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NUCLEAR ENERGY AGENCY
RADIOACTIVE WASTE MANAGEMENT COMMITTEE

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Working Party on Decommissioning and Dismantling (WPDD)

RELEASE OF SITES OF NUCLEAR INSTALLATIONS

Evaluation of a Questionnaire issued by the WPDD of the OECD/NEA and Other Background Information Final Report

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Release of Sites of Nuclear Installations
Evaluation of a Questionnaire issued by the WPDD of the OECD/NEA and Other Background Information
Final Report

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compiled for the Working Party on Decommissioning and Dismantling (WPDD)
of the OECD Nuclear Energy Agency (OECD/NEA)
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Aachen/Germany, 20 September 2004

FOREWORD

Decommissioning of nuclear facilities in OECD Member Countries is an activity already being carried out and is expected to reach a peak level after .2015. This increasingly important activity is currently being addressed by the Working Party on Decommissioning and Dismantling (WPDD) set up by the NEA Radioactive Waste Management Committee (RWMC). The WPDD has produced a document entitled "The Decommissioning and Dismantling (D&D) of Nuclear Facilities in OECD/NEA Member States - Status, approaches, issues" in which it recorded general agreement amongst various stakeholders on a number of key points. In particular it noted that:

- Techniques for D&D are already available, and valuable experience is being fed back to plant design and decommissioning plans.
- Many nuclear facilities have already been successfully decommissioned and dismantled.
- There is no unique or preferred approach to D&D of nuclear facilities.
- Current systems for protection of the safety of workers, the public and the environment are satisfactory for implementation and regulation of D&D.

In connection with the first point, the WPDD has held topical sessions covering information and experience on materials management and buildings and sites release and re-use. This background report to a Status Report which is intended to be issued in 2006, contains detailed information on the release of sites of nuclear installations from radiological control. The report is based on a questionnaire on site release distributed to relevant decommissioning projects.

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1. INTRODUCTION

This report contains recent information on the release of sites of nuclear installations from radiological control. The approach comprises the following elements:

- the procedures which were used for clearance/release of sites at decommissioning projects in various OECD Member States, based on a questionnaire on site release distributed to relevant decommissioning projects;

as well as on a compilation of information concerning:

- a compilation of information concerning the modelling for the derivation of suitable release limits;
- a compilation of information concerning measurement techniques for checking compliance with these release limits; and
- an attempt to draw conclusions from these information with respect to a more stringent and efficient implementation of site release in the future.

2. STRUCTURE

The overall structure of this report is as follows:

- The main body of the text presents all central data and conclusions in a condensed form;
- Annex I contains the details of the measurement methods and the radiological analyses;
- Annex II is a compilation of the answers from the decommissioning projects of the various OECD Member States.

The main body of the text addresses the following topics:

- Chapter 3: Overview of the questionnaire and other sources of information;
- Chapter 4: Release criteria and dose criteria;
- Chapter 5: Radiological models and the derivation of site release criteria;
- Chapter 6: Unconditional release vs. restrictions on land use after release;
- Chapter 7: Measurement techniques for site release.

3. OVERVIEW OF THE QUESTIONNAIRE AND OTHER SOURCES OF INFORMATION

The basis for the information on site release is the evaluation of a questionnaire on site release at decommissioning projects which was developed specifically for this purposes. This questionnaire was aimed at gaining a general overview of the situation instead of gathering a multitude of project specific information. It was hoped that reducing the number and complexity of questions would result in a higher readiness for filling out the questionnaire and responding. In fact, data from 11 countries with reference to 16 decommissioning projects as well as to generic regulations was available which allows quite a broad overview. A complete evaluation of the questionnaire can be found in Section 2 of Annex I. The compilation of the questionnaire and all answers which were received is available in Annex II.

In order to cover as broad a range of data as possible and readily available, an evaluation of proceedings of recent conferences where the issue of site release was touched was performed. The results supplement the answers to the questionnaire and are provided in Section 3 of Annex I.

4. RELEASE CRITERIA AND DOSE CRITERIA

Various approaches for the choice of the dose criterion for the release/clearance of sites have been described in the answers to the questionnaire as well as in papers presented at recent conferences. A comprehensive overview of this matter has been given in the summary report "Criteria for the Removal of Controls" presented by J.R. Cooper at the Conference *Berlin 2002* which draws the following conclusions:

- Some countries have used dose values up to 250 µSv/a, others prefer 100 µSv/a, a few even go to 10 µSv/a for sites. It must be taken into account that prescribing very low clearance levels for sites may result in unduly high efforts for site remediation, which would be in contradiction to the principle of ALARA (as low as reasonably achievable). From the USA there was an interesting approach for the clearance of sites that may show a solution for many challenges which clearance still provides: the use of a graded approach which uses different standards for different situations or types of facilities. This is also corroborated by the approaches taken, for example, in Spain and Germany, where different clearance options have been derived and/or implemented (general/unconditional clearance, clearance for specific purposes), all accompanied with their own sets of clearance levels.
- Obviously, the international community has accepted the fact that clearance is a necessary option and that it should be fostered internationally. It seems that the dose criterion of 10 µSv/a, or a range of that order, is now also agreed upon as far as materials like scrap, rubble, etc., are concerned. A general approach towards the clearance of sites is, however, only now emerging. It may require a flexible approach; there is no need to deprive oneself of the necessary flexibility by setting a too restrictive dose constraint for sites.
- Release of land versus release of materials: There is no unanimous opinion on whether the same criterion should be used for the clearance of land as for clearance of materials (10 µSv/a) or whether more flexibility should be allowed, leaving countries more freedom. After all, material can be traded across borders, land cannot. Compliance with 10 µSv/a in all cases might be a waste of effort; there are many types of installations which certainly could meet e.g. 250 µSv/a quite easily, while cleanup to a standard of 10 µSv/a would create excessive additional effort which would not be ALARA. Additionally, there are voices suggesting that internationally accepted models are needed for calculating doses from the residual contamination of sites. Furthermore, it should be kept in mind that it is hard to communicate to the public that different criteria are used in various states for land clearance.
- The general conclusion is therefore that countries should have flexibility to address the issue of land versus material as they see fit. It would also be useful to have internationally agreed models for assessing exposures from contaminated land.

These findings are in full agreement with the answers received to the questionnaire.

The two approaches for the dose criterion, i.e. the trivial dose range (~ 10 µSv/a) or a larger fraction of the individual dose limit (~ 100 to 250 µSv/a) can both be justified for site releases. While the first approach results in true clearance of the site, the second one takes into account the special nature of land (no movement possible) and the fact that anybody living on that land would be exposed to the natural activity in the soil anyway.

Therefore, it might currently be prudent to allow countries this flexible approach until more experience will be gained with site releases. This also applies to differences in the numerical values which are used as measurement limits. A flexible approach offers the possibilities for application of the ALARA principle and making best use of available resources.

The use of risk based criteria in relation to site release/clearance is only just emerging. Aspects of this approach have been outlined e.g. in section 3.4 of Annex I. It seems that there is not yet enough experience with this approach which might, however, be a reasonable basis for international harmonization and consensus.

5. RADIOLOGICAL MODELS AND THE DERIVATION OF SITE RELEASE CRITERIA

5.1 Pathways and Scenarios

A small overview of the radiological models which have been used for the derivation of site release/clearance criteria has been presented in section 4.3 of Annex I. Although only a few different approaches are currently available, it can be concluded that there is quite good agreement between the various models with respect to the pathways and scenarios. It is obvious that modellers use very similar pathways and scenarios although the parameter use may be different.

A reasonable minimum set of pathways may the following:

- External gamma exposure
- Inhalation of dust
- Ingestion of plant foods
- Ingestion of meat
- Ingestion of milk
- Ingestion of fish
- Ingestion of soil
- Ingestion of water

Radon inhalation may also be relevant in cases where a the residual contamination has a significant amount of U and Th and where the dose criteria above the trivial dose range of 10 μSv/a, i.e. 100 – 300 μSv/a are used. In all other circumstances the contribution Rn inhalation to the overall dose is negligible.

A reasonable minimum set of scenarios should be linked to the future use of the site. If it is by any means possible that a residential farmer and his family may live on the site later on, appropriate scenarios should be used, i.e. in the food and drinking water pathways mentioned above a reasonable amount of local produce should be included. If it is only conceivable that a normal residential area may be developed on the site with one-family houses and gardens, the amount of food and drinking water obtained from the site is considerably smaller or even zero.

5.2 International Harmonization on the Model Approaches

Some presentations e.g. at the conference *Berlin 2002* called for international harmonization between model approaches used for the derivation of release/clearance levels for sites (cf. e.g.

section 0 of Annex I and section 0 above). However, the view was also expressed that international harmonization might not be of great importance for sites because in this case international trade is not involved – as opposed to clearance of metal scrap or building rubble.

Yet, it might be beneficial for the acceptance of site release/clearance to have at least some guidance from international bodies how appropriate modelling should be carried out. Similar experience has been gained with international recommendations on clearance of metal scrap, buildings, building rubble etc. which have been issued by the European Commission for the EU Member States. Although these recommendations have been transposed into national legislation only in a small number of Member States, these recommendations serve as benchmarks in other countries for comparison with the national approaches. Therefore, international guidance on radiological modelling setting minimum standards might be a step worthwhile considering.

It should also be noted that the RESRAD code with its excellent implementation, description and quality assurance has become a quasi-standard over the past years for site release. This can be gathered from the fact that it has been applied not only in the US but also in Spain and that many operators of decommissioning projects are checking their approaches against RESRAD results.

6. UNCONDITIONAL RELEASE VS. RESTRICTIONS ON LAND USE AFTER RELEASE

The release of land (of a site) may be unconditionally or may be based on certain conditions. Especially the way in which compliance with such conditions may be ensured is a topic of ongoing discussions (as was e.g. addressed by some speakers at the Conference Berlin 2002). One of the main reasons for imposing restrictions after release is to reduce the number of possible exposure pathways (at least during the first decades during which part or all of the remaining activity will have decayed) in order to optimize the release/clearance criteria.

The question remains, however, under which part of the regulatory framework such a restriction could be achieved. There are two basic possibilities:

1. The restriction of land use after release of the site is controlled by the nuclear authority, i.e. the same body which was responsible for the nuclear installation. This, however, means that the site will remain under some kind of nuclear licensing regime which might be unfavourable for the future site development in the conventional (industrial) sector.
2. The restriction of land use is recorded in the land title register. It would then be the responsibility of the authorities competent in the building law together with the land registry office to ensure that the restrictions are complied with. This is a common procedure which is applied for any real estate where rights or titles are to be preserved etc. It would also not involve the nuclear authorities which only would have to ensure that the appropriate restriction is entered in the land title register.

The question of whether a site may be released with restrictions on its later use has been extensively discussed also in Germany when developing the Radiation Protection Ordinance (Strahlenschutzverordnung) of 2001. The general procedure was regarded as viable, but the competent authorities of the Federal States (Länder) expressed their view that a real estate which was to be released with any restrictions in place deriving from its former nuclear use was unacceptable to the public. So there might not be a general answer to the question whether land could be released with restrictions in place. It certainly should be determined on a site-specific level.

7. MEASUREMENT TECHNIQUES FOR SITE RELEASE

A considerable number of different measurement techniques are available for site release measurements (cf. the discussion in section 0 of Annex I). The detection limits of many techniques are low enough to allow for rapid and significant measurements. There is therefore no need for totally new techniques although of course the development continues.

However, some improvements should still be considered. As an example, automation of the application of measurement techniques would greatly enhance the speed of the measurements. As already used for clearance measurements in buildings, in situ gamma spectrometers could be mounted on self-propelling devices with GPS positioning which carry out and log measurements at least on smooth surfaces (like concrete surfaces, roads, ramps etc.). This would also reduce costs for the personnel.

There is general consensus that measurements for site release/clearance need not cover the entire surface. A reasonable measurement density needs to be derived from statistical considerations. Furthermore, the measurement density will also depend on the category of the surface, i.e. the contamination class. Such a categorization of sites according to their probability for contamination has been used in many decommissioning projects (cf. e.g. section 4.2 of Annex I).

Because of the complexity of the measurement process and the statistical evaluation, it might also be a reasonable approach to develop international standards for the application of these techniques. International standards for similar measurement techniques like e.g. the standard ISO 7503 on the evaluation of surface contamination have fostered harmonization in the application of measurements and have influenced national standardization projects.

Annex I

EVALUATION OF THE QUESTIONNAIRE AND BACKGROUND INFORMATION

1 OVERVIEW

This Annex evaluates and summarizes the full information from the questionnaire on release of sites which is presented in Annex II and also gives detailed information from other sources of information, e.g. recent publications. It gives the full technical details which are presented in condensed form in the main part of this document. Any references in this Annex refer to sections of this Annex, unless stated differently.

1.1 General Approach

This work is a summary of information on the release of sites of nuclear installations from radiological control. It incorporates various approaches:

- an evaluation of the answers to an extensive questionnaire which was circulated among decommissioning projects of nuclear power plants in OECD countries concerning the release of sites;
- an evaluation of recent publications on the release/clearance of sites of nuclear installations;
- an evaluation of radiological models on which criteria for release/clearance of sites are based;
- an overview of measurement techniques which are currently used for site release or which are being developed.

The reason for the development and distribution of a questionnaire on site release was to:

- gain more information on this comparatively young type of release from radiological control,
- get an overview of the dose criteria and the release criteria that are used in various countries,
- learn which measurement techniques for verification of compliance with the release criteria are preferred.

As only a few decommissioning projects of nuclear installations have advanced to the release of sites, there is less experience with site release than e.g. with clearance of metal scrap or concrete rubble from nuclear installations. Furthermore, there is a fundamental difference between clearance of materials and release of sites: While it is generally agreed that materials should be released (cleared) on the basis of a trivial dose criterion (doses of the order of 10 µSv per year), sites where people will be living or working might be released on the basis of considerably higher dose criteria – e.g. a fraction of the natural background dose. Therefore, the questionnaire focused both on technical as well as on radiological aspects of site release. The answers provided allow a meaningful and comprehensive evaluation of the release practices in various countries.

It is also very helpful and revealing for a complete understanding of site release to examine the radiological bases from which the release criteria are derived and how the assumptions differ between countries and even projects. This gives an indication how site release is viewed in various

countries: more as a special form of clearance implying the trivial dose regime or more as a kind of authorized release. An overview of the various approaches is therefore also provided in this report.

1.2 Structure of this Annex

This annex is structured as follows:

- Section 2 presents the questionnaire and gives an overview and evaluation of the answers that were submitted by various countries or decommissioning projects. It also tries to draw conclusions which directly relate to the questionnaire.
- Section 3 evaluates recent publications on site release which complement the information given in Section 0.
- Section 4 highlights features of radiological models which have been used for deriving release/clearance levels either on a generic level or for site-specific use. It also presents a discussion of the dose criterion on which release/clearance levels for sites are based.
- Section 5 gives an overview of the currently available measurement techniques for site release. As far as possible, the detection limits for in-field use are also presented.

2. THE QUESTIONNAIRE AND ITS EVALUATION

2.1 Intention of the Questionnaire

The intention of circulating a questionnaire on site release among various decommissioning projects in OECD countries was to gain more insight in the approaches, the release criteria, the measurement techniques, and other aspects of the release of nuclear sites and to draw comparisons between countries. The number of questions and therefore the scope of the questionnaire needed to be limited to stimulate answers to the questionnaire and to attain a short response time.

This section is structured as follows:

- Section 2.2 presents the original questionnaire and describes the reason why the particular questions were chosen.
- Section 2.3 gives an overview of the countries and the decommissioning projects which sent in answers to the questionnaire.
- Section 2.4 summarizes the features of the sites for which site release/clearance has been carried out.
- Section 2.5 examines the release criteria that were or would be used in site releases as well as the dose criteria on which the release criteria are based.
- Section 2.6 provides an overview of some of the radiological models which have been used for the derivation of release or clearance levels for sites.
- Section 2.7 gives an overview of the measurement techniques which have been used or are intended for the release of sites. It also provides a discussion of the various measurement techniques which are available for this task.
- Section 2.8 provides some conclusions on how the release of sites is carried out in various countries.

2.2 The Questions in the Questionnaire

The questionnaire focuses on radiological and technical aspects of the release of nuclear sites. The questionnaire encompasses all types of nuclear facilities, i.e. nuclear power plants, research reactors and fuel cycle installations. However, most of answers relate to nuclear power plants as this type of facility constitutes the majority of decommissioning projects.

The questions have been grouped into four groups as follows:

2.2.1 Description of the nuclear facility

Question 1: (Please describe the nuclear installation shortly – name etc. If more than one installation is described, please complete one form for each installation. Identify whether this decommissioning project is part of a larger decommissioning programme for which uniform standards are / will be applied.)

Question 2: What is the size of the site which was or will be released? Is this the entire site, or just part of it?

Questions 1 and 2 aim at gaining some information about the site or sites which is/are referenced in the following questions.

2.2.2 Site Contamination

Question 3: Is the radionuclide vector/radiological fingerprint representative for the site known? Is there a (standard) method to derive the radionuclide vector/radiological fingerprint from the measurements?

(Please give the list of relevant radionuclides such as Co 60, Cs 137, Sr 90 ...)

Question 4: In the case of a contamination of the site, is it known how deep the radionuclides may have penetrated into the ground?

Questions 3 and 4 relate to the radiological situation at the site(s): The knowledge of the radionuclides which are present in the contamination is essential for the radiological evaluation, for the choice of adequate measurement techniques and for the release criteria. The penetration depth is also relevant for the choice of measurement techniques and gives an indication of the chemical behaviour (K_d values) of various radionuclides which is important for modelling.

2.2.3 Regulatory Standards:

Question 5: On which basis has been or will the site be released?

(Please address the following points:

- describe the legal basis, for example, does the regulatory framework provide generic or site specific regulations,
- indicate the date when this regulation entered into force,
- indicate if radiological studies have been carried out which have served as a basis for deriving release levels.)

Question 6: Please give examples of release levels/clearance levels of the relevant nuclides which will be used / have been used for the release of (a) site(s).

Question 7: Are there special features of the site which have influenced the derivation of suitable release criteria or the release procedure itself?

(Such special features could be vicinity to dwellings, a nature reserve, a water body from which drinking water is taken – or the opposite: site surrounded by uninhabited land, no farming etc.)

Question 5 relates to the regulatory framework and the dose criterion which are the basis for the release of the site(s). This question also asks for details on the regulations, mainly if the release criteria have been derived on the basis of dedicated studies or if general release criteria have been applied. Question 6 asks for release (clearance) levels which have been used or which generally are/will be used in that country. Question 7 aims at details of the site which would be relevant for site-specific calculations of release criteria and which may have been taken into account. These information are relevant to assess the extent to which site specific release criteria differ from general criteria.

2.2.4 Implementation

Question 8: Which measurement techniques will be / have been applied for release / clearance measurements?

(Please indicate the relevant techniques, such as in situ gamma spectroscopy, soil excavation and bulk measurement etc.)

Question 9: What averaging criteria will be / have been used for the release of (a) site(s)?

(Please indicate averaging masses and/or surfaces, e.g. 100 m^2)

Additional references to open literature or the enclosure of internal information would be highly appreciated.

Questions 8 and 9 relate to the measurement techniques and the way in which measurements have been performed. Question 9 explicitly asks for the averaging criteria which is relevant for the implementation of site release measurements.

2.3 Participating Countries and Projects

Answers to the questionnaire have been received from the countries and projects within these countries listed in Table 2.1. The listing in this and the following tables is alphabetically by country.

Because there are no projects from Hungary and the Netherlands, those two countries are not included in the following tables evaluating the Questionnaire.

Table 2.1: **Countries and projects from which answers to the questionnaire were received**

Country	Projects	Remarks
Canada	Answer relates to four types of installations that would have sites suitable for release: ¢ U mines/mills, ¢ small research reactors, ¢ large power reactors, ¢ nuclear research establishments.	Answers are provided for the four types of installations separately where necessary. The answers are provided generically, i.e. not with respect to individual decommissioning projects.
Germany	KGR – NPP Greifswald	Answers are provided for each installation separately.
	HDR – NPP at Kahl (Heißdampfreaktor)	
	Fuel Element Facilities at Hanau	
Hungary	none	no decommissioning projects with any experience on site release in Hungary
Japan	JPDR	Answer relate to the JPDR only
The Netherlands	none	no decommissioning projects with any experience on site release in the Netherlands
Norway	Institute for Energy Technology (IFE), Kjeller with: ¢ research reactor JEEP1 ¢ research reactor JEEP2 ¢ zero-power reactor NORA	The answer does not relate to a nuclear area but to an area located in the Nielva River near the IFE. This area had been contaminated by the operation of the reactors on the IFE site.
Slovak Republic	NPPs at the Jaslovske Bohunice site	The answers relate to the Slovakian NPPs at both sites of which Bohunice 1 is in the decommissioning phase.
	NPPs at the Mochovce site	
Spain	Vandellós 1 NPP	The answer relates to the NPP Vandellós 1 which is under decommissioning.
Switzerland	KKB – NPP Beznau	The answers relate to the four NPP sites of Switzerland and are provided individually for each plant. Additional information is provided by answers to an OECD/NEA questionnaire on decommissioning strategies and costs
	KKG – NPP Gösgen	
	KKL – NPP Leibstadt	
	KKM – NPP Mühleberg	
United Kingdom	Winfrith	Answers for both sites are provided in the questionnaire. Information is split where necessary. Release of sites will take place at both sites.
	Harwell	
USA	none	generic answers on site release were received

2.4 Characteristics of the Sites of Participating Projects – Questions 1 and 2

The relevant data of the sites of the participating projects (plant type, size, other characteristics) are given in Table 2.2. It should be noted that the sizes of the sites vary considerably between around 1000 m² for research reactors and very small NPPs, a few 10,000 m² for fuel cycle facilities and several km² for NPP and U mine/mill sites. The size of the site has a large influence on the release measurements and the radiological modelling.

Table 2.2: **Characteristics of the sites of participating projects**

Country	Projects	Data
Canada	U mines/mills	generic: 3 – 5 km²
	large power reactors	generic: 1 – 10 km²
	small research reactors	generic: 400 – 1000 m²
	nuclear research establishments	generic: 40 km²
Germany	KGR – NPP Greifswald	entire area: 2.8 km², cleared area in first phase: 0.7 km²
	HDR – NPP at Kahl (Heißdampfreaktor)	total area around 6,000 m²
	Fuel Element Facilities at Hanau	NUKEM-A: 1.15 ha Siemens: 3.2 ha
Japan	JPDR	1 pilot NPP:
Norway	IFE, Kjeller	contaminated area in a river near a research centre contaminated area of sediments: 200 m²
Slovak Republic	NPPs at the Jaslovske Bohunice site	site with 3 NPP blocks: 720,000 m²
	NPPs at the Mochovce site	site with 2 NPP blocks in operation and 2 NPP blocks under construction: 1,292,237 m²
Spain	Vandellós 1 NPP	entire site: 153,000 m² released area in phase 1: 107,000 m² area remaining under control during NPP dormancy: 46,000 m² (for ~ 30 a)
Switzerland	KKB – NPP Beznau	2 units, each 380 MWe - size of site not given
	KKG – NPP Gösgen	1 unit, 1020 MWe - size of site not given
	KKL – NPP Leibstadt	1 unit, 1145 MWe - size of site not given
	KKM – NPP Mühleberg	1 unit, 355 MWe - size of site not given
United Kingdom	Winfrith	in phase 1: 4 ha, total site 40 ha
	Harwell	pilot area: 7 ha, total site 25 ha

2.5 Release Criteria and Dose Criteria

2.5.1 Overview of Nuclide Vector and Site Release Criteria – Questions 3, 4, 6

Table 2.3 gives an overview of the nuclides that were relevant for the various projects and the corresponding release or clearance criteria. If a penetration depth was included in the answers, it is also given in this table.

The data given Table 2.3 show that not many countries have established release or clearance levels for sites. If, however, levels exist, they are mainly expressed in a nuclide specific way. Only the UK uses generic levels in terms of total mass specific and total surface specific activity.

Table 2.3: **Nuclide vectors (NV) and site release/clearance criteria in terms of Bq/g or Bq/cm²**

Country	Projects	Data
Canada	U mines/mills	NVs are generally known – U with decay products Ra 226 may have been removed by leaching contamination profile known or will be established no generic release / clearance criteria
	large power reactors	fission products, few areas with activation near core, tritium penetration depth may be relevant no generic release / clearance criteria
	small research reactors	fission products, few areas with activation near core, tritium exact inventory has to be measured for actual decommissioning projects very small penetration depth no generic release / clearance criteria
	nuclear research establishments	inventory may be quite complex generally: fission products, tritium additional contamination from past practices possible penetration depth will be measured no generic release / clearance criteria
Germany	KGR – NPP Greifswald	Co 60 and Cs 137 only relevant nuclides average penetration depth of 5 cm used for radiological model clearance levels: Co 60: 0.2 Bq/cm² corresponds to 0.03 Bq/g Cs 137: 0.5 Bq/cm² corresponds to 0.1 Bq/g
	HDR – NPP at Kahl (Heißdampfreaktor)	Co 60 and Cs 137 only relevant nuclides average penetration depth of 10 cm used for radiological model clearance levels: sum of Co 60 + Cs 137 contamination: 0.5 Bq/cm² corresponds to 0.03 Bq/g
	Fuel Element Facilities at Hanau	Nuclides of U and Th decay chains plus fission products; mainly U 238, U 234, Th 234, Th 228, Pa 234m, Ra 228, Ac 228 penetration depth very variable, up to a few meters clearance levels (only for part of activity originating from plant operation): U 234: 10 mBq/g, Th 232: 2.4 mBq/g
Japan	JPDR	Co 60 was dominant nuclide, followed by Cs 137. Other nuclides not considered. Penetration depth not measured. No site release levels are established in Japan. Release criterion: no artificial radioactivity remaining, i.e. activity at background levels
Norway	IFE, Kjeller	Pu 239, Pu 240 main radionuclides in sediments, also Am 241 penetration depth: main contamination 50 – 60 cm below surface Sediment was removed if summed concentration of Pu 239, Pu 240 and Am 241 exceeded 10 Bq/g

Country	Projects	Data
Slovak Republic	NPPs at the Jaslovske Bohunice site NPPs at the Mochovce site	Co 60, Cs 137, alpha emitting nuclides around NPP A1 measurements for establishing penetration depth starting Clearance levels for all types of clears material with volume contamination (in Bq/g): 0.3 for Co 60, Cs 134, Cs 137, Pb 210, U 238, Am 241 and many others 3 for Co 58, Fe 59, Po 210 and others 30 for Cr 51, Co 57 I 129, Pu 241 and others 300 for C 14, Cl 36 and others 3000 for H 3, S 35, Ni 63 and others
Spain	Vandellós 1 NPP	relevant nuclides: H-3, C-14, Mn-54, Ni-59, Ni-63, Co-60, Sr-90, Nb-94, Sb-125, Cs-134, Cs-137, Eu-152, Eu-154, Pu-239, Am-241 no penetration detected, contamination purely surficial Clearance levels: H-3: 1,25E+02, C-14: 3,19E-01, Ni-59: 2,21E+02, Ni-63: 1,00E+02, Co-60: 4,95E-01, Sr/Y-90: 1,52E-01, Nb-94: 9,23E-02, Sb-125: 4,63E+00, Cs-134: 9,38E-01, Cs-137: 3,27E-01, Eu-152: 4,57E-01, Eu-154: 1,01E+00, Pu-239: 8,43E-01, Am-241: 8,22E-01
Switzerland	KKB – NPP Beznau KKG – NPP Gösgen KKL – NPP Leibstadt KKM – NPP Mühleberg	Relevant nuclides: H-3, C-14, Cl-36, Mn-54, Fe-55, Co-60, Ni-59, Ni-63, Sr-90, Nb-94, Tc-99, Ag-108m, Ag-110m, Sn-126, Sb-125, I-129, Cs-134, Cs-137, Ce-144/Pr-144m, U-235, U-238, Pu-238, Pu-239/240, Pu-241, Am-241, Cm-244 Measurements only performed for Co 60 and Cs 137 penetration depths not yet measured Total list of release / clearance levels are in the full answers in Annex A – levels for relevant nuclides: Co 60: 1 Bq/g, 3 Bq/cm² Cs 137: 0.7 Bq/g, 3 Bq/cm² (unclear whether those values apply to release of sites)
United Kingdom	Winfrith Harwell	activity ratio beta-gamma to alpha emitters: 99:1 main nuclides: Co 60, Cs 137, Am 241, Pu, U, H 3 contamination within the near-surface layer except for leaking drains clearance level: mass specific activity not more than 0.4 Bq/g above natural background level surface specific activity not more than 4 Bq/cm² (η/ν) and 0.4 Bq/cm² (ζ)
USA	sites under licence of US NRC or US EPA	activity values determined from dose or risk criteria described in **Table 2.4** on a site specific basis

2.5.2 Overview of Dose Criteria – Question 5

Table 2.4 provides an overview of the answers on the dose criterion that has been used in deriving release/clearance levels for the various projects. Answers on this point have usually been given under Question 5 of the Questionnaire.

