository**

In the second chapter of the present Regulation draft, in §3, in accordance to the recent EU recommendations, specific clearance levels for radioactive waste and the conditions for their application (e.g., averaging procedure, minimum surface area, etc.) are introduced:

- for metals, metal components, equipment or tools;
- for concrete and building rubble;
- for reuse or demolition of buildings.

In Appendix, Tables I and II summarise the corresponding levels for a number of important radionuclides. In these Tables only part of the established values is presented.
In addition to the above tabulated level values for metals and buildings, the draft also provides for clearance of the other waste types using the corresponding exemption levels [4.4] or a dose-assessment procedure basing on the generic levels [4.3]. The ERPC should be involved to authorise any act of clearance.

The §3 (8) of the regulation prohibits an addition of non-contaminated materials to the waste for their dilution in the averaging process.

The §4 introduces limitations to the activities of annual gaseous and liquid radioactive releases to the environment from one practice site, etc. The corresponding levels are tabulated in the draft. Detailed provisions imposed on a waste generator and its premises for safe management and storage of radioactive waste are given in §5. E.g. in the case of satisfactory storage conditions, the non-cleared waste may be stored up to 5 y in the waste generator premises. Any waste generator should register its stored waste, cleared waste, radioactive releases, transferred waste and spent sealed sources and maintain the records at least during 5 y after transfer. The provisions on documentation concerning the inventory of available waste and the transfer of waste to the radioactive waste management operator are listed in §7. The waste inventory at the generator's site and at the storage managed by the operator should be updated annually and the Estonian Radiation Protection Centre should be informed.

Requirements of § 8 address waste acceptance criteria (WAC) for storage and disposal of waste packages. Here generic provisions for WAC, including restrictions on mechanical, physical, chemical, biological and criticality properties of the waste packages, are listed. The elaboration of the detailed site-specific WAC is imposed on the radioactive waste management operator. The latter is also responsible for drafting the specifications and prescriptions concerning waste treatment/packaging, interim storage and disposal.

3.2.2 Assessment of population doses

Regulation of the Minister of Environment No 55 on the order of monitoring and assessment of population doses caused by natural radiation, radiation practices, sources of radiation and radiation accidents was issued in 1998 [4.3]. The regulation establishes internationally accepted terms, procedures, quantities and dose coefficient values needed for monitoring and assessment of population doses. The coefficients and the supporting data are given in a number of Appendices to the Regulation.

Paragraph 3 establishes the quantities, which should be determined for the assessment of doses from external exposure and from intake of radionuclides. The ERPC should determine the types and periods of measurement and analysis, manage the regular monitoring and assessment of population doses.

Pursuant to paragraph 5, any owner of a licence for radiation practice or any undertaker whose activities involve exposures to enhanced natural radiation levels should arrange monitoring and assessment of population doses caused by his/her practices/ activities and inform the ERPC.
Paragraph 5 establishes the above mentioned generic clearance levels and the investigation levels for specific activities of radionuclides in food products, for the specific activity indexes, I, in building materials, in drinking and house-hold water. The following tables are included in the Appendices to the Regulation:

Table 3. Parameters $\boldsymbol{A}_i$ for calculation of the specific activity index $\boldsymbol{I}$ for building materials

Material	Parameter A_i, $Bqkg^{-1}$			
	^{226}Ra	^{232}Th	^{40}K	^{137}Cs
Building materials	300	200	3000	--
Materials for building roads, streets, playgrounds	700	500	8000	2000
Ground filler materials	2000	1500	20 000	5000

α

Table 4. Parameters $\boldsymbol{A}_i$ for calculation of the specific activity index $\boldsymbol{I}$ for water

Material	Parameter A_i, Bql^{-1}						
	^{210}Pb	^{210}Po	^{226}Ra	^{228}Ra	^{234}U	^{238}U	^{222}Rn
Drinking and household water	0,5	3	3	2	20	20	300

Table 5. Specific activity limits in food products after a nuclear accident or any other radiological emergency

Group of radionuclides	Specific activity, $Bqkg^{-1}$			
	Infant food	Milk pro-ducts	Other food products	Liquid food products
Strontium isotopes, notably ^{90}Sr	75	125	750	125
Iodine isotopes, notably ^{131}I	150	500	2000	500
Alpha-emitting Pu and trans-Pu isotopes, notably ^{239}Pu and ^{241}Am	1	20	80	20
All other radionuclides with half-life > 10 d, notably ^{134}Cs and ^{137}Cs	400	1000	1250	1000

The established levels in Table 4 should be considered as a temporary attempt to regulate safety use of a material, radioactive properties of which are not known. Hopefully with a better knowledge about radioactivity in Estonian drinking water, more adequate values will be established as a drinking water standard in the future.

Table establishes maximum permitted levels for food stuffs and feeding stuffs following a nuclear accident or any other radiological emergency. Liquid foodstuffs include also household and drinking water.
The generic criteria, indexes, etc. can be used for clearance of sources, materials, including radioactive waste, from regulatory control or for the decision about occupational exposures in the case of enhanced levels of natural radiation pursuant to the paragraph 14 Natural exposure of the Radiation Act : "..

(1) Natural exposure is the exposure caused by cosmic radiation or radiation from natural radioactive substances which are not knowingly used as radiation sources.

(2) Natural exposure may be deemed to be occupational exposure:
- in work at mineral springs, in caves, mines or underground constructions;
- in work with usually non-radioactive substances which contain additions of natural radioactive substances;
- in the work of aircraft crews in high-altitude flights.

(3) If it is suspected that natural exposure is or may be detrimental to the health of workers performing work specified in subsection (2) of this section, the Minister of the Environment shall order an impact assessment on the proposal of the Radiation Centre.

(4) On the basis of the findings of an impact assessment specified in subsection (3) of this section, the Minister of the Environment shall decide whether or not the exposure is occupational.
(5) The expenses relating to the conduct of an impact assessment specified in subsection (3) of this section shall be borne by the employer."

In §§ 6-11 of the Regulation, the basic formula and the procedure for the evaluation of effective doses, to which the dose limits fixed by the Radiation Act apply, and for the selection of corresponding dose coefficients, are established.
Following the regulation, effective doses caused by radon/thoron progeny should be calculated using the following dose coefficients ($Sv\ (J\ h\ m^{-3})^{-1}$):

Radon ^{222}Rn	(homes)	1,1
Radon ^{222}Rn	(workplaces)	1,4
Thoron ^{220}Rn	(workplaces)	0,1

The above dose coefficients for radon/thoron progeny actually define the annual average concentration levels at workplaces, above which either remedial actions are needed or the work should be considered as a radiation practice. E.g., for radon concentration of 1000 $Bq\ m^{-3}$ an annual effective dose of 6 mSv can be estimated at workplace, as for the 2000 h occupational exposure and an equilibrium factor of 0.4 an annual dose of 20 mSv is equivalent to 6 x 10^6 Bq h m^{-3} radon gas [4.22].

3.2.3 Exemption levels

Regulation of the Government No 12 issued pursuant to the paragraph 6 of the Radiation Act in 1998 [4.4] establishes levels for total amounts and specific activities of radioactive substances exempted from licence requirements for radiation practice.

The Regulation presents a table of exemption level values, which coincide fully with the internationally accepted ones found in the Council Directive 96/29/EURATOM and in the International Basic Safety Standards.

A selection of the established values is presented below in Appendix 1 (Table III).

3.2.4 Requirements for areas, rooms and buildings housing a source and safety of sources

Regulation of the Minister of Environment No 82 [4.12] on requirements for the safe use of rooms and buildings housing a radiation source and of their construction elements and of a radiation source was issued in1999.

The Chapter I establishes a requirement to take specific safety measures in the working areas, where individual effective doses may exceed 1 mSv/a or if equivalent doses may exceed 10% of the

corresponding annual dose limits. Pursuant to §2, the safety measures should be appropriate to specifics of sources and to the magnitude of external and internal exposures.
In the next chapter (Chapter II), the terms "control area" and "supervised area" are defined. Their definitions differ from the internationally accepted ones and are clearly dose-biased, e.g., annual doses in the range from 1mSv/a to 20 mSv/a are defined for control areas and less than 1 mSv/a for supervised areas. As a result, it follows that the establishment of control areas is obligatory in any case, when annual effective doses slightly higher than 1 mSv should occur in working areas. This requirement, which differs also significantly from its initial formulation found in the draft Regulation, is obviously inadequate from the optimisation point of view and should be revised in the future upgrading. Provisions governing access, work activities, protective measures, supervision, zoning of control areas, personal monitoring, etc. implemented in control areas are laid down.

Subject to Chapter III on the requirements for safe use of sources, a license owner should ensure a safe storage of sources, a calibration of monitoring equipment at least once per year and an elaboration of detailed operational rules and the corresponding documentation for all operations involving a source. A list of items, which should be addressed in the documents, accompanies the requirement. Continuous radiation monitoring is made obligatory, while using a source.

3.2.5 Transport

Pursuant to § 27 of the Radiation Act, general requirements on transportation and provisions for hazard markings of radioactive substances, of radiation equipment containing radioactive substances and of radioactive waste are provided, as follows: "..

(1) The transportation of radioactive substances, radiation generators containing radioactive substances, and radioactive waste may only be conducted on the basis of a radiation practice licence.
(2) Upon transportation, exportation, import, sale or transfer, containers of radioactive substances shall be provided with hazard markings and documentation concerning radiation protection.
(3) The rules for transportation of radioactive substances, radiation generators containing radioactive substances and radioactive waste shall be established by a regulation of the Government of the Republic.
(4) The procedures for packaging of radioactive substances and labelling thereof with hazard markings and the requirements for accompanying radiation protection documentation shall be established by a regulation of the Government of the Republic".
Regulation of the Government No 162 issued in 1998 on implementation of Article 24, Article 27 (4) and Article 31 (1) of the Radiation Act [4.8] establishes detailed requirements on packaging, labeling and supplying safety means, on the enclosed safety documentation for transport of radiation sources containing radioactive materials and on issuance of export licences. The established provisions follow the accepted IAEA transport requirements.
In Chapter I GENERAL PROVISIONS, on the basis of activities or activity concentrations and their physical and chemical forms, radioactive materials are categorised into different groups: LSA-I, -II and –III; SCO-I, -II. A detailed specification for every group is defined.

The Chapter II PACKAGING REQUIREMENTS defines different package types for consignments: exempted, industrial type (IP-1, -2, -3) A-type and B-type packages. Maximum permissible dose rate and maximum permissible contamination values are set up for packages and for transport vehicles. Activity values, A_1 and A_2, for radionuclides are listed in Annex1. The next Chapter III MARKING AND LABELLING OF CONSIGNMENTS specifies their categories and labels needed. Chapter IV LABELLING OF A RADIATION SOURCE AND PROVISION OF SAFETY MEANS defines generic requirements for the installation of sources.

In Chapter V SAFETY LABELS different categories of safety labels and the corresponding required information in these labels are described. The requirements for locations of the safety labels are provided.

Chapter VI REQUIREMENTS FOR RADIATION SAFETY DOCUMENTATION OF A CONSIGNMENT sets up a list of information in the documents attached to a consignment.
Appendix 2 specifies activity limits for materials and articles sent in exempted packages, while Appendix 3 sets up those for consignments in industrial packages. Consignment-specific marking requirements are summarised in Appendix 4. Transport categories with their specific indexes and dose rate limits are given in Appendix 5 and the corresponding UN numbers – in Appendix 7.
The last part of the Regulation sets up an order for the issuance of export licences. The ERPC is authorised to issue and to suspend the licences and to supervise their application. A detailed procedure with the requirements for different parties is enforced. Appendix 1 and Appendix 2 present the valid forms for appropriate export documents.
Updating of the above Regulation is foreseen in the near future. The corresponding draft Regulation is already in its preparatory stage.

4 SELECTION AND DETAILED DEFINITION OF DISMANTLING OPTIONS

On the basis of the data collected in the first intermediate report (See Reference <1>), the aim of this section is to examine how the decommissioning strategies that was devised originally (See § 1.4) have to be adjusted.

This section is also based on the description of the Strategy that was selected to decommission the French Nuclear Submarines Reactors. This description will allow us to explain clearly and in an illustrated way the safety principles which have to be considered.

4.1 FRENCH NUCLEAR SUBMARINE REACTORS DECOMISSIONING STRATEGY

TECHNICATOME's experience in the field of submarines reactors decommissioning is based on the studies and works carried out for three reactors :

- The first French nuclear submarine named Le Redoutable was shut down in1991, and reached decommissioning level 1 in 1992, and level 2 in 1993.
- The first land based prototype named PAT, and located in Cadarache, in the south of France, was shut down in 1993, and reached decommissioning level 1 at the end of 1994.
- The second French nuclear submarine named Le Terrible was shut down in1996, and reached decommissioning level 1 in 1997, and the works to reach level 2 are in progress.

The strategy that was selected for the decommissioning of these nuclear reactors was the following :

1. Dismantle up to level 2 as soon as possible, to reduce operation costs and to reach a high safety level.
2. Wait 50 years (10 times Co-60 half period) in a temporary storage area, with a strict safety policy, in order to lower gamma dose rates in the compartment.
3. Dismantle up to level 3, sorting the arising waste and packaging radioactive waste in special containers.
4. Transport of these containers to the French repository named Centre de l'Aube (300 years near ground surface disposal).
5. After 300 years, free release of the Centre de l'Aube storage area.

4.1.1 Status of the reactor compartment corresponding to the IAEA decommissioning level 1

Safety rules are simplified, but the same confinement and monitoring procedures remain in force, The following work has been achieved :

- Reactor is defueled,
- Cover is welded to reactor vessel.
- Confinement barriers, ventilation systems, radiological measurement equipment, monitoring and alarm systems are maintained with no modifications,
- Radiation screens and markers are set on "hot" points,
- Pipes and tanks are emptied, except the shielding tank,
- All the electrical equipment and cables are removed.

4.1.2 Status of the reactor compartment corresponding to the IAEA decommissioning level 2

Inspection, maintenance and operation resources are notably reduced. The aim of decommissioning phase is to reduce to the minimum the volume of contaminated and activated material, and in the same time improving two successive confinement barriers tightness and the biological shielding. In this way, it's possible to reduce notably maintenance and monitoring.

Cutting/decontamination and disposal of small pipes (internal diameter below 26 mm).

Removal of all pieces of equipment that are not highly activated (pumps, auxiliary cooling systems).

- The contaminated and/or activated systems are totally dried up and sealed.

Restoring the first confinement barriers by sealing the primary circuit

(humidity level inside the primary circuit : < 40 %).

Restoring the second confinement barrier by sealing reactor compartment. Monitoring of humidity level inside the reactor compartment and maintaining below 40 % (access is provided to allow entry into the reactor compartments during the storage period, by the mean of an airtight airlock).

- The reactor compartment is separated from the adjacent compartments, and transferred in the storage building, providing an earthquake resistant foundation, protecting from bad weather.

Simplified ventilation systems, radiological measurement equipment, monitoring and alarm systems are maintained.

4.1.3 Level 3 of decomissioning

This status is equivalent to a free release of the site.

- After a 50 years decay period, all equipment is dismantled and conditioned in standard waste containers. Highly activated and/or contaminated pieces like reactor vessel and steam generator are conditioned into individual special containers.
- The containers are transported to the final disposal site.
- Reactor compartments are dismantled up, and the waste is sorted and characterized. The radioactive waste is conditioned and transported to the final disposal site.

4.1.4 Decay period

The 50-year decay period between level 2 and level 3 is justified by the followings concerns :

- Lowering the workforce dose, according to the ALARA principle : French Submarine Le Redoutable dismantling up to level 2 induced a total workforce dose of 330 mSv. This value, which is to be optimized for the next Nuclear Submarines to be dismantled, shows that further dismantling works can't be carried out now.

  Values of several Sv/h close to reactor vessel and IWS tank are too high for nearby human action. This dose rate is mainly due to Co-60 (half-life 5.3 years). It is necessary to wait radioactive decay of Co-60 for 30 to 50 years before there can be nearby human action using radiation screens.

- Postponing and lowering investments : Dismantling up to level 3 means handling big components like the reactor vessel or steam generators, cutting and/or decontaminating primary circuits pipework, and packaging the resulting waste. This work will induce the necessity of building a workshop with lifting, cutting, coating, and packing facilities. These heavy investments are to be written off with several NSSS.

- Enabling transport of highly activated components like the reactor vessel : the road transport of the waste packages towards the radioactive waste surface repository area must be done in accordance with the European transportation safety rules (ADR). It is necessary to wait about 50 years to be allowed to transport these particular big packages in type A containers, much less expensive than B type containers, which would be necessary if the transport was conducted now.

4.1.5 Safety policy

As said above, French strategy is to wait for 50 years before carrying out works to reach decommissioning level 3. During this period, radioactivity remaining in RC must be totally enclosed in two successive confinement barriers, fully protected against environmental risks:

These barriers must be kept in good condition whatever may happen during the decay period. They are designed to be airtight, corrosion and pressure resistant. They are regularly inspected and kept in good condition during the decay period.

4.1.5.1 First confinement barrier

It contains the whole radioactivity remaining in the RC. It is composed of the steel envelop of primary circuit and connected systems: reactor vessel, pressurizer, steam generator, activity filters, piping, tanks, etc.
All openings are welded. As we will see below, an air sampling piping is installed at several critical points for corrosion supervision.
The use of stainless steel (with chromium > 18%) as primary circuit main material and the use of an appropriate air conditioning system to maintain humidity level below 40% guarantees corrosion resistance of this fist barrier.