From the data in Table 2.4 it becomes clear that not all the release/clearance levels for sites have been based on a dose criterion but some have been taken directly from the existing regulation with no underlying models linking doses and activities. In other cases, clearance of sites have been regulated on the basis of 10 µSv/a (this relates only to the excess activity, not the natural background).

The USA uses risk based criteria (risk range of 10^{-4} to 10^{-6} total risk) for sites under sole USEPA authority and a dose criteria of 250 µSv/a for sites of USNRC licensees. Currently, a Memorandum of Understanding applies to the cleanup for those sites having overlapping jurisdiction under USNRC and USEPA.

Table 2.4: **Basic dose criteria used for the derivation of site release criteria**

Country	Projects	Data
Canada	U mines/mills	generally 50 µSv/a 10 µSv/a in cases where licensee chooses to derive own site specific clearance levels
	large power reactors	
	small research reactors	
	nuclear research establishments	
Germany	KGR – NPP Greifswald	10 µSv/a
	HDR – NPP at Kahl (Heißdampfreaktor)	10 µSv/a
	Fuel Element Facilities at Hanau	10 µSv/a
Japan	JPDR	not stated
Norway	IFE, Kjeller	not stated
Slovak Republic	NPPs at the Jaslovske Bohunice site	10 µSv/a individual dose, 1 manSv/a collective dose
	NPPs at the Mochovce site	
Spain	Vandellós 1 NPP	100 µSv/a proposed by ENRESA for Vandellós 1
Switzerland	KKB – NPP Beznau	not stated clearance levels are taken from Swiss Radiation Protection Act
	KKG – NPP Gösgen	
	KKL – NPP Leibstadt	
	KKM – NPP Mühleberg	
United Kingdom	Winfrith	not stated, clearance levels (**Table 2.3**) are not derived on the basis of radiological assessments
	Harwell	

USA	US EPA sites	nuclear facilities can be subject to clean-up under the Comprehensive Environmental Response, Compensation, and Liabilities Act utilizes a risk range of 10^{-4} to 10^{-6} to determine appropriate levels of cleanups for radioactive and hazardous materials at those sites under sole USEPA authority
	US NRC sites	upon decommissioning and license termination for USNRC licensees: fraction of the 1 mSv/yr public dose limit USNRC regulations in 10 CFR Part 20, Subpart E specify dose constraint for decommissioning and license termination of 0.25 mSv/yr and demonstration that the residual contamination levels are ALARA

2.5.3 Overview of the Regulatory Framework – Question 5

Table 2.5provides an overview of the regulatory framework for release/clearance of sites. This regulatory framework has been described mainly under question 5 or in a preamble.

The data given in Table 2.5 indicate that only in a few case there are parts of the regulatory framework dedicated to release/clearance of sites. In many cases, individual radiological assessments were performed for site release. Some countries indicate that suitable parts of the framework would be needed or would be beneficial. At present, site release is performed in most countries on a case-by-case basis.

Table 2.5: **The regulatory framework on which release / clearance of sites is based**

Country	Projects	Data
Canada	U mines/mills	Nuclear Safety and Control Act (NSC Act) and associated regulations Each nuclear facility licensee must apply for a licence to decommission a nuclear facility and the licensing process includes an environmental assessment carried out in accordance with the *Canadian Environmental Assessment Act (CEAA)* and its regulations. Following completion of decommissioning activities, a licensee may apply to the Commission for a Licence to Abandon the nuclear facility. The issuance of a Licence to Abandon demonstrates that the decommissioning has been successfully completed to the appropriate end-stage and that the facility is no longer under regulatory control by the CNSC.
	Large power reactors	
	small research reactors	
	nuclear research establishments	
Germany	KGR – NPP Greifswald	Individual radiological assessment in all cases, on behalf of the competent authority Meanwhile, generic clearance levels for sites exist in the German Radiation Protection Ordinance (Strahlenschutzverordnung)
	HDR – NPP at Kahl (Heißdampfreaktor)	
	Fuel Element Facilities at Hanau	

Japan	JPDR	The basic procedure for cancellation of radiation control areas was established in co-operation with the regulatory body
Norway	IFE, Kjeller	The river is no and has never been a restricted area Legal basis for site clearance: Act of 18 June 1938 on use of X-rays radium etc (now substituted by a new radiation protection act from 12 May 2000), Regulation from 23. January 1976 Individual radiological assessment performed
Slovak Republic	NPPs at the Jaslovske Bohunice site NPPs at the Mochovce site	Derived limits were issued in 2001 in accordance with IAEA /NEA guidance Act 470/2000 Coll. And Regulation 12/2001Coll. (valid since December 13, 2000) In 2001, the legal basis for remediation of contaminated sites was established first time. Any remediation activity needs a permission of the Regulatory Authority. The operator has to provide the evidence, that the way of remediation is the optimal from the point of the view of radiation protection.
Spain	Vandellós 1 NPP	CSN (Nuclear Safety Council) has not provided specific regulations with regard to the release criteria Relevant regulation: Act 15/1980 on 22 of April about the creation of the Nuclear Safety Council Decree 1836/1999 on 3 of December on Nuclear and Radioactive Facilities Regulations
Switzerland	KKB – NPP Beznau KKG – NPP Gösgen KKL – NPP Leibstadt KKM – NPP Mühleberg	Licence required according to Swiss Atomic Law Following the end of operation, the owner of the plant must prepare and submit for approval a detailed decommissioning project. This project must define the phases of decommissioning and the corresponding timetable. Radiological studies have been and will be carried out to establish the radiological fingerprint of the facility (see Question 3) but not for deriving release levels. These levels are regulated in the Swiss radiation protection act.
United Kingdom	Winfrith Harwell	The NIA(65) Act includes the conditions for the Variation or Cessation of a Licence (referred to as delicensing). The requirements for Variation or Cessation are the same, and are both described within the act as: *A variation to a nuclear site license may be granted by excluding any part of the licensed site which the licensee no longer needs for any use requiring a nuclear license And where no danger from ionising radiation exists on that part of the site.*

USA	US EPA sites	Nuclear facilities can be subject to clean-up under the Comprehensive Environmental Response, Compensation, and Liabilities Act.
	US NRC sites	USNRC regulations in 10 CFR Part 20, Subpart E specify a dose constraint for decommissioning and license termination
	US DOE sites	The U.S. Department of Energy has radiation protection regulations that are similar to those of the USNRC and USEPA, such as USDOE Order 5400.5, "Radiation Protection of the Public and the Environment." USDOE regulations are applicable to facilities and sites which they own.

2.6 Radiological Models, Derivation of Site Release Criteria

2.6.1 Introduction

The overview of some radiological models which have been used for the derivation of site release/clearance criteria. In the answers to the questionnaire, those models have been referenced only in a few cases which are described in section 2.6.2.

In order to present the general scheme of radiological modelling for site release, models from the USA and Germany have been provided. The US RESRAD model which is described in section 4.3.2 is a fully self-contained computer code for calculation of doses via all relevant environmental pathways for site specific calculations. The German model in section 4.3.3 has been designed for the derivation of generic clearance levels for sites for the German Radiation Protection Ordinance.

Section 2.6.3 tries to give a synthesis of both models with respect to relevant radiological pathways which should be taken into account for the derivation of release/clearance criteria for sites.

2.6.2 Radiological Models for Release of Sites used in the Participating Projects

Table 2.6 gives an overview of the radiological models used for the derivation of site release/clearance criteria. It must however be emphasized that information was available only in few cases because only few countries have release/clearance levels which were based on radiological assessments and even fewer countries provided details on that topic.

The general evaluation of the scenarios as pathways given in Table 2.6 is done in section 2.6.3 taking into account also other radiological models.

Table 2.6: **Radiological models used for the derivation of site release/clearance criteria**

Country	Projects	Data
Canada	U mines/mills	no information given
	large power reactors	
	small research reactors	
	nuclear research establishments	
Germany	KGR – NPP Greifswald	Large number of scenarios, i.a. construction workers on the site (construction of new buildings after clearance), persons in dwellings on the site (one-family houses with garden for food production), industrial use of the site, use of the site for recreational purposes, others. All relevant exposure pathways (external irradiation, inhalation, direct and secondary ingestion etc.) Details are also given in section 0.
	HDR – NPP at Kahl (Heißdampfreaktor)	Few scenarios for industrial uses: industrial use by a larger company, use as storage area (e.g. for construction material), use as public car park (which is not covered)
	Fuel Element Facilities at Hanau	Large number o Large number of scenarios, i.a. excavation of the soil and disposal on a landfill (workers, general public), leaving the soil on the site (general public: people living on the site, groundwater pathway, secondary ingestion)
Japan	JPDR	no information given
Norway	IFE, Kjeller	brief radiological assessment, no detailed information given
Slovak Republic	NPPs at the Jaslovske Bohunice site	no radiological analysis for existing release / clearance levels
	NPPs at the Mochovce site	
Spain	Vandellós 1 NPP	Exposure pathway modelling is used to calculate the radionuclide-specific concentration of specific nuclides that could result in a dose or specific risk equal to the release criterion. RESRAD and D&D codes have been used.
Switzerland	KKB – NPP Beznau	no information given
	KKG – NPP Gösgen	
	KKL – NPP Leibstadt	
	KKM – NPP Mühleberg	
United Kingdom	Winfrith	no information given
	Harwell	
USA		no specific information given exposure pathway modelling contained e.g. in RESRAD code, cf. section 0 of this report

2.6.3 Conclusions on the Relevant Exposure Pathways for Radiological Modelling

Information given in the questionnaires on details of the radiological modelling (section 2.6.2) was not very extensive[1]. The German models are described in reports while the Spanish models use the RESRAD models. Both approaches use similar pathways, and it is a common approach to consider a person or a family who live on the site and who get at least part of their diet from it. This may be a residential farmer where the amount of vegetable, meat, milk and drinking water originating from the site may be quite high. Such an assumption may be appropriate for a country like the USA where a site may be a large distance away from any other dwelling and therefore self-sufficiency of farmers is a possible situation. In countries with a much higher population density, like Germany, this would not be reasonable scenario. This is the reason why a family has been considered who only get part of their diet from their own garden on the cleared site.

Agreement is there also with respect to the exposure pathways. A general set like:

- external irradiation from the contaminated soil layer,
- dust inhalation,
- direct ingestion of contaminated soil,
- ingestion of vegetables grown in contaminated soil,
- use of water via groundwater pathway and via surface water pathway, for irrigation, watering of cattle and drinking water.

Other pathways like Radon inhalation may also be considered if appropriate. The ground water pathways generally needs considerable modelling effort in order not to be overly conservative.

2.7 Measurement Techniques for Site Release

2.7.1 Overview

Section 2.7.2 provides a compilation of those techniques which have been or will be actually used in the projects which have participated in the Questionnaire. The important subject of averaging criteria for these measurements is dealt with in section 2.7.3. A comparison of the techniques used in the participating projects with the generic recommendations of the USA (MARSSIM) and Germany (DIN) is drawn in section 2.7.4.

Measurement techniques in general for site release are described in section 5.

2.7.2 Techniques used in Site Releases of the Participating Projects – Question 8

Table 2.7 provides an overview of the measurement techniques (or measurement strategies) that were used or would be used at the various projects. It can be seen that the measurement techniques which have been or will be used are quite similar. The *in situ* gamma spectrometry is mainly used for measurements of surficial contamination in the field while sampling and lab analysis (mainly gamma spectrometry) are used to deal with deeper contamination or to establish the penetration depth. The MARSSIM procedure has been adopted by Canada (for research reactors) and Spain, cf. section 2.7.4.

1. Because of the lack of a larger number of answers, details of generic radiological models used for site release are presented in section 4.3.

Table 2.7: **Techniques used for verification of compliance with site release/clearance criteria**

Country	Projects	Data
Canada	U mines/mills	tailings areas are surveyed for gamma radiation as part of the decommissioning licence requirements gamma survey at one meter above the ground surface grid spacing is variable from 1 to 10 meters on centre, spacing depends on the site conditions
	large power reactors	CNSC has accepted plan, proposed by the licensee, for multiple testing protocols to confirm the absence of radioactive materials on the site, based on full site aerial gamma survey and ground-based gamma survey results used to develop biased / unbiased sampling campaigns for soil analysis biased samples taken at locations identified from the surveys, unbiased samples taken at random locations soil sampling campaign included in-situ gamma monitoring and radionuclide specific measurements
	small research reactors	MARSSIM approach adopted
	nuclear research establishments	no specific information given
Germany	KGR – NPP Greifswald	Division of site into grid areas collimated and uncollimated in-situ gamma spectroscopy, samples with laboratory gamma spectroscopy in compliance with DIN 25457 part 7
	HDR – NPP at Kahl (Heißdampfreaktor)	
	Fuel Element Facilities at Hanau	Division of site into grid areas of 1 x 1 m² Excavation of soil, depths of a few 10 cm Measurement of excavated soil by special facilities (belt conveyor before monitors) in compliance with DIN 25457 part 7
Japan	JPDR	Measurement techniques used in the building: gas flow type detector, sampling and ν-spectroscopy by Ge detector, NaI scintillation counter
Norway	IFE, Kjeller	gamma and alpha spectrometry for alpha spectrometry: radiochemical separation of Pu and Am
Slovak Republic	NPPs at the Jaslovske Bohunice site	various measurement methods, methodology under development
	NPPs at the Mochovce site	
Spain	Vandellós 1 NPP	Scanning survey of surface using portable radiation detection instrument assembled to a mobile device, sodium iodine survey meter 2"x 2" for ν, gas-flow proportional counter for ζ/η. Direct measurements of alpha, beta and photon surface activity Sampling and laboratory analysis: gross ζ/η measurements, ζ/ν spectrometry, low-energy η emitting radionuclides Procedure according to MARSSIM

Country	Projects	Data
Switzerland	KKB – NPP Beznau	not yet defined
	KKG – NPP Gösgen	
	KKL – NPP Leibstadt	
	KKM – NPP Mühleberg	
United Kingdom	Winfrith	in-situ gamma spectroscopy using high purity germanium detectors (HPGe) portable hand held gamma spectrometers (NaI detectors), for specific point surveys, supported by high sensitivity low resolution gamma surveying using NaI detectors supported with data loggers and global position (GPS) to provide high quality large surface area surveying. samples and lab analysis at places where contamination is present at depths
	Harwell	
USA		no specific information given

2.7.3 Averaging Criteria for Measurements

Averaging criteria specific to the release of nuclear sites have been reported only by Germany and Spain (Table 2.8). For sites with η/ν contamination, those criteria are in the range of 100 m² up to 10,000 m² and depend on the contamination level. For ζ contamination, smaller areas (range of 1 m²) are used.

Table 2.8: **Averaging criteria used in the measurements for site release**

Country	Projects	Data
Canada	U mines/mills	no fixed value, depending on variation of the activity level and site specific situations
	large power reactors	
	small research reactors	
	nuclear research establishments	
Germany	KGR – NPP Greifswald	site has been divided into zones according to contamination level averaging area: 100 m² and 1000 m²
	HDR – NPP at Kahl (Heißdampfreaktor)	no information given
	Fuel Element Facilities at Hanau	1 m²
Japan	JPDR	no information given
Norway	IFE, Kjeller	no averaging criteria, each sample had to be below release criterion
Slovak Republic	NPPs at the Jaslovske Bohunice site	no information given
	NPPs at the Mochovce site	
Spain	Vandellós 1 NPP	According to MARSSIM methodology (NUREG-1575) areas of the site are classified by contamination potential Class 2 land areas: 2000 – 10000 m², Class 3 land areas: no limit

Country	Projects	Data
Switzerland	KKB – NPP Beznau	not yet defined
	KKG – NPP Gösgen	
	KKL – NPP Leibstadt	
	KKM – NPP Mühleberg	
United Kingdom	Winfrith	averaging criteria: those defined for the release criteria
	Harwell	
USA		no specific information given

2.7.4 Comparison with MARSSIM and DIN Recommendations

There are two comprehensive sets of recommendations how to proceed with release of sites on a technical level. The MARSSIM document (USA) is a very comprehensive guide covering all aspects of performing an investigation of a contaminated site up to its release (section 2.7.4.1), while the DIN Standard (Germany) provides detailed guidance on application of measurement techniques for clearance of sites (section 2.7.4.2).

2.7.4.1 The MARSSIM document - USA

The MARSSIM document (Multi-Agency Radiation Surveys and Site Investigation Manual, NUREG 1575) provides information on planning, conducting, evaluating, and documenting building surface and surface soil final status radiological surveys for demonstrating compliance with dose or risk-based regulations or standards. The MARSSIM is a multi-agency consensus document that was developed collaboratively by four Federal agencies having authority and control over radioactive materials: U.S. Department of Defence (DOD), U.S. Department of Energy (DOE), U.S. Environmental Protection Agency (EPA), and U.S. Nuclear Regulatory Commission (NRC). The MARSSIM 's objective is to describe a consistent approach for planning, performing, and assessing building surface and surface soil final status surveys to meet established dose or risk-based release criteria, while at the same time encouraging an effective use of resources.

MARSSIM provides a statistical method for obtaining survey data and comparing them, on a pass-fail basis, to a release criterion or DCGL (Derived Concentration Guideline Level). It provides a process for collecting, organizing, and interpreting data and for making decisions about populations of data from samples. Statistics are useful for inferring population characteristics from a set of samples. Statistics also facilitate decision-making in conditions of uncertainty.

The MARSSIM procedure comprises three interrelated parts to demonstrate compliance with respect to conducting surveys:

1. Translate: Translating the cleanup/release criterion (e.g. mSv/y, mrem/y, specific risk) into a corresponding derived contaminant concentration level (e.g. Bq/kg or pCi/g in soil) through the use of environmental pathway modelling.
2. Measure: Acquiring scientifically sound and defensible site-specific data on the levels and distribution of residual contamination, as well as levels and distribution of radionuclides present as background, by employing suitable field and/or laboratory measurement techniques.

3. Decide: Determining that the data obtained from sampling does support the assertion that the site meets the release criterion, within an acceptable degree of uncertainty, through application of a statistically based decision rule.

The main topics addressed in MARSSIM are:

- Overview of the Radiation Survey and Site Investigation Process.
- Historical Site Assessment.
- Preliminary Survey Considerations.
- Survey Planning and Design.
- Field Measurement Methods and Instrumentation.
- Sampling and Preparation for Laboratory Measurements.
- Interpretation of Survey Results.
- Quality Assurance and Quality Control.

Additional information is provided in a number of appendices in the MARSSIM document which illustrate the application.

It is beyond the scope of this report to provide more information on the MARSSIM Guide.

2.7.4.2 The DIN Standard on Clearance of Sites – Germany

The German Institute for Standardization (DIN) has developed a series of Standards on the clearance of materials, buildings and sites from nuclear installations (DIN 25457: Activity measurement methods for the release of radioactive waste materials and nuclear facility components – part 7: Nuclear sites – DIN Deutsches Institut für Normung, Berlin).

DIN 25457 part 7 describes that the clearance process first needs a thorough investigation of the radionuclides which are involved, their spatial and depth distribution, the natural background (if relevant), and the choice of appropriate key nuclides which serve as the basis for measurements and from which the activities of other nuclides can be derived by multiplication with correlation factors.

Suitable measurement techniques for determining whether the contamination on a site is below clearance levels are the following:

- *in situ* gamma spectrometry, with collimator;
- *in situ* total gamma measurement;
- surface measurement with large-area proportional counters for ζ/η activity (only on sealed surfaces);
- sampling and lab analysis of samples;
- removal of material of defined area, decision measurement of the removed material (if material exceeds clearance levels, the material is removed and the process is repeated, otherwise the material is put back into place and the area is considered free).

DIN 25457 Part 7 also describes stochastical measurement and sampling strategies in detail.

2.7.4.3 Application of the Recommendations in Actual Projects

As shown in Table 2.7, some decommissioning projects make use of the available guidelines provided. Apart from the USA where it was developed and is widely applied, MARSSIM is also used in Canada and Spain for site releases. Germany, for example, has developed its own guidelines in DIN 25457.

The conclusion is that each country with ongoing decommissioning projects where the release of sites is about to be performed will need suitable guidelines for actually performing the measurements according to internationally agreed standards. Such guidelines are available, e.g. MARSSIM or DIN 24547 part 7.

2.8 Conclusions from the Questionnaire

The Questionnaire on the release of sites has yielded results with respect to the following topics:

- description of the nuclear facilities,
- site contamination (nuclides, activities),
- regulatory standards and radiological assessments,
- implementation (measurement techniques, averaging criteria).

Answers have been received from Canada, Germany, Hungary, Japan, the Netherlands, Norway, the Slovak Republic, Spain, Switzerland and the United Kingdom (alphabetical order). Those answers ranged from statements that no data were available for that particular country because no nuclear facility was under decommissioning yet over generic descriptions how release of sites in that country would be carried out to detailed project specific descriptions on actual site releases. It has therefore become clear from the Questionnaire that the overall experience with release/clearance of sites is not as comprehensive as with, e.g. clearance of materials.

The following is a compilation of results which have been derived from the Questionnaire and have been described in more detail in the previous sections of this report:

- Not all countries have dedicated regulations in place and have actually applied these regulations in decommissioning projects. It appears that imminence of site releases at several nuclear installations will promote development of such regulations.
- The dose criterion (individual dose) on which sites are released varies. Within the countries having participated in this Questionnaire, it ranges from 10 µSv/a, i.e. trivial doses which are also used for regulations on clearance of materials, to 100 µSv/a (Spain) and 250 µSv/a (USA, USNRC sites) – all values refer to doses above background, taking into account only that part of the activity originating from the facility. A risk based approach is used by the USEPA (sum of risk from radioactive and hazardous materials).
- Radionuclide vectors, i.e. the list of radionuclides together with their activity fractions, have been established for sites which were to be released. The most relevant nuclides for reactor sites are Co 60 and Cs 137. In the case of contamination containing U and Th, it is necessary to find a way to discriminated these nuclides from their natural background activities, or to find a suitable release level for the sum of activities originating from the facility operation and from natural background.
- A few countries have provided details of radiological assessments which link the basic release criterion in terms of dose to the mass or surface specific activity in the soil. These models

usually comprise all relevant exposure pathways and take into account a reasonable number of scenarios. In the USA, the RESRAD code for site release incorporates a large number of predefined scenarios which may be adapted to the specific site.

- There seems to be a tendency to take over existing guidance which is internationally available (like the MARSSIM Guide from the USA) which has proven applicable and comprehensive. This might help to avoid costly development of national guidelines on the release of sites.
- None of the countries participating in the Questionnaire has stressed the point that there should be international harmonization between the release/clearance procedures or the release/clearance levels used for sites. While this is a valid point which has been raised often in international discussion with respect to clearance of materials and transboundary movement, release/clearance of sites is a purely national problem. The radiological models to be used for dose assessments need to be adapted to the particular features of the specific country (climate, dietary habits, site specific features etc.).
- The measurement techniques which are or will be used for site release are quite similar between countries. They range from taking samples and performing laboratory analyses (gamma spectrometry or beta/alpha spectrometry after radiochemical separation, depending on the relevant radionuclides) to *in situ* measurements, e.g. with in-situ gamma spectrometers (with or without collimators). The fact that there is great similarity between the choice of techniques underlines that the equipment for release measurements is now commercially available. There does not seem to be much need for development of new measurement techniques, while there still may be room for improvement of existing techniques or equipment.
- For these measurements, some countries have installed averaging areas which are reasonably large (100 m² to 10,000 m²) and are thus adapted to the issue of site release.

The general conclusion can be drawn that release of nuclear sites is today a well established process in a number of countries. It has, however, also become apparent that countries where there will be no releases of sites (or only on a very small scale) do not foster any regulations or guidance on site releases prematurely.

3. EVALUATION OF RECENT PUBLICATIONS ON THE RELEASE/CLEARANCE OF NUCLEAR SITES

3.1 Overview

There have been an increasing number of papers on the release or clearance of sites (or parts thereof) of nuclear installations, presented at international conferences. Those reports either concentrate on describing the approach which was used in specific projects, or give a more general description of the issue in general.

Conferences of 2002 and 2003 where this subject has been treated are for example:

- The International Conference “Safe Decommissioning for Nuclear Activities” of the IAEA, Berlin, 14-18 October 2002. In the following, this conference is referred to as *Berlin 2002*.
- The Symposium “Freigabe von radioaktiven Stoffen aus dem Geltungsbereich des Atomgesetzes” (Clearance of radioactive material from the scope of the German Atomic Energy Act), Hamburg, 10-11 March 2003. In the following, this conference is referred to as *Hamburg 2003*.

- The International Symposium "Conditioning of Radioactive Operational & Decommissioning Wastes (KONTEC)", 19-21 March 2003, Berlin. In the following, this conference is referred to as *KONTEC 2003.*

The following discussion focuses on these conferences. It tries to group the key statements of the papers into topics which are similar to those used in the questionnaire (Section 2).

3.2 General Aspects of Site Release

The presentation "Safe Nuclear Decommissioning: Need for an International Common Approach" of A.J. González (*Berlin 2002*) provides an overview of the present system of radiation protection. Because of its relevance for Section 4, the contents of this paper is described in Section 4.1.

The dose concepts for site release which may be pursued by different countries is discussed by the paper "Co-operation and Consensus in the Development of Decommissioning Approaches" by G.J. Dicus (*Berlin 2002*). It is stated that approaches to conducting facility decommissioning could also vary depending on the state's interpretation and perspective. Ultimately, this may be the result of whether a national authority would adopt the international view that decommissioning is part of the overall pre-disposal radioactive waste management system, or view environmental restoration within the realm of an intervention. However, in same instances decommissioning and environmental restoration are viewed as both returning facilities and sites to uncontrolled or unrestricted use conditions, without distinguishing whether the activities are practices or interventions. This issue becomes more noteworthy when the criteria for compliance are examined. For example, restoration of a contaminated land area to the ICRP 82 suggested constraint of 0.30 mSv/a, as opposed to the target for clearance of 10 µSv/a, represents a vast difference, both in the resultant doses as well as in the resources needed to achieve compliance. The difference in the volume of radioactive waste inventories generated for disposal is very significant.

A related topic which is addressed in the paper "Magnitude of the decommissioning task in North America and Mexico, including the US Department of Energy" by P.M. Bubar and B.R. Clark (*Berlin 2002*) is the question "How clean is clean?". In that paper, this question is discussed in conjunction with the U.S. DOE, its regulators and its stakeholders. This can be rephrased as: do all sites have to be cleaned up to the same standards (i.e. green field or unrestricted use), or can some sites planned for restricted use such as reindustrialization be cleaned up to lesser (i.e. brown field) standards. It is argued that these decisions must be made very early on in the planning process associated with decommissioning and cleaning up a site. The amount of waste generated from various decommissioning strategies will be vastly different. For example, a site that will be under long term institutional control restricting site access could be expected to generate about 1 million m^3 of waste during decommissioning. The same site, if slated for brown field mixed land use (e.g. industry) may generate about 35 million m^3 of waste during cleanup. If the green field unrestricted land use approach is adopted (i.e. residential/agricultural), the waste volume would be expected to nearly triple to about 100 million m^3. The amount of time, work, and expense associated with cleanup to various land use standards also rises in a similar pattern. There must therefore be a balance between the costs involved with cleaning up a site and the amount and quality of land use needed or wanted. A graded approach to cleanup whereby "how clean is clean" is determined on a site by site basis would be a important to adopt.

3.3 Site Releases within Decommissioning Projects

3.3.1 JPDR and TPS, Japan

The release of the site of the JPDR (Japan Power Demonstration Reactor) and of the TPS (a gas-cooled reactor in Japan) are described in the paper "Magnitude of the Decommissioning Task in South and East Asia and Oceania" by M. Tanaka and S. Yanagihara (*Berlin 2002*).

- The JPDR was a BWR type reactor with a power of 90 M(th). It operated between 1963 and 1976. The decommissioning programme commenced in 1981 and ended in 1996 with achievement of green field conditions. (more details on site release can be found in Section 2).
- The TPS was a gas-cooled reactor with 166 MWe which was in operation between 1966 and 1998 as the first commercial NPP in Japan. The TPS decommissioning is in line with the government's policy. The site will be recovered to a green field condition for the next nuclear power plant. The decommissioning plan is divided into three phases. The dismantling activities started in December 2001. In the first phase, decommissioning activities concentrated on preparatory work and removal of conventional facilities such as cleaning of the spent fuel pond surface, reformation of utility systems and auxiliary cooling systems and removal of turbine systems and the fuel exchange machine. In the second phase, steam raising units will be dismantled over five years. In the third phase, the reactor area and the buildings will be dismantled and the site will be prepared for nuclear re-use. It is estimated that the decommissioning will be completed in 2017.

3.3.2 Approach taken in the UK

The approach to site restoration in the UK is briefly summarized in the paper "Regulatory challenges in the Decommissioning of Nuclear Facilities in the United Kingdom" by L.G. Williams (*Berlin 2002*). The Health and Safety Executive's Nuclear Installations Inspectorate (NII) has required all the nuclear operators to produce decommissioning strategies for their nuclear sites and to develop plans to implement those strategies. These strategies need to cover the task from start to finish, which ideally should be to clean up the sites to the extent that they can be released for other use, although this might not be possible in all cases.

3.3.3 Approach taken at Risø, Denmark

The approach taken in Denmark to site restoration is described in the paper "Decommissioning of the Nuclear Facilities at the Risø National Laboratory in Denmark" by M. Bagger Hansen and K. Larsen (*Berlin 2002*). After completion of decommissioning, the site may need to be restored and cleaned of the remaining contamination. The selection of restoration techniques, which can be appropriately applied, will depend upon a number of factors. The major factors include: (1) the scale of the contamination problem and the radionuclides involved; (2) the contaminated medium; (3) the location of the contaminated site with respect to the local population; and (4) the location of the contaminated site with respect to a suitable waste repository for any residues. The need for restoration will be based upon a comprehensive radiological survey of the site and a dose constraint of 50 μSv/a to the critical group. - A few alternative options to fast decommissioning to green field status have been considered. These include safe storage, where the nuclear plant is kept intact and placed in protective storage for several tens of years, and entombment, where the radioactive structures, systems and components are encased in a long-lived substance such as concrete. The latter is equivalent to establishing an on-site shallow land burial waste disposal facility. It is, however, very unlikely that any of these alternative options will be selected.

3.3.4 Vandellós and other Nuclear Installations in Spain

General aspects of site release in Spain are dealt with in the paper "Removal of Regulatory Control: The Spanish Experience" by J.C. Lentijo (*Berlin 2002*). As has been mentioned previously, there is no generally applicable radiological criterion to support cleanup restoration or site release in Spain. Some decommissioning projects already finished, like the stabilization of some uranium concentrate mill tailings and the restoration of old uranium mines sites, have been governed by particular criteria included specifically in the licence or authorization granted to each individual holder to whom the clearance or release applies.

The criteria that governed the decommissioning programme at the Andujar mill tailings stabilization project were taken from the standards given by the US Environmental Protection Agency for the rehabilitation of uranium mill tailings in the UMTRA programme and Spanish groundwater protection regulations. These criteria can be summarized as part of an effective equivalent dose to individuals in the critical group below 100 μSv, and an additional reduction in the residual concentration of Ra 226 on land, so the background level is not exceeded by more than 0.2 Bq/g (in the upper 15 cm of soil) and 0.6 Bq/g (in the 15 cm thick layers of soil more than 15 cm below the surface). It is worth noting here the establishment in 1995 of a CSN working group to derive radiological criteria for the decommissioning and restoration of uranium ore processing sites. The report, which included the criteria for site this restoration and site release, never came into force and never went beyond the draft stage. The proposed criteria were, nevertheless, subsequently included in later authorizations granted for new restoration and remediation projects. The document indicated, basically, that intervention to decontaminate the site was justified if the effective dose to individuals in the critical group is above 100 μSv/a. Intervention was not justified in any case below an effective dose of 10 μSv/a. Intervention in the range of 10-100 μSv/a will be necessary if the individual exposure in any hypothetical and conservative scenario implies an individual effective dose above 1 mSv/a, the dose limit for the public.

Consideration was given to suitable options for using the land after clearance, which must be realistic for the location in question, as well as to the relevant exposure pathways. This analysis stated that the agricultural/residential scenario (family farm) was the most restrictive scenario resulting in a guideline concentration for soil contamination of 0.1 Bq/g of U 238, in equilibrium with all the radionuclides of its natural decay chain. Higher values derived from other generic exposure assessments, requiring some special additional conditions, were established for three restricted and more plausible scenarios, as provided in Table 3.1.

Table 3.1: **Options for land use after clearance and release criteria used in Spain**

Scenario	Criterion
Agricultural/residential	up to 0.1 Bq/g
Forestry/grassland use	up to 1 Bq/g
Recreational area	up to 1 Bq/g and H < 0.1 Gy/h
Industrial use	up to 1 Bq/g and H < 0.3 Gy/h closed building only in soils < 0.1 Bq/g radon concentration inside buildings < 200 Bq/m³

Radiological criteria for the partial release that is being considered for the Vandellós 1 nuclear site have been proposed in the site restoration plan submitted by ENRESA to the CSN. The main features of the proposal can be summarized as follows:

- Relevant radionuclides that are considered in the analysis are: H 3, C 14, Ni 59, Ni 63, Co 60, Sr 90, Nb 94, Sb 125, Cs 137, Eu 152, Eu 154, Pu 239 and Am 241.
- Industrial scenario in the next 30 years: external exposure, inhalation and soil ingestion pathways.
- Residential scenario after 30 years: external exposure, inhalation and limited ingestion of vegetables and water, including inadvertent soil ingestion.

Dose release criteria (100 μSv/a) have been translated into corresponding derived concentration guideline levels using the RESRAD code for the two different scenarios. Site specific parameters have been used in the calculations and, for each radionuclide, the most restrictive concentration obtained in both scenarios is taken as the proposed concentration level. Typical values obtained for key radionuclides are as follows: 0.4 Bg/g for Co 60, 0.3 Bq/g for Cs 137, 0.15 Bq/g for Sr 90, and 0.3 Bq/g for C 14.

The radiological surveys to be conducted to demonstrate compliance with the derived concentration limits are based on the MARSSIM approach (NUREG-1700 and NUREG-1727) and include the planning, implementation, assessment and decision making phases required for a final status survey.

A historical site assessment and a scoping survey are initially performed to provide the necessary information to design the characterization survey. The characterization survey integrates scanning surveys with direct measurements and sampling and includes the classification of areas, the definition of survey units and the determination of the required data points. Appropriate statistical tests are finally used to demonstrate compliance for each survey unit.

Another project that is going to be implemented in Spain in the near future is the rehabilitation of the Centre for Energy-related Environment and Technical Research (CIEMAT), which is the main Spanish energy research centre and includes the former Spanish Nuclear Energy Board (JEN), created with a view to promoting the development and use of nuclear energy in Spain. The contamination existing inside the facilities, although low in level, means the continuation of a situation of risk without any benefit and causes difficulties to some other non-nuclear projects foreseen in the centre. The rehabilitation of the Centre requires an integrated safety improvement programme, part of which includes the restoration of pieces of land. According to the preliminary results of the characterization studies, there are some areas and buildings that have levels of contamination that are not acceptable for the activities to be performed at the Centre in the future. Different radiological criteria are proposed for the rehabilitation of areas with surface contamination and subsoil areas affected by activities carried out in the past that originated underground contamination. In this latter case, a set of actions that should be qualified as non-emergency interventions are being considered.

3.3.5 VKTA Rossendorf, Germany

The release of nuclear sites in the VKTA research centre at Rossendorf in Germany is described in the paper “Free release of the premises of a company according to the “de minimis”-concept – an application from the point of view of the operator and the consultant” by R. Knappik (*KONTEC 2003*) as well as in the paper “Erfahrungen mit der Freigabe radioaktiver Materialien

bei der Stilllegung in Rossendorf" by H.-D. Giera *et al*. (*Hamburg 2003*). The VKTA is a research installation near Dresden, Germany, which was formerly focussed on nuclear research.

A licence for clearance of a part of the site according to sect. 29 (2) StrlSchV (Radiation Protection Ordinance) was applied for in May 2002. This part of the site comprised a former controlled area for different liquid and solid radioactive materials which was situated partly outdoors. Radioactive materials were handled and stored in this controlled area since 1957. The storage installations have been decommissioned and emptied. A concept for site and soil remediation has been developed in the VKTA. This includes a site specific radiological evaluation for this particular site outdoors. The clearance levels which have been derived for the radionuclides Co 60, Cs 137 and Sr 90 are based on the 10 µSv concept. This means that a certain amount of residual contamination may remain in the soil. This possibly remaining contamination must correspond to an individual dose of at the most 10 µSv per calendar year even under pessimistic assumptions. This concept has been verified by an independent expert organization, the Öko-Institut e. V. Darmstadt.

3.4 Decommissioning Strategies and Public Acceptance

Some papers highlight the fact that decommissioning strategies and public acceptance of decommissioning are closely linked. According to the paper "Magnitude of the Nuclear Decommissioning Task in Africa and West Asia" of P.J. Bredell *et al*. (*Berlin 2002*), public opinion seems to favour early decommissioning, as these activities result in dismantling of disused facilities and the release of the land for non-nuclear use. A suitable process for the release of the formerly nuclear sites must therefore be available.

The magnitude of the task of site release in the USA and in Mexico is illustrated in the paper "Magnitude of the decommissioning task in North America and Mexico, including the US Department of Energy" by P.M. Bubar and B.R. Clark (*Berlin 2002*). Within the DOE's nuclear weapons complex, there are over 114 sites with a total area of 810,000 hectacres (8100 km²) that need to be cleaned up. The decommissioning strategies under consideration in this context are:

- Immediate dismantling (or early site release/decontamination) – DECON: After removal of the facility, the site is then available for reuse.
- Safe enclosure – SAFSTOR: The final removal is postponed for a longer period, usually 40 – 60 a during which the facility is placed into a safe storage condition. Finally, the facility will be removed and the site will then be available for reuse.
- Entombment – ENTOMB: The facility is placed into a condition that will allow the remaining on-site radioactive material to remain on-site without the requirement of ever removing it totally. After an appropriately long decay period, the activity is of no concern anymore. During the entombment time, the site cannot be released.

Currently, 14 NPPs in the US have chosen the SAFSTOR strategy, and 6 NPPs utilized DECON. From this, a certain tendency towards postponing large investments (implementation of decommissioning) can be derived. A further major factor in costs is the end state to be achieved for the site. For example, cleanup of a reactor site to an unrestricted usage or "green field" condition is more expensive than a site planned for restricted usage "brown field".

The general policy towards site release in the U.S. NRC is outlined in the paper "Removal of Controls for Decommissioning: A Graded Approach" by J.T. Greeves *et al*. (*Berlin 2002*). There it is stated that in the area of environmental remediation of sites and release of sites from regulatory control, the NRC established risk informed, dose based standards in 1997. The NRC requires that

residual radioactive material at sites not result in a dose to the average member of the critical group of more than 0.25 mSv/a from all environmental pathways and that these doses are as low as is reasonably achievable (ALARA). This approach is consistent with International Commission on Radiological Protection (ICRP) Publication 82. The NRC believes that remediation of sites to levels that permit unrestricted use is the preferable alternative. However, it is recognized that flexibility is needed in the remediation of nuclear sites because all sites may not be able to reach an unrestricted release status. Therefore, the NRC also provides for license termination with restrictions on future site use if it can be justified to the regulator. Restrictions can be graded according to the risk on a site specific basis. Separate environmental site cleanup and clearance standards are supported because experience indicates that a single standard for environmental remediation and clearance will not meet the need for adequate protection without undue burden on operator and regulatory resources. A graded risk informed/performance based approach should be used in conjunction with threshold limits. The NRC has made experience which supports the philosophy of not placing clearance and environmental remediation in the same dosimetric range, because the benefits may not support the environmental costs. For example, at sites with soil or groundwater contamination, remediation may cause excessive environmental damage to comply with a 10 µSv/a dose constraint. Finally, because there is an inherent difference in the implementation strategy between complex site cleanup and the release of commercial products that contain slight amounts of radioactivity, the use of a single standard is not appropriate. Cleanup is associated with land, natural resources and real estate.

A statement valid for the USA is made in the Panel Discussion during the Conference *Berlin 2002* by K.D. Crowley. These comments apply primarily to the situation in the USA, and most comments relate to the regulator-stakeholder (public, political bodies, other regulators) interface, which increasingly is seen to be the rate limiting part of the process. In the USA, criteria for the release of nuclear sites are defined using a dose based standard: 0.25 mSv per year with as low as reasonably achievable (ALARA). Most experts agree that this standard, if implemented appropriately, is protective of public health.

However, there is resistance to this expert view, and also resistance to efforts by US regulators to establish criteria for releasing radioactively contaminated materials. There are at least two reasons for this resistance, both of which are related to stakeholder trust and confidence:

- The criteria that have been implemented or proposed are difficult for most stakeholders to comprehend, and many are unwilling to rely on the assurances of regulators that the criteria are protective.
- Disagreements between the two primary regulatory authorities (the US Environmental Protection Agency and the NRC) over the numerical values of criteria for unrestricted site release (the so-called 15 versus 25 millirem debate) has not inspired public confidence. It also has led some to conclude that the numerical difference between these standards is significant.

This lack of confidence is being expressed, as it often is in democratic societies, through political action: Several US States have adopted unrestricted release standards that are more protective than the Federal standard, and the legislature of the largest State in the US (California) recently directed its regulators to develop such standards. The news media has reported that some California legislators favour a 1 millirem (10 µSv) per year standard. Implementing such a standard would be an expensive and technically daunting challenge.

The statement further addresses the question whether criteria can be defined in a way that will improve their likelihood of wider public acceptance. This issue has evolved considerably over the past decade, and several reports of the National Research Council have suggested that the explicit

use of risk based standards by regulators could improve stakeholder acceptance. The use of risk based standards has several distinct advantages:

- They provide opportunities for stakeholders to provide meaningful input because they allow the proposed standards to be compared with standards for other societal hazards. This provides for the possibility of harmonization.
- They require regulators to make explicit all of their assumptions about the relationships between radionuclide concentrations and human health risk.
- They also require regulators to acknowledge uncertainties in the knowledge base.

It should be pointed out that risk based standards can be expressed in terms of dose. The difference is the way in which the standard is derived and presented to stakeholders: A risk based standard is derived from an explicit analysis of the relationship between concentrations and health risk, information about the analysis is provided to stakeholders (preferably in an understandable form), uncertainties are acknowledged, and the final standard is set by taking into consideration acceptable risks for other societal hazards through a broad based stakeholder participation process. Regulators in the USA are taking some steps in the right direction in this regard. The NRC, for example, is adopting what it calls a risk informed approach for regulating nuclear activities. The US Department of Energy is also beginning to show an interest in using risk based approaches for decision making in its cleanup programme. It is therefore suggested that careful consideration is given to the use of risk based approaches in the future.

The paper “The IAEA’s Decommissioning Concept” by D. Reisenweaver (*Berlin 2002*) discusses i.a. the relation between the Safe Enclosure decommissioning strategy and site release. The facility is placed into a long term storage condition for up to 50 years, followed by the final decontamination and dismantling of the facility to allow removal of all regulatory control. To allow this storage period to occur, all of the liquids are drained from the systems, any operational waste that has been collected during the operational period is removed and areas not normally in need of access during the storage period are secured. This option does also allow for the decay of radionuclides. There are many advantages to this option. Some minor decontamination may occur and allow the boundary or 'footprint' of the controlled area to be significantly reduced, which will save money and other resources over the enclosure period. Portions of the facility or site may be used for other purposes. Large exclusion or buffer zones are no longer needed.

Problems with using different radiological criteria for clearance (10 μSv/a) and for site restoration (e.g. 0.3 mSv/a) are discussed in the paper “Co-operation and Consensus in the Development of Decommissioning Approaches” by G.J. Dicus (*Berlin 2002*). It is stated that separating decommissioning and restoration could prove to be counterproductive if different dose levels are utilized. For instance, different dose protection levels create a perception of non-uniform levels of protection, which in turn may be perceived to correspond to significantly inconsistent risk levels. Furthermore, in cases where significant efforts would be needed to comply with inordinately stringent dose constraints, alternative strategies could raise the impression of regulatory disparity or environmental inequity. The perception may be that cleanup is held hostage to economics and, as a result, national authorities would more likely rely on the use of institutional controls. If a more realistic level was used, remediation could be accomplished by utilizing the ICRP optimization approach, which would foster safe, environmentally sound and more feasible cleanup levels. As a result, the path to co-operation and consensus will need to be pursued actively from both the national and international deliberation arenas.

3.5 Other Topics and Background

The aspect of imposing (conventional) site use restrictions for sites which have been released is discussed in the panel discussion statement by J. Averous (*Berlin 2002*). Recent experience involving the nuclear industry as well as the conventional industry has shown the necessity to keep track of past uses of land and to at least define the minimum use restrictions when a facility handling hazardous materials has been occupying the site. This conclusion is based on technical considerations (how far can it be proven that a piece of land has been absolutely cleaned of all hazardous contaminants). as well as on social considerations (cases when observation of a cluster of some sickness is automatically linked to past uses of the land, even if the link between this sickness and potential contamination cannot be proven). Basic precautionary use restrictions should include minimum measurement requirements when digging or performing any civil engineering works (in particular digging and earthworks), and a prohibition against erecting buildings involving potentially more sensitive occupants, like schools. Of course, the application of these use restrictions has to be taken into account in the urbanization plans for the vicinity in order to optimize land use.

The question whether contaminated soil may be left in place is discussed by J. Averous and V. Štefula in panel discussion statements during the conference *Berlin 2002*.

- J. Averous expresses the opinion that Contaminated soil can be left in place provided that optimzation has been done and that it is not justified to intervene on a cost/benefit basis. This topic has to be dealt with in an open discussion taking in stakeholders. An impact study has to be provided and land use restrictions as well as, as a precaution, possibly continuous monitoring of the site have to be put in place.
- V. Štefula provides the example of the NPP Bohunice A1: The contaminated soil problem is typical for nuclear power plants shut down after an accident. Such is the case of the Al nuclear power plant at the Bohunice site. The safety principles for liquid waste storage were introduced only in 1987, and thus there are hot spots of contaminated soil, predominantly because of poor liquid waste storage practices in the past. The worst contaminated soils have already been removed and are stored in drums nowadays; however, some hot spots still remain. As a mitigation measure, water is being pumped out of the aquifer underneath and discharged into the river. This questionable decision is to prove to the public how environmentally considerate the operator is because the groundwater is seen as a potential source of drinking water and discharge into the river within the concentration limits is not in conflict with legislation. In this respect, a project for contaminated soil landfill has been launched with the intention to collect all the contaminated soil stored in drums, remove the hot spots from around the old leaky tanks and dispose of them in a controlled manner. The location for the landfill has been chosen within the Bohunice nuclear site. From the assessor's point of view, the above mentioned decision on groundwater extraction and discharge into the river can be challenged. The safety analysis proves that there is sufficient dispersion and retardation of the contaminants in the aquifer before it gets to the nearest well. Quite a large amount of the contaminated soil removed during excavation of the groundworks for the new waste processing centre in Bohunice has been tentatively stored in old unused concrete basins and landscaped to a “green field” condition. Yet, it is still part of a nuclear facility under regulatory control. A side effect of this safety analysis is that it proved that even the tentative solution to the contaminated soil problem can be requalified as a final solution with no further remediation activities, and the basins can be taken out of regulatory control. So the answer to the question posed above in Slovakia is: Yes, the contaminated soil can be left “in place” if it is done in a controlled manner with sufficient safety measures in place. That means that the soils will be accumulated in a single place - the landfill with appropriate cap, and some

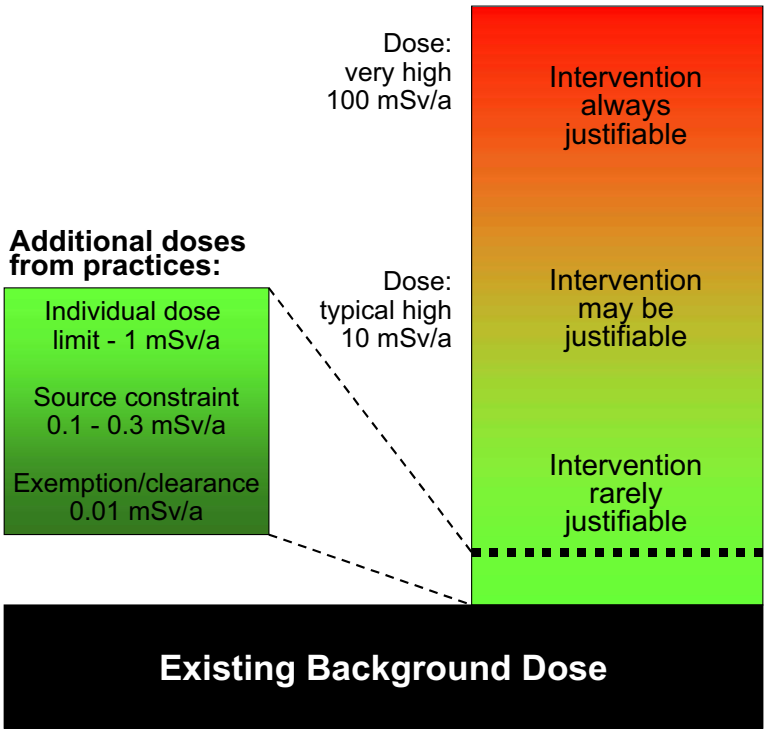
institutional control will be introduced to prevent human intrusion for a certain period of time. In the case described here the minimum institutional control period being considered is 100 years, which correlates approximately with the time required for decommissioning all the existing nuclear facilities on the site.

4. EVALUATION OF RADIOLOGICAL MODELS FOR RELEASE/CLEARANCE CRITERIA OF NUCLEAR SITES

4.1 Underlying Dose Criteria

An outline of the present system of radiation protection can be found in the presentation "Safe Nuclear Decommissioning: Need for an International Common Approach" of A.J. González (*Berlin 2002*). It is summarized in Figure 4.1.

Figure 4.1: **The present system of radiation protection**



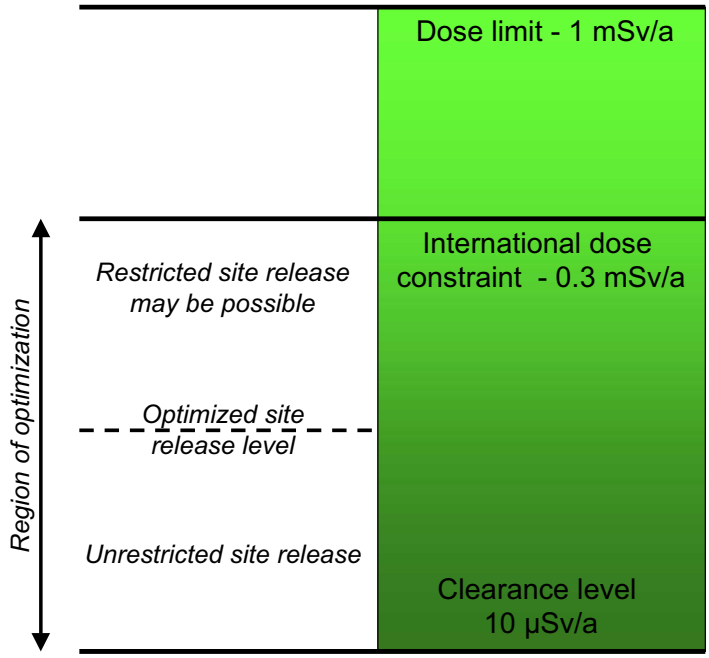
This figure highlights the fact that there are various regions where different aspects of the system of radiation protection apply. The existing background dose may amount on average to several mSv/a (e.g. 2.4 mSv/a world-wide average background), but may reach typical high values of around 10 mSv/a and in some places very high values of around 100 mSv/a. Measures for intervention may be justifiable from the range of 50 to 10 mSv/a, while they will always be justifiable in the range of several 10 to 100 mSv/a.

This figure also makes clear that any doses from practices are to be understood as being above the existing background dose. While the limit for individual doses is 1 mSv/a, source related dose constraints (e.g. 0.1 – 0.3 mSv/a) may be used. In addition, the regime of exemption and clearance is governed by the trivial dose range, i.e. the range on the order of 10 µSv/a or several 10 µSv/a.

The release of sites from control is situated in this system of radiation protection on the left side in the context of additional doses from practices. The appropriate dose criterion is, however, not *a priori* clear. If the site release is understood as clearance in a specific country, the dose criterion would be 10 µSv/a; if it is understood as a kind of authorized release, an appropriate dose criterion would be a value between 0.1 and 0.3 mSv/a.

Other fundamental aspects of site release are discussed in the paper "International Criteria for the Termination of Practices: Release of Materials and Sites from Regulatory Control" by G. Linsley (*Berlin 2002*). It focuses on that part of the scheme of radiation protection which is depicted on the left-hand side of Figure 4.1, i.e. the range between dose constraints for clearance (10 µSv/a) and the public dose limit (1 mSv/a) with respect to site release. This is depicted in Figure 4.2.

Figure 4.2: **A scheme for the release of sites**



The ICRP, in its Publication 82, has provided advice on the release of sites of formerly operating practices from regulatory control. Essentially, the guidance is that sites may be released from regulatory control if predicted doses from all plausible future uses of the site are within the dose constraint. If the dose constraint is not satisfied, site release may still be possible, but under conditions which restrict the potential for public exposure, for example, by limiting the use of the site to industrial operations. As yet the IAEA has not developed guidance on this subject.

The paper "International Criteria for the Termination of Practices: Release of Materials and Sites from Regulatory Control" further outlines the fundamentals for the development of guidance on the release of sites of formerly operating practices from regulatory control:

A comparison between the safety impact of the release of materials from control and of the release of sites from control provides insights which are useful in the development of release policies. In the case of release of materials from control, because of the potential for the released materials to be used for a wide variety of purposes, for there to be multiple exposures to cleared materials and for them to be subject to movements within and outside the country of origin, the radiological

criteria for establishing the clearance levels have been set very low. In contrast, sites, by their nature, are fixed in position and the potential exposures from future use after release can be predicted with some certainty. No transboundary movements of the sites are possible and, furthermore, in the event of some unforeseen event remedial actions are possible (unlike in the case of released materials). It is also necessary to take account of other related practices involving the release of radioactive materials to ensure that there is overall coherence in the local radiation protection policy. The practice of authorized discharge of radioactive materials to the environment has the potential to contaminate areas surrounding the discharge point, including, possibly, the site under consideration for release from control. The criteria for site release must therefore be compatible with the discharge criteria and vice versa.

When all of these factors are taken into account it may be concluded that:

- In the context of practice situations, the overriding policy for the release of materials and sites from control should be based on constrained optimization as established in the BSS for the control of the exposure of the public.
- There should be coherence in the policies for release of materials, of sites and of discharges.

The scheme outlined in Figure 4.2 has been developed from the basic scheme for protection of the public from single sources. For the release of sites, the "optimized site release level" should be determined by a process of optimization, taking into account on the one hand the benefits, in terms of dose reduction, of remediating the site and, on the other hand, the costs of the remediation activity. The process should be constrained by the dose constraint of 0.3 mSv per year. Sites at which the residual level is below the optimized site release level may be released for unrestricted use. For sites at which it is deemed not to be cost effective to reduce the contamination levels to below the optimized site release level, restricted site release may be possible. This may involve to use restrictions being placed on the site by the Regulatory Authority and records being kept to ensure that restrictions are complied with in the future. In setting the optimized site release level, account should be taken, where necessary, of the criteria established for local practices involving the discharge of radionuclides, especially where such discharges could affect radiation levels at the site under consideration.

It is further noted that this approach allows site specific or country specific values to be adopted. The optimized site release level has been fixed at various levels in different countries ranging from 10 μSv/a to 0.25 mSv/a. The proposed scheme has the advantage of being flexible and allows, within dose limits and dose constraints, account to be taken of the difficulties of remediation in different situations and of variations in economic circumstances.

4.2 Categorization of Areas

It is a very reasonable approach to categorize the areas of the site which is to be released according to their radiological relevance. A broadly used approach is presented in Chapter 2 of the MARSSIM Document [MAR 02]:

- *Class 1 Areas*: Areas that have, or had prior to remediation, a potential for radioactive contamination (based on site operating history) or known contamination (based on previous radiation surveys) above the clearance/release criteria. Examples of Class 1 areas include: 1) site areas previously subjected to remedial actions, 2) locations where leaks or spills are known to have occurred, 3) former burial or disposal sites, 4) waste storage sites, and 5) areas with contaminants in discrete solid pieces of material and high specific activity.

- *Class 2 Areas*: Areas that have, or had prior to remediation, a potential for radioactive contamination or known contamination, but are not expected to exceed the clearance/release criteria. To justify changing the classification from Class 1 to Class 2, there should be measurement data that provides a high degree of confidence that no individual measurement would exceed the DCGLw. Other justifications for reclassifying an area as Class 2 may be appropriate, based on site-specific considerations. Examples of areas that might be classified as Class 2 for the final status survey include: 1) locations where radioactive materials were present in an unsealed form, 2) potentially contaminated transport routes, 3) areas downwind from stack release points, 4) upper walls and ceilings of buildings or rooms subjected to airborne radioactivity, 5) areas handling low concentrations of radioactive materials, and 6) areas on the perimeter of former contamination control areas.
- *Class 3 Areas*: Any impacted areas that are not expected to contain any residual radioactivity, or are expected to contain levels of residual radioactivity at a small fraction of the clearance/release criteria, based on site operating history and previous radiation surveys. Examples of areas that might be classified as Class 3 include buffer zones around Class 1 or Class 2 areas, and areas with very low potential for residual contamination but insufficient information to justify a non-impacted classification.