4.1.5.2 Second confinement barrier

It is composed of the steel envelop of the reactor compartment shell completed with bow and stern welded bulkheads. All openings are welded (access is provided to allow entry into the reactor compartments during the storage period, by the mean of an airtight airlock).
Black steel of the RC shell is protected by anti-corrosion paint. Internal and external air is recycled with the same air conditioning system.

4.1.5.3 Safety provisions

The Reactor Compartment is placed into a ventilated light building. This building could not be considered as a third confinement barrier, but its main purpose is to protect RC against environmental risks :
- Earthquake: the concrete slab and the vault structure behind the RC are designed to resist against probable local earthquakes.
- Storm: The roof and walls of the storage building are designed to resist against probable local storm. Note that they don't need to be earthquake resistant thanks to the presence of the two RC confinement barriers.
- Flood: the concrete slab is built higher than the highest probable local water level.

Additional provisions are necessary to protect the confinement barriers against fire risk and vandalism :
- Fire: the dismantling of combustible material inside RC reduces fire risk. There is no fire detection and extinction system installed.
- Vandalism: the storage shed is under surveillance, with controlled and restricted access.

4.1.5.4 Safety checking

All the above measures could be inadequate or might be unsuccessful. For this reason, it is essential to check regularly that they are sufficient to ensure the confinement of radioactive materials for the whole storage period. The following periodic inspections allow to check the integrity of the two barriers :
- Monthly: external round and recording of RC temperature, pressure, and hygrometry. Hygrometry analysis inside the second barrier will allow early detection of corrosion risk. If necessary, the ventilation is switched on to circulate the air through an air dryer until hygrometry inside RC decreases to 30 to 40%.

- Quarterly: air sampling from Reactor Compartment (by the mean of the ventilation system) and chemical and radiological analysis. A leak in the first confinement barrier would bring radioactive contamination into the RC. So the air inside the RC is moved and passed through filters by the mean of ventilation systems, using radiological measurement system to detect potential air contamination (late detection of corrosion).

- Yearly: human entry inside RC, for internal round and air sampling from primary circuit in view of chemical and radiological analysis. In normal circumstances, the reactor compartment is air tight and isolated from external air. At the occasion of human entry, air is then changed and filtered and the RC and a barometric depression of 10 mbar compared with storage building atmosphere is established inside.
  - Detection of confinement barriers potential corrosion trace.
  - Detection of water condensation traces inside RC.
  - Checking if the RC is still airtight by means of a pressure test : a barometric depression of 100 mbar must induce an air leak lower than that corresponding to a pressure drop of 10 mbar/h.
  - Updating of the RC radiation map.
  - Measurement of air hygrometry in the primary and secondary circuit (steam generator). If necessary, it is possible to set out the ventilation system in order to circulate the air of the primary circuit through the air dryer until hygrometry decreases to 30 to 40%.
  - Air sampling is done by means of piping fitting out several critical points of the first barrier. (Reactor vessel, Steam generator, etc.)
  - Checking if the primary circuit is still airtight by means of a pressure test : a barometric depression of 100 mbar must induce an air leak lower than that corresponding to a pressure drop of 10 mbar/h.

4.2 EVALUATION OF THE CURRENT STORAGE CONDITION OF THE PALDISKI REACTOR COMPARTMENTS

On the basis of a comparison between :

- the above description of the storage conditions of the decommissioned French nuclear reactors, and compartments,
- the data about the Paldiski Reactor Compartments collected in the course of the first task of the project,

this paragraph gives an evaluation of the current storage conditions of the Paldiski reactor compartments, highlighting possible deficiencies as far as safety is concerned.

We will see that these reflections are significant to define suitable dismantling strategies.

4.2.1 Physical characteristics

The size and mass of the reactor compartments are similar.

Characteristics		RC #1	RC #2	French RC
Compartment diameter, m		7	9,45	11,6
Compartment length, m		12.35	12.3	7,5
Shell thickness, mm		27	20	40
End bulkheads thickness, mm		10	12	40
Compartment mass, tons		922.65	1040,7	800
Materials:	- Reactor vessel	Black steel buttered with stainless steel	Black steel buttered with stainless steel	Stainless steel
	- Primary circuit	Stainless steel	Stainless steel	Stainless steel
	- IWS tank	Stainless steel	Stainless steel	Painted black steel
All stainless steels have chromium > 18%				

There is no important difference between the materials of the reactor compartments s as far as corrosion is concerned.

4.2.2 Storage arrangements

Storage conditions of French Submarines nuclear reactors compartments are briefly described above in § 4.1. Although storage arrangements of Paldiski and French reactors have been both designed for a 50 years storage period, there are important differences.

These differences are significant regarding the risk of radioactivity dispersion.

4.2.2.1 RCs contents

French nuclear reactor compartments have been emptied as much as possible so as to lower fire risk and to confine radioactivity inside the primary circuit.
In contrast, Paldiski RCs contain miscellaneous contaminated and activated waste as tools, calibration sources, rags, etc. The waste includes flammable material like rags, PVC, clothes, filters, wood in significant quantity.

A part of this waste is coated with concrete poured into compartments, but the most part of it was only piled up in the compartments.

The fire risk is not minimised.

4.2.2.2 Remaining water

Paldiski reactors primary systems were not totally dried up. Remaining water was not neutralized :
- About 1370 liters remains inside the RC#1, of which near 360 liters of borated water in the primary circuit.
- About 2280 liters remains inside the RC#2, of which near 600 liters of water in the primary circuit.

The presence of water in primary circuits may trigger off corrosion of primary systems materials.

4.2.2.3 Air conditioning

The Paldiski reactor compartments have not been fitted out with ventilation and air conditioning system.
Humidity absorbers have been set into RCs to maintain humidity under 40%, but this system might not 50-year efficient.
Furthermore, the air between RCs and sarcophagi is moved by natural convection through openings that are too small to eliminate water condensation on external side of RC shells.

The provisions that have been made against humidity effects regarding primary circuit and reactor compartment corrosion risk are not sufficient.

4.2.2.4 Monitoring and inspection

The Paldiski reactor compartments have not been fitted out with radioactivity and humidity monitoring inside or outside RCs.

Due to the absence of ventilation system, it’s not possible to check reactor compartments airtightness.

Furthermore, there are no regular inspection inside RCs.

There’s no way to check up the efficiency of radioactivity confinement provisions.

4.2.2.5 Storage building

The Paldiski reactor compartments have not been moved into a special building. Two sarcophagi have been erected on the spot around the RCs.

Although support structures of the RCs were reinforced to resist against earthquake, *it seems that no arrangements have been made against flood risk.*

4.2.3 Radiological status

4.2.3.1 Activation

A comparison of total activation values of the four reactors is given in the table below. The values mentioned have been calculated or estimated, and rounded (See First intermediate report reference <1>).

Activation (TBq) Year	Paldiski RC#1	Paldiski RC#2	Le Redoutable	PAT
1999	360	144	70	60
2030	93	22	15	12
2050	69	14	10	8
2100	47	10	8	7

This table shows that Paldiski RC#1 enclosed much more activation than the three other reactors. Taking into account the high level of activation of RC#1 and the lack of arrangements to control corrosion and flood risks, we must notice that:

RC#1 represents a potential danger of radioactivity dispersion for the next 50 years.

4.2.3.2 Irradiation and contamination

The Gamma exposure dose rates given in table below result from radiological inspection carried out five years or so after the reactors shut down.

Location of measurements (µSv/h unless mentioned)	**RC#1**	**RC#2**	**Le Redoutable**	**PAT**
Piping between SG and vessel (hand-on dose rate)	230-360		400	80-90
SG room (average dose rate)	8	2.5		
SG (hand-on dose rate)			65-270	230
Under reactor vessel	1900-5500	2.5	<1	<1
Near reactor vessel (Sv/h)	> 10	> 10	> 10	> 10
Upper floor (average dose rate)	0.15	0.15	20-100	70
Mid floor (average dose rate)	2-110	0.3-10	40-70	50
Lower floor (averagedose rate)	110-280	2.5	160-400	110
Internal contamination (Bq/cm^2) (miscellaneous wastes included)	70	130	Undetectable	Undetectable
External Contamination (Bq/cm^2)	Undetectable	Undetectable	Undetectable	Undetectable

Considering Paldiski reactor compartments, the given dose-rates values have resulted from a radiological inspection carried out seven years ago : today, dose-rates values might be two times lower (half-life of Co-60 : 5.3 years).
Values are very difficult to interpret and to compare from one reactor to another, for the following reasons :

- The reactors design is different.
- Radiological screens have been set on "hot points" inside French RCs to lower dose rate in the accessible room at the maximum value of 400 µSv/h. The aim was to limit personal exposure during decommissioning level 2 works, and during inspections inside RCs.
- It is necessary to take into account hand-on dose rate close to activated equipment in oder to estimate dismantling works resulting exposure. But the data resulting from the radiological inspection in 1994 in the Paldiski reactor compartments gives mainly average dose-rates in the different compartments rooms.

However, considering men exposure inside reactor compartments, the tables below shows that the conditions of Paldiski RC is quite similar to the conditions of French RC. Our conclusion is that :

Human work is at the moment possible everywhere except nearby reactor vessel. But Values of several Sv/h close to reactor vessel and IWS tank are too high for nearby human action. Further dismantling works can't be carried out now, as it is necessary to wait radioactive decay of Co-60 for 30 to 50 years before there can be nearby human action using radiation screens.

4.2.3.3 Miscellaneous waste

As already said, miscellaneous additional radioactive waste have been stored in the Paldiski RCs. Most of these wastes are low level wastes (rags, metallic wastes, tools), with surface contamination. But some radioactive sources were also put into concrete poured into RC #1 compartment. These radioactive sources are neutron, alpha, beta, and gamma type. Most of them remain inside their container placed inside RC#1 before it was partially poured with concrete.

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The detailed list of radioactive waste stored into RC #1 and #2 is given in first intermediate report (See reference <1>), appendix 4.

Type	**Total mass or activity**	**γ Dose rate (µSv/h)**	**β Contamination (Bq/cm²)**
Miscellaneous waste stored in RC #1	14 tons	0.3-5700	1.7-67
Small radioactive sources stored in RC#1	4-5 TBq (95)		
Miscellaneous waste stored in RC #2	2.5 tons	0.6-5700	1.7-133

Due to the presence of this waste, *Radioactivity is not confined into primary circuit.*

Moreover, this waste include burnable elements like wires, plastic, rags, wood, etc. that arise fire risks.

Waste also includes several tons of lead, which is considered as a chemical poison. (Saturnism)

4.2.4 Consequences regarding the definition of confinement barriers

According to European safety standards radioactivity remaining in RC must be totally enclosed in two successive confinement barriers.

On the basis of the previous paragraphs, two suitable confinement barriers have to be selected for the Paldiski reactors. After that, we will see what additional works should be carried out to guarantee confinement efficiency.

4.2.4.1 First confinement barrier

Primary circuit
Can we consider primary circuit and connected systems as a possible reliable first confinement barrier? As seen before :

- The presence of remaining water, arise primary circuit corrosion risk for the next 10 years.
- Furthermore, airtightness of the primary circuit cannot be periodically checked.
- At last, small sources and contaminated waste have been set inside RCs, and outside primary circuit or connected systems.

In order to dry up primary circuits systems now, because it would be necessary to carry out heavy dismantling works to gain access to the primary circuit systems (because of the presence of concrete). And extracting radioactive sources of the concrete poured into RC#1 without deteriorating them would be high labor consuming, and so would result in high personal exposure.

In conclusion we think that, it is impossible to consider Paldiski reactors primary circuits as reliable confinement barriers.

RC shell
However, if we consider the steel shell of RCs :

- The reactor compartments are airtight and have been tested, but airtightness must be regurly checked up;
- The air inside RCs has been dried thanks to humidity absorbers, but these systems might not be efficient for a 50-years period.
- The shell thickness is higher than 10 mm and is externally covered with anti-corrosion paint. It should resist to corrosion action during a few years.

We conclude that RCs shell could be a reliable first confinement barrier for a few years, but at the present time, it's not possible to guarantee the reliability of this barrier for 50 years. However, we strongly think that the efficiency of this confinement barrier could be guaranteed for 50 years if a few complementary safety provisions are implemented.

These additional provisions are the following :

- Providing access inside the RCs in order to be able to periodically look for RC shell corrosion, and repair if necessary (by scrapping + sanding, + anti-corrosion paint on every corrosion stain detected). The access door must be an airtight door.
- Fitting out the RCs and sarcophagi with a ventilation system that would allow to :
  - check up periodically the airtightness of the RCs (and immediately repair if necessary by patch welding),
  - monitor humidity inside and outside the RCs,
  - circulate the air on additional humidity absorbers if necessary.

The above provisions would allow to make sure that the first confinement barrier resists to corrosion action through the time. They are easily feasible and not expensive.

4.2.4.2 Second confinement barrier

If RC shell is defined as the first barrier, sarcophagus must logically be the second confinement barrier. But unfortunately its partly pre-cast constitution does not totally guarantee a very efficient confinement (Walls of Sarcophagi #1 are made of in-situ cast reinforced concrete, while roof is made pre-cast slabs, covered by 20 cm thick concrete grouting. As Sarcophagi #1 is using partly construction elements of the building #301, its western and eastern walls are 50 cm thick brick walls). Indeed, concrete plates are probably not systematically sealed with joints. In addition, cracks may appear in the walls during the next years because of little underground movements (variations of underground water level, earth tremor, etc…).
To define sarcophagus as the second confinement barrier, it will be indispensable to periodically check up airtightness and improve it if necessary. The maximum admissible leak must be lower than 0,1 vol./h under 100 mbar. This check up could be done by the mean of a ventilation system, connected to sarcophagi convection openings.

It will also be necessary to measure underground water level and to improve resistance against flood, and sarcophagi resistance against earthquake should be evaluated and improved if necessary.

The absence of water inside the sarcophagi would be periodically checked, especially in case of unusual precipitation. Possible infiltration water or condensation should be collected in a sump to be created in the slab.

4.3 SELECTION OF STRATEGIES AND OPTIONS

On the basis of :
- the dismantling strategies that were devised in the beginning of the project,
- the above evaluation of the Paldiski Reactor compartments storage conditions, it's now possible to select suitable dismantling options,

this paragraph explain how the dismantling options fully developed in the following of this study were selected.

4.3.1 Original strategies

The dismantling strategies that were devised in the beginning of the study to reach decommissioning level 3 are given below. These dismantling options were more or less theoretical :

- Dismantling strategy #1 :
  Removal of the reactor compartments followed by their final disposal in a near-surface facility or in deep geological formation.

– Dismantling strategy #2 :
  Decontamination/cutting of the reactor primary systems into small pieces, and disposal of the resulting radioactive wastes.
– Dismantling strategy #3 :
  Melting of the metallic parts of the reactors with the view to volume reduction and/or recycling.

Each strategy is to be optimized in terms of cost and workforce dose taking into account a storage period of 10, 50 or even 100 years before starting operations.

4.3.2 Type and location of repository

4.3.2.1 Transport of the RC as a whole towards another nuclear area

Another project is to be launched to define the criterion in order to choose a suitable site and to decide what type of repository would be constructed. Six possible locations have been identified, all located along Estonia northern cost.

The use of a B type container is obligatory beyond an “A2” activity limit imposed by European transportation rules for radioactive goods (ADR).
This A2 value depends on the radionuclides attached to the waste transported. We calculated in the table below for the year 1999, 2050, and 2100, in the case of RC#1 and RC#2 :

Radionuclide proportion $\%_i$	**Year 1999**	**Year 2050**	**Year 2100**	**$A2_i$ (GBq)**
55 Fe (T=2.7 years)	34	0	0	4^{E}+04
59 Ni (T=7.5 E+04 years)	0.3	0.3	0.3	4^{E}+04
63 Ni (T=96 years)	26	97	97.6	3^{E}+04
60 Co (T=5.3 years)	31.6	0.3	0	4^{E}+02

A2= $\sum_i (\%_i / A2_i)^{-1}$ (GBq)	1 238	25 370	30 660
Total activity of **RC#1** (GBq)	360 000	68 800	47 000
Total activity of **RC#2** (GBq)	144 000	13 300	9 600

In accordance with the European transport regulations, we can see in the above table that Paldiski RCs must be transported at the moment in a B type container, due to their total enclosed activity. But designing a B type container for components as big as reactor compartments is quite impossible.
– RC#2 may be transported in a container A type after the year 2030.
– RC#1 may be transported in a container A type only after the year 2165.

Furthermore, the size (8 x 12 x 9 m) and weight (1000 tons or so) of reactor compartments are very high, and it appears that Pakri peninsula roads characteristics are not convenient to transport the reactor compartments as a whole from Paldiski site to a disposal site which would be located in another area. And there's no suitable infrastructure on the Paldiski site to allow the shipping of the reactor compartments.

For these reasons, we will assume to develop the first dismantling strategy that the repository is located on the Paldiski site.

4.3.2.2 Type of the repository

The geological data of the survey reference 5 shows that implementing a repository in deep geological formation in the Pakri Peninsula might not be possible, because of the groundwater characteristics, seems not to be suitable.

Moreover, the investments necessary to implement a repository in deep geological formation would be very high, because this technology is not fully mastered today. Feasibility of waste final disposal in deep clay layers is to be studied by ANDRA in France, at being built Meuse subterranean laboratory. Other research programs have been launched at an international level to study the feasibility of waste final disposal in deep granite layers (Sweden, Finland, Canada, Switzerland).