Class 1 areas have the greatest potential for contamination. These areas therefore require the highest effort for release (highest number of sampling/measurement points etc.).

Table 4.1: **Recommended Survey Coverage for Land Areas (from [MAR 02])**

Area Classification	Surface Scans	Soil Samples
Class 1	100 %	Number of data points from statistical tests; additional measurements may be necessary for small areas of elevated activity
Class 2	10 to 100% - systematic and judgmental	Number of data points from statistical tests
Class 3	Judgmental	Number of data points from statistical tests

For Class 1 areas, scanning surveys are designed to detect small areas of elevated activity that are not detected by the measurements using the systematic pattern. For this reason the measurement locations, and the number of measurements, may need to be adjusted based on the sensitivity of the scanning technique. This is also the reason for recommending 100%.

Approaches similar to those cited above from the MARSSIM document have been adopted in various site release projects. An approach like this has for example been used in the release of sites from the NPP Greifswald (KGR) in Germany (cf. section 2). The definitions of the site categories used there are reproduced in Table 4.2 for comparison with the MARSSIM approach:

Table 4.2: **Definitions of the site categories for the NPP Greifswald (KGR) in Germany**

Category	Description	Measurement requirements
I	uncontaminated – no contamination with artificial nuclides	no clearance measurements only control measurements for verification purposes
II	suspicion – a contamination with artificial nuclides cannot be excluded	the clearance procedure (incl. measurement density) is chosen according to the nature of the site to be cleared
III	contaminated - contamination with artificial nuclides is present	like category II; increased measurement density, especially at points where activity near clearance levels has been detected

4.3 Radiological Pathways and Scenarios

4.3.1 Introduction

In general, radiological pathways and scenarios and the combination of all such pathways and scenarios, the radiological model, establish a link between the hypothetical doses to members of the public (including non-radiation workers working on the site to be released) and the residual radioactivity which may remain in the soil or on the site.

It must further be distinguished between cases where site specific evaluations are to be used and general approaches e.g. where a country establishes generic release/clearance levels for sites which can be used nation-wide:

- Site specific approaches will usually concentrate on a smaller number of exposure pathways and scenarios which are tailored to the conditions of the site. Site specific models will take account of site specific parameters, like the size of the site, the exact nuclide vector, known details of the future use of the site, meteorological, hydrological and other parameters relating to the site etc.
- Generic approaches need to accommodate a larger number of different sites the details of which are not known a priori and can therefore not be incorporated into the models. Generic models have to include all pathways and scenarios which might become relevant for any site in the country or in the region for which the derived release/clearance levels shall be valid. Such models therefore may have a tendency towards the conservative side when compared with site specific approaches.

The following two examples from the German and the US approach concentrate on generic models because information on site specific approaches is scarce. At this point, only an overview of the pathways and scenarios can be given.

4.3.2 USA – RESRAD Code

The RESRAD code is described in detail in the User's Manual for RESRAD [RES 03], issued by the Environmental Assessment Division of Argonne National Laboratory, United States Department of Energy. RESRAD is a computer model designed to estimate radiation doses and risks from RESidual RADioactive materials. RESRAD represents a continuously developed computer code which was issued in its first version in 1989. Since this time, RESRAD has been used widely by the U.S. Department of Energy (DOE), its operations and area offices, and its contractors for deriving limits for radionuclides in soil. RESRAD has also been used by the U.S. Environmental Protection Agency (EPA), U.S. Army Corps of Engineers, U.S. Nuclear Regulatory Commission (NRC), industrial firms, universities, and foreign government agencies and institutions.
RESRAD takes into account a detailed radiological model which establishes the link between the fundamental dose criterion (usually values between 100 and 300 μSv/a are used in the USA) and the mass or surface specific activity levels for the relevant nuclides. The radiological model comprises i.a. the following scenarios:

- resident farmer;
- suburban resident;
- industrial worker;
- recreationist.

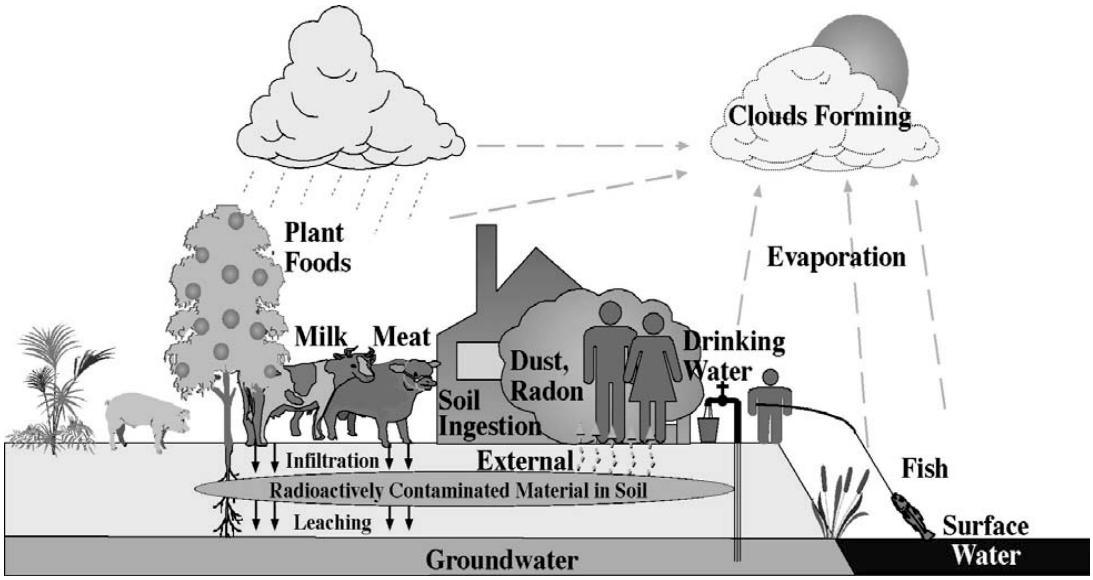
For each scenario, a number of exposure pathways is considered. The following table gives an overview of pathways and scenarios.

Table 4.3: **Pathways to be Considered for the Resident Farmer, Suburban Resident, Industrial Worker, and Recreationist Scenarios**

Exposure pathway	Resident Farmer	Suburban Resident	Industrial Worker	Recreationist
External gamma exposure	Yes	Yes	Yes	Yes
Inhalation of dust	Yes	Yes	Yes	Yes
Radon inhalation	Yes	Yes	Yes	Yes
Ingestion of plant foods	Yes	Yes	No	No
Ingestion of meat	Yes	No	No	Yes
Ingestion of milk	Yes	No	No	No
Ingestion of fish	Yes	No	No	Yes
Ingestion of soil	Yes	Yes	Yes	Yes
Ingestion of water	Yes	No	No	No
Resident Farmer: water used for drinking, household purposes, irrigation, and livestock watering is from a local well in the area. Suburban Resident: no consumption of meat and milk obtained from the site, and the water used for drinking is from offsite sources. Industrial Worker: no consumption of water or food obtained on the site. Recreationist: no consumption of food except meat (game animals) and/or fish obtained from the onsite pond, and the water used for drinking is from offsite sources.				

An overview of the pathways which are used in the model is given in Figure 4.3. It should be noted that not all pathways depicted in that figure are used for all site assessments. The default assignment between pathways and scenarios is given in Table 4.3, but for site specific assessments only a subset of these scenarios may be used according to the local circumstances.

Figure 4.3: **Pathways used in the RESRAD Code [RES 03]**



The critical group of persons is obviously a family who lives on the released site and who produces the major part of their foodstuff on this site (including cattle, milk and drinking water).

The RESRAD code has been used for derivation of clearance levels e.g. for the Cintichem site (former research reactor site). The results are presented in Table 4.4.

Table 4.4: **Release levels for the Cintichem site derived with the RESRAD program – values relate to soil**

Nuclide	Release level [Bq/g]	Nuclide	Release level [Bq/g]
Mn 54	0.1	Cd 109	2
Fe 55	20,000	Cs 134	0.07
Co 60	0.03	Cs 137	0.1
Sr 90	0.6	Eu 152	0.07
Tc 99	17	U 238	0.8
Ag 110m	0.03	Pu 241	1

The RESRAD code and its application is very complex. Further discussion at this point is beyond the scope of this report.

4.3.3 Germany – Derivation of Clearance Levels for Sites

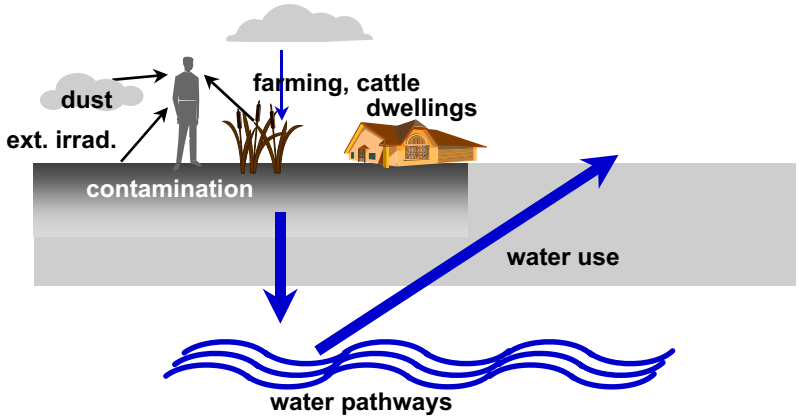
The clearance levels for sites of nuclear installations in Germany have been derived on the basis of a radiological modelling starting from the dose criterion for triviality (10 µSv/a). A report is available which describes the model and the results in detail (Thierfeldt, S.; Nüsser, A.; Deckert, A.; Kugeler, E.; Schramke, M.; Neuhaus, I.: Stillegung von Kernanlagen - Freigabe von Bodeflächen kerntechnischer Standorte; Endbericht zu Vorhaben SR 2271 des Bundesministeriums für Umwelt, Naturschutz und Reaktorsicherheit; Brenk Systemplanung, Aachen, September 1999 – in German only).

The radiological model which is developed in this report has lead to the clearance levels for sites which are provided in the German Radiation Protection Ordinance (Strahlenschutzverordnung).
The model contains scenarios for:

- workers on the construction site (conventional demolition of the old nuclear facility, construction of new buildings, landscaping) and
- persons of the general public (all age groups: < 1, 1-2, 2-7, 7-12, 12-17, >17 a): people living on the site in one-family houses with garden, with age dependent parameters for exposure times, breathing rates, ingested quantities etc.

The general overview of these pathways is given in Figure 4.4.

Figure 4.4: **Pathways used in the German model for derivation of clearance levels for sites**



The following exposure pathways are considered for each scenario (where appropriate):

- external irradiation from the contaminated soil layer
- dust inhalation
- direct ingestion of contaminated soil
- ingestion of vegetables grown in contaminated soil
- use of drinking water via groundwater pathway and via surface water pathway

Table 4.5 gives an overview of clearance levels calculated with the model outlined above. They are for generic use at any site. If these values are not applicable, site specific investigations and analyses have to be performed.

Table 4.5: **Examples of Clearance levels for sites (values relate to soil, in Bq/g)**

Radionuclide	Clearance level	Radionuclide	Clearance level
H 3	3	Cs 137	0.06
C 14	0.04	I 131	0.2
Fe 55	6	Pu 242	0.04
Co 60	0.03	Am 241	0.06

5. OVERVIEW OF MEASUREMENT TECHNIQUES FOR SITE RELEASE

5.1 Overview of Measurement Techniques

This section gives an overview of measurement techniques used for alpha, beta and gamma emitting nuclides. The overviews presented here only serve as indicators which techniques are available. For the application in a particular site release/clearance process, the choice of appropriate techniques depends on many factors, like e.g. the conditions of the surface, the penetration depth, the nuclide vector etc.

5.5.1 Measurement Techniques used for alpha surveys

Table 5. 5.1 gives an overview of measurement techniques which can be used for the detection of alpha emitting nuclides in the soil or on surfaces.

5.1.2 Measurement Techniques used for Beta Surveys

Table 5.2 gives an overview of measurement techniques which can be used for the detection of beta emitting nuclides in the soil or on surfaces.

5.1.3 Measurement Techniques used for Gamma Surveys

Table 5.3 gives an overview of measurement techniques which can be used for the detection of beta emitting nuclides in the soil or on surfaces.

Table 5.1: **Radiation Detectors with Applications to Alpha Surveys (from [MAR 02])**

Detector Type	Detector Description	Application	Remarks
Gas Proportional	<1 mg/cm² window; probe area 50 to 1000 cm² <0.1 mg/cm² window; probe area 10 to 20 cm²	Surface scanning; surface contamination measurement Laboratory measurement of water, air, and smear samples	Requires a supply of appropriate fill gas
	No window (internal proportional)	Laboratory measurement of water, air, and smear samples	
Air Proportional	<1 mg/cm² window; probe area - 50 cm²	Useful in low humidity conditions	
Scintillation	ZnS(Ag) scintillator; probe area 50 to 100 cm²	Surface contamination measurements, smears	
	ZnS(Ag) scintillator; probe area 10 to 20 cm²	Laboratory measurement of water, air, and smear samples	
	Liquid scintillation cocktail containing sample	Laboratory analysis, spectrometry capabilities	
Solid State	Silicon surface barrier detector	Laboratory analysis by alpha spectrometry	
Passive, integrating electret ion chamber	<0.8 mg/cm² window, also window-less, window area 50-180 cm², chamber volume 50-1,000 ml	Contamination on surfaces, in pipes and in soils	Useable in high humidity and temperature

Table 5.2: **Radiation Detectors with applications to beta surveys (from [MAR 02])**

Detector Type	Detector Description	Application	Remarks
Gas Proportional	<1 mg/cm² window; probe area 50 to 1,000 cm²	Surface scanning; surface contamination measurement	Requires a supply of appropriate fill gas
	<0.1 mg/cm² window; probe area 10 to 20 cm² No window (internal proportional)	Laboratory measurement of water, air, smear, and other samples Laboratory measurement of water, air, smear, and other samples	Can be used for measuring very low-energy betas
Ionization (non-pressurized)	1-7 mg/cm² window	Contamination measurements; skin dose rate estimates	
Geiger-Müller	<2 mg/cm² window; probe area 10 to 100 cm²	Surface scanning; contamination measurements; laboratory analyses	
	Various window thickness; few cm² probe face	Special scanning applications	
Scintillation	Liquid scintillation cocktail containing sample	Laboratory analysis; spectrometry capabilities	
	Plastic scintillator	Contamination measurements	
Passive, integra-ting electret ion chamber	7 mg/cm² window, also window-less, window area 50-180 cm², chamber volume 50-1,000 ml	Low energy beta including H-3 contamination on surfaces and in pipes	Useable in high humidity and temperature

Table 5.3: **Radiation Detectors with Applications to Gamma Surveys (from [MAR 02])**

Detector Type	Detector Description	Application	Remarks
Gas Ionization	Pressurized ionization chamber; Non-pressurized ionization chamber	Exposure rate measurements	
Geiger-Mueller	Pancake (<2 mg/cm² window) or side window (~30 mg/cm²)	Surface scanning; exposure rate correlation (side window in closed position)	Low relative sensitivity to gamma radiation
Scintillation	NaI(Tl) scintillator; up to 5 cm by 5 cm	Surface scanning; exposure rate correlation	High sensitivity; Cross cali-brate with PIC (or equivalent) or for specific site gamma energy mixture for exposure rate measurements.
	NaI(Tl) scintillator; large volume and „well“ configurations	Laboratory gamma spectrometry	
	CsI or NaI(Tl) scintillator; thin crystal	Scanning; low-energy gamma and x-rays	Detection of low-energy radiation
	Organic tissue equivalent (plastics)	Dose equivalent rate measurements	
Solid State (in situ gamma spectrometer)	Germanium semiconductor	Laboratory and field gamma spectrometry and spectroscopy	
Passive, integrating electret ion chamber	7 mg/cm² window, also window-less, window area 50-180 cm², chamber volume 50-1,000 ml		Useable in high humidity and temperature

5.1.4 Detection Limits for Alpha and Beta Survey Instruments

Table 5.4 provides some indication of the detection limit for various instruments for alpha and beta surveys for static one minute counts on soil contaminated with U 238.

Table 5.4: **Examples of Estimated Detection Sensitivities for Alpha and Beta Survey Instrumentation (Static one minute counts for U 238, calculated; from [MAR 02])**

Detector	Probe area (cm^2)	Background (cpm)	Efficiency (cpm/dpm)	Detection Limit (Bq/m^2)
Alpha proportional	50	1	0.15	150
Alpha proportional	100	1	0.15	83
Alpha proportional	600	5	0.15	25
Alpha scintillation	50	1	0.15	150
Beta proportional	100	300	0.20	700
Beta proportional	600	1500	0.20	250
Beta GM pancake	15	40	0.20	1800

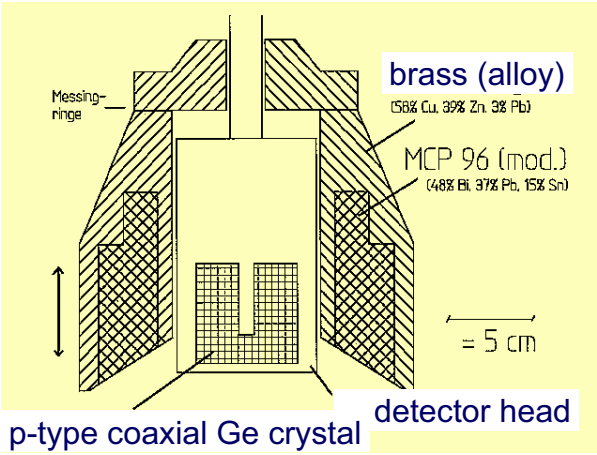
5.2 *In situ* Gamma Spectrometry

5.2.1 General Description

The *in situ* gamma spectrometry is usually applied for measurement from which the dose commitment to the general public is calculated. This non-destructive measurement technique discriminates the energies of the gamma radiation and can therefore be used for identification of the radionuclides present. It can also be used for detecting radioactivity beneath shielding layers.
The *in situ* gamma spectrometry can be applied in the usual uncollimated way where no spatial resolution of the activity distribution is possible and only the integral activity value can be calculated, and in a collimated way where suitable shielding screens out radiation outside the opening angle of the collimator.

The collimated in situ gamma spectrometry or the collimated gamma scanning (this term is used for devices which can be easily moved for scanning large surfaces) is based on the use of suitable collimators made from lead or a suitable brass alloy. This collimator is depicted in Figure 5.1. It focuses the area of measurement to e.g. 1 m^2, depending on the opening angle and the height in which the spectrometer is placed above ground.

Figure 5.1: **Detector and collimator used for collimated in situ gamma spectrometry (from [TÜV 95]**



The collimated in situ gamma spectrometry is suitable for rapid measurements on larger areas - stochastically distributed measurements on a grid or measurements covering the entire area. It can also detect that part of the activity which has penetrated into the ground. This is particularly useful for areas where an uncontaminated new layer of soil or concrete has been added after an initial contamination. The maximum depth of detection depends on the radionuclide (the gamma energies) and may reach several 10 cm.

Nowadays, commercial systems are available which work on the basis of collimated in situ gamma spectrometry with sodium iodide (NaI) or germanium (Ge) detectors. Those systems have been widely applied for release measurements of buildings and sites at many decommissioning projects.

5.2.2 *In situ* Gamma Spectrometry with NaI Detector

The advantages of NaI detectors consist of the high efficiency and the rather low dependence of the efficiency on the position of the source and the photon energy. That means that short measurement times (a few minutes) yield good measurement statistics. Furthermore, the instruments are robust and require only a small effort for maintenance.

A severe disadvantage, however, is the low energy resolution. As a consequence, it is impossible to perform a depth correction by measuring the peak ratios for Am 241 or other nuclides in the low energy range.

5.2.2.1 Calibration of a NaI(Tl) Detector

The calibration of measurement systems with NaI detectors requires more effort because of the rather low energy resolution of the NaI crystal compared to the Ge detector. For each standard geometry which is to be used in measurements, the calibration comprises determination of the efficiency as well as of the stripping factors which describe the influence of other gamma lines on the energy window in question. The corresponding activities are calculated with a matrix method. This requires the choice of fixed energy windows.

Such a calibration procedure is described here using the example of the NaI system GeoSCAN [GEN 98]. The energy stabilisation is performed using a Ba 133 source. Eight fixed energy

windows are defined for evaluation of the spectra (Table 5.). The standard calibration is performed with reference sources consisting of concrete cylinders of 0.7 m diameter and 0.32 m height which are contaminated with fixed amounts of K 40, Ra 226 and Th 232 as well as with surface samples of 0.7 m diameter contaminated with Co 60 and Cs 137. The background countrate is determined from similar concrete samples that do not contain any added radioactivity. From these measurements, calibration factors specific for the GeoSCAN system are calculated.

Table 5.5: **Energy windows of the measurement system GeoSCAN with collimated in situ gamma spectrometry using a NaI detector**

Window name	Nuclide	Energy [keV]
BA	Ba 133	356
RA1	Bi 214 (Ra 226)	609
CS	Cs 137	662
TH1	Tl 208 (Th 232)	911
CO	Co 60	1173 and 1333
K	K 40	1461
RA2	Bi 214 (Ra 226)	1765
TH2	Tl 208 (Th 232)	2615

Particular attention must be paid to the width of the energy windows. If these windows are chosen too narrow, the energy drift will reduce the counting statistics, yielding low count rates compared to broader energy windows. Using broad energy windows, on the other hand, will increase the count rates, but it will also increase the danger that gamma lines from adjacent windows affect the results. The optimum choice of the energy windows can be determined from calibration spectra.

5.2.2.2 Calculation of Mass or Surface Specific Activities

The net count rate is calculated from the gross count rate by subtracting the background count rate and any interferences from adjacent energy windows. This latter effect is calculated using the stripping factors which have been derived from the calibration.

The net count rate is proportional to the activity of the radionuclide. The mass or surface specific activities of a particular nuclide are determined by multiplication of the net count rate with the appropriate calibration factor. Any shielding layer in the case of activity penetration is taken into account by using a correction factor which is determined from measurements at the calibration samples with similar shielding.

5.2.2.3 Detection limits

Table 5.6 provides an overview of detection limits for scanning measurements for common radionuclides and radioactive materials in soil. It is important to note that the variables used in the above examples to determine the detection limits for scanning for the 1.25 inch by 1.5 inch NaI(Tl) scintillation detector — i.e. the $MDCR_{surveyor}$ detector parameters (e.g. cpm per µR/h), and the characteristics of the area of elevated activity — have all been held constant to facilitate the calculation of scan MDCs provided in Table 5.6. The benefit of this approach is that generally applicable scan MDCs are provided for different radioactive contaminants. Additionally, the relative detectability of different contaminants is evident because the only variable in Table 5.6 is the nature of the contaminant.

Table 5.6: **Minimum detectable concentrations in soil for *in situ* gamma spectrometers with NaI detector (from [MAR 02])**

Radionuclide/Radioactive Material	1.25 in. by 1.5 in. NaI Detector		2 in. by 2 in. NaI Detector	
	Scan MDC (Bq/kg)	Weighted cpm/µR/h	Scan MDC (Bq/kg)	Weighted cpm/µR/h
Am-241	1,650	5,830	1,170	13,000
Co-60	215	160	126	430
Cs-137	385	350	237	900
Th-230	111,000	4,300	78,400	9,580
Ra-226 (in equilibrium with progeny)	167	300	104	760
Th-232 decay series (Sum of all radionuclides in he thorium decay series)	1,050	340	677	830
Th-232 (In equilibrium with progeny in decay series)	104	340	66.6	830
Depleted Uranium[b] (0.34% U-235)	2,980	1,680	2,070	3,790
Natural Uranium[b]	4,260	1,770	2,960	3,990
3% Enriched Uranium[b]	5,070	2,010	3,540	4,520
20% Enriched Uranium[b]	5,620	2,210	3,960	4,940
50% Enriched Uranium[b]	6,220	2,240	4,370	5,010
75% Enriched Uranium[b]	6,960	2,250	4,880	5,030

Table 5.7 shows typical detection limits for the measurement system GeoSCAN with collimated in situ gamma spectrometry. The calibration type „worst-case“ means that a point source is assumed to be present at 5 mm depth in the far corner of the measurement area of 1 m x 1 m. The homogeneous calibration corresponds to a surface of similar area on which a homogeneously distributed activity is present.

Table 5.7: **Detection limits for the measurements system GeoSCAN**

Geometry	Co 60	Cs 137
calibration worst-case collimator with 90° opening angle, meas. time 10 min	0.33 Bq/cm²	0.61 Bq/cm²
calibration: homogeneous activity distribution collimator with 90° opening angle, meas. time 10 min	0.41 Bq/cm²	0.64 Bq/cm²

5.2.3 *In situ* Gamma Spectrometry with Ge Detector

The main advantages of measurements with Ge detectors with a crystal of high purity are the very good energy resolution and the high information contents in the spectra. It is possible to distinguish the gamma lines of natural radionuclides from those of anthropogenic nuclides. In comparison to NaI detectors, measurements with HPGe detectors allow to establish depth profiles of Am 241 in soil or concrete using a comparison of the peak areas of the Am 241 peaks.

A commonly used measurement system with collimated in situ gamma spectrometry and HPGe detector is the ISOCS (**I**n **S**itu **O**bject **C**ounting **S**ystem). The detector efficiency usually is the range of 40 %.

Figure 5.2: **The ISOCS system in preparation for use (left) and in use at the fan building of the Brookhaven Graphite Research Reactor**

5.2.3.1 Calibration of a HPGe Detector (ISOCS)

The calibration of the ISOCS measurement system is done using a mathematical model which simulates measurement geometries and conditions. This model calculates the detector efficiency for the specific geometry and conditions using a Monte Carlo approach.

5.2.3.2 Detection Limits

Table 5.8 gives an overview of detection limits of the ISOCS system for various common measurement geometries. The detection limits may be as low as a few mBq/g for Co 60 and Cs 137 with reasonable measurement times of about 15 min.

Table 5.8: **Detection limits for the ISOCS system for some common geometries**

Geometry	Co 60	Cs 137
Calibration worst-case, 90° collimator, measurement time 5 min	0.29 Bq/cm² *	0.24 Bq/cm² *
Homogeneous activity distribution, 90° collimator, meas. time 5 min	0.06 Bq/cm² *	0.08 Bq/cm² *
Detector 1 m above ground, homogeneous distribution, meas. time 15 min	0.8 Bq/kg **	1.1 Bq/kg **
Wall of a room 3 · 3 m², distance wall-detector 2.5 m, meas. time 60 min	0.031 Bq/cm² **	0.035 Bq/cm² **
*: detection limit as calculated by the system; **: data of the manufacturer		

The detection limits for other in situ spectrometers with Ge detector are summarized in Table 5.9. A comparison with Table 5.8 shows agreement between both sets of values.

Table 5.9: **Detection limits for *in situ* spectrometers with Ge detector for soil and other materials, measuring time 10 min, normal background dose rate (80 nSv/h) [TÜV 02]**

Nuclide	uncollimated	collimated	
	concrete, activity only in the uppermost cm	concrete, asphalt surface contamination only	soil homogeneous depth distribution
Cs 137	0.008 Bq/cm²	0.012 Bq/cm²	0.0013 Bq/g
Co 60	0.004 Bq/cm²	0.008 Bq/cm²	0.0008 Bq/g
Ba 133	0.008 Bq/cm²	0.021 Bq/cm²	0.0026 Bq/g
U 235	0.015 Bq/cm²	0.024 Bq/cm²	0.0045 Bq/g

5.3 Comparison of Collimated *In Situ* Gamma Spectrometer and Other Procedures for Release/Clearance Measurements

5.3.1 Sampling and Laboratory Measurements

The comparison between ex situ measurements of samples with laboratory gamma spectrometry and measurements with collimated in situ gamma spectrometry needs to take into account the measurement results, their accuracy and the time required to perform the measurements. The time required for taking the samples, sample preparation for measurement including homogenization, and finally the measurement itself is substantial, so that the results of ex situ measurements will be available only a considerable time after sampling has taken place.

Measurements taken with *in situ* gamma spectrometry, on the other hand, are available instantaneously. The time required for setting up the measurement system (around 30 min) and for taking each measurement (around 5 min for HPGe systems and 10 min for NaI systems) is rather small. In addition, no preparation of the object to be measured is necessary (or even allowed) so that areas can be used again immediately after release/clearance.

Sets of measurements from the same area which have been acquired from laboratory measurements and from collimated in situ gamma spectrometry reveal considerable differences between both measurement methods [HUM 97], [GEN 98]. This may be due to differences in the real contamination depth, the depth to which the result of the *in situ* gamma spectrometry is related, and the depth down to which samples have been taken. The collimated *in situ* gamma

spectrometry can always be calibrated in such a way that the activity which is really present is conservatively measured, i.e. will not be underestimated.

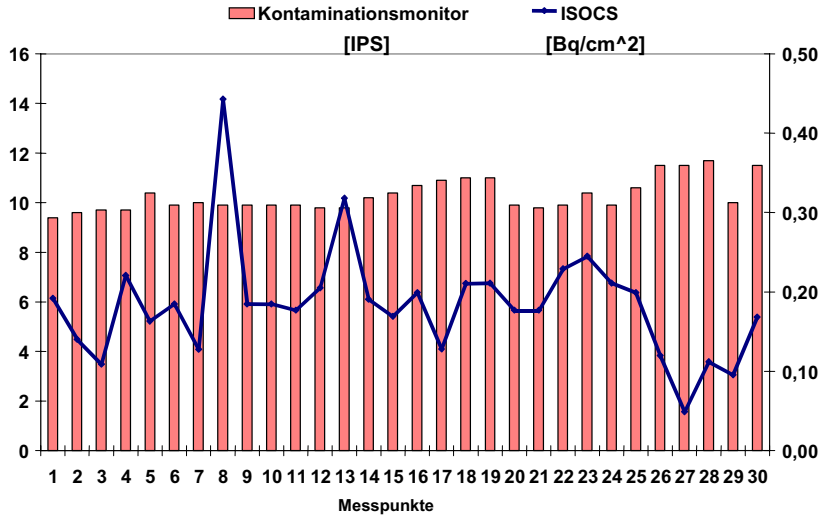
5.3.2 Contamination Monitors

The comparison between measurements taken with ζ/η contamination monitors and collimated in situ gamma spectrometry [HUM 97] [GEN 98] reveal that those nuclides which are not quite near to the surface will not be detectable with a contamination monitor. In the study [GEN 98], measurements with both methods were conducted on a larger area (around 30 m²). The contamination monitor which was calibrated to Co 60 and used background subtraction showed 0.0 Bq/cm² on the entire surface. The in situ gamma spectrometer measured Co 60 activities up to 0.2 Bq/cm² using a calibration for homogeneous activity distribution.

This example clearly demonstrates that in many cases η contamination cannot be correctly measured with contamination monitors. Vegetation on the soil or some re-distribution of the contaminated parts of the soil by wind may significantly disturb the measurements. In addition, deeper penetration of the radionuclides into the soil with rain water will also lead to errors.

Furthermore, the K 40 content in soil and especially in concrete may prevent the measurement of anthropogenic nuclides with a contamination monitor. The example shown in Figure 5.3 [GEN 98] reveals that the in situ gamma spectrometry measurements are able to resolve the variations of the Co 60 activity while the measurements taken with a contamination monitor are dominated by the natural activity and cosmic background radiation showing no variation which could be attributable to Co 60.

Figure 5.3: **Comparison of measurement results obtained with a contamination measurement device with results obtained with the ISOCS measurement system for Co 60 (from [GEN 98])**



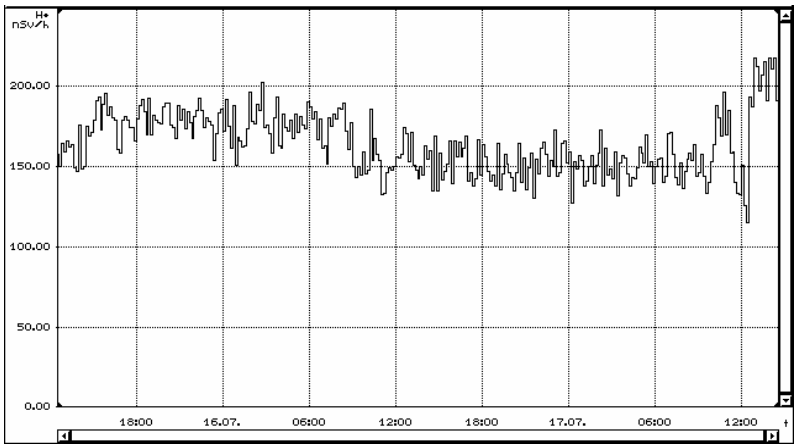
5.3.3 Dose Rate Measurements

The variability of the dose rate at a single place is illustrated by dose rate measurements which have been reported in the study [GEN 98]. Those measurements have been conducted using a dose rate meter GammaTRACER of Genitron Instruments which possesses two Geiger Müller counters

with a measurement range of 20 nSv/h to 10 mSv/h. Figure 5.4 presents the results as a time series of measurements taken every 10 min. The mean value of the dose rate was 163 nSv/h with a minimum of 114 nSv/h and a maximum of 215 nSv/h. The 2ω interval of the measurement results was 10 %.

The measurements show a strong variation of the dose rate at a single spot. This variability makes it clear that measurements which rely on the dose rate alone could not be used for release or clearance of a nuclear site.

Figure 5.4: **Timeline of dose rate measurements [nSv/h] obtained with the GammaTRACER device (from [GEN 98])**



(*the dose rate increase at the right end is caused by moving a waste drum near the probe)*

5.3.4 Variation of the Natural Activity in Concrete

In addition to the factors listed in the previous sections, variations of the elemental and therefore the radionuclide composition of the natural activity in concrete and soil disturb all non-spectroscopic measurement techniques. This effect obscures any effects from the contamination itself. As an example, Figure 5.5 and Figure 5.6 show the distribution of count rates at a floor and at a wall, both totally uncontaminated, measured with a contamination monitor.

The floor area was around 13 m². As shown in Figure 5.5, the measurements varied between 10.2 and 15.4 ips, the difference corresponding to a (hypothetical) Co 60 activity of 0.31 Bq/cm². Similar results have been obtained for a wall (area around 3 m²) where the count rate varied between 12.2 and 18.4 ips (Figure 5.6). This difference would correspond to a (hypothetical) Co 60 activity of 0.36 Bq/cm².

Figure 5.5: **Distribution of count rates from a floor measurement taken with a contamination monitor (from [GEN 98])**

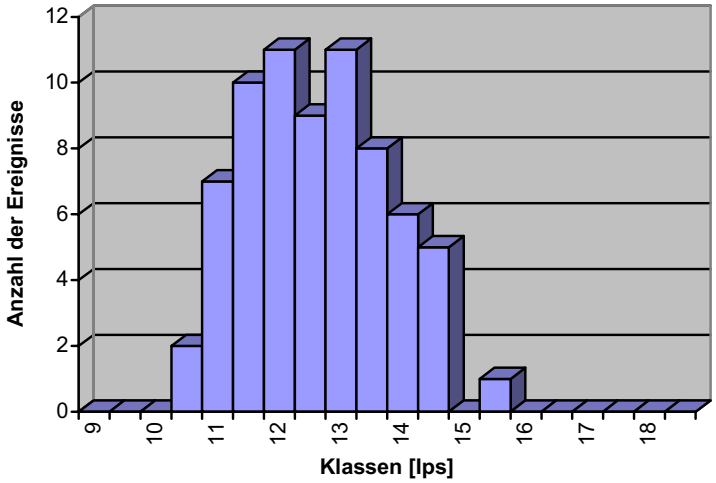
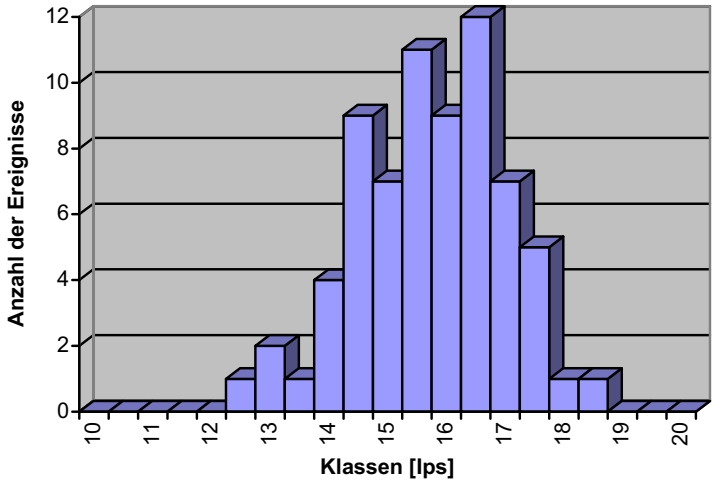


Figure 5.6: **Distribution of count rates from measurements at a concrete wall taken with a contamination monitor (from [GEN 98])**



If in addition the contamination monitor subtracts a fixed background value (as is usually the case), systematic errors cannot be avoided. Therefore, a release measurement of a surface with a limit value for Co 60 of, say, 0.5 Bq/cm² and a background variation corresponding to a Co 60 activity of 0.36 Bq/cm² is almost impossible.

5.4 Comparison of the Measurement Techniques

The discussion so far has shown that suitable measurement techniques for release/clearance measurements of soil and building surfaces are available. Table 5.10 gives a general overview of these techniques and the tasks for which they are best applied. The figures for detection limits and

costs are given only for orientation purposes and may vary significantly between measurement tasks and between countries.

Table 5.10: **Overview of measurement techniques suitable for release/clearance measurements for soil and building surfaces**

Area	Technique	Preferential use	Detection limits	Costs
uncovered soil	collimated in situ gamma spectrometry with NaI	for radiological characterization and release measurements; percentage of ν activity must be high enough	0.3 Bq/cm² Co 60, 0.6 Bq/cm² Cs 137	50 € / measurement point
	collimated in situ gamma spectrometry with HPGe	very suitable for any radiological characterization and release measurements; percentage of ν activity must be high enough	3 mBq/g Co 60 4 mBq/g Cs 137 (homogeneous activity distribution)	80 € / measurement point
	collimated in situ gamma spectrometry with HPGe	auxiliary technique for the ν component of the nuclide vector, not suitable for quantitative results	1 mBq/g Co 60 1 mBq/g Cs 137	120 € / measurement point
	sampling and ex situ gamma spectrometry	especially for nuclide vectors without sufficient gamma component; high costs	0.5 Bq/kg Co 60 0.5 Bq/kg Cs 137	120 € per nuclide; costs for sampling not included
	contamination monitor	for radiological characterization; less suitable for clearance/release measurements	0.1 Bq/cm² Co-60	10 € / measurement
	Dose rate measurements	auxiliary technique for radiological characterization; not suitable for clearance/release measurements	> 50 nSv/h	10 € / measurement
covered areas, concrete surfaces (streets, parking places etc.)	collimated in situ gamma spectrometry with NaI	for radiological characterization and release measurements; percentage of ν activity must be high enough	0.3 Bq/cm² Co 60, 0.6 Bq/cm² Cs 137	50 € / measurement point
	collimated in situ gamma spectrometry with HPGe	very suitable for any radiological characterization and release measurements; percentage of ν activity must be high enough	0.06 Bq/cm² Co 60, 0.08 Bq/cm² Cs 137 (homogeneous activity distribution)	80 € / measurement point
	collimated in situ gamma spectrometry with HPGe	auxiliary technique for the ν component of the nuclide vector, not suitable for quantitative results	0.02 Bq/cm² Co 60, 0.03 Bq/cm² Cs 137 (homogeneous activity distribution)	250,- DM pro Meßpunkt
	sampling and ex situ gamma spectrometry	less suitable for concrete surfaces, sampling requires high effort	0.5 Bq/kg Co 60 0.3 Bq/kg Cs 137 1 Bq/kg Am 241	120 € per nuclide; costs for sampling not included
	contamination monitor	for radiological characterization; also for clearance/release measurements if natural activities low enough and limit values sufficiently high	0.2 Bq/cm² Co 60, 0.1 Bq/cm² Cs 137 < 0.5 Bq/cm² for other nuclides	10 € / measurement
	Dose rate measurements	auxiliary technique for radiological characterization; suitable for clearance/release measurements in special cases	> 50 nSv/h	10 € / measurement

5.5 Release Measurement Facilities (RMF)

5.5.1 General Description

RMFs are suitable for clearance measurements of material which is placed in boxes or drums. The use of a large number of detectors (e.g. 24 in total, four detectors on each of the six sides - floor, ceiling, both side walls and doors) allow for a 4ϕ geometry. RMFs have a high throughput (up to a few Mg per hour) combined with short measurements times (e.g. one minute per box filled with several 100 kg of material). Figure 5.7 shows an example of such an RMF.

Figure 5.7: **Example of release measurement facility: Front part: conveyor with material to be measured (box), background: measurement chamber with detectors (floor, ceiling, both side walls and doors) surrounding the material**

Measurements with RMFs are gross gamma measurements, i.e. it is necessary that the nuclide vector contains a suitably high percentage of gamma emitting nuclides. Furthermore, the measurement does not discriminate the gamma energies. Therefore, a careful calibration for each nuclide vector is necessary. Usually, the nuclides Co 60 and Cs 137 are taken as key nuclides to which the other radionuclides are correlated via correlation factors. The detection limit is in the range of 0.01 Bq/g if only artificial radionuclides are present. If, however, a significant amount of natural radionuclides are present, the detection limit increases accordingly. Concerning release/clearance measurements for soil, the variation of the contents of natural radionuclides must be kept in mind and must be included in the calibration.

The main use of RMFs for the release of sites is for procedures where the soil is excavated down to a certain depth. The soil is excavated in layers of a certain thickness (e.g. 5 – 10 cm) and the excavated soil is then measured in the RMF. Release/clearance levels for this soil have to be derived previously from the general release/clearance levels pertaining to the site. The procedure is repeated. If the excavated soil exceeds the clearance levels, it is removed and may be cleared for disposal on a landfill site or may be treated as radioactive waste, depending on its residual radioactivity. If, however, it complies with the release/clearance levels, it can be placed back onto the area from where it had been extracted and this particular area may then be considered free.

Such a procedure is particularly effective if real contamination has been found or is strongly suspected. It is not recommendable for the release of sites where no contamination is present or suspected and where the measurements only serve for verification. In such cases, the *in situ* gamma spectrometry is much more effective.

5.5.2 Application for Soil Measurements

The general applicability of RMFs for soil measurement is very good. The measurement time is reasonably fast and large quantities can be assessed at a time. Typical detection limits for RMFs are shown in Table 5.11.

Table 5.11: **Detection limits for a typical large RMF; for Co 60 and Cs 137, various measurement times and material quantities; for homogeneous activity distribution [RAD 03]**

Net mass [kg]	Nuclide	Detection limit [Bq] for measurement time		
		30 sec	60 sec	180 sec
100	Co 60	75	55	40
300	Co 60	100	70	50
600	Co 60	130	100	70
100	Cs 137	230	170	120
300	Cs 137	280	210	140
600	Cs 137	380	280	190

Another example of the application of a RMF (model RTM 642, RADOS) for the clearance of soil is described in the paper "Erfahrungen mit der Freigabe radioaktiver Materialien bei der Stilllegung in Rossendorf" by H.-D. Giera *et al*. (*Hamburg 2003*). This paper describes the clearance of soil from a site in the research centre VKTA (Verein für Kernverfahrenstechnik und Analytik Rossendorf e.V. - Nuclear Engineering and Analytics Rossendorf Inc.). It is suitable for measuring 500 l boxes as well as 200 l and 240 l drums. Calibration factors are available for the radionuclides Co 60, Cs 137, Eu 152 and Ba 133. The instrument has been calibrated so that it is independent of the material to be measured. This makes it possible to measure diverse materials like soil, building rubble, cables, tubes, steel or aluminium scrap or mixed material. The condition is that the boxes or drums are at least half filled and that the maximum mass does not exceed 1,000 kg. It is possible to measure up to 30 boxes or drums a day.

Because of the large quantities of soil to be measured and because of the high standards to which the radiological characterization is performed, a new measurement strategy was introduced for bulk material. Instead of measuring the entire material quantity, the RMF measurement were performed only at representative masses. This required an appropriate treatment of the material. The only calibration geometry which is reasonable for and applicable to bulk material is that for a homogeneous activity distribution. Tests were necessary to establish this procedure: homogenisation of a sample mass of 10 m^3 and subsequent RMF measurements clearly showed that the variation of the activity which is measured in the RMF is small (a few percent). The results of this test enabled the VKTA to establish a procedure where under normal circumstances only one box needs to be measured per 10 m^3 of homogenized material. If, however, there is reason to believe that the material will not comply with the clearance levels, the entire quantity is measured in the RMF.

Regarding the calibration of the RMF for those measurements, the very conservative point calibration is now changed to the more realistic volume calibration which more accurately describes the true activity contents in the boxes. It had to be checked carefully how much deviations from a homogeneous activity distribution in the (real) material would affect the measurement results. It was possible to establish a quantitative criterion for homogeneity by Monte Carlo calculations.

5.6 Other Measurement Techniques

5.6.1 Sampling and Laboratory Analysis

Activity measurements based on soil or concrete samples taken from the sites to be measured yield very accurate results with a low limit of detection and a small error range. However, each sample is only representative for a small part of the entire area to be measured. It is therefore difficult to achieve representative sampling for the entire site. Taking enough samples to characterize larger areas at a given confidence level (e.g. 95%) by sampling and laboratory analysis alone would therefore a high effort. In addition, soil samples may require significant preparation time (mainly for homogenization) as well as long measurement times. Therefore, sampling and laboratory analysis is mainly used to compare and support measurements taken with other techniques, especially at those spots which show some elevated level of contamination.

5.6.2 Contamination Detectors

Contamination monitors can be used for alpha and beta measurements. The devices are inexpensive and are widely available. Those devices can be used for measuring pure surface contaminations with a reasonably small limit of detection. As shown in Table 5.10, detection limits of 0.1 Bq/cm² can be achieved if the contamination is situated only on the surface. However, the distance between the window of the instrument and the surface must be rather small (not more than a few centimetres). This technique can therefore be used only on paved surfaces while it is unsuitable for normal soil.

The main disadvantage of this technique is its inability to detect activity in deeper layers of soil. A penetration depth of only a few millimetres leads to a decrease in the efficiency to around 10%. If contamination is present on nuclear sites, it will not be limited to such a thin layer. A possible solution is to determine the ratio between the purely surficial part of the activity and the entire activity (including the part which has penetrated to a larger depth). Depending on the characteristics of the site, this may allow a (very conservative) estimate of the total activity from the surficial activity.

Release of clearance of an entire site therefore cannot be achieved using only contamination monitors. This technique may only be usable if additional information from a characterization of the site are present.

5.7 Choice of Sampling/Measurement Points and Density

It is generally not necessary to perform measurements or a sampling programme which cover the entire area of the site to be released. Measurements or sampling will rather be performed on specific spots the density and position of which have been determined by stochastic methods. Details on methods how the density can be derived to comply with stochastic requirements and how the pattern of measurement / sampling positions can be established are described in [MAR 02]. An example for a random pattern of measurement points is shown in Figure 5.8 which is taken from [MAR 02].

Another example for a measurement pattern is shown in Figure 5.9. Here, the task is to place a given number of measurements per grid cell onto the entire grid of the area to be measured. The rectangular grid may have cells of 10 x 10 m² while each measurement which is carried out with collimated in situ gamma spectrometry covers an area of 1 m². This area may be a class II area (cf. section 4.2) so that the measurement density may have been determined to 5%, i.e.

5 measurements of 1 m² per 100 m² grid cell. The measurements may then be distributed as shown in the figure.

Figure 5.8: **Example of a Random Measurement Pattern (from [MAR 02])**

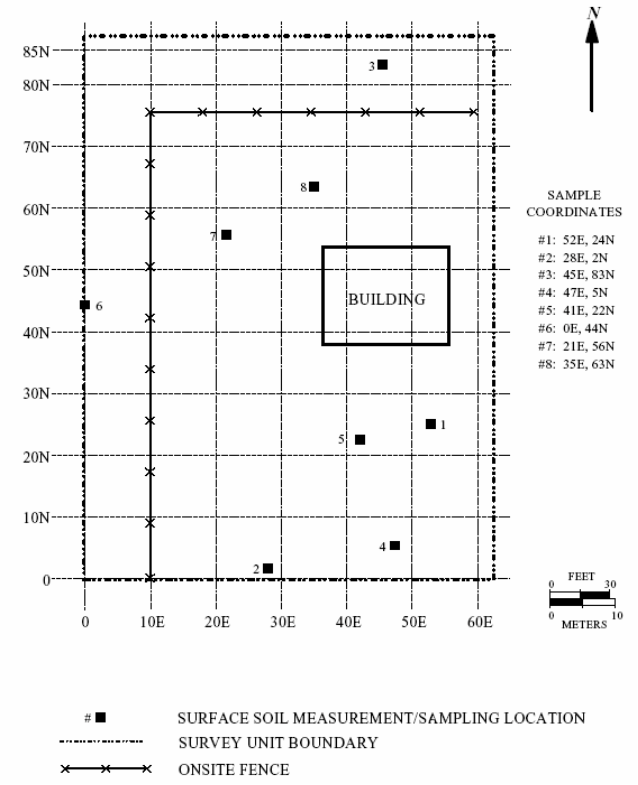
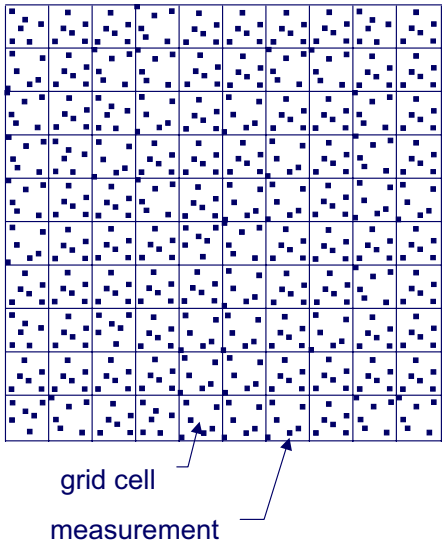


Figure 5.9: **Example of a measurement pattern with a given number of measurements per grid cell (from [DIN 25457])**



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Annex II

THE QUESTIONNAIRES

This Annex first presents the Questionnaire as it was distributed to selected OECD Member Countries and then circulated among relevant decommissioning projects. It then gives the answers of the various participating projects and/or countries in full text without the introductory parts of the questionnaire.

1. THE QUESTIONNAIRE AS CIRCULATED

Questionnaire: Clearance of Sites of Nuclear Installations in OECD Countries
- 30 May 2003 –

An important number of nuclear installations is presently in the decommissioning phase, and this number will be increasing with time. This questionnaire deals with the release of nuclear sites from regulatory control. Intention is to collate the information that is received and to discuss it within the WPDD Task Group on Release and Reuse of Materials, Buildings, and Sites. The raw information will also be tabulated before compilation and distributed to all respondents to the questionnaire.

BACKGROUND

The background for this questionnaire can be summarized as follows:

- The decommissioning of nuclear installations, mainly nuclear power plants (NPP), in OECD countries is progressing. A number of smaller and a few larger plants have already been completely dismantled and the site has been released from regulatory control, e.g. for reuse for other purposes. This means that there is considerable experience available on the release of nuclear sites that could now be summarized and evaluated for the benefit of future projects.
- Although the release of a site is the very last step in de-regulating nuclear installations, it should be carefully planned and integrated in the overall waste management strategy from the start: Actual radionuclide-specific levels for the release of a site determine the percentage of soil which will have to be removed and will therefore also impact the final costs.
- The procedures used for the release of a site vary from country to country and they may also be site specific. Furthermore, they may change with time[2]. Therefore, it is relevant to gain an overview of the current procedures.
- Radiological scenarios establish a link between the maximum permissible activity in the soil of a site and the dose which may eventually result to members of the public. Release levels can be "calculated" via release scenarios and radionuclide-specific release levels can be derived on the basis of applicable dose "limits". In order to assess the meaning of release levels for sites that may be used in a country on a generic basis or may be applied to a specific project, it is

2. In Germany, for example, quantitative, radionuclide-specific levels were introduced by the new Radiation Protection Ordinance that entered into force in 2001. The release of the KKN (NPP Niederaichbach) site from regulatory control in 1994/95 was made on the basis of a site-specific assessment, a procedure that is different from the generic one given in the new Radiation Protection Ordinance, which has been applied for parts of the KGR (NPP Greifswald) site in 2001/02.

important to know which dose "limits" and which scenarios have been used for the derivation of release levels for nuclear sites.

- Furthermore, such release levels have to be verified by measurements and/or by (additional) evaluation of the contamination history. Measurements do not necessarily have to cover 100% of the surface of the site or the volume of the relevant soil. Evaluations might be performed on the basis of a statistically determined grid of measurements. Averaging criteria (mass, surface) may be used.

PRACTICAL INFORMATION

The questionnaire has been tested, and it should take no more than a day to fill.
We ask you to provide your answers to the Secretariat by 31 July 2003.
Please e-mail your answers to: suzanna.grant@oecd.org ; s.thierfeldt@brenk.com

QUESTIONNAIRE

We ask for responses to the following questions. If not all questions can be answered, answers to a subset of questions could also be of great help.

Description of the nuclear facility

Question 1: Please describe the nuclear installation shortly – name etc. If more than one installation is described, please complete one form for each installation. Identify whether this decommissioning project is part of a larger decommissioning programme for which uniform standards are/will be applied.)

Question 2: What is the size of the site which was or will be released? Is this the entire site, or just part of it?

Site Contamination

Question 3: Is the radionuclide vector/radiological fingerprint representative for the site known? Is there a (standard) method to derive the radionuclide vector/radiological fingerprint from the measurements?

(Please give the list of relevant radionuclides such as Co 60, Cs 137, Sr 90 ...)

Question 4: In the case of a contamination of the site, is it known how deep the radionuclides may have penetrated into the ground?

Regulatory Standards

Question 5: On which basis has been or will the site be released?

(Please address the following points:

- describe the legal basis, for example, does the regulatory framework provide generic or site specific regulations,
- indicate the date when this regulation entered into force,

- indicate if radiological studies have been carried out which have served as a basis for deriving release levels.)

Question 6: Please give examples of release levels/clearance levels of the relevant nuclides which will be used / have been used for the release of (a) site(s).

Question 7: Are there special features of the site which have influenced the derivation of suitable release criteria or the release procedure itself?

(Such special features could be vicinity to dwellings, a nature reserve, a water body from which drinking water is taken – or the opposite: site surrounded by uninhabited land, no farming etc.)

Implementation

Question 8: Which measurement techniques will be/have been applied for clearance measurements?

(Please indicate the relevant techniques, such as in situ gamma spectroscopy, soil excavation and bulk measurement etc.)

Question 9: What averaging criteria will be / have been used for the release of (a) site(s)?

(Please indicate averaging masses and/or surfaces, e.g. 100 m^2)

Additional references to open literature or the enclosure of internal information would be highly appreciated.

ANSWERS FROM CANADA

Description of the nuclear facility

Question 1: In Canada, a nuclear facility is defined under the Nuclear Safety and Control Act and includes a nuclear reactor, particle accelerator, uranium mine or mill as well as processing plants for the manufacture of products from uranium and other fuel materials. The definition also includes facilities for the disposal of nuclear substances which are generated at another nuclear facility. Nuclear facilities are further identified as either Class I facilities, Class II facilities and uranium mines/mills. Class I nuclear facilities include power and research reactors, uranium processing facilities and waste disposal facilities. Class II nuclear facilities are typically accelerators and irradiators. Each type of facility (Class I, Class II or uranium mine/mill) is covered by a separate set of regulations.

There are four generic types of nuclear facilities in Canada that would have sites that would be considered for release following the completion of decommissioning activities. These include uranium mines/mills, small research reactors, large power reactors and nuclear research establishments.

In Canada, uranium mines are divided into two classes, the currently operating mines and mills and the non-operating, historic mines. Historic mines are those which were closed prior to the 1970's and operated under a different set of standards and management practices. The CNSC is currently evaluating the historic mines to determine what additional licensing and regulatory controls are required for these sites.

Operating uranium mines and mills can encompass significant land areas, ranging about 3 to 5 square kilometres, particularly for the mine and the associated waste rock piles. All currently operating uranium mining facilities are located away from urban areas although some facilities in the past have had urban facilities located in close proximity to support the worker population. While the mill itself will occupy a smaller tract of land, the tailings ponds associated with its operation may occupy large sections of land and the tailings may be deeply distributed in that area. This includes deposition of tailings in natural features like valleys or in artificial features like mined-out open pit mines. It should also be noted that the licensed area may include large tracts of land that are not affected by the mining or milling operations. The unaffected lands could be available for release, through established regulatory procedures, following termination of the operations of the facility.

Small nuclear research reactors are generally located within research institutions such as universities. These nuclear facilities generally occupy a few rooms of a building that may also house other types of non-nuclear uses. One exception is a larger research reactor which is located in a purpose-built building. These facilities range in size from a few hundred square meters in size up to approximately 1000 square meters. The decommissioning plans for the smaller of these facilities generally call for the nuclear-related components to be removed and the rooms released for unrestricted use by the institution.