In the other hand, TECHNICATOME has gained a great experience in near surface repository for low and medium level short-lived radioactive design (acting as prime contractor and designer of the Centre de l'Aube in France and El Cabril in Spain). And this type of repository is convenient to store the radioactive waste resulting from nuclear reactors dismantling operations.

So the Estonian final repository will be assumed to be of the same type as Centre de l'Aube in France, made for low and medium level short-lived radioactive waste.

4.3.2.3 Another alternative regarding the final disposal of the reactor compartments as a whole

The main difficulties to remove and to transfer the reactor compartments as a whole to the repository are :
- the opening of the sarcophagi without any damage for the reactor compartments,
- the handling of the reactor compartments from the sarcophagi to the disposal site.

The handling of the reactor compartments (weight : 1000 tons or so) requires special devices. Such handling devices do exist, but their operation would require to fit out the Paldiski site with heavy infrastructure.
These difficulties induce us to consider another strategy, which consists in reinforcing sarcophagi with the aim to transform them in small final disposal sites, in accordance with Estonian regulations. This strategy would avoid handling the reactor compartments. We strongly think that this strategy is worth going into more details, and we suggest to study it as an alternative to the first strategy.

This strategy will be based on civil engineering techniques also envisaged in the course of other projects : the ICC Consortium, that involves TECHNICATOME, has studied possible scenarios for reinforcing and stabilising Chernobyl sarcophagus, in the course of SIP Project (Shelter Implementation Program) supported by G7.

4.3.3 Influence of the storage Period

According to the study contract, each strategy is to be optimized in terms of cost and workforce dose taking into account a storage period of 10, 50 or even 100 years before starting operations. As reactors were shut down in 1989, we can consider now that the period of 10 years is achieved.

Radioactive decay is more or less important according to :

- the need of work near gamma radioactive sources,
- the classification of radioactive waste.

4.3.3.1 Total men exposure

Regarding total men exposure, waiting until the year 2050 could result in a total men exposure up to one thousand times lower. This is due to the composition of the gamma radioactivity spectrum. Spectrum repartition shows that Co-60 is the more significant gamma high-energy radionuclide. Its half-life is 5.3 years.

In the year 2050, Co-60 activity will have decreased one thousand times, and gamma dose rates due to Co-60 will have decreased in the same ratio.

If we consider the first dismantling strategy, the influence of the storage period is not really significant regarding men exposure, as dose rates outside the reactor compartment is quite low, except under the reactor vessel. However, after a 50 years storage period, hand-on gamma dose rate under the reactor vessel should be lower than 2 mSv/h, and so the transport of the reactor compartment as a whole would be possible in total accordance with the transport regulations regarding dose rate, without fitting out the RC with biological shielding.

But if we consider the second and third dismantling strategies, that implies heavy dismantling works near gamma radioactive sources, the influence of the storage period is actually very important. This will allow manual working near the most activated and contaminated areas with rather light gamma screens, instead of using very expensive remote technologies. Nevertheless total men exposure could be less than ten times lower. For this reason, these dismantling options will be developed taking into account a 50 or 100 years storage period.

4.3.3.2 Radioactive wastes volume

If we consider the first dismantling strategy, the influence of the storage period is not significant at all regarding waste volumes, as the waste volume is equal to the reactor compartments volume.

Considering, the second and third strategies, the situation seems to be quite different. But due to the presence of Ni-63 (half-life : 100 years) and Ni-59 (half-life : 75000 years), there will be no significant change in the total activity-weight ratio of the resulting waste between 50 years and 100 years.

Therefore, as the waste categories and exemption and clearance levels in force in Estonia are based on total activity-weight ratio, there will be no significant change in the management of the waste resulting from dismantling operations after 50 years.

However, we will try to estimate the difference in terms of waste volumes for full dismantling operations carried out in 2050 or 2100.

4.3.4 Full dismantling of reactor compartments : compromise between lowering men exposure and lowering wastes volumes

Actually, the second and third dismantling strategies are close one from the other : the main difference between these two options is the waste management strategy. Indeed, both strategies implies to open the sarcophagi, and then the reactor compartments, in order to extract the main components.

After that, a selection (or a compromise) has to be made between two main concerns :
1. Minimising cutting works in order to lower men exposure.
2. Decontamination and cutting components into small pieces in order to sort wastes, in view to reduce the resulting waste volume.

4.3.4.1 Minimising cutting works in order to lower men exposure

This strategy is the closest to the one that was chosen by TA to carry out level 3 dismantling of the French nuclear submarines. At the end of the 50 years storage period, the main components (reactor vessel with its welded cover, steam generators and pressurizers) will be extracted of the reactor compartments and conditioned into special containers. After that, these containers will be transferred to the disposal site (Centre de l'Aube), and filled with special concrete before being installed into the disposal site.

We're going to transpose this strategy for the Paldiski reactors dismantling.

4.3.4.2 Decontamination and cutting components into small pieces in order to sort wastes, in view to reduce the resulting waste volume.

After the extraction of the main reactors components, decontamination operations will be carried out before cutting components into small pieces in order to sort wastes and to reduce the resulting waste volume. Additional decontamination could be carried out after cutting components into small pieces. The use of melting techniques in view to further reduction of the waste volumes will be studied

This strategy is very interesting because the resulting waste volume would probably be much lower than in the case of the dismantling option described paragraph 4.3.4.1, but men exposure could be higher, and more equipment would be required.

4.3.5 Selected decommissioning strategies and options

According to the previous paragraphs, the dismantling strategies that will be fully developed in this study are the following :

- **Strategy #1** : **Final disposal of the RCs as a whole**
  - Option #1 : in situ, in the sarcophagi, avoiding RC removal operations.
  - Option #2 : in a near surface disposal facility located on the Paldiski site.
- **Strategy #2: Full dismantling of the RCs**
  - Option #1 : Minimizing cutting works in order to lower men exposure.
  - Option #2 : Decontamination and cutting components into small pieces in order to sort wastes, in view to reduce the resulting waste volume (using or not melting devices).

According to paragraph 4.3.3, the first decommissioning strategy will be considered for operations performed immediately or after 50 years, the second one will be considered for operations performed after 50 years or after 100 years.

5 FIRST DECOMMISSIONING STRATEGY : FINAL DISPOSAL OF THE RC AS A WHOLE

The aim of this strategy is to study if the final disposal of the reactor compartments as a whole is a feasible dismantling route according to the Estonian regulations, and how this dismantling route could be carried out from a technical angle. The disposal site should be under monitoring for a few centuries, and after that period, the disposal site should be free released.

5.1 RADIOACTIVE WASTE STORAGE REGULATIONS AND RECCOMMENDATIONS FOR RADIOACTIVE WASTE FINAL DISPOSAL

Reactor compartments disposal must be done in accordance with Estonian radioactive waste disposal regulations.

Several requirements arise from the IAEA nuclear waste management recommendations regarding radioactive low and intermediate level waste in a near surface facility. The main requirements are given below :

- Surface disposal areas must be put under surveillance for a minimal period of 300 years.
- Surface disposal areas must be protected for at least 300 years against external aggressions (earthquake, bad weather, flood).
- Waste disposal must be done providing two efficient confinement barriers for 300 years : the first barrier is the waste package, and the second one is the disposal facility.
- For each waste package, specific activity must be below an acceptance limit depending on the toxicity and the half-life of contained radionuclides. For instance, The Centre de l'Aube acceptance limits are given below for a few radionuclides :

Radionuclide (half life)	Maximum specific activity (GBq/t)
Fe-55 (2.6 years)	10^6
Co-60 (5.3 years)	51800
Ni-63 (100 years)	11840
Mo-93 (3500 years)	0,44
Ni-59 (75000 years)	63
Tc-99 (21500 years)	1

- Radioactive waste must be chemically stabilized, in particular the ratio of remaining water must be below <1% before conditioning, an protected from oxygen.
- Radioactive waste must be solidified and immobilized by embedding (usually with light concrete), in order to reduce the potential for migration and dispersion of radionuclides. Empty spaces volume must be below 3% of the package volume.
- Waste package activity must be as homogeneous as possible. In particular, punctual radioactive sources are not allowed.

Considering sarcophagi as disposal site and the RCs as waste packages, it appears that neither they are consistent to the above requirements :

- There is no monitoring of the sarcophagi.
- The sarcophagi are not designed to resist against earthquake, storm, flood during the next 300 years.
- Grouting the reactor compartments with concrete in order to solidify and immobilize waste, limiting empty spaces below 3% seems not to be realistic.
- Radioactive waste enclosed inside containers is not chemically stabilized, due to the presence of water and oxygen, arising corrosion risk
- Radioactivity in the RC is not homogenous: 95% is inside the reactor vessel. Furthermore, small radioactive sources are dispersed into concrete.

5.2 FIRST STRATEGY - OPTION #1 : FINAL DISPOSAL OF THR RC IN THEIR SARCOPHAGI.

The aim of this section is to define the works to be carried out in order to transform the sarcophagi in small final disposal sites, in accordance (as much as possible) with Estonian regulations and IAEA recommendations.

5.2.1 Soil, geological, geotectonic and hydrological surveys on the possibility of permanent in-situ disposal

For final disposal with a monitoring facility for a time span of the order of 300 years, the soil should present the following main characteristics. The data summarised in the survey reference <5> shows that the Pakri Peninsula could be a convenient area, but this fact should be confirmed on the basis of a specific survey.

Geology:
The region chosen must present geological characteristics with satisfactory impermeability at very great depth. This is generally the case for deep formations of the Lower Cretaceous Age, which are most often covered with layers of soil of the marl-limestone, clay marl and clay-sand types.

Geotectonic:
A structural sketch map will be drawn to define the organisation of the major underlying structural fields. The region and site must not present major accidents, faults, cleavage, tilting of strata, or

potential landslides likely to result in the disintegration of the bedrock and underlying soil formations.
Earthquake:
The stability of the disposal structures and their dimensions will be defined and drawn up on the basis of a maximum intensity revised upwards to VII MSK. Although this intensity does not give rise for alarm with this type of structure, all verifications will be made to establish the mechanical withstand, the maintenance of structural integrity and adequate performance in relation to cracking.

Hydrology:
The position of the upper bed of the water-bearing system must be below the clay layer covering the draining layers of sand and gravel. The arrangement of layers in relation to one another reduces the permeability of the upper part of the subsoil, so that both water originating from infiltration and either surface run-off water or groundwater likely to rise inside the disposal structures can be confined and drained away.
Water currents in the groundwater or streams must provide draining for excess water due to variations in the groundwater level or streams of run-off water resulting from soil infiltration. It is essential that these natural outflows exist so that the groundwater level remains relatively constant. In any case the groundwater must not be confined. Piezometers will have to be installed to routinely check water level variations.

5.2.2 Fitting out RCs with Fire extinguishing systems

RCs compartments contains miscellaneous waste, and notably burnable elements like wires, plastic, rags, wood, etc, arising fire risk. For this reason, RCs should be fitted out with fire estinguishing systems. However, if most burnable wastes were extracted from the RCs it may be possible to avoid providing such systems.

5.2.3 Fitting out RCs and sarcophagi with an air conditioning system

The presence of oxygen and moisture inside and outside the RC may generate corrosion. It is essential to provide against first barrier corrosion risk. In order to carry out corrective actions if air hygrometry is too high, it's necessary to fit out RCs and sarcophagi with an air ventilation, filtration and dryer system. The ventilation system allows to:

- circulate the air through an air dryer until hygrometry inside RC and inside sarcophagi so as humidity level decreases to 30 to 40%,
- air sampling from Reactor Compartment (by the mean of the ventilation system) and chemical and radiological analysis. The air inside the RC could be circulated through filters, using radiological measurement system to detect potential air contamination (late detection of corrosion).
- Checking if the RC are still airtight by means of a pressure test : a barometric depression of 100 mbar must induce an air leak lower than that corresponding to a pressure drop of 10 mbar/h.
- Checking if the sarcophagi are still airtight by means of a pressure test : a barometric depression of 100 mbar must induce an air leak lower than that corresponding to a pressure drop of 10 mbar/h.

It is much better if this system is able to start and stop automatically, setting off a remote alarm to the surveillance team. The system could be maintained during monthly inspections.

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This system could be designed as follows :

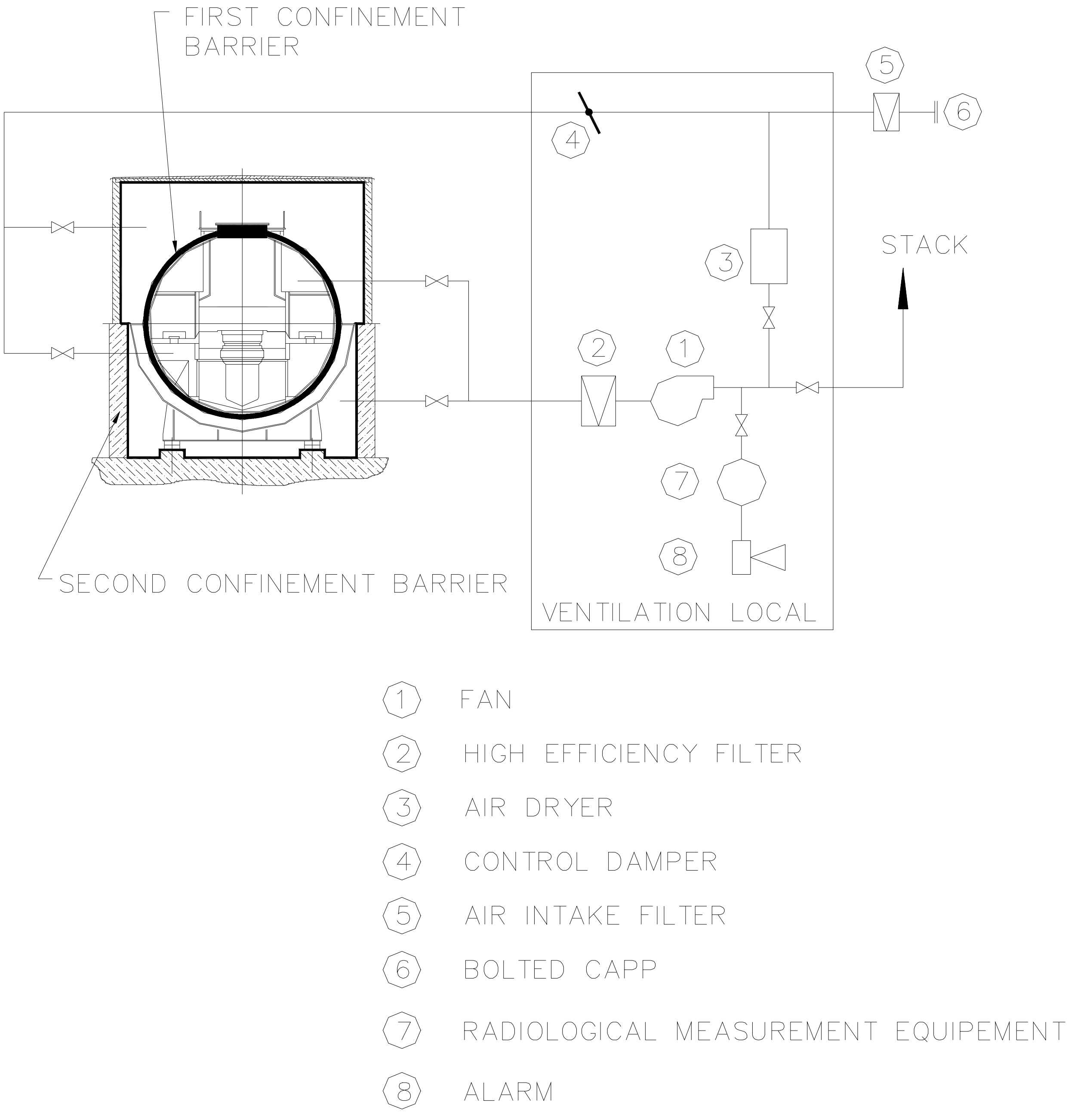

Figure 2 – RC and sarcophagus ventilation system process flow diagram

5.2.4 Filling the primary circuit with concrete

The presence of oxygen and acid water inside the primary circuit may generate corrosion. As the primary circuit contains 95 % of the radioactivity inside the RC, it is logical to treat the primary circuit as much as possible as a proper waste package, as if it were a very first confinement barrier.

We saw above that the primary circuit cannot be considered as a reliable first barrier, but we have to make it as reliable as possible in order to prevent radioactivity release inside RC.

According to the IAEA radioactive waste storage regulations, we saw in § 5.1 that :

- Radioactive waste must be chemically stabilised (ratio of remaining water must be below <1%), an protected from oxygen.
- Radioactive waste must be solidified and immobilised by embedding, in order to reduce the potential for migration and dispersion of radionuclides (empty spaces volume must be below 3% of the package volume).

If we consider RCs as waste packages, we should envisage to grout them with concrete, in order to fulfil the above requirements. However, this hypothesis seems not to be technically realistic.

For these reason, we suggest to grout only the primary circuit. This operation might be rather difficult to achieve and expensive, but not impossible. It will be necessary to study, referring to as-manufactured documents, where the potential air retention are located. Then will be defined where and how many holes are to drill, and what will be the sequence of concrete filling and air emptying. The concrete would be liquid, and without shrinkage. After operations, a little quantity (about 1 %) of remaining air or water is usually admitted.

5.2.5 Sarcophagi strengthening

First of all, specific civil engineering surveys should be carried out to estimate precisely the actual state of sarcophagi :

- Checking the sarcophagus structure calculations and surveys, both above grade and at raft levels, on the following criteria:
  - Containment and/or the state of wall and roof structure cracking,
  - Containment of the low walls and the raft at soil level,
  - The state of the soil props,
  - The groundwater table level,
- Survey of existing buildings and possible consolidation to provide the sarcophagi protection from adverse climatic conditions (wind, rain, snow, ice, etc) for the estimated time required to complete the sarcophagus filling operation for permanent disposal.

In a very first phase the sarcophagi should be consolidated undertaking the following works:

- Injections of calcium silicate at the sites of cracking,
- Surface application of thick cement mortar reinforced with fiber-glass mesh to the inside walls,

- Injections of fine rendering cement mortar and bentonite-cement grout on the underside of the raft through previously inserted casings,
- Sarcophagi should be made airtight as much as possible by setting joints between concrete plates, by sticking bitumen plates on the cracks, etc. Every opening devoid of high efficient filtration should be sealed.

In a second phase, in the light of survey results, major development work should be carried out to guarantee the isolation and final mechanical resistance properties of the above-grade structures, raft and ground :

- <u>Raft and earth works (See Appendix 2 page 2):</u>
  As the groundwater level is practically at ground level, and bearing in mind the seasonal variations, the raft is permanently immersed.
  A waterproofing system is proposed for the layer below the raft consisting of injecting fine rendering cement mortar and bentonite cement grout to a thickness of roughly 5 m under the raft. The pressure-, volume- and flow-controlled injections are made through a casing network crossing the raft and injection pipes of variable length and fall. The pipes and casing are laid out in grid format according to the nature of the soil and over a surface area equal to that of the sarcophagi extended by a minimum additional projection of 3 m at the edges. This injection operation would be carried out by lowering the groundwater level in two or three stages. It is presumed that the layers of sand or other matter that make up the soil are sufficiently consolidated so that no compacting will take place when the groundwater level is lowered.
  To complete this waterproofing system, we recommend that a continuously cast reinforced concrete slab is laid on top of the existing raft with an HDPE (High-Density Polyethylene) membrane separating them.

- <u>Works to reinforce the sarcophagi superstructure (See Appendix 2 page 1):</u>
  The sarcophagi walls should be lined with 45 cm thick skins from base to roof level. These skins would be supported and anchored at the base in the rafts and on the supporting ledges of the sarcophagi runners. The upper part of the existing roof cover should be finished with a reinforced concrete slab fixed at the top of the skin stays by the existing slab reinforcements. The upper side of this slab should offer a slope of 5 mm per meter to enable rainwater to run off that might otherwise stream across the roof cover.

- <u>Impermeability (See Appendix 2 page 1):</u>
  The roof cover should be made impervious to rainwater by a built-up system comprising an HDPE membrane (3 mm thick) laid on the existing roof slab separated by a geotextile film. This membrane covers the upper part of the existing skin and goes down over the outer skin mantle.
  A second HDPE membrane is laid on the new additional containment slab separated by a pure bitumen separating film. This membrane skirts the perimeter ledge housing and is heat-sealed to the vertical walls with a double lap. The holes that were drilled previously to feed the reinforcement through prior to bonding the slabs and skins, are made watertight using bitumen plugs. Surface drainage could be provided by a layer of pebbles roughly 20 cm thick retained by a concrete groove.

- <u>Containment (See Appendix 2 page 1):</u>
  Containment should be provided by 2 HDPE films applied either side of the sarcophagus walls.

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5.2.6 Dealing with the problem of small radioactive sources scattered in concrete inside RC#1

Several small radioactive sources (neutron, alpha, beta, gamma) had been scattered in concrete inside RC#1 during the works carried out after the reactors shutdown and core unloading. These sources were used for calibrating radiological measurement equipment :

– Neutron sources : Pu238-Be, Cf252.
– γ-radiation sources : Co60
– β-radiation sources : Cl Sr90, Sr90 + Ittrium 90, Cl Na22, Tallium 204, Cs 137.
– α-radiation sources : Pu 239.

Most of these sources are small sources. Plutonium and Cesium sources are very small sources (from a few kBq to a few MBq). Their activity is given in waste lists above mentioned (See first intermediate report reference <1>, appendix 4). The total activity of the radioactive sources that were on site and had or might have been placed into RC #1 was to 4.4 TBq or so in 1995 (main radionuclide : Co60), and can be estimated to two thousand times lower after fifty years, and to a few MBq in 2100.

In accordance with Estonian radioactive waste management regulations and IAEA recommendations, it is in theory necessary to extract these sources from the RC#1 before disposal. Actually, these radioactive sources contained long lived radionuclides, and their specific activity is far above exemption and clearance levels, as they cannot be considered as an homogeneous activity scattered into concrete poured into RC#1.

But if the extraction works were conducted now with the present radiation conditions, the arising men exposure would be quite high as these works would be highly labor consuming : 200 mSv or so (average hand-on dose rate in the primary circuit room assuming concrete is removed :0,1 mSv/h ; average hand-on dose rate above the reactor cover, assuming concrete is removed : 1 mSv/h).

For this particular point, we think that as each source contains a rather little radioactivity of its own, it could be possible to deviate temporarily from the regulations regarding radioactive waste management. The volume of concrete in which sources have been poured has to be correctly marked out so that it will be possible to easily extract the sources in 50 years if necessary, when the radiation dose rate in the reactor compartments is lower.

5.2.7 Surveillance program

The aim of the surveillance program is to prevent against risk of radioactivity leakage, and to check that safety provisions remain efficient. As said in the previous chapters, we consider that the RC steel shell is the first confinement barrier against radioactivity leakage.
The sarcophagus is then the second confinement barrier. The sarcophagus also has the role to protect the RCs against physical and chemical external aggressions.
The main purpose of the surveillance program is to periodically check the integrity of these barriers and as a priority to prevent corrosion of the RCs. Taking into account that the works detailed in §5.2.3 to 5.2.4 are supposed to have been carried out, the surveillance procedure could be slightly less restricting that the one detailed in § 4.1.5.4 :

- **Monthly**: inspection inside building and outside sarcophagi: structures, detection of roof leakage and wall cracking.
- **Quarterly**: inspection inside sarcophagi and outside RCs: measurement of air hygrometry and air contamination, detection of possible water condensation and corrosion traces.
- **Annual**: inspection inside RCs: measurement of air hygrometry and air contamination, detection of possible water condensation and corrosion traces, measurement of RCs and sarcophagi confinement airtightness.
- **Every five years**: environment inspection: detection of contamination of water and clay samples from the soil around and under sarcophagi.

If any abnormal event is detected, corrective actions are to be carried out.
Manpower and equipment required to lead the above surveillance program are rather light and cheap.

This surveillance procedure should be in force for a minimal period of 300 years.

5.2.8 Advantages and drawbacks of this decommissioning option

Advantages	Drawbacks
– This decommissioning strategy is quite simple, as it does not requires heavy dismantling works. – As a consequence, the global cost would be quite low. – Except radioactive sources extraction, the works to be carried out would not imply high men exposure. – No radioactive waste transport is required. – The risk of radioactivity release into the environment is reasonably low.	– Waste packages (RCs) are not consistent to the IAEA recommendations regarding waste management – Waste packages are not totally immobilized into the final disposal.

5.3 FIRST STRATEGY - OPTION #2 : FINAL DISPOSAL OF THE RCS AS A WHOLE IN A REPOSITORY LOCATED ON THE PALDISKI SITE

The aim of this section is to define the works to be carried out in order to transfer the reactor compartments as a whole in a near surface repository designed for low and intermediate level radioactive waste, in accordance (as much as possible) with Estonian regulations and IAEA recommendations.

This decommissioning option rests on the following steps :

- Carrying out preliminary works in the RCs
- Erecting a new building on Paldiski site.
- Building a special heavy-duty route between the sarcophagi and the new building.
- Transferring the RCs as a whole from the sarcophagi to the new building.

5.3.1 Carrying out preliminary works inside the RCs

We saw in §5.2.4 that we have to make the primary circuits it as reliable as possible in order to prevent radioactivity release inside RCs, as it contains 95 % of the total radioactivity.

As the presence of oxygen and acid water inside the primary circuit may generate corrosion, we suggest to grout the primary circuits systems internal parts (See §5.2.3).

The RCs should be provided with a provisional ventilation systems before these works to be performed.

Considering fire risks, as seen in §5.2.2, if most burnable elements like wires, plastic, rags, wood, etc. were extracted from the RCs before their transfer, it may be possible to avoid installing a fire extinguishing system inside RCs.

5.3.2 Erecting a new disposal building on Paldiski site

As seen in § 5.2.1, the soil of the area chosen for erecting a final disposal site must present main typical characteristics. The data summarised in the survey reference 5 shows that the Pakri Peninsula could be a convenient area, but this fact should be confirmed on the basis of a specific survey.

This building would have to be built as an extension (North side) of the building #301/302, to reduce the length of the heavy-duty way as much as possible.

This building must be designed for both reactor compartments disposal. Its design goes behind the limits of this project. However, the main safety provisions are given below.

The guarantee of the confinement lies on the guarantee of the integrity of the two barriers; the first one being the RC steel cover and the second one being the new disposal building. The building must give this guarantee by protecting the RCs against all external aggressions like corrosion, earthquake, storm, flood, fire and intrusion :

Earthquake:

The building must be designed to guarantee air confinement in case of earthquake. That means it would be necessary construct a kind of blockhouse with very thick slab and walls.

Flood and bad weather:

The building must be erected upper than the highest water level, and resist against wind, rain and snow. Moreover, it should be designed making use of very efficient underground water proofing systems. The figure below shows the principles for underground waterproofing that were implemented for the French repository for low and intermediate level waste Centre de l'Aube.

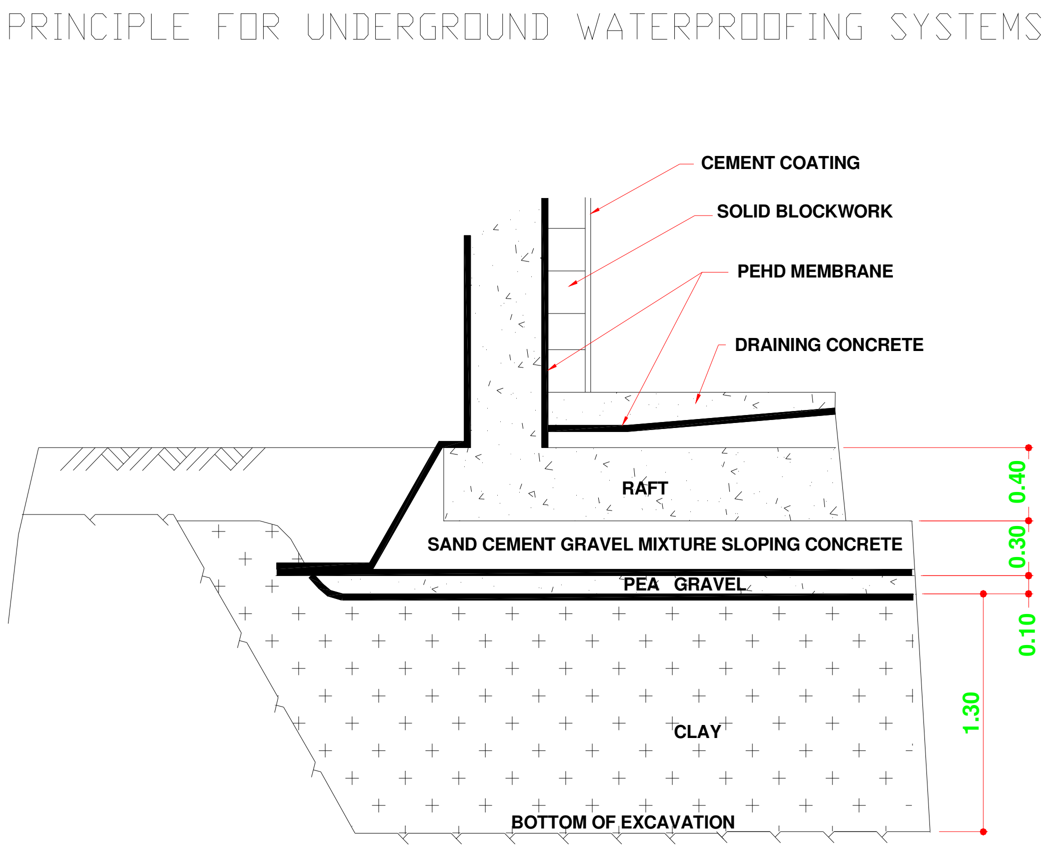

Figure 3

Providing against Corrosion Risks and Radioactivity confinement :
It is essential to install an air conditioning system in the new building with ventilation, filtration, dryer and hygrometry analysis. This air conditioning system would treat the air inside and outside RCs to ensure the air-tightness of the first barrier constituted by the RC shell (See §5.2.3).

The disposal building could be fitted out with a system that could be designed as shown in § 5.2.3.

5.3.3 Building a heavy-duty route

After erecting a new disposal building, a heavy-duty route is to be constructed between the sarcophagi and the new building.
This route could be designed for railway or wheel transfer. It must be as plane as possible to facilitate the transfer.
The following work is to be done, to end to a route which constitution is similar to an airport takeoff and landing runway:

- Set reinforcements to support the building upper structures,
- Dismantle the low structures of the building along the axis of the route, 10 m wide,
- Excavate 2 to 4 meters under the future road level (0.0 m)
- Filling the openings with a stable bed of clay, gravel and rocks,
- Setting a steel reinforcement structure between 0.0 m and –0.5 m,
- Pouring concrete on the structure; with expansion joins every 10 meters,
- If rail transfer is choosen, fitting out the road with railways.

5.3.4 Transferring the RCs to the disposal building

Transferring the RCs in their entirety will be a difficult operation (each one weights about 1000 tons), that has been studied by VNIPIET institute.

The work is similar from one RC to the other. Reactor Compartment #2 would be transferred first to the disposal building. After that, Sarcophagus #2 would be completely dismantled, and the heavy-duty road would be prolonged towards Sarcophagus #1. Then Reactor Compartment #1 would be transferred to the disposal building, according to the sequence resumed below (see schematic flowcharts in appendix 3) :

- Prior to carry out dismantling works, some temporary systems of technical support should be provided :
  - Ventilation and dust exhauster equipment,
  - Compressed air for pneumatic equipment,
  - Power supply for welding and electric tools,
  - Gas supply for gas-cutting operations,
  - Fire fighting systems.
- Dismantling of a part of the lateral reinforced concrete walls below RC, to allow man, material and tools access under RC.
- As dose rates under the reactor vessel is quite high, fitting out RCs with a biological shielding (except if the transferring works are carried out after storage period long enough so as the hand-on gamma dose rate under the reactor vessel is lower than 2 mSv/h, and dose rate at a distance of 2 meters lower than 0.1 mSv/h).

- Dismantling the roof of the sarcophagi to permit crane access upon RC.
- Dismantling the end wall of the sarcophagi (the one connected to the heavy-duty route).
  Prolonging the heavy-duty road and railways inside sarcophagus.
  Removal of the movable concrete biological shielding,
- Dismantling the biological shielding and tanks under RC, and the steel ladders, gangways, etc.
- After the required operating space is provided under RC, mounting and welding a set of support pieces under the RC (close to fr. 82-85 and fr.103-105).
- Removal of RC foundation support in the area of fr. 100-102, and fr. 98-99.
- Installing safety mechanical support under the RC shell.
- Introducing hydraulic jacks under the support pieces (capacity : 300-350 tons),
- Cutting off the RC foundation supports close to fr. 88-96, and lifting RC by stages by the mean of special purpose hydraulic jacks. As RC foundation is located below one meter the building floor, the lifting height should be 2.5-3 meters or so. In order to ensure synchronic lifting, the working fluid should be supplied to jacks from one hydraulic station. The mechanical safety supports should be adjusted after each lifting stage to ensure safety of operations carried out under RCs : holding up the RC only by the mean of hydraulic jacks should not be allowed.
- Removal from under RC cut-off sections of RC foundation supports,
- Introducing a special support in the free space released by RC lifting, which purpose is to artificially upper ground level up to the main building floor, as well as laying railways for slipway trolleys.
- Introducing several mobile trolleys under RC special supports that are welded to the RC shell.
- Leaning the RC down on the trolleys by means of hydraulic jacks.
- Transferring the RC towards the new disposal building by means of the trolleys. Prior to transportation, the compartment should be fastened to the beams of slipway trolleys.
- Leaning RC on its new support structure by the mean of the same sequence in the opposite order.

This whole operation is rather complex and will need a significant amount of labour.

As explained in § 4.3.2.1, the Reactor Compartments transfer as described above is not fully consistent to the radioactive waste transport regulation due to their total enclosed activity, but as the waste packages are transported on tens of meters with no exit of the Paldiski site, we think that this option could be reasonable. This option is the one that was selected by TECHNICATOME to transfer in 1993 the reactor compartment of the first French nuclear submarine called Le Redoutable to the building dedicated to its temporary storage.

5.3.5 Dealing with the problem of small radioactive sources scattered in concrete inside RC#1

The problems arising from the presence of small radioactive sources is exactly the same as for option #1 (See § 5.2.6) : in accordance with Estonian radioactive waste management regulations and IAEA recommendations, it is in theory necessary to extract these sources from the RC#1 before disposal.

For the same reasons, we think that it could be possible to deviate temporarily from the regulations regarding radioactive waste management : the volume of concrete in which sources have been poured has to be correctly marked out so that it will be possible to easily extract the sources in 50 years if necessary, when the radiation dose rate in the reactor compartments is lower.

5.3.6 Surveillance program

The surveillance program of Reactor Compartments and Disposal building should be the same as for Option #1 (See § 5.2.7), taking into account that the second confinement barrier is now the disposal building instead of the sarcophagi.