In Canada, most power reactors are located at multi- unit stations, either four or eight units per station. Two provinces, New Brunswick and Quebec, have single unit stations although the latter province has one additional power reactor that has been shut down and placed in a monitoring with surveillance phase. The Province of Ontario has three operating stations, an eight unit station at Pickering, an eight unit station at Bruce and a four unit station at

Darlington. Two older power reactors, one at Rolphton and one at Douglas Point (near Bruce) have been shut down and are currently in a monitoring with surveillance phase. Collectively the three shut down reactors are managed as nuclear waste facilities pending resumption of decommissioning activities following the monitoring with surveillance phase.

The power reactor sites tend to occupy large land areas, up to approximately 10 square kilometres for a large, multi- unit station and these sites may also have associated waste management facilities for the materials produced during operation and rehabilitation of the station equipment. This includes dry storage of irradiated nuclear fuel as well as low and intermediate level wastes. Each power reactor station is considered a separate nuclear facility as are the co- located waste management facilities. Decommissioning plans for such sites will include removal of the nuclear facilities and eventual release of the land, generally for restricted re- use at some future date. No power reactors have obtained a decommissioning licence in Canada at this time.

There are two nuclear research establishments located in Canada. The primary site is located in Ontario at Chalk River and the other site is located in the adjacent province of Manitoba at Whiteshell. Both facilities include one or more nuclear research reactors as well as other licensed nuclear facilities. The Chalk River site also includes an isotope production facility utilizing two purpose-built reactors. Each site also has a waste management area for storing wastes produced during the operation of the various facilities. At Chalk River, some waste from external producers of radioactive wastes are also accepted for storage while at Whiteshell only the waste from on-site activities is accepted for storage.

Both nuclear research establishment sites are approximately 40 square kilometres in size but that also includes a substantial portion of land that has been unaffected by operations and past practices. The licensee may propose to remove the unaffected lands from the licence at the start of decommissioning in order to reduce the licensed area. The release of those lands would have be in accordance with the practices and procedures as noted later in this questionnaire.

The Chalk River site is expected by the licensee to remain operational for the next 100 years, whereas the Whiteshell site has been recently licensed for decommissioning which is scheduled to occur over the next 60 years. The decommissioning of the Whiteshell site is projected by the licensee to occur in several phases as the separate nuclear facilities are gradually shut down, allowed to remain in a monitoring with surveillance phase and then finally decommissioned when suitable nuclear waste depositories are available.

Question 2: What is the size of the site which was or will be released? Is this the entire site, or just part of it?

As noted above, the following site sizes are applicable:

Uranium mines/mills 3 – 5 km²
Small research reactors 0.004 to 0.001 km² (400 – 1000 m²)
Power reactor sites 1 – 10 km²
Nuclear research test establishments 40 km²
It is possible for a licensee of a large nuclear facility site to apply for an amendment of the licence to remove sections of the listed facility from the licence and hence, from regulatory control.

Although it has not yet been requested by a licensee in Canada, the application would have to demonstrate, to the satisfaction of the Commission, that the lands to be released have been monitored and meet the relevant criteria for release.

In the case of some older mining sites, the land not directly involved with mining or milling operations (the 'unaffected land') may be considered for release provided the licensee can demonstrate that there has been no impact and this is verified by CNSC staff. A proposal is currently under consideration by CNSC staff for areas of land associated with an older mining site in south-eastern Ontario.

In the case of the decommissioning plan at Whiteshell, the licensee is currently intending to release all lands at approximately the same time, once all decommissioning activities have been completed. This will follow after the issuance of a licence to abandon the facility has been granted by the Commission and based on the final radiological monitoring report.

Site Contamination

Question 3: Is the radionuclide vector/radiological fingerprint representative for the site known?

For uranium mines and mills, the radiological inventory is known and includes contamination products from the uranium series in the waste rock and the tailings management area. Radium-226 may or may not have been preferentially removed as a result of leaching and this may alter the radiological composition of the resulting waste materials in the tailings management area.

In small research reactor facilities, the primary radionuclide composition will be fission products present as contamination while activation products, associated with the neutron flux of the reactor, may be found in the surrounding concrete and earthen structures. The inventory of the radionuclides found at the time of decommissioning will depend on the operating history of the reactor and the time period between last operation and the decommissioning as shorter- lived radio nuclides will be removed by decay. The exact composition of the radiological inventory is required as part of the detailed decommissioning plan for all components and structures in the facility at the time of decommissioning. Research reactors involved in isotope production may also have contamination associated with those processes as well and information on this inventory will also be required at the time of decommissioning.

Power reactors will also have similar radiological inventories although the total amounts will be larger due to the higher operating levels, neutron fluence and the operating history of the reactor. Again, it is expected that fission products will exist as contamination and activation products will be found in concrete and earthen structures of the facility. Of particular note to the CANDU stations (pressurized heavy water moderated, natural uranium reactors) used in Canada, tritium is produced in the moderator and primary coolant due to neutron activation of the deuterium. The tritium can become mobile within the reactor structure and result in localized contamination of the facility.

For nuclear research test establishments, the radiological inventory can be quite complex, based on current and past operations. Associated with the large research reactors will be the fission products and tritium in the form of contamination, as noted earlier. Activation products will be found located near the shielding and structures of the research reactors as well as around particle accelerators that may also be utilized at the site. Other radioisotopes may be created in the reactor for research, teaching or industrial purposes and these may be transported within the site to other facilities for use or study, thus increasing the number and type of locations where radiological material can be found during decommissioning. In addition, past experimentation with fuel processing and re-processing can result in areas with fission product as well as transuranic contamination.

The actual inventory and composition will depend heavily on the facility being decommissioned and the time of that decommissioning relative to the last operation of the facility.

Question 4: In the case of a contamination of the site, is it known how deep the radionuclides may have penetrated into the ground?

In most cases, for the facilities described above, the contamination profile is known at the time of decommissioning.

Historic uranium mines are an exception due to the different standards and management processes under which these facilities operated. In addition, it is not known whether any engineered barriers that may have been put into place at the time of operation are still intact. Therefore, for these facilities, the depth and extent of contamination is generally unknown at this time. However, the CNSC is taking steps to characterize the sites but this is proceeding slowly.

For operating uranium mines and mills, it is not expected that the radiological inventory of the tailings ponds will have migrated into the surrounding land. Current design procedures for uranium mines and mills are set to minimize this leaching and surface water from the site may have slightly elevated levels of some radionuclides but the surface water must meet provincial water quality objectives. At this time, there are no standards for groundwater.

In reactor facilities, both research and power, the contamination by tritium and fission products is generally on the surface only and readily removed with normal decontamination techniques. Activation products are generally located deeper in shielding materials due to the penetration by neutrons. The depth of penetration depends on the reactor power and design as well as the operating history. For research reactors used in Canada, experience has shown this depth of penetration to be small and the radiological inventory is small. The problem is more acute for power reactors and they generally employ long periods of storage with surveillance specifically to reduce the radiological inventory that must be managed during decommissioning.

The complex and varied nature of the nuclear research and test establishments provides for a number of potential ground contamination situations to exist. Some of these are due to past practices while others are the result of abnormal events or upset conditions. In most cases, the extent and radiological inventory of the ground contamination is known through sampling and analysis, primarily of groundwater. In some of these cases, the licensee has already implemented interception and mediation strategies aimed at preventing further release and ground contamination. For these types of facilities, it is vitally important that the licensee provide appropriate information on the radiological characterization of the property before a decommissioning licence is provided. This will work towards ensuring that all sources and areas of contamination have been properly identified and will document this work so that the appropriate remedial actions can be taken during the decommissioning activities.

Regulatory Standards

Question 5: On which basis has been or will the site be released?

In May 2000, the former Atomic Energy Control Board became the Canadian Nuclear Safety Commission (CNSC) with the proclamation of the *Nuclear Safety and Control Act* (NSC Act) and associated Regulations. This Act of the Canadian Government established the Commission, which is a quasi-judicial tribunal that regulates the use of nuclear energy

and materials to protect health, safety, security and the environment and to respect Canada's international commitments on the peaceful use of nuclear energy. Under Section 26(e) of the NSC Act, no person may decommission or abandon a nuclear facility, except in accordance with a licence issued by the Commission. The requirements for obtaining licences for these facilities are set out in the *General Nuclear Safety and Control Regulations* and there are additional requirements under the specific regulations for the type of facility (Class I, Class II or uranium mine/mill).

The current regulatory framework utilized in Canada does not provide generic criteria for release of nuclear facilities from licensing. Each nuclear facility licensee must apply for a licence to decommission a nuclear facility and the licensing process includes an environmental assessment carried out in accordance with the *Canadian Environmental* Assessment *Act (CEAA)* and its regulations. The Commission may not proceed to a licensing decision on a nuclear facility unless the requirements for an environmental assessment have been satisfied.

Following completion of decommissioning activities, a licensee may apply to the Commission for a Licence to Abandon the nuclear facility. The issuance of a Licence to Abandon demonstrates that the decommissioning has been successfully completed to the appropriate end-stage and that the facility is no longer under regulatory control by the CNSC. The licensee may continue to own and carry out activities on the site of the former nuclear facility for purposes that do not require a licence from the Commission. Any future re- use of the site for a nuclear facility would require a new licence from the Commission.

The regulatory framework does provide for generic assessments of contaminated land where a person or the Commission believes, on reasonable grounds, that such contamination may exist. This is covered under Sections 45 and 46(1) of the NSC Act and permits the Commission to conduct public hearings if contamination is suspected in any place. Under Section 24 of the *General Nuclear Safety and Control Regulation*s, the prescribed limit of radioactive nuclear substance that would permit such consideration is that quantity which may, based on the circumstances, increase a person’s effective dose by 1 mSv or more per year over the normal background radiation for the place. While these sections would not likely apply to currently licensed nuclear facilities, they could apply to former nuclear facilities that were abandoned prior to the legislation coming into force.

It should be noted that in order to avoid inconsistency in the application of regulation, it would be reasonable to assume that before issuing a licence to abandon, the Commission would wish to be satisfied that there would be very little likelihood of any person receiving an effective dose close to or above 1 mSv per year in the absence of any further regulatory control.

Uranium mines are different than other nuclear facilities in that the current policy in Canada does not generally provide for the affected area to be released for unrestricted use following decommissioning. The current policy is that the sites will be managed in the long-term under institutional control through continued licensing, usually by the federal or a provincial government agency. While there may be exceptions to this protocol, none has been proposed or accepted in Canada.

Uranium mines must meet several decommissioning criteria for worker and public dose limits, effluent release criteria and the stability of any remaining structures (e.g. dams, etc). Once these objectives have been met, the decommissioning activities are complete, the site is stable and no active treatment is required, it may be possible for the site to be turned over to a federal or provincial agency for the long-term institutional control.

While it may be possible for a site to be released from licensing in the future, the criteria for obtaining a licence to abandon for uranium mines has not been established at this time.

Question 6: Please give examples of release levels / clearance levels of the relevant nuclides which will be used/have been used for the release of (a) site(s).

During the decommissioning planning process, licensees have to develop criteria for the final end-state of the facility. This includes the final radiological inventory and is based largely on sampling and to a much smaller degree on inference. The criteria are also based on a maximum individual dose criterion of 50 μSv/annum. Licensees then propose to the CNSC the radiological limits that will apply at the facility following the completion of decommissioning.

For solid materials, CNSC staff has accepted the clearance levels provided in IAEA TECDOC-855 (Interim report for Comment) in applications, generally by small licensees. The TECDOC-855 levels are based on a generic pathways analysis leading to an individual dose criterion of 10 μSv/annum, which minimizes the likelihood of exceeding the 50 μSv/annum criteria. Licensees could apply to the Commission for unconditional release of a former nuclear facility based on the derived unconditional clearance levels that are provided in Table I of TECDOC-855. This approach has been used for a small research reactor in Canada.

Alternatively, licensees are permitted to derive their own clearance levels based on site-specific approach using an individual dose criterion of 10 μSv/ annum. This approach may be considered by a licensee where it can be demonstrated by the licensee that clearance levels higher than those provided in TECDOC-855 may be acceptable, based on a complete pathways analysis specific to the nuclear facility.

CNSC staff has been evaluating the values provided in the IAEA Draft Safety Report *Radioactivity in Material not requiring Regulation for Purposes of Radiation Protection* (DS161). These values, along with other information, are being considered by CNSC staff during the preparation of a Canadian regulatory standard on clearance and exemption levels which is currently is preparation.

Once accepted by the CNSC, the radiological limits proposed by a licensee are then incorporated into the decommissioning licence by reference. These limits then become legally binding on the licensee and the licensee must demonstrate that these levels have been met following completion of the site decommissioning.

Question 7: Are there special features of the site which have influenced the derivation of suitable release criteria or the release procedure itself?

The current Canadian policy on release of sites formerly used as nuclear facilities is that they must meet the site-specific limits that are set in the decommissioning licence. Licensees are required to include the limits for unconditional release of the site in their application for a decommissioning licence. By necessity, these criteria will be influenced by the site, including its location, use, proximity to potential recipients and the material used at the site. Remote mining sites may have different release criteria than a research reactor located in a municipal area. In all situations, protection of the public and the environment is the paramount concern and the potential dose through the evaluation of the pathways analysis must be less than 50 μSv per annum.

Implementation

Question 8: Which measurement techniques will be/have been applied for clearance measurements?

CNSC staff has accepted the use of the *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)* as an acceptable protocol for confirming the presence or absence of radioactive nuclear substances. This approach has been used in the evaluation of the end-state condition of a sub-critical assembly and a small research reactor.

For a large multi- nuclear facility site, CNSC staff has accepted a plan, proposed by the licensee, for multiple testing protocols to confirm the absence of radioactive materials on the site. The tests utilized included a full site aerial gamma survey and a ground-based gamma survey. The results from these surveys were then used to develop both biased and unbiased sampling campaigns for soil analysis. Biased samples were those taken at locations that were identified from the surveys while unbiased samples were taken at random locations. The soil sampling was carried out at these sites, with some samples retained for confirmatory monitoring by CNSC staff. The soil sampling campaign also included the provision for in-situ gamma monitoring and radionuclide specific measurements.

In the case of a uranium mine, the tailings areas are surveyed for gamma radiation as part of the decommissioning licence requirements. At this time, the criteria used are a gamma survey at one meter above the ground surface. The grid spacing is variable from 1 to 10 meters on centre and the spacing depends on the site conditions, including the survey results obtained. No single spot may exhibit a reading in excess of 1.5 μSv/hr and the average grid area survey may not exceed 1 μSv/hr. The objective is to achieve a facility that will result in an annual dose of less than 50 μSv to the most likely exposed individual.

Question 9: What averaging criteria will be / have been used for the release of (a) site(s)?

Sites may or may not retain some structures at the time of completion of decommissioning activities. This decision is determined at the time of application for a decommissioning licence where the licensee must describe the intended end-state of the facility as well as demonstrate that it meets the requirements of the landowner or agency who will take possession once the former facility has been released from regulatory control. Therefore the discussion on averaging criteria should include surveys of buildings and other artificial structures.

Generally for surface measurements, gamma radiation is measured on a grid system, using a spacing of between one and ten meters, at a height of one meter from the ground. The spacing of the grid is dependant on the terrain and other site conditions as well as the information gathered during the survey. If the field is found to be highly variable and dependant on location, then the grid spacing may need to be reduced accordingly. The average of the grid readings must not be greater than 1 μSv/hr and no single spot may exceed 1.5 μSv/hr.

Surface contamination measurements are normally taken at biased and unbiased locations and the results are averaged over the area of the wipe, normally taken to be 100 cm^2 . For structures on nuclear facility sites, the type of survey will depend on the nature, history and use of the facility. For smaller structures, a grid pattern of one to five meters is usually acceptable while larger facilities can have a larger spacing on the grid, depending on the variability of the readings.

In the case of volume contaminated material, the CNSC has accepted situations where the contamination is higher in a layer near the surface but where it is not feasible to remove it,

the entire volume of the structure or portion of the structure can be used in calculating the release concentration. For example, a large shielding block may only be contaminated in the first several centimetres from the surface but removal of this material may be impractical. Therefore, the CNSC may accept, depending on the circumstances, the averaging of the concentration over the entire block of concrete.

ANSWERS FROM GERMANY

KGR – NPP Greifswald (Germany)

Description of the nuclear facility

Question 1: The NPP site at Lubmin (near Greifswald) at the coast of the Baltic Sea was planned to consist of 8 NPP blocks of Russion WWER type (PWR) with 440 MWe each. 4 blocks had been in permanent operation (block 1: 1973 – 1990, block 2: 1974 –1990, block 3: 1978 – 1990, block 4: 1979 – 1990), while block 5 was in trial operation 1989 – 1990. Block 6 was in the final phase of construction, while blocks 7 and 8 were still in the construction phase.

Question 2: What is the size of the site which was or will be released? Is this the entire site, or just part of it?

Part of the site have been released in 2001/2002. This part of the site had not been in use by buildings of nuclear installations but were a spare area covered with grass/vegetation.

The entire KGR site had an initial area of 2.8 km². The released part of the site had an area of 700.000 m².

Site Contamination

Question 3: Is the radionuclide vector/radiological fingerprint representative for the site known? Is there a (standard) method to derive the radionuclide vector/radiological fingerprint from the measurements?

The only way in which the released part of the site could have been contaminated was by gaseous emissions during operation. The nuclide vector for these emissions was (simplified): noble gases: 80 % Kr 85, 10 % Xe 133 and 10 % Xe 135; aerosols: 80 % Co 60 and 20 % Mn 54; I 131 25 % elementary and 75 % organic.

Only at few parts of the site was the possibility for contamination from liquid effluents for which the typical nuclide vector was: longer-lived radionuclides: 22,5 % Cs 137, 16,2 % Cs 134, 14,5 % Ag 110m, 13,6 % Co 60, 8,1 % Co 58 and 9,5 % Mn 54 as well as I 131 with 15,6 %.

The possible radionuclide vector for the purposes of radiological modelling had been derived from the following table which gives assumptions on the annual deposition of radionuclides and calculates an upper estimate for the deposited activity for each radionuclide.

Nuklide	deposition rate [Bq/cm²/a]	deposition after 16 a of operation [Bq/cm²]
I 129	9,7E-14	1,6E-12
I 131	4,6E-03	7,4E-02
I 132	9,0E-03	1,4E-01
I 133	1,8E-02	2,9E-01
I 134	9,0E-03	1,4E-01
I 135	1,8E-02	2,9E-01
Cr 51	9,0E-05	1,4E-03
Mn 54	3,6E-05	5,8E-04
Fe 59	3,6E-05	5,8E-04
Co 58	1,8E-04	2,9E-03
Co 60	5,3E-04	8,6E-03
Sr 89	1,7E-06	2,7E-05
Sr 90	1,7E-06	2,7E-05
Zr 95	1,8E-05	2,9E-04
Nb 95	1,8E-05	2,9E-04
Sb 124	3,6E-04	5,8E-03
Te 123m	3,6E-05	5,8E-04
Cs 134	1,4E-04	2,3E-03
Cs 137	3,2E-04	5,1E-03
Ba 140	3,6E-05	5,8E-04
Pu 238	7,3E-08	1,2E-06
Pu 239	7,3E-09	1,2E-07
Pu 241	3,6E-06	5,8E-05
Am 241	3,4E-09	5,4E-08
Cm 242	9,7E-07	1,6E-05
Cm 244	7,3E-08	1,2E-06

The actual radionuclide vector has been established on the basis of extensive measurements of soil samples and in-situ gamma spectroscopy. The only nuclides which had been found to be of relevance were Co 60 and Cs 137.

Question 4: In the case of a contamination of the site, is it known how deep the radionuclides may have penetrated into the ground?

The penetration depth had been measured at some places on the site. A penetration depth of 5 cm to 10 cm for Co and Cs has been found as a reasonable estimate during the time from start of operation to the date the assessment had been carried out (1998-99). For the radiological model, depths of 1, 5 and 20 cm have been used (calculation of dose coefficients for external irradiation). The final results have been based on a penetration depth of 5 cm for Co and Cs.

Regulatory Standards

Question 5: On which basis has been or will the site be released?

A radiological assessment has been carried out to derive suitable clearance levels for a number of nuclides which may have been relevant in the contamination of KGR. This assessment was performed on behalf of the regulatory authority (Umweltministerium

Mecklenburg-Vorpommern – Environmental Ministry of the Federal State in which KGR is situated) as part of the licensing procedure.

The radiological assessment has been based on an individual dose of 10 µSv/a. The assessment comprised a number of radiological pathways such as:

- construction workers on the site (construction of new buildings after clearance);
- persons in dwellings on the site (one-family houses with garden for food production);
- industrial use of the site;
- use of the site for recreational purposes;
- others.

All relevant exposure pathways (external irradiation, direct ingestion of soil, dust inhalation, secondary ingestion – water pathways, plants grown in soil and others) have been taken into account.

At the time the assessment was done, there was no general clearance regulation for sites in place yet. This is now the case with Art. 29 para. 2 no. 1 of the Radiation Protection Ordinance (Strahlenschutzverordnung of July 20, 2001). However, the same model has been used for deriving the generic clearance levels of the Radiation Protection Ordinance and the site-specific levels for KGR.

Question 6: Please give examples of release levels/clearance levels of the relevant nuclides which will be used/have been used for the release of (a) site(s).

In the end, only clearance levels for Co 60 and Cs 137 were used because these were the only nuclides which had been detectable in the field.

The clearance levels have been expressed in terms of Bq/cm² and Bq/g. One value is convertible into the other taking penetration depth and soil density into account. Compliance with these clearance levels allow the site to be re-used for any purpose (unconditional clearance).

Radionuclide	Clearance level in Bq/cm²	Clearance level in Bq/g
Co 60	0.2	0.03
Cs 137	0.5	0.1

Question 7: Are there special features of the site which have influenced the derivation of suitable release criteria or the release procedure itself?

The Chernobyl fallout has lead to a problem with the distinction between Cs 137 which may have originated from the operation of the KGR plant and Cs 137 from the fallout (background). A method for background subtraction was established.

Implementation

Question 8: Which measurement techniques will be/have been applied for clearance measurements?

Various measurement techniques had been used, mainly:

- *In situ* gamma spectroscopy with collimators (so that the measurement was confiend to an area of around 1 m² only) and without collimators (so that the measurement was taken to represent a larger area, e.g. 100 m²).

- Excavation of the soil and laboratory analysis with gamma spectrometry.

At the largest part of the area to be released, it was not required to perform measurements which covered the entire area because of the only possible source of contamination (deposition of gaseous releases) leading to homogeneous contamination. At these areas, a formula for deriving the measurement density has been used which was also took into account the general contamination level.

Only at few places with known contamination or where there was a strong possibility for contamination, 100 % measurements were required.

Question 9: What averaging criteria will be / have been used for the release of (a) site(s)?

The averaging criterion has been 100 m² for areas were contamination was suspected to be present and 1,000 m² on areas which were assumed to be free of contamination. A value of 100 m² corresponds to the minimum area on which any person will stay at random in any reasonable scenario so that there is no need to average over smaller areas.

HDR – NPP Heißdampfreaktor Karlstein (Germany)

Description of the nuclear facility

Question 1: The HDR at Karlstein near Kahl at the Neckar river was a BWR with 25 MWe. It started operation in 1969 and was shut down due to fuel element failure in 1971. The main feature of the plant in comparison to an ordinary BWR was the nuclear super-heating of the steam which aimed at improving the operation of the turbine.

The shut-down plant was later used for experiments in reactor safety (mainly simulations of incidents) between 1975 and 1992. The plant has been fully removed ("green field"). The site has been released in 1998.

Question 2: What is the size of the site which was or will be released? Is this the entire site, or just part of it?

The HDR site was rather small in comparison to sites of larger NPPs. It comprised 3,500 m² open space and 2,300 m² of covered areas (buildings, roads, sealed areas).

Site Contamination

Question 3: Is the radionuclide vector radiological fingerprint representative for the site known? Is there a (standard) method to derive the radionuclide vector/radiological fingerprint from the measurements?

The nuclide vector was derived from measurements. It was concluded that only Co 60 and Cs 137 were relevant for clearance of the site. The percentages of the activity of these nuclides in the nuclide vector has been derived on the basis of measurements.

Natural radionuclides were measured but considered no further because they did not result from the operation of the facility.

Question 4: In the case of a contamination of the site, is it known how deep the radionuclides may have penetrated into the ground?

The penetration depth has been assumed to 10 cm for the radiological modelling/calculations.

Regulatory Standards

Question 5: On which basis has been or will the site be released?

The site has been released on the basis of a separate radiological assessment which has been carried out 1996 on behalf of the licensing authority, the Bavarian State Ministry for the Environment (Bayerisches Staatsministerium für Landesentwicklung und Umweltfragen). At that time, there was no generic basis in the regulation for clearance of sites.

The assessment has been carried out using 10 µSv/a as the basis for the radiological modelling. The model comprised the following scenarios for industrial reuse:

- industrial use by a larger company: inhalation of dust during earth works, external irradiation from soil deposited above ground, external irradiation from soil on the ground, ingestion of radionuclides which have been leached by rainwater

- use as storage area (e.g. for construction material): inhalation of dust from the soil, external irradiation from the ground, ingestion of radionuclides which have been leached by rainwater
- use as public car park (which is not covered): external irradiation while staying on the car park, inhalation of dust from the soil, ingestion of radionuclides which have been leached by rainwater

Question 6: Please give examples of release levels/clearance levels of the relevant nuclides which will be used / have been used for the release of (a) site(s).

The clearance levels have been:

- 0.5 Bq/cm² for the sum of the surface specific activities of Co 60 and Cs 137,
- 0.03 Bq/g for the sum of the mass specific activities of Co 60 and Cs 137.

Question 7: Are there special features of the site which have influenced the derivation of suitable release criteria or the release procedure itself?

The initial use of the site for industrial purposes was known and was taken into account in the radiological models.

Implementation

Question 8: Which measurement techniques will be/have been applied for clearance measurements?

Compliance with clearance levels has been established using collimated in-situ gamma spectroscopy with HPGe detector. Additionally, soil samples for gamma spectrometry in a laboratory were taken and measurements with contamination monitors on sealed surface areas of the site were performed.

The Cs 137 from Chernobyl fallout has been deducted from the measurements because the clearance levels relate only to the part of the activity originating from the plant.

Question 9: What averaging criteria will be/have been used for the release of (a) site(s)?

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Literature: L. Hummel, B. Sitte (TÜV Süddeutschland, München): New Strategies for Assessment of Achieving the Release Criteria for Building Structures of NPPs with in-situ Gamma Spectrometry – 2nd EC/ISOE Workshop on Occupational Exposure Management at NPPs, Tarragona, Spain, 5-7 April 2000

Fuel Element Production Facilities at Hanau (Germany)

Description of the nuclear facility

Question 1: These answers relate to the two fuel element production facilities:

- NUKEM-A; and
- Siemens Brennelementwerk Betriebsteil Uran.

which were located at Hanau (Hessen) on adjacent sites. Because the decommissioning projects were carried out in parallel to a large extent and similar radiological criteria were used, both facilities are described in a single questionnaire.

Question 2: What is the size of the site which was or will be released? Is this the entire site, or just part of it?

- NUKEM-A: 1,15 ha (135 x 85 m²).
- Siemens Brennelementwerk Betriebsteil Uran: total area of site 3.2 ha (32,000 m²).

Site Contamination

Question 3: Is the radionuclide vector / radiological fingerprint representative for the site known? Is there a (standard) method to derive the radionuclide vector / radiological fingerprint from the measurements?

The radionuclide vector was derived from the nuclide mixture which has been handled in the facilities during operation. It was corrected for radioactive decay (to account for the ingrowth of daughters from Th and U in long-term scenarios).

The (inital) nuclide vectors in the two facilities comprise mainly the following radionuclides (bold: nuclides with activity percentages above 10 %):

- NUKEM-A: **Ac 228**, Bi 212, Ce 144, Cs 134, Cs 137, Pa 234m, Pb 212, Po 212, Po 216, Pu 238, Pu 239, Pu 240, Pu 241, Ra 224, **Ra 228**, Rn 220, Ru 106, Sr 90, **Th 228**, Th 230, Th 231, Th 232, Th 234, Tl 208, U 232, U 233, **U 234**, U 235, U 236, U 238
- Siemens Brennelementwerk Betriebsteil Uran: Ba 137m, Bi 212, Cs 137, Pa 234, **Pa 234m**, Pb 212, Po 212, Po 216, Ra 224, Rn 220, Th 228, Th 230, Th 231, **Th 234**, U 232, U 233, **U 234**, U 235, U 236, **U 238**.

Question 4: In the case of a contamination of the site, is it known how deep the radionuclides may have penetrated into the ground?

The penetration depth varied considerably because the contamination to a large extent originated from leakages of pipes under the building floors. It could reach a depth of 2 m, at some places even more.

Regulatory Standards

Question 5: On which basis has been or will the site be released?