This surveillance procedure should be in force for a minimal period of 300 years.

5.3.7 Advantages and drawbacks of this decommissioning option

Advantages	Drawbacks
– Except radioactive sources extraction, the works to be carried out would not imply high men exposure. – Reactor compartments transfer to the disposal building could be considered as a radioactive waste transport, but this the radioactive waste (the reactor compartments) remains on the Paldiski site. – The risk of radioactivity release into the environment is reasonably low.	– This decommissioning strategy is quite simple, but requires heavy works to transfer the RCs into the disposal building. – As a consequence, the global cost would be much higher than for option #1. – The Reactor Compartments transfer as described above is not fully consistent to the radioactive waste transport regulations due to their total enclosed activity, but as the waste packages are transported on tens of meter with no exit of the Paldiski site, we think that this option could be reasonable. – Waste packages (RCs) are not consistent to the IAEA recommendations regarding waste management. – Waste packages are not totally immobilised into the final disposal.

5.4 OPTIMIZATION TAKING INTO ACCOUNT A STORAGE PERIOD OF 10 OR 50 YEARS BEFORE STARTING OPERATIONS.

As said in § 4.3.3, the influence of the storage period is not really significant regarding men exposure considering this first dismantling stategy, as dose rates outside the reactor compartment is quite low, except under the reactor vessel. However, after a 50 years storage period, gamma dose rate under the reactor vessel should be lower than 0.1 mSv/h, and so the transport of the reactor compartment as a whole would be possible in total accordance with the transport regulations regarding dose rate, without fitting out the RC with biological shielding.

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But as explained in §4.2, we think that it's not possible to guarantee the reliability of the confinement barriers for 50 years without implementing complementary safety provisions from now on. The works to be carried out are described in § 6.2.

Moreover, the influence of the storage period is not significant at all regarding waste volumes, as the waste volume is equal to the reactor compartments volume.

For these reasons, we think that waiting fifty years before implementing this first decommissioning strategy brings no significant advantage, but implies additional cost to guarantee the reliability of the confinement barriers for the whole storage period.

As a consequence, we recommend that this first decommissioning strategy would be implemented as soon as possible.

6 STRATEGY # 2: COMPLETE DISMANTLING OF THE RCS

6.1 PRESENTATION OF THIS DISMANTLING STRATEGY

The aim of this dismantling stategy is to release the site as quickly as possible, by performing complete dismantling of reactor compartments, packaging the resulting waste in accordance with Estonian regulations and IAEA recommendations, and transporting the waste packages to a final disposal near-surface facility, being located in Estonia.

As seen in § 4.3.3.1, a storage period is essential before performing this dismantling strategy, that implies heavy dismantling works near gamma radioactive sources. For this reason, these dismantling options will be developed taking into account a 50 or 100 years storage period. In the following, this storage period will be assumed to last 50 years. The advisability of extending the storage period to 100 years will be examined at the end of this section (See § 6.9).

The framework can be resumed as follows :

1. Restoration of standardised storage conditions for the storage period :
   - Improvement of RCs resistance against corrosion.
   - Sarcophagi strengthening and improvement of sarcophagi confinement properties.
   - Improvement of sarcophagi resistance against flood.
   - Implementing the surveillance program.

2. During the storage period, the following works should be performed :
   - Building of the Estonian radioactive waste surface storage site.
   - Transfer of the waste already stored in building #301/302 to this storage site.
   - Building of a “packaging workshop” in building #301/302.
   - Upgrading of the 50 tons crane lift of this building.

3. Dismantling of the RCs into pieces to be transferred to the “packaging workshop”.

4. Packaging of the arising radioactive waste. Two different options will be considered for radioactive waste packaging :
   - Option # 1: Making special waste packages minimising cutting works.
     Special waste packages are prepared in the “packaging workshop” in accordance with the storage rules. These packages are made from whole NPU systems like reactor vessel, steam generator vessels, etc , minimising cutting works.
     The final volume of waste might be high, but the dismantling and packaging operations are simplified and men exposure minimised according to the A.L.A.R.A. principle.
   - Option # 2: Minimising the volume of definitive wastes.
     The aim of this packaging option is to minimise the final volume of waste, using techniques like decontamination, compaction, recycling by the mean of melting devices, etc. Most of the additional work is done in the “packaging workshop”.

5. Release of the site after dismantling and decontamination of the sarcophagi.

6.2 50 YEARS CONSERVATION PERIOD

This conservation period is necessary to optimise the doses of the workers during the dismantling works, as seen in § 4.3.3.1.
During this conservation period, it is necessary to restore standardised storage conditions. The arrangements to be implemented are in principle similar to those already proposed for the 300-year definitive storage in situ (first decommissioning strategy – option 1, see § 5.1). However, they are generally much lighter.

6.2.1 Improvement of RC resistance against corrosion

The reliability of the first confinement barrier constituted by the RC shell must be ensured for 50 years. As the presence of oxygen and moisture inside and outside the RC may generate corrosion, it is essential to provide against first barrier against this risk. In order to carry out corrective actions if air hygrometry is too high, it's necessary to fit out RCs and sarcophagi with an air ventilation, filtration and dryer system. The ventilation system allows to:

- circulate the air through an air dryer until hygrometry inside RC and inside sarcophagi so as humidity level decreases to 30 to 40%,
- air sampling from Reactor Compartment and chemical and radiological analysis,in order to detect potential air contamination (late detection of corrosion).
- Checking if the RC are still airtight by means of a pressure test.
- Checking if the sarcophagi are still airtight by means of a pressure test.

This system could be designed as shown in § 5.2.2.

6.2.2 Sarcophagi strengthening and improvement of sarcophagi confinement properties

The sarcophagi should be consolidated by implementing the following works (See appendix 5) :

- Injections of calcium silicate at the sites of cracking,
- Surface application of thick cement mortar reinforced with fiber-glass mesh to the inside walls,
- Additional concrete skins poured in situ and anchored on the existing walls,
- Injections of fine rendering cement mortar and bentonite-cement grout on the underside of the raft through previously inserted casings,
- Sarcophagi should be made airtight as much as possible by setting joints between concrete plates, by sticking bitumen plates on the cracks, etc. Every opening devoid of high efficient filtration should be sealed.

6.2.3 Improvement of sarcophagi resistance against flood

In order to upgrade sarcophagi resistance against flooding, following measures should be taken, depending on the occurrence of flooding in Paldiski :

- The concrete floor slab of sarcophagi may be waterproofed by covering it with bitumen.
- A peripheral drainage may be implemented around sarcophagi to reinforce the slab dryness at usual underground water levels.
- Cofferdams may be installed to prevent from sudden flood in case of temporary high water levels.

The existing main buildings (301/302) protect the sarcophagi from adverse climatic conditions (rain, wind, snow, etc). Following the roof and load-bearing wall surveys, some consolidation work may be carried out to guarantee these functions over a time span equal to the sarcophagus operating period for final disposal.

6.2.4 Surveillance program

It is essential to check regularly that the above arrangements are sufficient to ensure the confinement of radioactive materials for the whole storage period. The following periodic inspections allow to check the integrity of the two confinement barriers :

- **Monthly**: inspection inside building and outside sarcophagi: structures, detection of roof leakage and wall cracking.
- **Quarterly**: inspection inside sarcophagi and outside RCs: measurement of air hygrometry and air contamination, detection of possible water condensation and corrosion traces.
- **Annual**: inspection inside RCs: measurement of air hygrometry and air contamination, detection of possible water condensation and corrosion traces, measurement of RCs and sarcophagi confinement airtightness.

If any abnormal event is detected, corrective actions are to be carried out. If any leakage, condensation, corrosion, abnormal hygrometry is detected, corrective actions are to be carried out rapidly.

The above surveillance procedure is similar to that in force for storage of reactor compartments of French submarines (see § 4.1.5.4).

6.3 PREPARATIVE WORKS

6.3.1 Detail studies

Prior to start dismantling works, studies have to be carried out to define:

- Techniques for dismantling operations,
- Study of risks (safety studies),
- Safety arrangements to install (defense against radiation, fire detection, etc.),
- Physical and radiological inventories and zoning,
- Technical support that should be provided :
  - Ventilation and dust exhauster equipment,
  - Compressed air for pneumatic equipment,
  - Power supply for welding and electric tools,
  - Gas supply for gas-cutting operations,
  - Fire fighting systems.
  - Building 301/302 lighting upgrading.
- Upgrading of the 50 tons crane,
- Building of the "packaging workshop", according to the option of packaging # 1 or 2
- Workers changing rooms, bathroom and toilet, personal and material accesses
- Tools and special equipment supplying for dismantling operations.

The required time to carry out studies may be one year or so.

6.3.2 Physical and radiological inventories and zoning

These inventories represent the major entry data of the dismantling studies. They should be carried out not only on the basis of the data collected in the course of the first task of this project, but also owing to on-site inspections, especially inside reactor compartments.

Taking into account the operation history of the nuclear steam supply systems, the inventories will allow to define a zoning connected with radiological risks and waste categories. This zoning has to be marked up on the field. Thanks to this zoning, the criteria for the setting of individual and collective radiological protections will be defined.

These inventories have to be fully detailed, as on they will be the basis for quantifying, defining and planning the work to do, and optimising human and material resources and doses.

6.3.3 Upgrading of the 50 tons crane

This is the most significant tool of the dismantling works. Its upgrading to about 50 tons lifting capacity is essential.
The 301/302 building structures may be reinforced. The rolling track of the crane should be lined up. A new beam and a new trolley should be mounted on it. Its electric power supply should be renovated.

6.3.4 Auxiliary systems

Studies will define what are the needs in terms of safety arrangements like defense against radiation, fire detection, etc.
They will also define what are the needs in terms of power supply for welding and electric miscellaneous tools, compressed air for pneumatic tools, breathable air for workers, water supply, used water treatment, lighting, ventilation, compartmentalisation, etc..

6.4 BUILDING OF THE PACKAGING WORKSHOP

Big nuclear powered unit components will come one by one from the reactor compartments during dismantling operations, by the means of the 50 tons crane.
The workshop size should be about 10 m wide x 20 to 40 m long (depending on the packaging option selected) x 5 m high. It should be located in the building #301/302, between the two sarcophagi.

It will be equipped with the following utilities, depending on the packaging option selected (See § 6.6 and 6.7) :

6.4.1 Waste reception area

The waste is introduced into the workshop through large openings located in its roof. These openings are normally closed. The waste is then diagnosed and orientated towards the different processing areas.

6.4.2 Transfer of waste inside workshop

Transfer of waste from a processing area to another is made by the means of the 50 tons crane if the piece is particularly big and/or heavy (reactor vessel), or by the means of ground rolling trolleys or forklift trucks in the case of little and light pieces.

6.4.3 Thermal cutting room

A thermal cutting and welding room is equipped with an oxyacetylene torch, with plasma arc cutting, and arc welding.
This area is fully closed and separated from the rest of the workshop. It is equipped with an adequate ventilation, pre-filtration and filtration device, and with fire and explosion detection.

6.4.4 Dismantling area

It is a big open place where actions of volume reduction of waste are made. The area can be divided into small rooms if several operations are made simultaneously and if necessary (because of noise, dust, etc.)
Manual dismantling, radiological and material sorting, mechanical cutting operations are made in this dismantling area. It is equipped with several adequate tools like saw, shears, moving ventilation and filtration device, etc..).
The dismantling area is also equipped with 2 longitudinal monorail cranes, which capacities are respectively about 10 and 2 tons, and 4 m high under hook.

6.4.5 Decontamination room

A decontamination room is equipped with basic mechanical equipment like brushes, scrapers, etc. and with a spray-paint device for labile contamination fixing.
This room is fully closed, and separated from the rest of the workshop. As for the thermal cutting room, it is equipped with an adequate ventilation, pre-filtration and filtration device.

6.4.6 Packaging area

In this area, waste is put into definitive packages and blocked. This area is equipped with a cement mixer and a cement-grouting device.
Several packaging containers are available, corresponding to the material and radiological sorting imposed by waste management rules.

6.4.7 Radioactivity Measuring room

In this room, total radioactivity of packages is controlled by direct measurement. The room is equipped with radiological probes adapted to the geometry of the several packages like parallelepiped cement containers, metallic drums, etc.
The ground, roof and walls of this room are made of thick concrete to prevent from measuring background radioactivity from the other workshop areas.

Measurements of external labile contamination and irradiation of containers are also made in this room, in accordance with transport rules. Containers are labelled before their transfer to the dispatching area.

6.4.8 Dispatching area

In this area, we find a storage zone for empty and full containers. Trucks can deliver and load the empty and full containers through a big airlock opening, by means of the 50 tons hoisting device.

6.5 DISMANTLING WORKS

6.5.1 Usable Techniques to carry out dismantling operations

As every intervention in the sarcophagi arises risks for workers health or for environment, the dismantling sequence should be carefully studied before acting.

The usual dismantling techniques are used, but nuclear safety provisions are taken into account.

6.5.1.1 Concrete dismantling techniques

Generally, concrete and steel-concrete are dismantled with diamond cable saw, circular saw, jackhammers, core drill, hydraulic jacks and shears.

If it is sure that steel concrete is not contaminated, it may be dismantled with oxygen lance or jackhammers in addition with oxyacetylene torch.

These operations are to be carried out very carefully due to the fire risk, the classical handling risks, and the contamination dispersion risk related to smoke, gas and dust, or water-cooling.

6.5.1.2 Metal cutting

Generally, black steel and other metals are dismantled with oxyacetylene torch, plasma arc cutting, and abrasive circular saw.
If the waste to cut is contaminated, or activated, "hot" cutting may generate radioactivity dispersion due to the smoke, gas or dust. This problem can be treated as follows:
- Local decontamination of the cutting zones (drawing of a "cutting map" on the piece) for contaminated waste pieces,
- Mechanical "cold" cutting with jigsaw, scroll saw, band saw, hydraulic shears, etc. for activated waste pieces.

6.5.1.3 Local confinement

Every dismantling work that arises a risk of radioactivity dispersion is carried out inside a local plastic sheet tent, with the help of a local mobile ventilation and high efficiency filtration device. (2 000 to 4 000 m3/h for each tent)

6.5.1.4 Handling risks

The preparatory studies will have to verify the safety of each handling operation:
- RC stability while removal of heavy components,
- Existence of handling points on components, and their test before use,
- Estimation of the weight of the pieces taken by the crane,
- Etc…

6.5.2 RC #1 Dismantling sequence.

The sequence of dismantling works to be carried out in order to perform the complete removal of unit 346 A reactor compartment enclosed equipment an shell is given in appendix 4. Every waste extracted from the RC is brought to the packaging workshop with the help of the 50 tons crane.

This dismantling sequence is made up of 11 successive stages, that are numbered from 0 to 10 in appendix 4.

6.5.2.1 Stage 0 : Replacement of the sarcophagus roof

See Appendix 4 folio 2/27 and 3/27.

In order to facilitate waste evacuation towards packaging workshop, the present concrete roof is to be dismantled.
This work is done from a big scaffolding structure mounted inside the sarcophagus, under the roof, and with the help of the 50 tons crane.
A new metal structured roof, with appropriate sliding trap doors, are set instead of the old one.
Several openings should be available on the whole surface of this new roof; they can be closed when they are not in use, restoring a good confinement.

6.5.2.2 Stage 1 : Dismantling upper structures of RC

See Appendix 4 folio 4/27 to 7/27.

The upper metal gangways, stairs, etc. are dismantled with the help of the scaffolding structure used for roof replacement.
The upper half part of the RC is cut into several pieces and evacuated. The concrete poured on reactor vessel cover and on hatches is demolished for man and material access to lower compartments and reactor vessel. Small sources are localised with very much care, separated little by little from concrete, and then sent to a special workshop to be destroyed or stored separately.

Miscellaneous waste is gradually evacuated to clear up RC upper roof. Then, the scaffolding structure is dismantled.

6.5.2.3 Stage 2 : Extracting Pressurizers

See Appendix 4 folio 8/27 and 9/27.

The biological shielding located on the floor above pressurizers room is dismantled (lead). The floor is cut to create a hatch for waste evacuation.
Biological shielding around circuits is removed (carborite). Circuits around pressurizers are drilled at lower points to remove the potential remaining primary water.
Piping is cut at critical points to separate the six pressurizers. Plugs are welded on openings. Pressurizers elements are unscrewed from their supports and evacuated one by one, so as for the connected piping.

6.5.2.4 Stages 3 and 4 : Extracting Steam generators and activity filters

See Appendix 4 folio 10/27 and 11/27 (stage 3).
See Appendix 4 folio 12/27 and 13/27 (stage 4).

The same operating process as above is used to extract steam generators and activity filters.

6.5.2.5 Stage 5 : Removal of starboard side biological shielding

See Appendix 4 folio 14/27 and 15/27.

Blocks of biological shielding on the starboard side and around reactor vessel are dismantled and evacuated. Operations around reactor vessel and Iron-water shielding tank are carefully prepared. Exposure time to radiation, operating procedures, and biological screens are systematically defined before starting operations. Workers are prepared and trained before operations.

6.5.2.6 Stage 6 and 7 : Extracting reactor vessel

See Appendix 4 folio 16/27 and 17/27 (stage 6).
See Appendix 4 folio 18/27 and 19/27 (stage 7).

After removal of reactor upper biological shielding, primary piping around vessel is cut at critical points, opening are immediately sealed by plug welding. Primary circuits piping connected to the vessel are cut. Reactor vessel fastenings are cut.

A cylindrical biological shielding is prepared in the workshop to receive reactor vessel as soon as it arrives. A circular plate biological shield is also prepared to fill the hole left by the reactor vessel in the IWS tank.

Reactor is evacuated from RC to the workshop while biological shielding is set on hot points. During this particular handling, the number of workers is limited to the strict necessary and they keep as far as they can be from central part of the vessel. The weight of the reactor with internal parts inside is 40 tons. Once in the workshop, the reactor vessel is filled with light concrete as soon as possible, to lower its radiation dose rate.

Primary circuit piping that was connected to the vessel is evacuated in 4 to 6 pieces. The longitudinal bulkheads between SG rooms and reactor room are dismantled.

6.5.2.7 Stage 8 : Extracting Iron-water Shielding tank

See Appendix 4 folio 20/27 and 21/27.

The same operating process as above for reactor vessel is used to extract iron-water shielding tank. The weight of IWS tank is 52 tons.

6.5.2.8 Stage 9 : Dismantling U shaped room, current converter and pumps

See Appendix 4 folio 22/27 and 23/27.

After removal of the steel and lead plate located above the pump enclosure, the concrete monolith block is cut into several pieces. Each block weight is less than 50 tons. Cuts are made at a distance from primary piping and from radioactive sources poured in concrete, if possible.
Once in the workshop, if necessary, pieces are cut again up to conform to the containers dimensions. Small sources are localised with very much care, separated little by little from concrete, and then sent to a special workshop to be destroyed or stored separately.

6.5.2.9 Stage 10 : Dismantling RC lower structures

See Appendix 4 folio 24/27 to 26/27.

Remaining structures like lower half part of RC, support structure, tanks, biological shielding, metal passageways, stairs, scaffolding structures, etc. are gradually dismantled.
Walls and soil of sarcophagus are controlled and cleaned if necessary using jackhammers and aspirators with high efficiency filters and decanting pots for contaminated rubble and dust.

The final situation is shown on Appendix 4 folio 27/27 : the last step is the dismantling of remaining sarcophagus structures, with the same techniques as for any other concrete building.

6.5.3 <u>RC #2 Dismantling sequence</u>

The dismantling of RC #2 should be carried out with the same methodology, the same tools and the same techniques as those described above for RC #1.
RC #2 is less activated than RC #1. As its design is more recent and modular, it will be easier to dismantle. Differences between RC #1 and 2 are not sufficient to justify a complete change in methodology.

All big components like pressurizer, steam generators/primary circuit pumps, reactor vessel, filters, can be vertically removed from the shielding structure with a hoisting device. Classical biological screens made of lead plates or special cylindrical and circular plane steel biological shielding are to be utilised when working on high-activated material.

The dismantling sequence of RC #2 enclosed equipment was studied by VNIPIET. The corresponding section of the VNIPIET report is attached in appendix 6.

6.6 PACKAGING OPTION # 1: MAKING A FEW BIG DEFINITIVE PACKAGES

6.6.1 <u>Packages definition</u>

Special waste packages are prepared in the “packaging workshop” in accordance with the disposal regulations. These waste packages are made from special steel containers designed to contain reactors pieces of equipment (reactor vessel, steam generator vessels, etc.) without performing cutting works.

This option is similar to the one selected to carry out future level 3 dismantling works of the French nuclear submarines reactor compartments. The final volume of waste is quite high, but the dismantling and packaging operations and investments are lighted. In other respects, avoiding cutting works allows to lower men exposure.

The dismantling of both reactor compartments would result in 36 special packages :

- Reactor vessel (x2)
- Steam generator vessel (x13)
- Pressurizer vessel (x9)
- Activity filter vessel (x4)
- RC #1 primary circuit room (x4)
- Heat exchanger (x2)
- RC #1 Shielding tank / RC #2 biological shielding (x2)

Four of these containers will weight approximately 60 tons : those containing the reactor vessels, and those containing RC #1 Shielding tank and RC #2 biological shielding.

As seen in § 4.3.2.1 :
RC#2 may be transported in a container A type after the year 2030.
- RC#1 may be transported in a container A type only after the year 2165.

As a consequence the special container aimed to contain the reactor vessels must be designed as A type container (as it seems not to be realistic to design a B type container for components as big as reactor compartments). Regarding RC #1 vessel waste package, the total enclosed activity will remain after 50 years more than two times higher than the transport limit. If the final disposal site is located on the Paldiski site, we think that this option could be reasonable as this waste package is transported on tens of meter. If the final repository is not located on the Paldiski site, it will be necessary to study if it is possible to deviate from the regulations regarding radioactive waste management, provided that complementary safety measures could be adopted for this particular transportation.

About 1000 tons of additional low contaminated radioactive waste (concrete, steel, lead, carborite, etc.) will be packed in about 300 standard parallelepiped concrete containers. Waste volumes will be estimated in the course of task 3 of the project.

6.6.2 Packaging procedure

The special container is placed in the waste reception area (See §6.4.1) before extracting the reactor component from the RC. Scaffoldings are erected beside the container in order to be able to carry out the following works.

As soon as extracted from the RC, the reactor component is put into the container. The container is immediately grouted with light concrete up to 80 % to lower men exposure. Drills are performed in the component surface to allow filling it with light concrete. Container grouting is then ended, and the container cover is put in its place. The container is then transferred in a waiting area (Dispatching area : See § 6.4.8), for concrete solidification.

Regarding standard containers for miscellaneous radioactive waste, like pipework for instance, they are placed in the dismantling and waste sorting area (See §6.4.5) and transferred when full to the packaging area (See §6.4.6) where they are grouted, before being transferred to the dispatching area.

This packaging option allows to simplify the packaging workshop : there is no need for a large dismantling area, nor for a specialised decontamination room.

6.7 PACKAGING OPTION # 2: MINIMIZING THE VOLUME OF DEFINITIVE WASTE

The aim of this option is to minimise the final volume of waste, using all the techniques like decontamination, compaction, recycling using melting devices, etc…

It addresses the feasibility of deploying a decontamination strategy involving the:

- Use of In-Situ re-circulation decontamination of the reactor primary coolant circuits to assist with the dismantling and disposal of the reactor components by reducing the associated dose levels of the coolant system.

- Use of tank decontamination of reactor components following their decommissioning to further reduce dose or to recategorise the wastes produced

- Use of Melting to volume reduction of wastes and or free release of some reactor components

In reviewing the decontamination strategy the report seeks to identify the additional requirements and related problems associated with this strategy.

6.7.1 Design Parameters

The In-Situ decontamination plant design has been based upon the use of 'local decontaminant re-circulation using processes which could involve strong mineral acids as the decontaminating agents.

Data (Ref. 1) shows that approximately 99% of the radioactivity is within the reactor vessels with the remaining 1% in other materials affected by intensive neutron exposure and on the surfaces of equipment and pipework within the primary cooling circuit where active corrosion products have been deposited.

It is assumed that only 0.1% of the total radioactivity is due to radioactive corrosion products deposited as a thin film on the internal surfaces of the reactor pressure vessel, primary cooling circuit components and piping (Ref. 6.2).

The use of decontamination agents would assist in removing the radioactive corrosion products from the reactor primary coolant circuit components but will not remove neutron activation products in the cooling circuits themselves or the reactor structures.

In-Situ decontamination would therefore have little effect on the dose within the reactor compartments but will help to minimise the spread of contamination during dismantling and size-reduction of the coolant circuits.

Following 'In-Situ' decontamination the plant and equipment would have to be dismantled and size-reduced. This would have to be by either semi-remote or manual means. The actual approach to dismantling would have to be dictated by the radiation dose that could be collected during the work. A series of As Low As Reasonably Practicable (ALARP) assessments would have to be carried out to determine the correct approach.

The size-reduced waste from the reactor cooling circuits would then have to be categorised by radiation measurement and sorting. Following this there is the option to re-categorise this waste by further decontamination. This would involve the use of tank immersion decontamination.

The use of either the In-Situ or tank decontamination will generate liquid effluent, which contains 'spent' decontaminants. This effluent will require further treatment before it can be discharged to sea. These 'spent' decontaminants could contain high levels of radioactivity (mainly Co^{60}), heavy metals (Iron, Nickel, Chromium) and anions (Nitrate, Fluoride, etc. dependent on the decontamination process used).

Along with the ALARP dose assessments the benefits of both of these decontamination steps would have to be viewed as to whether it was the Best Practicable Environmental Option (BPEO) for the disposal of the wastes. This study would have to take into account several key factors including the:

- Estonian waste disposal criteria
- Availability of a Estonian repository
- Effluent treatment routes and discharge limits
- Associated costs of further treatment

The requirements for the decontamination are further developed later in this report.

Lastly the use of melting as a means to reduce the volume of certain categories of solid waste consigned to an Estonian repository and to maximise the recycling of materials for re-use would require a similar BPEO study.

The requirements for the use of melting are further developed in Section 6.7.12 and Appendix 7.2.

6.7.2 <u>Duty</u>

The following primary circuit components could be considered available for the use of In-situ decontamination. Following their dismantling, size reduction and sorting they would then be available dependent on dismantling and disposal strategy for further decontamination and or melting.

6.7.2.1 Unit 1

Steam generators (8 off)

Pressurisers (6 off)

Coolers (3 off)

Main primary pump

Auxiliary pump

Activity filter

From current information all the pipe connections to the primary cooling circuit and the associated draining/drying system are plugged or welded. It is believed that the primary cooling circuit is complete but its ‘hold-up’ volume is not known. Currently it is not known whether it is possible to make ‘In-Situ’ decontaminant flow and return connections to the primary circuit from outside of the Reactor chamber. An additional complicating factor to the use of In-Situ decontamination is that the primary circuit room is grouted with concrete.

6.7.2.2 Unit 2

Steam generators/primary coolant pumps (5 off)

Pressurisers (3 off)

Filter cooler

Ion-exchange filter

Shutdown cooling circuit pump

Valves & piping

The primary cooling circuit is all located within the Shielding tank. The biological shielding plus a layer of concrete are located above the Shielding tank. It is not known whether it is possible to make ‘In-Situ’ decontaminant flow and return connections to the primary circuit from outside of the Reactor chamber. It is believed that the primary cooling circuit is complete but the ‘hold-up’ volume is not known.

Data on the hold-up volumes for all the vessels in units 1 & 2 would have to be supplied. This would allow the establishment of the hold-up volumes of the vessels and associated pipework, this information is key to the size of In-Situ decontamination plant required. Work being carried out by VNIPIET is seeking to determine this information.

6.7.3 Radiation levels

The gamma-radiation levels in the areas where work would be carried out on primary circuit equipment dismantling or where connections would be made for 'In-Situ' decontamination have been estimated from the data supplied by VNIPIET (Refs. 6.1 & 6.3). Overall the radiation doses in the reactor compartments according to this data appear to be very low. The information is summarised as follows:

6.7.3.1 Unit 1

The primary circuit equipment is all located on the first floor. The primary circuit room was grouted with concrete. Before grouting the gamma-radiation level was 110-280 μSv/h (microSievert/hour).

6.7.3.2 Unit 2

1 μSv/h - 70 μSv/h (microSievert/hour) assuming that the Biological shielding concrete has been removed and the Shielding tank refilled with water. The higher dose is associated with work local to the reactor. Other working areas are typically 1-10 μSv/h (microSievert/hour).

6.7.3.3 Impact of Doses

In one current BNFL project the aim is to reduce the dose levels by decontamination to below 50 μSv/h (microSievert/hour) to allow the decommissioning of the equipment. This work as with all decommissioning work needs to be supported by ALARP assessments. These assessments need detailed information given by radiation dose maps. These dose maps could be developed either from detailed monitoring or by the use of RADSCAN techniques.

Using the 50 μSv/h (microSievert/hour) criteria above and from the limited dose information it appears that Unit 1 may benefit from some In-Situ decontamination if the dose is believed to be associated with the primary coolant system. However it would appear that with Unit 2 there is less justification for the use of In-Situ decontamination as the doses are already relatively low.

6.7.4 Decontamination Processing Loops

According to information supplied by VNIPIET the primary circuits are complete (Ref. 6.9) but it is not known whether the primary circuit valves are open or not. A complicating factor that was previously noted in Section 3 of this report was that concrete had been poured into the primary circuit rooms. This means that the operation of the valves in this area is likely not to be possible. Overall it would indicate an altered approach to the In-Situ decontamination of the primary circuits one which would require the connection to individual vessels.

The internal details of the primary circuit vessels are not available but based upon external dimensions it is estimated that the liquor hold-up in the largest loop, assuming each vessel is decontaminated separately, is $1m^3$. If the primary circuits are intact the liquor hold-up could be as much as $7\text{-}10m^3$. Again this information is key to the correct design size of the In-Situ Decontamination Plant.

6.7.5 Decontamination Processes.

The selection of a decontamination process for use either as the In-Situ or tank decontamination of materials requires the understanding of the materials to be treated.

With the primary material of construction used for the active process pipework and vessels being stainless steel (18Ni.9Ti.1Cr.). Further when this is subjected to extreme conditions, e.g. high temperatures, corrosion, abrasion, etc. and contact with active materials, its effective decontamination is only likely to be achieved by the removing the surface of the substrate metal. This has to be carried out with an aggressive chemical, electrochemical or physical process

Previous work by both BNFL decontamination group and MSC Tennessee a subsidiary of BNFL (Ref. 6.4) has surveyed the literature to identify suitable decontamination technologies. Strong mineral acids have been identified as potential ‘hard’ chemical processes for decontamination of stainless steel and mild steel. This is because they are sufficiently aggressive to give a relatively high decontamination factor based upon a high dissolution rate accompanied by a low processing time. Additional factors in their favour are that they :

- Are relatively cheap
- Have well defined chemistry
- Have well proven methods of treatment when exhausted. They are also effective for a wide variety of feedstocks.

Additional factors, which have to be taken into account in the selection of a process, are that it:

- Be effective and capable of reducing the dose uptake local to the equipment decontaminated
- Reduce the safety risks to personnel by reducing the activity/contamination present during dismantling/size reduction
- Minimise the production of secondary wastes.
- Be relatively simple to use and minimise overall lifetime costs.
- Be compatible with existing or proposed liquid effluent disposal routes and treatment facilities. Alternatively be capable of being treated locally or regenerated or recycled

Laboratory development work at BNFL Sellafield, operational experience at MSC, a BNFL subsidiary, and data from the literature has identified a number of mineral acids which are currently being assessed for their suitability for future use in the design of the BNFL In-Situ decontamination plant. Ongoing laboratory testwork and full-scale design activities will determine if they all remain appropriate as decontaminants. This work will also determine the optimum concentrations of the decontaminants, operating temperature, for use on the full-scale plant.

BNFL is also reviewing decontamination processes with the potential to decategorise decommissioning wastes. This includes the development of a strategy involving the building of a pilot plant for immersion decontamination. This strategy could provide valuable data on the decontamination of Plutonium Contaminated Material (PCM) or Intermediate Level Waste (ILW) to Low Level Waste (LLW) also the conversion of LLW to background levels. The overall driving force behind this work is to help in minimising the overall waste management costs.

Further detail on decontamination processes can be found in Appendix 7.1 of this report.

6.7.6 Effluent Treatment

The treatment of the liquid effluent generated from either In-Situ or tank decontamination treatment must be considered on an overall site basis. This would involve the evaluation of each effluent stream. Any new effluent discharges can then be assessed for compliance with this strategy and an appropriate treatment determined.

In the absence of information about:

- An overall strategy for effluent treatment
- Knowledge of the overall site effluent discharges
- Integrated site waste management strategy
- Associated time scales

It is not possible to comment in detail, on the integration of the effluent treatment policy for the decommissioning of the sarcophagi, with the overall site policy. This integration would be required to produce an overall optimised strategy. Therefore the effluents generated from decontamination of the sarcophagi are addressed separately.

The possible outline of the treatments for the liquid effluents are given in Appendix 1 of this report. However the treatment of these effluents will be dependent upon the following factors:

- Decontamination process selected
- When the decommissioning project takes place
- Compatibility with the local site drains and treatment facilities

- Routing options
- Site liquid effluent discharge limits.

If significant re-categorisation of wastes is to take place or dose rate reduction is to be achieved then an aggressive chemical decontamination process is needed.

Chemical treatment alone may not produce an effluent which is likely to comply with the Estonian sea discharge criteria and therefore a multistage treatment is likely to be required involving :

- Sampling
- Chemical precipitation for the removal of Iron/Nickel/Chrome as the hydroxide
- Settling & Filtration
- Encapsulation of the settled filtered precipitates.
- Ion exchange treatment of the filtrate
- Encapsulation of the ion exchanger.
- Reuse of the treated liquor to make-up additional decontaminant or its discharge to sea.

In the liquid effluent from the decontamination the following corrosion products, are expected to contain the following nuclides (Ref. 6.3):

- Iron (Fe^{55})
- Cobalt (Co^{60})
- Nickel (Ni^{59} & Ni^{63}).

These nuclides can all be removed by neutralisation and hydroxide precipitation. This method routinely gives Decontamination Factors (DFs) or removal of radioactivity of up to100 times.

It is expected that the most effective treatment would be in a small local plant and disposal to sea of the treated effluent would be the most cost-effective option.

6.7.7 Secondary Wastes.

Radioactive secondary wastes will be generated from the decontamination activities; these secondary wastes are likely to be of the following types :

- Filtering systems

- Encapsulated metal hydroxide sludge
- Encapsulated Ion exchangers
- Filters from active ventilation systems

Eventually the treatment equipment will have to be disposed of or if possible reused. However it is also a potential secondary waste stream.