The clearance criteria for both sites have been established on the basis of expert assessments on behalf of the Environmental Ministry of the Federal State Hessen (Hessisches Ministerium für Umwelt, Energie, Jugend, Familie und Gesundheit). At that time (1998/99),

there were no generic clearance criteria for sites which have meanwhile been established in the Radiation Protection Ordinance (Strahlenschutzverordnung of July 20, 2001).

The radiological models comprised a large number of scenarios which took into account the following pathways:

- excavation of the soil and disposal on a landfill:
  - workers on the landfill (external irradiation, direct ingestion of soil, dust inhalation)
  - general public (secondary ingestion via water pathways, dust inhalation; including long-term analysis over several 1000 a)
- leaving the soil on the site
  - general public: people living on the site
  - general public: groundwater pathway, secondary ingestion

Question 6: Please give examples of release levels/clearance levels of the relevant nuclides which will be used/have been used for the release of (a) site(s).

The clearance levels relate to that part of U 234 and Th 232 which originated from the operation of the plants.

For NUKEM-A, the clearance levels were 10 mBq/g for U 234 and 2.4 mBq/g Th 232, based on the assumption of homogeneous contamination distribution. The natural backgroud of both nuclides were 45 mBq/g für U 234 and 20 mBq/g for Th 232. Starting from these considerations and taking the actual contamination distribution into account, an average *total* activity of 100 mBq/g for U 234 was derived. When the site was remediated to conform with this U 234 clearance level, it was concluded that the Th contamination would automatically be reduced to such an extent that the total dose from both contaminants would be below 10 µSv/a.

Question 7: Are there special features of the site which have influenced the derivation of suitable release criteria or the release procedure itself?

The U contamination has penetrated far into the ground and has reached groundwater. These aspects had a considerable relevance for the radiological analysis, but it is not possible to discuss this in detail here.

Implementation

Question 8: Which measurement techniques will be/have been applied for clearance measurements?

The sites were divided into grid areas of 1 x 1 m². On each grid area, the soil was excavated with depths of a few 10 cm. The excavated soil was then measured. If it exceeded the clearance levels for site release, it was removed and brought to landfill disposal or was treated as radioactive waste, otherwise it was put back into place and the grid area was considered free.

Question 9: What averaging criteria will be / have been used for the release of (a) site(s)?

The averaging area corresponded to the grid areas, i.e. 1 m².

ANSWERS FROM HUNGARY

Gabor Bacsko of PURAM, Hungary, wrote:

> As I mentioned earlier by our personal meeting, for the time being we have no decommissioning project in progress, thus I can answer only to the first question of the questionnaire which refers to the plant site, and I wonder if there is any meaning in sending you a detailed description of the Paks plant site.
>
> Referring to the casual soil contamination below the units I can report only the results of water sample analysis, taken from the observation wells on the site. These analyses indicated the tritium that appeared in some of the boreholes. Its concentration is continuously measured. We have no information from any other isotope as up to now; no samples have been taken from the soil. According to our preliminary decommissioning plan that is now in the approval stage, release of the plant site could be expected only about in 2013[3], i.e. in 100 years, after 70 years of safe enclosure period. That is why no Authority regulations are available. Data applied in the decommissioning cost scheduling are based only on technical estimation.

3. probably meant to be 2103.

ANSWERS FROM JAPAN - JPDR

Description of the nuclear facility

Question 1: Name: Japan Power Demonstration Reactor (JPDR)

The JPDR was a boiling water reactor (90 MWt) that was constructed first in Japan as a power reactor. It was located in Tokai Research Establishment of Japan Atomic Energy Research Institute and operated from 1963 to 1976; it was finally shutdown due to several problems. The JPDR decommissioning program was a demonstration project for future decommissioning commercial nuclear power plants in Japan, which include technology development and actual dismantling of the facilities to verify the developed technologies as well as availability of conventional dismantling tools. The dismantling activities were conducted from 1986 to 1996 to recover green field conditions.

Question 2: What is the size of the site which was or will be released? Is this the entire site, or just part of it?

The size of the JPDR site is approximately 6,600 m², which located inside of Tokai Research Establishment. The JPDR site has been released, however it is still in the part of Tokai Research Establishment.

Site Contamination

Question 3: Is the radionuclide vector / radiological fingerprint representative for the site known? Is there a (standard) method to derive the radionuclide vector / radiological fingerprint from the measurements?

Before demolishing the JPDR buildings, radioactive vectors were identified by systematically measuring radioactivity of the building inner surface. It was proved by the measurement, that Co-60 was a dominant radionuclide and Cs-137 in small part.

The contaminated parts were removed to be equal to background levels for cancellation of radiation control areas. After demolishing the buildings, radioactivity on the ground was measured to confirm that these was no effective radioactivity at the site. No contamination was measured on the ground.

Question 4: In the case of a contamination of the site, is it known how deep the radionuclides may have penetrated into the ground?

No.

Regulatory Standards

Question 5: On which basis has been or will the site be released?

In the case of JPDR decommissioning program, a basic procedure for cancellation of radiation control areas was established in cooperation with the regulatory body; the cancellation of radiation control areas followed by building demolition resulted to release of the JPDR site. The level for cancellation of radiation control areas was decided to be that there was no artificial radioactivity remaining in the areas, i.e., background radioactivity.

Question 6: Please give examples of release levels/clearance levels of the relevant nuclides which will be used/have been used for the release of (a) site(s).

There is no site released levels in Japan.

Question 7: Are there special features of the site which have influenced the derivation of suitable release criteria or the release procedure itself?

None.

Implementation

Question 8: Which measurement techniques will be/have been applied for clearance measurements?

Various measurement techniques were applied to radioactive inventory evaluation in the building after removal of components.

These measurement techniques were as follows:

- gas flow type detector;
- sampling and ν-spectroscopy by Ge detector
- NaI scintillation counter.

Question 9: What averaging criteria will be/have been used for the release of (a) site(s)?

There is no criteria for site release in Japan.

ANSWER FROM THE NETHERLANDS

Hans Codee of COVRA writes:

> "With reference to your questionnaire 'Clearance of Sites of Nuclear Installations in OECD Countries I can be rather short in my answer for the Netherlands. In the Netherlands there is no nuclear installation yet, that is in the phase of releasing the site. Thus there is no practical experience at all on this subject. Of course there are some small installations dismantled inside buildings, but never was a complete building dismantled yet."

ANSWERS FROM NORWAY

Description of the nuclear facility

Question 1: Institute for Energy Technology (IFE), Kjeller. Contaminate area were located in the nearby Nielva River, the recipient for discharges of liquid waste.

Nuclear activities began in Kjeller in 1951 with the start up of the first research reactor "JEEP1" This reactor was in operation until 1967. The 2 MW(t) JEEP 2 research reactor was commissioned in 1966 and are still in operation. Since 1952, radionuclides have been produced at the site. Most of them are used for medical purposed. The area also houses some non-nuclear activities.

In the period 1961-68 the zero power reactor "NORA" was also in operation, during this periode the site also hosted a small plant for reprocessing and purification of uranium. During the subsequent decommissioning of the purification plant, discharges of plutonium and americium to the local environment increased substantially although it should be noted that levels fell within the then-existing limits.

Contaminate area were located in the nearby Nitelva River, the recipient for discharges of liquid waste. The contaminated sediments were removed by IFE, as a requirement from NRPA, in 2000.

The subsequent information deals not with the decommissioning of the plant itself but with the concomitant contamination of the Nitelva River and the practices employed during a subsequent clean-up operation.

Question 2: What is the size of the site which was or will be released? Is this the entire site, or just part of it?

An area of relatively contaminated sediment was concentrated in an area of approximately 200 m² close to the outlet of the "NALFA" discharge pipeline.

Site Contamination

Question 3: Is the radionuclide vector/radiological fingerprint representative for the site known? Is there a (standard) method to derive the radionuclide vector/radiological fingerprint from the measurements?

Activity concentrations of beta, gamma and alpha emitting radionuclides are routinely monitored and reported by IFE Kjeller. The radiological fingerprint, in this sense, is well defined. In the case of the contaminated area on the Nitelva River, ^{239,240}Pu was the main radionuclide of concern.

Question 4: In the case of a contamination of the site, is it known how deep the radionuclides may have penetrated into the ground?

In the contaminated part of Nitelva River, the depth of contaminated sediment is in the order of 1 m. However the main contamination were found 50-60 cm below the surface.

Regulatory Standards

Question 5: On which basis has been or will the site be released?

It should be noted that this river site has never been restricted and has been a public unrestricted area. When the contamination was discovered the area were fenced until the contamination were removed.

Contamination in the sediment of Nitelva were of concern and remediation measures were incumbent upon IFE. The legal basis for site clearance, was:

- Act of 18 June 1938 on use of X-rays radioum etc.
- Regulation from 23. January 1976.

The act from 1938 is now substituted by a new radiation protection act from 12 May 2000.

The legal basis provides opportunities for the radiation protection authority to regulate radioactive contamination as a result of a plant activity. In this case an assessment was required by the Norwegian Radiation Protection Authority.

Brief radiological assessments, involving sampling measurement and habit survey and impact assessment were undertaken with the aim of deriving potential doses to human in the area. One important aspect was also the uncertainty about potential future use of the site e.g. building activities, dredging etc which may lead to other conditions than the present situation. No explicit consideration was made in relation to impact directly on the environment – this was assumed to be negligible

Question 6: Please give examples of release levels/clearance levels of the relevant nuclides which will be used/have been used for the release of (a) site(s).

All sediment with a summed concentration of 10 Bq g^{-1} plutonium (^{239}Pu, ^{240}Pu) and americium (^{241}Am) were removed from the contaminated site.

Question 7: Are there special features of the site which have influenced the derivation of suitable release criteria or the release procedure itself?

The contaminated are was in close proximity to human habitation. Conservative assessments were made by considering a protracted occupancy of the contaminated area and the subsequent potential for inhalation of contaminated sediments.

Implementation

Question 8: Which measurement techniques will be/have been applied for clearance measurements?

Gamma spectrometry, and alpha spectrometry (radiochemical separation of Pu and Am followed by alpha counting) were employed. 10s of samples were analysed in duplicate, by NRPA and IFE, from the contaminated area before and after site remediation.

Question 9: What averaging criteria will be / have been used for the release of (a) site(s)?

Essentially no averaging criteria were employed. The release criteria simply required all sampled material to fall below the 10 Bq g^{-1} limit following remediation.

References to open literature

Further details can be found in the following reference:

- Norwegian Radiation Protection Authority (2001). Radioactivity in the marine environment. StrålevernRapport 2001:9, NRPA, Østerås, Norway pp.39.
- Norwegian Radiation Protection Authority (1999). Radioaktive utslipp fra IFE Kjeller til Nitelva.
- StrålevernRapport 1999:11 (in Norwegian).

In addition to this specific case in the Nitelva River another operation involving removal of contaminated soil were performed. It was decided to remove containers containing solid radioactive waste from wich was stored underground at the IFE, Kjeller site. The containers should be removed to the new storage for low and medium active waste in Himdalen. At that operation a clearance level of 100 Bq/g for ^{137}Cs was used.

ANSWERS FROM SLOVAK REPUBLIC

Description of the nuclear facility

No site in Slovak Republic was released up to now as well as it is not planned for next decade. In Slovakia, there are only three sites, one with LILW near surface repository and two with NPPs and other NIs. Each site is described individually.

Question 1: **Site Jaslovské Bohunice** is compact site with one common fence. It contents three NPPs (NPP A1 with HWGCR, NPP V1 (EBO1,2)with WWER 440,V230 and NPP V2 (EBO 3,4) with WWER 440, V213), Wet Interime Spent Fuel Storage (ISFS), Technologies for radioactive waste treatment and conditioning TSURAO (2 bituminization plants including evaporators, Bohunice Conditioning Centre with evaporation, incineration,supecompaction, fragmentation and cementation). NPPA1 is under decommissioning, NPP V1 is planned to be shut down in 2006 (1.unit) and 2008 (2.unit) with the goal to decommission it. There is supposed to ask for life extension for NPP V2 after 2014. Each NPP has individual decommissioning plan. ISFS was recently reconstructed with life time to 2050, it has also individual decommissioning plan. TSURAO is completed to serve for treatment and conditioning of all radwaste produced at the site. Decommissioning plan is prepared for the all technologies. Two experimental tretment technologies (bituminisation and incineration) are placed in the buildings of NPP A1.

Site Mochovce contents NPP Mochovce 1,2 (EMO1,2) with WWER 440, V213 commissioned in 1998,1999, and Mochovce 3,4 with WWER 440, V213 (construction interrupted). Radioactive waste treatment technology at this site is under preparation to contruction.

Question 2: **What is the size of the site which was or will be released? Is this the entire site, or just part of it?**

Site Jaslovské Bohunice represents 720 000 m².

Site Mochovce represents 1 292 237 m².

Site Contamination

Question 3: **Is the radionuclide vector/radiological fingerprint representative for the site known?**

YES, preoperational measurement as well as regular measurement during operation is carried.

Is there a (standard) method to derive the radionuclide vector/radiological fingerprint from the measurements?

YES for preoperational measurement and operation of all NI with exemtion of NPP A1 (shut down after accident) – methodology of sampling (penetrated activity) is under evaluation.

Please give the list of relevant radionuclides such as Co 60, Cs 137, Sr 90 ...

Management of Contaminated soil (Co 60, Cs 137, alpha) around NPP A1/remediation is under preparation.

Question 4: In the case of a contamination of the site, is it known how deep the radionuclides may have penetrated into the ground?

The content of radionuclides is measured in underground water (regular measurement at several places around source of contamination). Deatiled measurement of penetration into the ground for purpose of remediation is starting.

Regulatory Standards

Question 5: On which basis has been or will the site be released?

Until 1996 no clearance levels were established in Slovak Republic; zoning principle was only used for release of potentially clean materials.

The basic clearance principle issued in 1996 is based on limit of effective dose for individual (an average member of the critical group) 10 micro Sv/y as well as collective effective dose 1 man Sv/y from each exempt/clearance act. Supporting derived limits for surface and mass specific activity of released metal materials were also issued in 1996.

New supporting derived limits were issued in 2001 in accordance with IAEA/NEA guidance. Nuclides are grouped to 5 classes regarding their radiotoxicity. A permit of Regulatory Body for release is necessary.

Question 6: Please give examples of release levels/clearance levels of the relevant nuclides which will be used/have been used for the release of (a) site(s).

Clearance levels for all type of materials were issued in Act 470/2000 Coll. and Regulation 12/2001Coll. (valid since December 13, 2000):

Clearance levels for all types of cleared material:

Type of radioactive contamination	**Radiotoxicity class**				
	1	2	3	4	5
Materials, solid substances and items contaminated/activated through the whole volume	Clearance levels for specific activity [$Bq.g^{-1}$]				
	0.3	3	30	300	3 000
Materials and items contaminated on the surface	Clearance levels for surface activity [$Bq.cm^{-2}$]				
	0.3	3	30	3 00	3 000

Nuclides are grouped to 5 classes regarding their radiotoxicity. A permission of Regulatory Authority under Ministry of Health (Radiation protection) for each type of release act is necessary (with exemption of individual small tools /items with surface less then 150 cm² and mass less then 10 kg). Measurement system including instrumentation and calculation of some nuclides activities based on nuclides vector must be justified and approved.

Radiotoxicity classes

Class	Radionuclides
1	Na-22, Na-24, Mn-54, Co-60, Zn-65, Nb-94, Ag-110m, Sb-124, Cs-134, Cs-137, Eu-152, Pb-210, Ra-226, Ra-228, Th-228, Th-230, Th-232, U-234, U-235, U-238, Np-237, Pu-239, Pu-240, Am-241, Cm-244
2	Co-58, Fe-59, Sr-90, Ru-106, In-111, I-131, Ir-192, Au-198, Po-210
3	Cr-51, Co-57, Tc-99m, I-123, I-125, I-129, Ce-144, Tl-201, Pu-241
4	C-14, P-32, Cl-36, Fe-55, Sr-89, Y-90, Tc-99, Cd-109
5	H-3, S-35, Ca-45, Ni-63, Pm-147

Activity is averaged over 1 t or 1m² for homogenous volume/surface activity and over 0.3 t or 0.1 m² for non-homogenous one. Non homogenous activity can accede the levels given in table three times, general activity limit averaged over 1 t or 1 m² must be kept.

If material/items are contaminated in both volume and surface ways, the evidence that both limits are kept has to be done. If material/items are contaminated by more nuclides, sum of their aliquot portions cannot accede one.

In 2001, the legal basis for remediation of contaminated sites was established first time. Any remediation activity needs a permission of the Regulatory Authority. The operator has to provide the evidence, that the way of remediation is the optimal from the point of the view of radiation protection.

Question 7: Are there special features of the site which have influenced the derivation of suitable release criteria or the release procedure itself?

No.

Implementation

Question 8: Which measurement techniques will be/have been applied for clearance measurements?

All described methods are used for measurement and there are evaluated. Methodology is under development.

Question 9: What averaging criteria will be/have been used for the release of (a) site(s)?

see answer to Question 8.

ANSWERS FROM SPAIN

Description of the nuclear facility

Question 1: Station Name: Vandellós 1 NPP

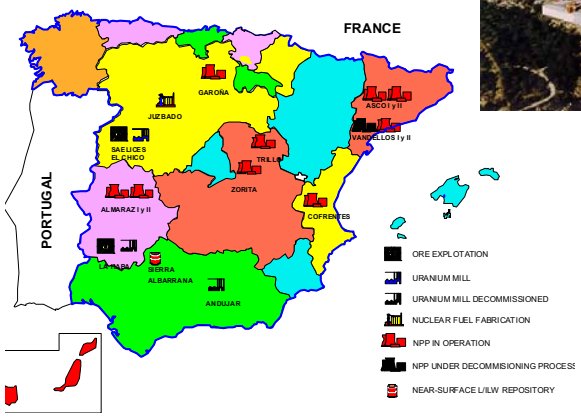
Location: Vandellós, Tarragona – Spain

Electric power: 500 MWe

Reactor Type: Gas/graphite reactor

Start operation: 1972

Shut down: 1989

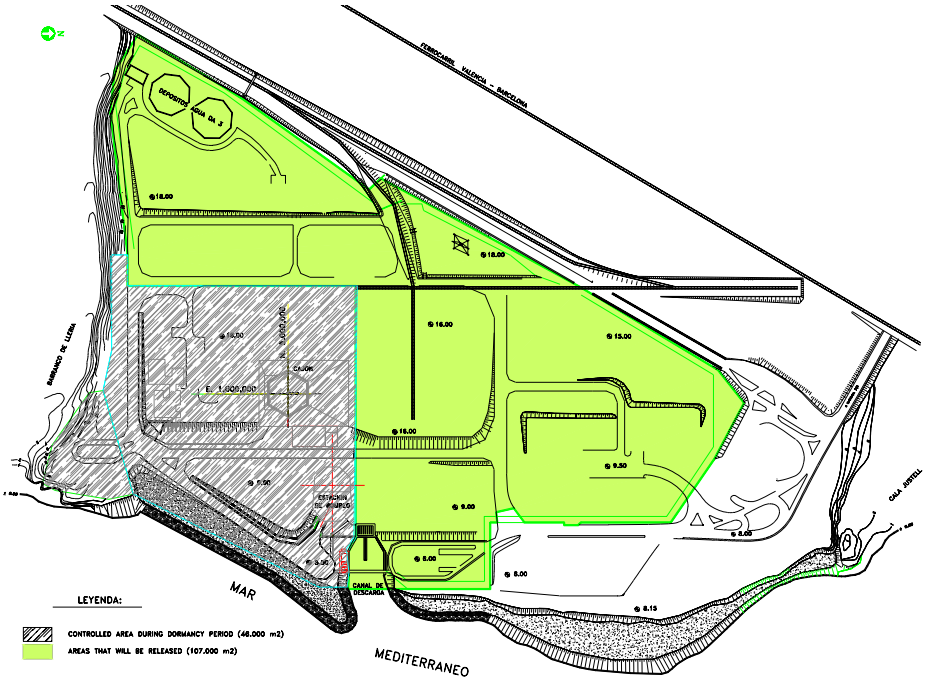


Question 2: What is the size of the site which was or will be released? Is this the entire site, or just part of it?

The entire size of the site is about 153,000 m²

It will be released about 107,000 m² of land surfaces, soils and roads, concrete pads. It will remain under controlled area about 46,000 m² over 30 years of dormancy.

A general plan of the site is presented in the following figure, including areas that will be released.



Site Contamination

Question 3: Is the radionuclide vector/radiological fingerprint representative for the site known? Is there a (standard) method to derive the radionuclide vector/radiological fingerprint from the measurements?

Correlation studies have been carried out to obtain a radionuclide vector representative for the site. However, the results obtained from the samples measurements data do no permit to define a single vector for the site.

Based on operational data and site characterisation samples and measurements, the relevant radionuclides for potential contaminated soils have been established as representative of soil contamination (see Table).

Relevant radionuclides are as follows: H-3, C-14, Mn-54, Ni-59, Ni-63, Co-60, Sr-90, Nb-94, Sb-125, Cs-134, Cs-137, Eu-152, Eu-154, Pu-239, Am-241.

Pathway / Issue	Radionuclides considered	Radionuclides not considered
Deposition	H-3, C-14, Co-60, Cs-137	(I-131 and Mn-54 not considered because $T_{1/2}$ < 1 year
Resuspension+Deposition	Sb-125, Cs-134, Cs-137	Sb-124, Nb-95, Zr-95, Se-75, Zn-65 not considered because $T_{1/2}$ < 1 year
Retrieval of ILW storaged in Silos	Fe-55, Ni-59, Ni-63 (steel wires). Sr-90, Nb-94 (graphite and spent fuel)	
Others Measured in Soils	Eu-152, Eu-154, Pu-239 , Am-241	

Question 4: In the case of a contamination of the site, is it known how deep the radionuclides may have penetrated into the ground?

All soil samples taken in the initial site characterisation indicate that contamination in the area to be released is only surficial and has not penetrated into the ground. There are no sources of potential groundwater contamination in the area to be released.

In addition, contamination levels above the released limits of soils have not been reported.

A characterization borehole is planned to define the actual vertical profile of radionuclide distribution in the first 15 cm of depth.

Regulatory Standards

Question 5: On which basis has been or will the site be released?

The Spanish regulatory system related to decommissioning process of nuclear facilities is based in the following regulations:

Act 15/1980 on 22 of April about the creation of the Nuclear Safety Council. The CSN acts as the only responsible body all over the state in nuclear safety and radiation protection matters (regulatory body).

Decree 1836/1999 on 3 of December on Nuclear and Radioactive Facilities Regulations. This legal basis establishes:

- the issuance of a Decommissioning Authorization and a Termination of the License are required to carry out the decommissioning of a nuclear power plant.
- for the granting of the decommissioning authorization, the applicant shall submitted among others documentation, a Restoration Plan, addressing the surveys plans to establish the radiological status of the site to be released and the final status survey to be carried out to demonstrate compliance with the release criterion.

At present, CSN does not provide specific regulations with regard to the release criterion.

A release criterion, expressed in terms of dose, of 100 μSv/y has been proposed by ENRESA for Vandellós 1, taking into account the operating documentation of the exploitation of the plant, Spanish and international reference guidelines applied in different projects, and related literature. Figure 2 shows the reference values considered.

Exposure pathway modelling is used to calculate the radionuclide-specific concentration of specific nuclides that could result in a dose or specific risk equal to the release criterion. RESRAD and D&D codes have been used.

Two specific scenarios have been considered. An Industrial scenario representative of the 30 years of dormancy, for a restrictive use of the site only for industrial activities, and a Residential scenario for the period beyond 30 years.

Question 6: Please give examples of release levels/clearance levels of the relevant nuclides which will be used/have been used for the release of (a) site(s).

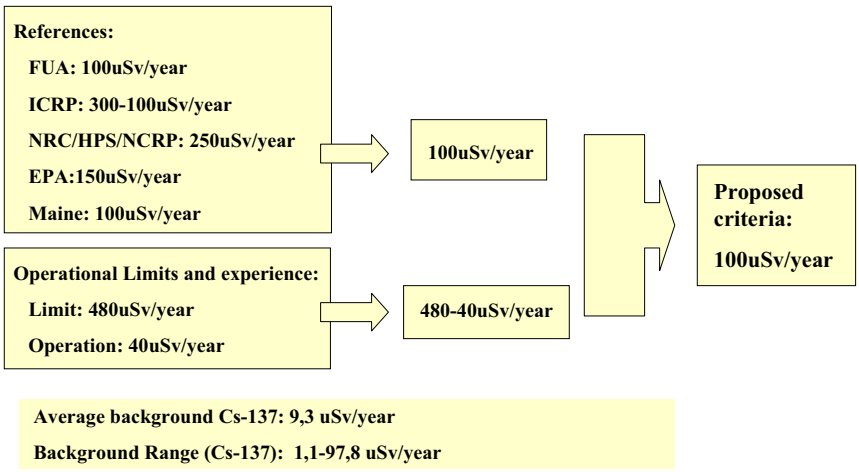
Clearance levels of the relevant nuclides, which have been proposed for the release of the site, are as follows:

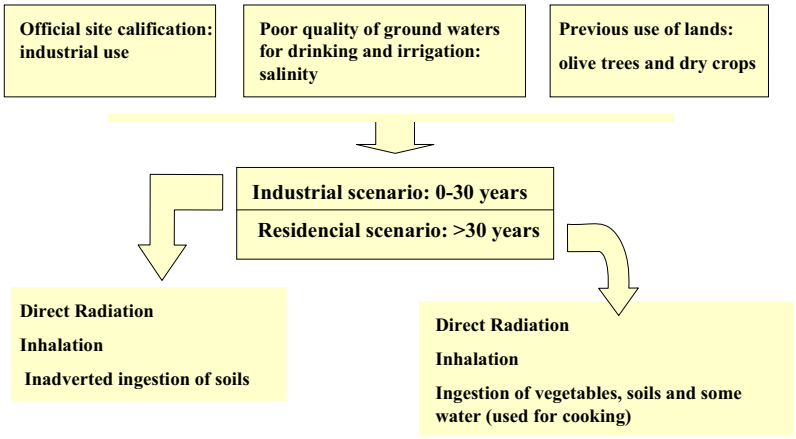
Radionuclide	Clearance level (Bq/g)
H-3	1,25E+02
C-14	3,19E-01
Ni-59	2,21E+02
Ni-63	1,00E+02
Co-60	4,95E-01
Sr-90 (Y-90)	1,52E-01
Nb-94	9,23E-02
Sb-125	4,63E+00
Cs-134	9,38E-01
Cs-137	3,27E-01
Eu-152	4,57E-01
Eu-154	1,01E+00
Pu-239	8,43E-01
Am-241	8,22E-01

Question 7: Are there special features of the site which have influenced the derivation of suitable release criteria or the release procedure itself?

The following figure details exposure pathways that have been considered in the scenarios proposed.

There are not special features of the site. In fact, all water pathways can be eliminated due to salinity of ground waters, in the residential scenario.





Implementation

Question 8: Which measurement techniques will be/have been applied for clearance measurements?

- Scanning survey of surface using portable radiation detection instrument assembled to a mobile device (see photos).
  - For photon emitting radionuclides – sodium iodine survey meter 2"x 2"
  - For beta/alpha emitters – gas-flow proportional counter.
- Direct measurements of alpha, beta and photon surface activity.
- Sampling and laboratory analysis:
  - gross alpha/beta measurements.
  - Alpha/gamma spectroscopy.
  - Low-energy beta emitting radionuclides (e.g. H-3, C-14, Ni-63) and difficult detection radionuclides (e.g. Sr-90).

Question 9: What averaging criteria will be/have been used for the release of (a) site(s)?

Based on MARSSIM methodology (NUREG-1575) areas to be released in the site are classified by contamination potential taking into account both the level of contamination and the distribution of the contamination.

Survey units are limited in size based on classification:

- Class 2 land areas: 2,000 – 10,000 m²
- Class 3 land areas: no limit

No Class 1 areas have been identified.

ANSWERS FROM SWITZERLAND

KKB – NPP Beznau (Switzerland)

Description of the nuclear facility

Question 1: KKB – Nuclear Power Plant Beznau, Switzerland.

Type: PWR, 2 units, each 380 MWe net, in operation since 1969/1971.

Shutdown not before 2009 (this date assumed for costing purposes in decommissioning studies from 2001).

Question 2: What is the size of the site which was or will be released? Is this the entire site, or just part of it?

Weight of materials for decommissioning (both units): appr. 280,000 tonnes.

Weight of Radwaste (both units): appr. 5,200 tonnes.

These numbers are from decommissioning studies 2001.

Site Contamination

Question 3: Is the radionuclide vector/radiological fingerprint representative for the site known? Is there a (standard) method to derive the radionuclide vector/radiological fingerprint from the measurements?