6.7.7.1 Amounts of Secondary Wastes

The amounts of this will be determined by the amount of material dissolved from in the treatment processes. Primarily the amount metal dissolved from the surface of the waste. Currently there is insufficient data is available to determine the surface area of the primary coolant system to be decontaminated. This information is essential to assist in providing an order of estimate for the amount secondary wastes produced. Other information would include the detailed process design associated with the decontamination system and the subsequent effluent treatment systems.

In addition to the radioactive wastes there will be the non active wastes associated with the treatment options these will also have to be accounted for in the overall disposal costs.

6.7.8 Safety & Design Issues

The basic plant elements and design issues are considered below :

6.7.8.1 Plant Containment system

The primary plant containment comprises the vessels, pipe work, pumps etc of the decontamination system. These would be constructed from materials, which are resistant to the corrosive effects of the decontamination agents.

Corrosion resistant sumps would provide the secondary containment of the major vessels of the decontamination plant. The boundary of the system would provide the secondary containment of the pipe work.

6.7.8.2 Plant Shielding

The overall construction of the decontamination system would be determined by the radiation dose associated with the material to be removed by the decontaminants. If the dose is low as indicated by the current dose information on the primary coolant systems, then bulk shielding in the form of massive concrete shielding is not likely to be needed.

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The actual amount of activity that will be removed could only be determined by a sampling campaign of the vessels to determine the amounts of deposited activity. This combined with the information about the decontamination process to be used would enable a good estimate of levels of activity likely to be released and hence the shielding design requirement of the plant.

6.7.9 Hydrogen evolution

If strong mineral acids are used in the decontamination process then Hydrogen will be generated.

The safety of the process relies entirely upon the dilution of the hydrogen to a safe concentration by a flow of air induced a vessel ventilation system. This vessel ventilation system must include the ventilation of the reactor coolant system being decontaminated. Failure of the ventilation system would have the potential to cause the hydrogen concentration to rapidly reach the LFL of 4%. The risk of an explosion if this Hydrogen concentration rises above this level.

There are two fault conditions, which could result in an increase in the hydrogen concentration these are:

- Failure of the ventilation air flow.
- Increase in hydrogen generation rate.

6.7.9.1 Failure of Ventilation

The acid decontaminates by dissolving metal from the surface of the pipe work and vessels and will therefore release hydrogen.

The rate of metal dissolution in the In-Situ decontamination plant pipe work and vessels, assuming the use of Hastelloy will be extremely low. However the decontaminant will dissolve the metal, which is being decontaminated, and hydrogen will therefore be produced in the decontamination loop.

Under normal operating conditions the concentration of hydrogen in the decontamination system ullages should be controlled to no greater than, say, 0.4%, i.e. 10% of the Lower Flammable Limit (LFL).

Detection of the failure of the ventilation would by both airflow and vessel depression instrumentation based upon diverse monitoring systems built into the design of the decontamination system.

When either the flow meter or the differential pressure gauge detects loss of ventilation, the decontaminant system is shut down. The use of an emergency air purge to the system is automatically started to maintain a hydrogen concentration of less than 4%, i.e. the LFL.

6.7.10 Increase in Hydrogen Generation

The rate of hydrogen generation could increase due to any of the following faults:

- Acid too concentrated.
- Acid too hot.
- Presence of a reactive metal.

A dangerous increase in the rate of hydrogen generation is to be detected by two diverse hydrogen monitors located in the ventilation duct on the downstream, 'clean', side of the Vessel ventilation scrubber.

If the hydrogen concentration exceeds 20% of the LFL (0.8% H_2) the monitors which are linked to a control system which would shut down the decontaminant system. This could involve:

- Turning off the heating in the treatment tank.
- Stop the acid re-circulation pump.
- Activate an increased ventilation flow.

Additional actions to reduce the hydrogen generation could be carried out:

- Forcing water into the decontaminant loop to dilute the remaining acid
- Emptying the decontaminant loop by using compressed air

As additional protection to prevent a build-up of static leading to production of a spark or corona discharge which would be extremely hazardous in the presence of Hydrogen. The decontaminant system and all elements of the treatment loops must be earthed.

6.7.10.1 Material Selection Issues

Materials that have been considered for the design duties in the construction of this type of decontamination plant are:

- High nickel alloys (C22 & C276 Nickel/Chromium/Molybdenum alloys)
- PolyVyliDene Fluoride (PVDF)
- PVDF lined stainless steel.
- PolyTetraFluoroEthylene (PTFE)
- PolyPropylene (PP)
- Methylpentene polymers (TPX)
- FKM (i.e. Viton fluoro elastomer)
- FPM (i.e. Kalrez, Chemraz etc)
- CSM (Hypalon)

6.7.10.1.1 Metallic Materials

Nickel-Chromium-Molybdenum alloys i.e. Hastelloy C22 & C276, have been identified as suitable candidate materials (Ref. 6.8) for the manufacture of vessels and pipe work. However even with these materials some corrosion is likely to occur with the use of these decontaminants, so a corrosion allowance is required in the design. Since Hastelloy is relatively expensive compared to stainless steel, the costs could be reduced by the use of polymeric lined metals i.e. PVDF lined stainless steel. The use of PVDF lined Hastelloy may also help with an increase in the operational life of the plant and possibly reduce the overall cost of the use of this material. One of the advantages of selecting metallic materials is their ability to be welded easily. However the use of Nickel-Chromium-Molybdenum alloys does introduce difficulties in welding. Recent experience in the use of these materials within BNFL has led to a large reduction in the failure rate of welds. This requires a both modified welding techniques and a comprehensive QA system to be employed.

The BNFL fabrication standard for Nickel-chromium-molybdenum alloys contains three levels of acceptance based upon the integrity requirement of the fabrication. All aspects of fabrication, welding and associated inspection with this range of alloys are contained in this standard.

6.7.10.1.2 Polymers

Polymeric materials as noted above could be used to line tanks and vessels to improve their corrosion resistance. More importantly they will be required for use in the pumps, valves and seals associated with the decontamination system. The range of viable polymer materials have been identified by references to Schweitzer reference tables, PLASCAMs database and trade literature.

The selection of suitable elastomer and seal materials is particularly difficult because of the need to meet dynamic mechanical requirements. Performance testing or operational data from a similar plant/process will be required to determine life expectancy of an item.

PTFE, PVDF and PP are commonly available pump and valve materials. PVDF and PP may be obtained in a wide range of product forms, pipe, fittings, sheet, plate and bar and are weldable.

Methylpentene polymers (TPX) are available in transparent grades but a major disadvantage is that they are susceptible to environmentally assisted stress cracking in a wide range of fluids.

FKM (i.e. Viton fluoro elastomer) is resistant to the acid fluids but may be attacked by sodium hydroxide if this is used to neutralise the decontaminants. FPM (i.e. Kalrez, Chemraz etc) is resistant to these fluids but is extremely expensive. CSM (Hypalon) and EPDM, have some resistance to all of the environments at ambient temperatures. Testing will be required and care must be taken to ensure that the appropriate grade of the elastomer selected is used.

The mechanical properties of polymers are a significant factor when considering their suitability for use. From experience gained on the decontamination pilot plant and the B229 Lab 193 projects, these materials are very sensitive to handling and mark extremely easy. Scoring of pipe surfaces is a particular problem and PVDF exhibits relatively poor impact resistance. Although impact properties of the other polymers are somewhat better, protection against the event of impact must be allowed for during handling, fabrication and in particular during plant operation.

Whilst the welding of polymers is fairly straight forward the proof of weld quality is a more difficult task. At present there are no established techniques for non-destructive (radiography, ultrasonic, etc.) inspection of polymers and the verification of acceptable welds are proven by pre-production welder test and subsequent visual and spark testing of production welds.

At present there are BNFL Company standards for the fabrication of polymeric materials and for projects which are currently using these materials the tendency has been to write specifications dedicated to the application.

Similarly polymer fabrication is not very well represented in the UK National standards and Germany is the only country that has embarked on the production of a range of standards designed for commercial use. These standards are the DVS series and they cover design, welding, testing and inspection.

6.7.10.2 Design of Decontamination System

The design of the decontamination system would be dependent on numerous factors, which can only be confirmed in more detailed design studies. Major impacts on its design have been noted in other sections of this report these would include:

- Radiological duty which would effect safety design with respect to :
  - Plant shielding requirements
  - Plant containment
  - Plant ventilation system
  - Waste type, disposal route and volume
- Hydrogen production
- Size and duty of the ventilation system
- Emergency ventilation system
- Control system
- Plant duty which would effect design with respect to:
  - Size of vessels and ‘foot print’ of plant
  - Type of material of construction for vessels, pipe work and pumps
  - Control system

Some of the key information particularly about the state of the primary coolant systems could only come from records collation and surveys. This information gathering is likely to include :

- As built drawings

- Modifications
- Plant connections
- Photographs
- Process history
- Post Operative Clean Out (POCO) records
- Sampling
- Radiation dose mapping (RADSCAN)

Current information on the internal details of the primary circuit vessels is not available but based upon external dimensions it is estimated that the liquor hold-up in the largest loop, assuming each vessel is decontaminated separately, is approximately $1m^3$. This would then set a limit on the size of the decontamination plant required. The information on the internal details of the primary circuit vessels should be obtained to confirm this. See Sections 6.7.10.2.2 & 6.7.10.2.3
For the purposes of this report it is assumed that the decontamination system will have the following components.

6.7.10.2.1 Reagent Receipt

A system for the receipt of reagents will have to be included in the design this could be based upon the use of International Bulk Containers (IBC) to deliver cubic metre quantities of reagents which would then be pumped into the decontamination re-circulation tank for further dilution before use.

6.7.10.2.2 Re-Circulation Tank

It has been estimated that the vessel and piping loops to be decontaminated have total hold-up volumes of between $7\text{-}10m^3$ and therefore the combined volume of the Re-circulation tank plus Sentencing tank should be at least $12m^3$. In a current design being developed by BNFL the Re-Circulation tank is going to be approximately $7\ m^3$.

6.7.10.2.3 Sentencing Tank

Again in the current BNFL design the Sentencing tank will be identical to the Re-circulation tank in size, duty and material.

6.7.10.2.4 Manifold

Flow and return manifolds transfer the decontaminant between the Decontamination Plant containment and the reactor chambers.

The manifolds are likely to be constructed of either 25 to 50 mm pipes and valves and must be robust and ideally should be fully welded to minimise the potential for breaking containment. The safety case may require the use of coaxial pipes. This would be required to contain leaks from the primary containment of the pipe work if the consequences of a leak were thought to be unacceptable due to release of radioactivity (dependent on activity levels) or the release of aggressive mineral acids into working environments.

6.7.10.2.5 Decontamination Plant Pipe work

It is expected that the tanks within the Decontamination Plant, along with the flow and return manifolds between the plant and the reactor chambers will be constructed of C22 Hastelloy or an equivalent high nickel alloy. The piping and valves within the plant will be constructed of the same material. C22 is available in schedule 80 pipe and fittings are commercially available.

6.7.10.2.6 Valves

Process valves are expected to be 25mm ball valves and service and ventilation valves 15 - 50mm ball valves.

Possible materials of construction are high nickel alloys, polymeric materials and or lined stainless steel. Polymeric or lined valves introduce the risk of potential leak paths at the flanges. This could be addressed by the use of 2-way and 3-way one-piece ball valves with welded connections. These are also commercially available in C22 Hastelloy.

It is expected that all valves will be installed and maintained during operations within secondary containment.

The valves controlling the flow and return to the decontamination loops will be manually operated and will be normally locked shut. The valves will only be opened after the connections have been made to the loop to be decontaminated and this will be done under the control of detailed operating procedures.

6.7.10.2.7 Pipe work within the reactor chambers

It is understood that the primary cooling circuit pipework may have been partially removed with the connections to vessels cut and plugged/capped.

If vessels have been disconnected and capped each will require a separate ALARP evaluation to determine whether the dose which will be incurred in making new connections is justified by the dose savings from the decontamination.

It is expected that the connections from the manifolds to the individual vessels and pipe work loops in the primary coolant system will be via flexible loose. These flexibles loses will be reinforced Viton, PTFE, PVC or other compatible material.

6.7.10.2.8 Pumps

Centrifugal pumps are available in C22 alloy and could be used. However there is always a danger of leakage at the rotating drive shaft and maintenance is complex and difficult to carry out in a glovebox.

BNFL has must experience in the use of double diaphragm pumps particularly in its THORP reprocessing plant. They are commercially available in C22 alloy with PTFE diaphragms and hence should be investigated for use in the design.

6.7.10.2.9 Ventilation

The Decontamination Plant will be connected to a self-contained modular ventilation and filter system comprising a fan and a two stage HEPA filter station.

A vessel vent system will be required for the ventilation of the primary coolant system so that any hydrogen can be removed by dilution with air. Since this vessel ventilation will contain hydrogen and the appropriate safety systems will be required. Entrained aerial activity would be removed first via a vessel ventilation scrubber, if the radioactive duty of the ventilation system required it and then via two stage HEPA filtration.

This scrubber will generate a source of liquid effluent.

6.7.10.2.10 Instrumentation

It is proposed that the decontamination plant would be designed for manual operation with control via detailed operating procedures. However key systems such hydrogen detection and plant shutdown etc. would have to be incorporated in a computer controlled system.

Automatic temperature, level and differential pressure controls will be provided where appropriate. All pumps, heaters, etc. will normally be operated manually with safety controls and over-rides fitted where appropriate.

The controls/instrumentation will be hard-wired to a control panel located external to the Decontamination Plant containment.

Stand-by equipment and back-up systems will only be installed if considered necessary from a safety standpoint. In the event of a breakdown the plant will be shutdown until the necessary repairs are carried out.

6.7.11 Costs Associated with a Decontamination Plant for the Paldiski Sarcophagi.

Detailed design and costing work is being carried out in BNFL on the use of In-Situ decontamination systems in preparing plants, which are to under go decommissioning. In addition an inactive plant has been built to prove the process of tank decontamination in relation to PCM and ILW wastes. This provides a reasonable basis for the approximation of the associated Design and engineering and the capital costs of the equipment. Additional work would be required to define the operation and decommissioning costs of this type of plant.

6.7.11.1 In-Situ Decontamination

If an 'In-Situ' decontamination plant were built on the Paldiski site the lifetime costs of this facility would need to be assessed including:

- Design
- Construction
- Commissioning
- Operation
- Decommissioning

The lifetime costs would have to take into account the down stream treatment of liquid effluents and the generation of secondary wastes.

The volumes of active waste for decontamination by the use of In-Situ decontamination within the Paldiski sarcophagi are relatively small and it is possible that it is not cost effective to build new plants unless:

- The system is designed based around mobile rigs with the aim of minimising capital, operational and decommissioning costs.
- The use of a system of mobile rigs would enable their use to decontaminate other wastes elsewhere on the Paldiski site, or elsewhere within Estonia.

The estimated costs of the design and construction of an In-Situ decontamination plant based upon the current BNFL designs. The first is one that will serve as the decontamination facilities for a major active cell which has handled Medium Active (MA) or possible Highly Active (HA) process liquors

The cost of this type of plant is of the order of EUR 13 to EUR 16 Million.

It has been estimated in a recent exercise that approximately EUR 0.8M to EUR 1.2M would be required for detailed up front design investigations. This is work that is prior to the detailed design being undertaken.

Significantly the cost range of this decontamination plant has in part been effected by the need to:

- Modify what was a major active but now redundant plant
- Provide significant shielding since the decontamination liquors once they have been through the process should contain a relatively large amount of entrained activity.

This cost does not include the cost of :

- Upgrading the existing plant to allow for the additional ventilation systems to maintain plant safety.

Another design study for In-Situ decontamination has recently been carried out. This design would be for the decontamination of part of a much newer plant, which has evaporated MA liquors. This part of the plant has vessels and pipe work which is several times larger in volume than in the first plant described above. However it is likely it would not have to undertake the major cost of upgrading its ventilation system.

The cost of the In-situ decontamination plant for this type of installation has been given a base cost order of cost estimate of EUR 35 Million.

6.7.11.2 ILW Tank Decontamination

The facility to carry out extensive trials to look at the use of tank decontamination for PCM and ILW treatment to reduce them to LLW classification has been built at Sellafield. This facility has been built of materials, which will also allow their testing to support their use in other decontamination facilities. The cost of this facility and its operation will be approximately EUR 3 – EUR 5 Million.

6.7.11.3 Effluent Treatment

It is important to note that the estimated cost of these plants does not include the necessary effluent treatment facilities, which on the Sellafield site are provided by site effluent treatment plants. It is therefore not appropriate to include the large capital costs of these facilities since they serve the entire Sellafield site for the treatment of certain effluent streams and are not just dedicated to In-Situ decontamination.

Until the duty of the Decontamination Plant can be established the cost of the effluent treatment can not be established.

Cutting and Size Reduction

To make the approach to the project ALARP 'In-Situ' decontamination is used to help reduce dose rates coming from the primary coolant system. However it is likely that the majority of the dose will be coming from activation of structures associated with the reactors. So to reduce the overall time spent in the working area where the background radiation levels are high, the items should be removed in large pieces to be further size reduced in a dedicated waste treatment facility.

6.7.11.4 Waste Treatment Facility

In this facility the following activities would be carried out.

- Cutting up of large items to allow:

- ➢ Access to enclosed surfaces for subsequent monitoring to allow accurate waste inventory and consignment
- ➢ Segregation of different materials or components of differing activity levels and waste category
- ➢ To provide piece sizes suitable for waste packaging

- Additional treatment could also be carried out in this dedicated facility to reduce dose and or re-categorise waste. This could involve:

- ➢ Use of immersion decontamination process
- ➢ Use of melting equipment (see Section 6.7.2 and Appendix 7.2)

The waste treatment facility would have to provide the following:

- Lay down and sorting areas

- Heavy item handling equipment
- Ventilation system with filtration
- Mobil local shielding for ‘hot spots’
- Cutting equipment including both hot and cold cutting
- Wash down and effluent control and collection system
- Services
- Waste packaging and loading facilities

The actual removal strategy of the equipment from the reactor including the primary coolant system will have to be addressed in future design studies. However it would be sensible to site the waste treatment facility adjacent to the reactor compartment and within the coverage of an overhead crane to avoid double handling of the waste pieces. This would allow the waste to be transported from the reactor compartments in large pieces.

It is assumed that the more radioactive components such as the reactor vessels and other items, which are highly activated, will be disposed of as one piece items and that, therefore, the waste treatment facility would not require remote equipment. However, semi-remote equipment could be deployed for the more active items, which require size reduction. Such equipment could be set up manually and allowed to operate automatically thereafter. This would help reduce doses to the labour force.

Some of the Plant items from Decommissioning the reactors will only have localised contamination over a relatively small area, the rest of the items being relatively clean. This would be shown after removal from the reactor compartment by detailed monitoring. Where this is the case, consideration will be given to removing or cutting out the more contaminated part for treatment / disposal thus releasing the rest of the item for release as a lower waste category or, possibly, free release.

Before designing the waste treatment facility, the type and characteristics of the waste would have to be modelled this would include its:

- Sizes
- Weights
- Thickness
- Materials of construction
- Likely dose and contamination levels

It is recognised that some items may still have water within them, in dead spaces, and facilities to capture and dispose of this as noted above should be incorporated into the facility.

6.7.11.5 2 Size Reduction Techniques

There are various size reduction techniques, which could be employed, some of which can be employed in place whilst others are more suited to a dedicated cutting facility. The technique must be selected according to the waste piece concerned.

The level of radiation dose and associated contamination will effect the safety of removal and the requirements for Personal Protective Equipment (PPE). Some of the plant items will be 'clean' and can be size reduced with basic clothing augmented by standard protective gear such as goggles and dust mask.

Where airborne contamination is a risk, a higher degree of PPE may be required such as a PVC suit with respirator to reduce internal dose or even a pressurised suit. Where this level of PPE has to be used there will also have to be segregation of the cutting activities from the main waste treatment facility and also dedicated ventilation to this area.

The manpower support for such protective clothing teams is necessarily higher and more costly, and the efficiency of the operator is much reduced due to the encumbrance of the clothing.

6.7.11.6 Cold Cutting Techniques

These are mechanical techniques such as:

- Power nibblers
- Shears
- Reciprocating saws

These usually don't cause an unacceptable increase in airborne contamination levels however given that they are relatively slow they have an additional cost associated with them in terms of man-hours. There is also the issue of dose uptake for the operators since they have to be in the working environment for a longer period.

6.7.11.7 Hot Cutting Techniques

The cutting techniques that come into this category are

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Mechanical (Abrasive disc cutters)

Gas (Oxyacetylene and Oxybutane/propane)

Plasma arc

These cutting techniques can exacerbate the spread of contaminants by causing vaporisation and additional dedicated ventilation will be required. In particular plasma arc cutting tends to create a lot of fume which can blind conventional HEPA filters, therefore additions to the ventilation systems, such as cyclone separators, electrostatic precipitators, may be required. An assessment must be made as to whether the costs of the additional ventilation equipment is justified by the benefits that such techniques provide in terms of speed of cut. This speed will also have an effect on the dose uptake based upon operator proximity to the work face.

It is possible to use gas or plasma arc cutters on parts of components, which are known to be free from contamination. This would help speed up decommissioning operations.

Where cutting of a particular item is likely to cause contamination spread use of a modular containment system; local tenting and the use of strippable coatings should be considered.

Overall detailed ALARP dose assessments are required to develop which technique, cold or hot cutting is the most effective in terms of dose uptake.

6.7.12 Melting

Following decontamination, the waste from the reactors may still be above the limits for free release. A further technique, melting, could be considered which could reduce the amount of waste required to be sent for disposal.

Melting of decommissioning metal wastes can give the following benefits:

- As a method of decontamination, even for activated components, as the radionuclides are preferentially trapped in the slag.
- For volume reduction of the wastes for disposal or storage.
- Fixing of the activity, the surface activity is distributed into the mass and fixed thus reducing the risk of re-suspension if the material is free released.
- Melting provides an easier method of sampling for free release than sampling the original component, which may have complex geometry.
- Sale of recycled metals can earn useful revenue to offset some of the costs.

There are two melting techniques which are currently being used within BNFL for the recycling of radioactive materials. These are the:

- Reverberatory Furnace
- Induction Furnace

6.7.12.1 Melting technology at BNFL's Capenhurst plant

The Melting Facility at Capenhurst in England essentially consists of three furnaces, two of which are electric induction furnaces and one oil fired Reverberatory furnace. The facility also has a :

- Comprehensive ventilation system
- Computer monitoring system for the operation of the furnaces, ventilation system and airborne contamination levels
- Short term buffer storage of feedstock,
- Temporary storage of the melted ingots
- Temporary dross and slag storage to allow monitoring and sampling prior to its disposal

The decommissioning feedstock is stored on the site, within adjacent buildings prior to it being transported to the Melting Facility.

The facility has operated successfully using the above basic layout since November 1994 melting >2500 tonne of Aluminium and 270 tonne of steel and cast iron. During this time significant operational experience has been gained and the lessons learned used to improve the plant or to revise procedures.

Reverberatory melting has continued without major incident and has recently been refurbished to more efficiently accommodate larger items of feedstock.

6.7.12.1.1 Process Ventilation System

The process ventilation system serves to remove active and inactive particulate together with potentially harmful combustion gases and associated heat from the furnaces, taking them away from the operating environment.

With the reverberatory furnace which is a suitable melting technique for materials with admixed plastic or other non metallic materials, the combustion products are passed through an afterburner, where necessary, to satisfy requirements of the Environmental Protection Act (EPA) 1990 and the Capenhurst site gaseous discharge authorisation.

The extracted particulate and acidic gases are removed from the air by means of cyclones, bag filters, HEPA filters and a water scrubber system. These in combination ensure that environmental discharges are minimised.

6.7.12.1.2 Dross and Slag Removal and Handling.

Activity concentration is anticipated in the dross (from aluminium) or the slag (from steel). Dross and slag are removed from the induction furnaces by manual skimming into ventilated containers.

The slag or dross containers are to be changed after each melt batch. The content of these is then sampled and, subjected to radiochemical analysis. The material is bulked as necessary in 210 litre drums for storage prior to disposal.

6.7.12.1.3 Furnace Pouring.

When sufficient material has been melted and the slag or dross removed, the molten metal is poured from the furnace into suitable moulds. The moulds are positioned beside the furnace by the use of a hydraulically operated trolleys on steel tracks. Moulds are preheated and dried to prevent ejection of the melt due to wet moulds and other causes. Additional precautions are taken to prevent local melting of the mould when steel is poured.

6.7.12.1.4 Ingot Handling, Monitoring and Storage

The mould contents are allowed to solidify before being moved, and then moved to the ingot handling area of the facility. Ingots are tipped out of the moulds by dedicated turnover equipment, the weight of these ingots can vary between 0.5 Tonne and 1.6 Tonnes, depending on size and feed material density.

Ingots are transferred to the covered buffer store for cooling, prior to sampling, obtaining swarf by drilling and then using radiochemical techniques on the sample quantities, backed up by low level alpha and beta counting of additional samples taken during pouring and by gamma spectroscopy. Ingots are coded and masked with the melt number for traceability and control.

The metal ingots, after radiological examination and appropriate clearance, are transported to other suitable interim areas prior to their free release or disposal to waste disposal sites if appropriate.

6.7.12.2 Other European Melting Facilities

Other experience of melting techniques is available in Europe. This is summarised below:

6.7.12.2.1 Germany

Siempelkamp CARLA plant, a specially designed melting facility for treatment of radioactive contaminated and or activated steel and non-ferrous metals such as zinc-plated materials, copper, aluminium and brass. This plant incorporates an induction furnace modified to include ventilation systems and manipulator systems to allow remote control of the process.

The Eiram facility, located at the Nuclear Research centre Karlsruhe, has an induction furnace with a ceramic inner container. The furnace was used to melt steel scrap from the decommissioning of the Niederaichbach Nuclear Power Plant (KKN).

6.7.12.2.2 Sweden

The Studsvik plant in Sweden recycles contaminated metallic scrap from nuclear facilities. It consists of an induction furnace with an associated hall for waste handling. This melting plant has been operating since 1987.

The approach used at this facility is to achieve free release of metals from components at a suitably low level of contamination. Some items with a higher degree of contamination are decontaminated prior to melting. If the resulting ingots are above free release levels they are stored until the activity has decayed to suitable levels. Secondary waste in the form of slag is returned to the owners for disposal.

6.7.12.3 Quantities and Types of Metals at Paldiski

It is assumed that the most radioactive components such as the reactor vessels following decommissioning will be disposed of as one-piece items directly to the Estonian repository.

It is also assumed that the bulk of the structural steel, which makes up the reactor compartments, could be decontaminated to free release. The items left, which might require melting, are therefore the plant items contained within the compartments such as pressurisers, steam generators, etc.

Materials range from carbon steels, stainless steels, titanium alloys, cupronickel and heat resistant steels.

The total weight is difficult to establish but in terms of order of magnitude, amounts to a few hundreds of tons prior to decontamination, which might suggest 50-100 tonnes requiring melting.

6.7.12.4 Economics of Melting

Whether or not melting should be pursued for the Paldiski wastes depends largely on the costs. The economics of melting vs. disposal depend on many factors. The costs of each alternative need to be understood before a decision can be made.

To carry out an economic evaluation for Paldiski on the lines above would require much more information than is currently available. The economics of melting requires an assessment based around the following areas:

- Definition of the Estonian disposal limits
- Volumes of waste arising in each disposal category following decontamination
- Overall waste disposal costs including
  - Packaging
  - Storage
  - Transport
  - Repository costs
- Potential to ship and treat at another facility
  - Waste packaging requirements (International standards)
  - Waste packaging costs
  - Waste transport costs
  - Waste treatment costs
- Lifetime costs of a new melting facility
  - Design
  - Construction
  - Commissioning
  - Operation
  - Decommissioning
- Revenue income from sale of free release of materials

A complicating factor in this assessment is that there is no repository in Estonia, and so these costs will be difficult to establish. Much will depend on the free release criteria adopted for Estonia, and on the success of any decontamination process chosen to be used before the melting option.

However, evidence from studies in other countries does give some encouragement that the economics could be favourable. There are two examples, which are cited from the following paper:

Economic Aspects of Melting and/or Recycling of Waste Metals from Decommissioning, Technical Seminar on Melting and Recycling of Metallic Waste Materials from Decommissioning of Nuclear Installations, Krefeld, Germany, Oct. 1993. The examples are found in Appendix 7.2 of this report.

6.8 ADVANTAGES AND DRAWBACKS OF THIS DECOMMISSIONING STRATEGY

Advantages	Drawbacks
– Waste packages are consistent to the IAEA recommendations regarding waste management. – Waste package (parallelepipeds) will be totally immobilised into the final disposal. – Waste packages transport to the disposal site is fully consistent with Estonian regulations and IAEA recommendations regarding radioactive waste transport, <u>except for RC#1 reactor vessel</u> (but as explained in §6.6.1, RC#1 reactor vessel transport could be reasonably performed, providing eventually some complementary safety measures).	– This decommissioning strategy requires heavy works to dismantle the Reactor compartments. These works imply a very high labour, and a quite high total men exposure. – As a consequence, the global cost would be higher than for first strategy. – The risk of radioactivity release into the environment is reasonably low, but higher than for the first strategy.

6.9 SECOND STRATEGY OPTIMIZATION TAKING INTO ACCOUNT A STORAGE PERIOD OF 50 OR 100 YEARS BEFORE STARTING OPERATIONS.

As said in § 4.3.3, the influence of the storage period is significant regarding men exposure considering this first dismantling strategy, as hand-on dose rates close to reactor compartment components is quite high, especially regarding the reactor vessel. However, after a 50 years storage period, the decrease of gamma dose rate will be much slower, as spectrum repartition shows that Co-60 is the most significant gamma high-energy radionuclide (half-life is 5.3 years). In the year 2050, Co-60 activity will have decreased one thousand times, and the total men exposure for the dismantling operations will be less than ten times lower. Between 2050 and 2100, the decrease of gamma dose rate will be ten times lower than between reactors shutdown and 2050.

Regarding the transport RC#1 vessel to the disposal site, we saw in § 4.3.2.1 that if the disposal site is not located on the Paldiski site, this operation can't be carried out in strict accordance with the Estonian regulations and IAEA recommendations before the year 2165. The total enclosed activity will remain after 50 years more than two times higher than the transport limit, and more than 1.5 times higher than the transport limit after 100 years. So from this angle there's no benefit in prolonging the storage period to 100 years.

And as explained in §4.2, we think that it's possible to guarantee the reliability of the confinement barriers for 50 years if the works that are described in § 6.2 are carried out as soon as possible. But if it comes to extend the storage period up to 100 years or more, the safety provisions described in § 6.2 won't be sufficient : the works to be carried out would be similar to those described as First Strategy – Option 1 in § 5.2 (Final disposal of the RC in their sarcophagi). These works are much heavier and much more expensive.

Moreover, the influence of the storage period might not be really significant either regarding waste volumes, as all activated steel equipment contain long life radionuclides (Ni59 and Ni63) that won't allow free release of the waste, even if the total activity has strongly decreased. So the total volume of waste assuming that dismantling works are carried out in 2100 should be only 10% less than if we assume that dismantling works are carried out in 2050.

For these reasons, we think that waiting 100 years before implementing this second decommissioning strategy brings no significant advantage, but implies additional cost to guarantee the reliability of the confinement barriers for the whole storage period.

As a consequence, we recommend that this second decommissioning strategy would be implemented after a storage period of 50 years.

7 DECOMMISSIONING OPTIONS ADVANTAGES AND DRAWBACKS SUMMARY

The main advantages and drawbacks of each decommissioning strategy is summarised in the table below :

Decommissioning Strategy	Advantages	Drawbacks
First Decommissioning Strategy Disposal of the RCs as a whole	– Except radioactive sources extraction, the works to be carried out would not imply high men exposure. – The risk of radioactivity release into the environment is reasonably low.	– Waste packages (RCs) are not consistent to the IAEA recommendations regarding waste management – Waste packages are not totally immobilized into the final disposal.
Decommissioning Option 1 In situ disposal in the sarcophagi	– This option does not require heavy dismantling works. – As a consequence, the global cost would be quite low. – No radioactive waste transport is required.	
Decommissioning Option 2 On-site near surface disposal at Paldiski	Reactor compartments transfer to the disposal building could be considered as a radioactive waste transport, but this radioactive waste (the reactor compartments) remains on the Paldiski site.	– This decommissioning option requires heavy works to transfer the RCs into the disposal building. – As a consequence, the global cost would be much higher than for option #1. – The Reactor Compartments transfer is not fully consistent to the radioactive waste transport regulations due to their total enclosed activity, but as the waste packages are transported on tens of meters with no exit of the Paldiski site, this option seems to be reasonable.
Second Decommissioning Strategy Full Dismantling of the RCs	– Waste packages are consistent to the IAEA recommendations regarding waste management. – Waste package (parallelepipeds) will be totally immobilised into the final disposal. – Waste packages transport to the disposal site is fully consistent with Estonian regulations and IAEA recommendations regarding radioactive waste transport, <u>except for RC#1 reactor vessel</u> (but as explained in §6.6.1, RC#1 reactor vessel transport could be reasonably performed, providing eventually some complementary safety measures)	– This decommissioning strategy requires heavy works to dismantle the Reactor compartments. These works imply a very high labour, and a quite high total men exposure. – As a consequence, the global cost would be higher than for first strategy. – The risk of radioactivity release into the environment is reasonably low, but higher than for the first strategy.
Packaging Option 1 Disposal of big components as specific waste packages	– Total men exposure is lower than for packaging option #2	– The resulting waste volume is higher than for packaging option #2
Packaging Option 2 Minimising the waste volume by decontamination, recycling, etc ..	– The resulting waste volume is lower than for packaging option #1	– Total men exposure is higher than for packaging option #1

8 APPENDIX 1 : ESTONIAN LEGISLATIVE ACTS IN RADIOACTIVE WASTE MANAGEMENT

9 APPENDIX 2 : FIRST DECOMMISSIONING STRATEGY – OPTION #1 - SCHEMATIC FLOWCHARTS

10 APPENDIX 3 : FIRST DECOMMISSIONING STRATEGY – OPTION #1 - SCHEMATIC FLOWCHARTS

11 APPENDIX 4 : SECOND DECOMMISSIONING STRATEGY - SCHEMATIC FLOWCHARTS

12 APPENDIX 5 : SECOND CONFINEMENT BARRIER REINFORCING FOR A STORAGE PERIOD LESS THAN 50 YEARS

13 APPENDIX 6 : DISMANTLING SEQUENCE FOR RC#2 ENCLOSED EQUIPMENT

14 APPENDIX 7 :

14.1 APPENDIX 7.1 : DETAILS OF POTENTIAL DECONTAMINATION PROCESSES

14.2 APPENDIX 7.2 : COST COMPARISON ON MELTING