The radiological fingerprint for contamination and waste from the cleaning circuits (resins, slurries, concentrates) is determined on a routinely basis by measurements of nuclides which are relevant for the operational phase and long term safety of a final repository. Typical relevant nuclides are H-3, C-14, Cl-36, Mn-54, Fe-55, Co-60, Ni-59, Ni-63, Sr-90, Nb-94, Tc-99, Ag-108m, Ag-110m, Sn-126, Sb-125, I-129, Cs-134, Cs-137, Ce-144/Pr-144m, U-235, U-238, Pu-238, Pu-239/240, Pu-241, Am-241, Cm-244. Based on these measurements of periodically taken raw waste samples correlation factors to Co-60 and Cs-137 are established and total waste inventories of drums/containers calculated. Activated components (reactor internals, RPV, Bioshield) are characterized by activation calculations (developed to work in areas of infinite dilution, e.g. not ORIGEN 2.1); periodically samples are taken to validate these calculations.

Question 4: In the case of a contamination of the site, is it known how deep the radionuclides may have penetrated into the ground?

No (not investigated up to now).

Regulatory Standards

Question 5: On which basis has been or will the site be released?

The present Swiss Atomic Law (AtG) itself makes no specific reference to decommissioning. It states that each change in the state of a nuclear installation needs a license. This means that for decommissioning and dismantling a license is needed. A Federal

Act on the Atomic Law establishes that the general license for a nuclear installation can only be issued if a concept for decommissioning and eventual dismantling of the facility at the end of its life has been prepared. Studies on the feasibility of decommissioning have been accepted as sufficient proof in this context. Following the end of operation, the owner of the plant must prepare and submit for approval a detailed decommissioning project. This project must define the phases of decommissioning and the corresponding timetable.

Radiological studies have been and will be carried out to establish the radiological fingerprint of the facility (see Question 3) but not for deriving release levels. These levels are regulated in the Swiss radiation protection act.

Question 6: Please give examples of release levels/clearance levels of the relevant nuclides which will be used/have been used for the release of (a) site(s).

Nuclide	Bq/g	Bq/cm²	Nuclide	Bq/g	Bq/cm²
H-3	300	1'000	Sb-125	10	10
C-14	20	30	I-129	0.08	1
Cl-36	10	3	Cs-134	0.5	3
Mn-54	10	100	Cs-137	0.7	3
Fe-55	70	300	Ce-144	1	10
Co-60	1	3	U-235	0.4	1
Ni-59	200	1'000	U-238	0.4	1
Ni-63	50	1'000	Pu-238	0.02	0.3
Sr-90	0.3	3	Pu-239	0.02	0.3
Nb-94	5	3	Pu-240	0.02	0.3
Tc-99	20	3	Pu-241	0.9	10
Ag-108m	5	30	Am-241	0.02	0.3
Ag-110m	3	10	Cm-244	0.03	0.3
Sn-126	2	3			

The sum rule applies!

Question 7: Are there special features of the site which have influenced the derivation of suitable release criteria or the release procedure itself?

No, universal release criteria and limits for all types of material/waste are defined in the Swiss radiation protection act.

Implementation

Question 8: Which measurement techniques will be/have been applied for clearance measurements?

Not yet defined, but probably a combination of all these measurements.

Question 9: What averaging criteria will be/have been used for the release of (a) site(s)?

Not yet defined, to be discussed with the regulator before a real decommissioning project (see Question 5).

Additional references to open literature or the enclosure of internal information would be highly appreciated.

- Questionnaires on Decommissioning Strategies and Costs/Switzerland.
- Presentation KONTEC 2001.

KKG – NPP Gösgen (Switzerland)

Description of the nuclear facility

Question 1: KKG – Nuclear Power Plant Gösgen, Switzerland.

Type: PWR, 1 unit, 1'020 MWe net, in operation since 1979.

Shutdown not before 2019 (this date assumed for costing purposes in decommissioning studies from 2001).

Question 2: What is the size of the site which was or will be released? Is this the entire site, or just part of it?

Weight of materials for decommissioning: appr. 420,000 tonnes.

Weight of Radwaste: appr. 3,000 tonnes.

These numbers are from decommissioning studies 2001.

Site Contamination

Question 3: Is the radionuclide vector/radiological fingerprint representative for the site known? Is there a (standard) method to derive the radionuclide vector/radiological fingerprint from the measurements?

See Questionnaire for KKB, Nuclear Power Plant Beznau.

Question 4: In the case of a contamination of the site, is it known how deep the radionuclides may have penetrated into the ground?

See Questionnaire for KKB, Nuclear Power Plant Beznau.

Regulatory Standards

Question 5: On which basis has been or will the site be released?

See Questionnaire for KKB, Nuclear Power Plant Beznau.

Question 6: Please give examples of release levels/clearance levels of the relevant nuclides which will be used/have been used for the release of (a) site(s).

See Questionnaire for KKB, Nuclear Power Plant Beznau

Question 7: Are there special features of the site which have influenced the derivation of suitable release criteria or the release procedure itself?

See Questionnaire for KKB, Nuclear Power Plant Beznau.

Implementation

Question 8: Which measurement techniques will be/have been applied for clearance measurements?

See Questionnaire for KKB, Nuclear Power Plant Beznau.

Question 9: What averaging criteria will be/have been used for the release of (a) site(s)?

See Questionnaire for KKB, Nuclear Power Plant Beznau.

Additional references to open literature or the enclosure of internal information would be highly appreciated.

- Questionnaires on Decommissioning Strategies and Costs/Switzerland.
- Presentation KONTEC 2001.

KKL – NPP Leibstadt (Switzerland)

Description of the nuclear facility

Question 1: KKL – Nuclear Power Plant Leibstadt, Switzerland.

Type: BWR, 1 unit, 1'145 MWe net, in operation since 1984.

Shutdown not before 2024 (this date assumed for costing purposes in decommissioning studies from 2001).

Question 2: What is the size of the site which was or will be released? Is this the entire site, or just part of it?

Weight of materials for decommissioning: appr. 527,000 tonnes.

Weight of Radwaste: appr. 7,400 tonnes.

These numbers are from decommissioning studies 2001.

Site Contamination

Question 3: Is the radionuclide vector/radiological fingerprint representative for the site known? Is there a (standard) method to derive the radionuclide vector/radiological fingerprint from the measurements?

See Questionnaire for KKB, Nuclear Power Plant Beznau.

Question 4: In the case of a contamination of the site, is it known how deep the radionuclides may have penetrated into the ground?

See Questionnaire for KKB, Nuclear Power Plant Beznau.

Regulatory Standards

Question 5: On which basis has been or will the site be released?

See Questionnaire for KKB, Nuclear Power Plant Beznau.

Question 6: Please give examples of release levels/clearance levels of the relevant nuclides which will be used/have been used for the release of (a) site(s).

See Questionnaire for KKB, Nuclear Power Plant Beznau

Question 7: Are there special features of the site which have influenced the derivation of suitable release criteria or the release procedure itself?

See Questionnaire for KKB, Nuclear Power Plant Beznau.

Implementation

Question 8: Which measurement techniques will be/have been applied for clearance measurements?

See Questionnaire for KKB, Nuclear Power Plant Beznau.

Question 9: What averaging criteria will be/have been used for the release of (a) site(s)?

See Questionnaire for KKB, Nuclear Power Plant Beznau.

Additional references to open literature or the enclosure of internal information would be highly appreciated.

- Questionnaires on Decommissioning Strategies and Costs/Switzerland.
- Presentation KONTEC 2001.

KKM – NPP Mühleberg (Switzerland)

Description of the nuclear facility

Question 1: KKM – Nuclear Power Plant Mühleberg, Switzerland.

Type: BWR, 1 unit, 355 MWe net, in operation since 1972.

Shutdown not before 2012 (this date assumed for costing purposes in decommissioning studies from 2001).

Question 2: What is the size of the site which was or will be released? Is this the entire site, or just part of it?

Weight of materials for decommissioning: appr. 123,000 tonnes.

Weight of Radwaste: appr. 3,000 tonnes.

These numbers are from decommissioning studies 2001.

Site Contamination

Question 3: Is the radionuclide vector/radiological fingerprint representative for the site known? Is there a (standard) method to derive the radionuclide vector/radiological fingerprint from the measurements?

See Questionnaire for KKB, Nuclear Power Plant Beznau.

Question 4: In the case of a contamination of the site, is it known how deep the radionuclides may have penetrated into the ground?

See Questionnaire for KKB, Nuclear Power Plant Beznau.

Regulatory Standards

Question 5: On which basis has been or will the site be released?

See Questionnaire for KKB, Nuclear Power Plant Beznau.

Question 6: Please give examples of release levels/clearance levels of the relevant nuclides which will be used/have been used for the release of (a) site(s).

See Questionnaire for KKB, Nuclear Power Plant Beznau

Question 7: Are there special features of the site which have influenced the derivation of suitable release criteria or the release procedure itself?

See Questionnaire for KKB, Nuclear Power Plant Beznau.

Implementation

Question 8: Which measurement techniques will be/have been applied for clearance measurements?

See Questionnaire for KKB, Nuclear Power Plant Beznau.

Question 9: What averaging criteria will be/have been used for the release of (a) site(s)?

See Questionnaire for KKB, Nuclear Power Plant Beznau.

Additional references to open literature or the enclosure of internal information would be highly appreciated.

- Questionnaires on Decommissioning Strategies and Costs/Switzerland.
- Presentation KONTEC 2001.

ANSWERS FROM THE UNITED KINGDOM

UKAEA RESPONSE TO THE QUESTIONNAIRE PREPARED FOR THE WPDD OF THE OECD/NEA

Preamble to the Questionnaire

UKAEA currently operates five sites located across the United Kingdom, four located in England and one in Scotland. The sites although diverse in location have all historically undertaken research and development into nuclear energy, including the operation of a number of prototype power reactors, material test reactors, post irradiation fuel examination cave lines, fuel manufacturing facilities, large low and intermediate level waste stores, active chemistry laboratories and other support sub systems.

Today UKAEA's mission is the restoration of these sites and facilities in a safe, cost effective, environmentally and publicly acceptable manner. Each of the four main sites has an established a site restoration programme that considers the appropriate end point for each site or part there of; the time period in which this can be achieved; any period during which a care and maintenance regime is in place and any opportunities where facilities and or land could be released from regulatory control for re-use by others not related to the nuclear industry. These site restoration programmes are progressing satisfactorily with sites such as Dounreay receiving much credit for the vision and extents of its long term restoration programme, whilst Harwell and Winfrith are well advanced in their programme of releasing ex-nuclear buildings for marketing opportunities. The Winfrith site has recently achieved 'delicensing' status on 4ha of land, and with Harwell have further applications awaiting determination by the HSE Nuclear Installations Inspectorate.

UKAEA's response to these questions are focussed on the work undertaken at the Harwell and Winfrith site where active programmes exist to release buildings and land from nuclear regulatory control. These applications have been built on an iterative communication process with the UK nuclear regulator. Each site application has been allowed to address the specific technical issues relevant to each site, within the requirement of compliance with a UKAEA Corporate procedure. In turn the UK nuclear regulator has considered each case on its individual merit, and has not applied 'blanket industry standards or precedence based criteria'. This has allowed the regulator full flexibility in addressing the unique complex issues that surround releasing land from nuclear regulatory control.

These programmes are sub tasks within overall programmes assessing land and building quality. It must be noted that successful exclusion of an area from nuclear regulatory control does not qualify that land as 'clean' simply that it complies with a release criteria established by the licensee, tested and approved by the nuclear regulator. To fully characterise and quantify land for re-use requires a range of additional tools and techniques to ascertain the condition and status of other contaminants (non-radiological) that may exist. Further details can be found by accessing web sites listed in the references section.

Description of the nuclear facility

Question 1: Description of the nuclear installation

WINFRITH

Delicensing at the Winfrith site is based on a phased approach releasing land from regulatory control from the east to the western part of the site. The current delicensing programme is phased over a ten-year period. Although the programme remains sufficiently flexible to be modified if required. The recent successful deregulation of land required UKAEA to decommission a number of active facilities and sub systems. These facilities included alpha material chemistry laboratories, active material stores, active discharge ventilation systems together with extensive below ground drainage systems that handled active waste, together with large open plan land areas.

HARWELL

Currently it is planned to delicence the North Eastern Area of the Harwell site. This area has been sub-divided to allow for decommissioning activities to be completed. The first area, referred to as the Pilot Area, has a history of use from 1946. Around 44 buildings have been constructed and demolished since that time and have had a variety of uses. Only nine of them involved the use of radioactive materials and only one of those remains in use today but not for work with ionising radiations.

Only the delicensing case for the Pilot Area has been submitted to the regulator.

Question 2: What is the size of the site which was or will be released? Is this the entire site, or just part of it?

Winfrith: Phase 1 delicensing is 4ha. The remaining phases total approximately 40ha.

Harwell: The Pilot Area is a 7ha area. The North Eastern Area is a 25ha area.

Site Contamination

Question 3: Is the radionuclide vector/radiological fingerprint representative for the site known?

Assuming that nuclide vector refers to the radiological fingerprint of the proposed delicensing area, then in the first part of the question the answer is Yes. However there are two parts to a radiological fingerprint. The first part being the assumed fingerprint that drives the radiological monitoring regime, and the latter being the actual fingerprint derived from physical sampling and analysis. In the first part the radiological fingerprint was determined not only from the current and historic operations within the area, but also considered the radioisotopes in use in facilities that border onto the delicensing zone. Accidents and incidents that may have occurred both within and outside the facilities with non-building specific RAM being moved around the area were also considered. The fingerprint also took cognisance of the decommissioning that was taking place and the extensive data that confirmed that the ex-lab areas satisfied the delicensing criteria and that decommissioning did not give rise to any spread of contamination beyond the building demise.

WINFRITH: Within this specific case the assumed radiological fingerprint gave rise to a fingerprint ratio of beta/gamma to alpha of 99:1% contribution, with a composition of isotopes including Co 60, Cs 137, Am 241, Pu, U and Tritium.

At the completion of the field assessment surveys and sampling the final ratio of beta / gamma to alpha was 99.9:0.1%.

HARWELL: The principal nuclides are the same as for Winfrith.

Question 4: In the case of a contamination of the site, is it known how deep the radionuclides may have penetrated into the ground?

WINFRITH: With reference to the facility current and historic operations it was determined that the contamination, if present, would lie within the near surface layers of the land, as the drains within the area had been previously cctv surveyed, radiologically monitored and then removed. After removal the adjacent soils were sampled and surveyed to prove compliance with the delicensing criteria.

HARWELL: Experience has shown that contamination when present is at the surface or at depth only if a drain has leaked or when residual contamination is associated with infill material obtained when demolishing a building.

Question 5: On which basis has been or will the site be released?

Within the United Kingdom a nuclear licence is required for all nuclear sites that undertake prescribed processes such as the operation of nuclear reactors; the storage, processing or disposal of nuclear fuel or other materials produced or irradiated as a consequence of the production of nuclear fuel (excludes fusion). The requirement and conditions for this licence are given in the Nuclear Installations Act 1965 and as amended. The NIA(65) Act includes the conditions for the Variation or Cessation of a Licence (referred to as delicensing).

The requirements for Variation or Cessation are the same, and are both described within the act as: -

NII65, Section 3 - Grant and Variation of Nuclear Site Licenses

Sub section

(6)A variation to a nuclear site license may be granted by excluding any part of the licensed site :-

(a) Which the licensee no longer needs for any use requiring a nuclear license

(b) And where no danger from ionising radiations exists on that part of the site.

No further definition or clarification is available within the UK Nuclear Regulatory framework. NII65,S3,(6)(b) introduces the requirement for 'no danger', currently there have been no judicial reviews of what 'no danger' means, current interpretation by both UKAEA and the Nuclear Regulator utilises a second piece of legislation The Radioactive Substances Act 1999 (RSA99), and the Substances of Low Activity Exemptions Orders (SoLA) to determine no danger, supported by further requirements to demonstrate local contamination levels are within a defined criteria and that no anomalous variation in the background radiation dose level exists.

Question 6: **Please give examples of clearance levels of the relevant nuclides which will be used / have been used for the release of (a) site(s).**

UKAEA Corporate Criteria

Current levels utilised by UKAEA in recent applications have been derived from the RSA / SoLA regulations where by total activity measured within the delicensing zone must not be greater than 0.4Bq/g in excess of the natural background activity level. In addition surface contamination must not be greater than 4Bq/cm² beta / gamma and 0.4Bq/cm² alpha, finally all radiation levels must be within background levels.

Question 7: Which measurement techniques will be/have been applied for clearance measurements?

A number of techniques have been used, these include in-situ gamma spectroscopy using high purity germanium detectors (HPGe), portable hand held gamma spectrometers (NaI detectors), for specific point surveys, supported by high sensitivity low resolution gamma surveying using NaI detectors supported with data loggers and global position (GPS) to provide high quality large surface area surveying.

Where land is considered not to have been impacted on i.e. 'no disturbance / excavations throughout lifetime of site' then multiple samples were bulked to give background values, other wise all physical samples were unique to that location and were recorded with positioning data from either conventional surveying or via global positioning system (GPS)

Where there was the potential for contamination at depth (from historical records) trenches / trial pits were excavated and samples were surveyed in the field using the above techniques as well as being radiochemical analysed in the laboratory.

Laboratory analysis was undertaken by external organisations that could demonstrate appropriate quality standards such as ISO9000 and UKAS accreditation. Where a process was not accredited UKAEA undertook technical audits using its own in-house experts to assess the quality and integrity of the techniques used.

Question 8: What averaging criteria will be / have been used for the release of (a) site(s)?

No application specific averaging criteria was cited, other than within the criteria of Bq/g or Bq/cm² as defined in the release criteria.

Additional references to open literature

Robinson I F and Simister D N (1999) – A regulatory view of Delicensing at UK nuclear sites. Proceedings of the 6th Society of Radiological Protection International Symposium, Southport, June 1999.

Robinson I F (1999) - A nuclear inspectors perspective on decommissioning at UK nuclear sites. Journal Radiological Protection Vol. 19 No. 3 p203 - 212

Robinson I F (2000) Delicensing Nuclear Sites in the UK. Proceedings of the 10th International Radiation Protection Association Congress, Hiroshima, May 2000.

Simister D N and Denison W (1999) - Delicensing of the F2 Chemicals limited Facilities from the Nuclear Site at BNFL Springfields, Knoxville Conference Proceedings 1999.

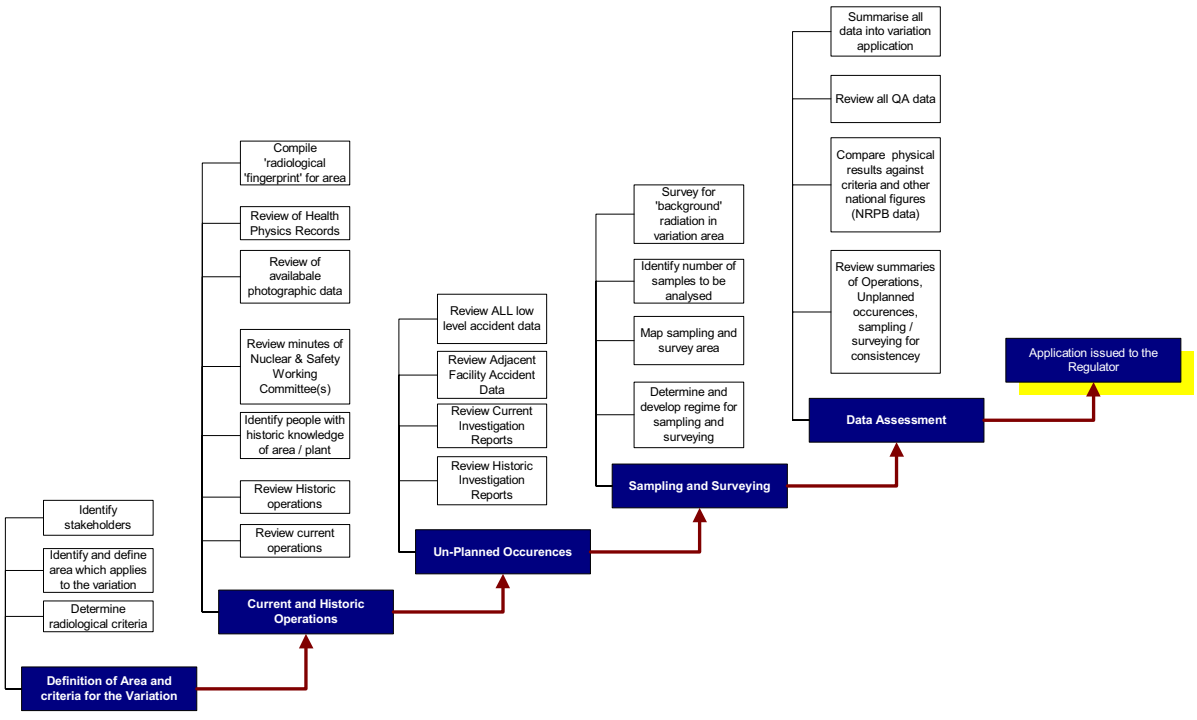
Burgess P H (1998) - Handbook on measurement methods and strategies at very low levels and activities, PH Burgess, published in Nuclear safety and the environment, EU17624, 1998.

Jessop G (2002) - Recent UKAEA Success In Partial Site Deregulation (Delicensing), Proceedings of the EPRI International Decommissioning and Radioactive Waste Workshop, Dounreay, September 2002.

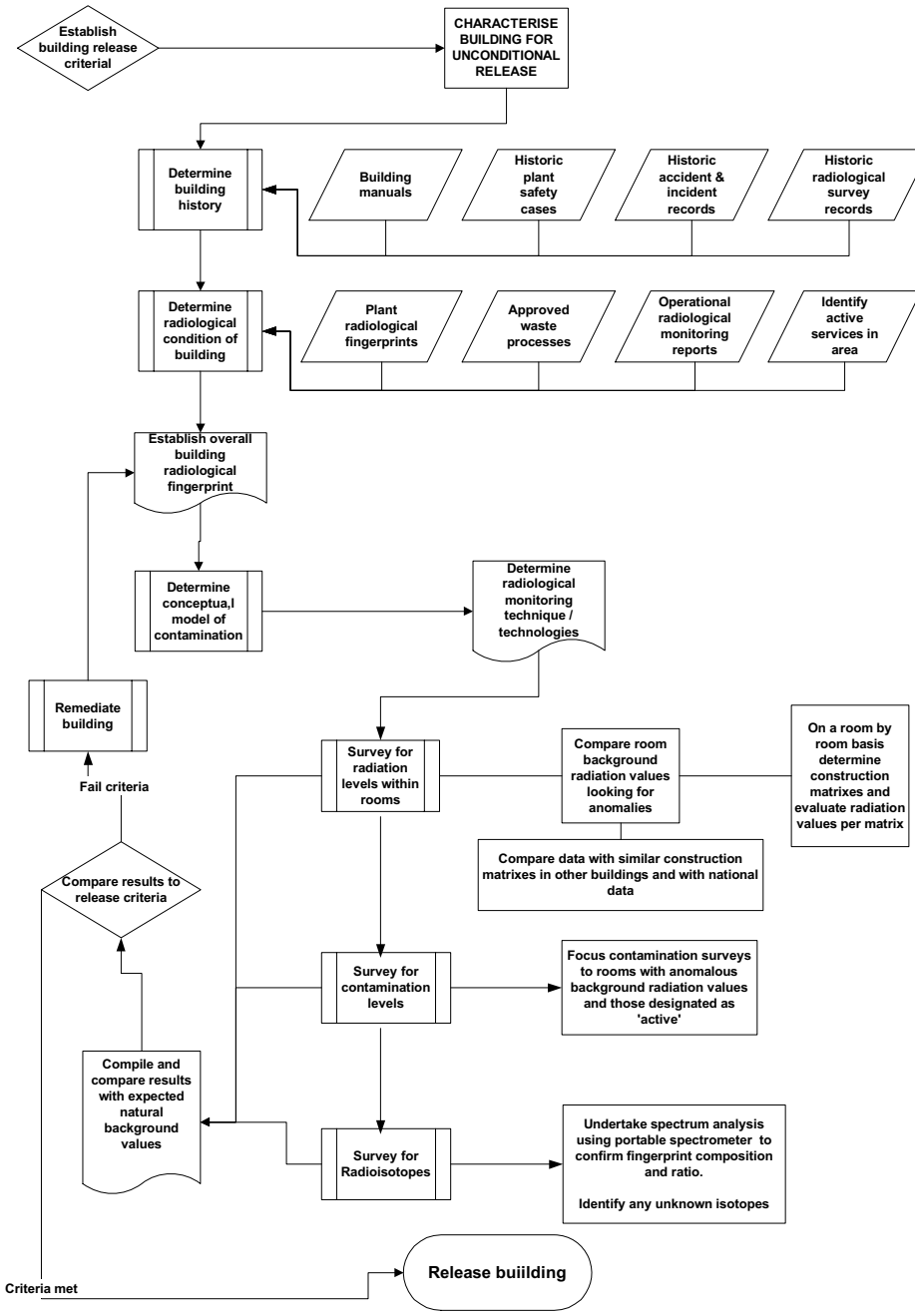
Web Sites

http://www.dti.gov.uk/nid	UK Government approach to nuclear installations within the UK
http://www.ukaea.org	UKAEA web site
http://www.nrpb.org.uk/	Independent reports on distribution of isotopes and their levels within the UK pre and post Chernobyl
http://www.hse.gov.uk/nsd	
http://www.safegrounds.com/guidance	UK Nuclear Industry Working group tasked with identifying acceptable principles and processes to characterise ex-nuclear facilities for the presence of, or management of potentially contaminated sites (rad and chem).
http://www.environmentagency.com	Source of available environmental regulatory guidance, or signpost to other web sites for threshold levels for contamination within soils / groundwater

PROJECT WORK BREAKDOWN STRUCTURE FOR A DELICENSING PROJECT



METHODOLOGY FOR THE RELEASE OF BUILDINGS WITHIN A DELICENSING ZONE



ANSWERS FROM THE USA

Although no specific answers to the questionnaire on site release were received from the USA, there is still a summary on site release which has been compiled for the document "Removal of Regulatory Controls for Materials and Sites – Topical Discussion held at the 6th Meeting of the RWM-RF in March 2003" of the Radioactive Waste Management Committee of the Nuclear Energy Agency. The summary on site release provided for this document is reproduced here because it responds to a significant part of the questionnaire.

- The U.S. Congress designated the U.S. Environmental Protection Agency (USEPA) as the primary federal agency for establishing environmental standards for the protection from harmful and avoidable exposures to radiation. Nuclear facilities can be subject to clean-up under the Comprehensive Environmental Response, Compensation, and Liabilities Act, which utilizes a risk range of 10^{-4} to 10^{-6} to determine appropriate levels of cleanups for radioactive and hazardous materials at those sites *under sole USEPA authority.*
- The U.S. Nuclear Regulatory Commission implements standards by promulgating safety regulations for source, special nuclear, and byproduct material that are expressed in annual total effective dose equivalents, as well as air and liquid effluent concentrations for restricted and unrestricted areas. In general, NORM and TENORM are not within USNRC's jurisdiction.
- Under USNRC regulations for protection of individual members of the public, in general, the total effective dose equivalent from all licensed sources and practices is not to exceed 1mSv/yr (0.1 rem/yr). This dose is exclusive of background radiation, any medical therapy contributions, a licensee's disposal of radioactive material into sanitary sewerage, or other contributions not attributable to licensed operations.
- USEPA regulates some specific TENORM wastes, and has regulatory standards and guidance which are applicable to limiting public and environmental exposures, site cleanup, and waste disposal for other forms of NORM and TENORM. These regulations and guidance are utilized in most States to limit exposures to these wastes.
- Upon its decommissioning and license termination, a USNRC licensee is held to a fraction of the 1 mSv/yr (100 mrem/yr) public dose limit. USNRC regulations in 10 CFR Part 20, Subpart E specify a dose constraint for decommissioning and license termination of 0.25 mSv/yr (25 mrem/yr) and demonstration that the residual contamination levels are ALARA. This dose constraint is considered protective of public health and safety and the environment, and the appropriate allocation of the public dose limit for license termination, which is assumed to occur once for a NRC-licensed facility.
- Currently, a Memorandum of Understanding applies to the cleanup for those sites having overlapping jurisdiction under USNRC and USEPA.
- Under the authority of the Atomic Energy Act, the USNRC has relinquished regulatory authority for non-nuclear power plant licenses to many States through written agreement, and this includes site cleanup. Many States have comprehensive radiation control programs that are considered adequate to protect public health and safety.
- Sources of NORM and TENORM may be covered in States by either specific regulations for those materials, or general radiation protection standards.

- The U.S. Department of Energy has radiation protection regulations that are similar to those of the USNRC and USEPA, such as USDOE Order 5400.5, "Radiation Protection of the Public and the Environment." USDOE regulations are applicable to facilities and sites which they own.