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RADIOACTIVE WASTE MANAGEMENT COMMITTEE

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Working Party on Decommissioning and Dismantling (WPDD)

Topical Session on Materials Management

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WPDD TOPICAL SESSION ON MATERIALS MANAGEMENT

I INTRODUCTION

Set up by the Radioactive Waste Management Committee (RWMC), the WPDD brings together senior representatives of national organisations who have a broad overview of Decommissioning and Dismantling (D&D) issues through their work as regulators, implementers, R&D experts or policy makers. These include representatives from regulatory authorities, industrial decommissioners from the NEA Co-operative Programme on Exchange of Scientific and Technical Information on Nuclear Installation Decommissioning Projects (CPD), and cross-representation from the NEA Committee on Nuclear Regulatory Activities, the Committee on Radiation Protection and Public Health, and the RWMC. The EC is a member of the WPDD and the IAEA also participates. This ensures co-ordination amongst activities in these international programmes. Participation from civil society organisations is considered on a case by case basis, and has already taken place through the active involvement of the Group of Municipalities with Nuclear Installations at the first meeting of the WPDD

At its second meeting, in Paris, 5-7 December 2001, the WPDD held two topical sessions on the D&D Safety Case and on the Management of Materials from D&D, respectively. This report documents the topical session on the management of materials.

Engenio Gil Lopez, Deputy Director for Environmental Radiological Protection Consejo de Seguridad Nuclear (CSN), Spain, served as Session Chair. Richard Ferch, Director, Wastes and Decommissioning Division Canadian Nuclear Safety Commission, Canada, served as the rapporteur for the Topical Session.

Presentations during the topical session covered key aspects of the management of materials and meant to provide an exchange of information and experience, including:

Experience and lessons learnt from VLLW and non-radioactive material management in Spain and Germany with special attention to recycling

- How specific solutions came about?
- Are there ‘generic’ examples for wider adoption?

Risk assessment of recycling and non-recycling : a CPD study

Waste acceptance issues within different national contexts

- What constraints are there on the waste receiving body and what flexibility can the latter have?
- What constraints does this impose on D&D implementors ?
- What about wastes are without current solution? What needs to be done ?

- What about large items and “difficult” waste in general?

Radiological characterisation of materials during decommissioning, particularly difficult situations (large volumes, large items, α/β wastes, heterogeneous streams)

- What examples of established practice?
- What are the approaches or aspects that set the regulatory requirements?
- How can the flow rates be large but the answers acceptable?
- How much is needed to be known for later action, e. g., disposal, release, protection of worker, etc.

Radiological characterisation of buildings as they stand, in order to allow conventional demolition.

- What are strategies for optimisation of characterisation?
- How much needs to be known to take action later? e.g. for storage, disposal, release, cost estimation and ALARA?
- What needs to be done in advance and after decommissioning/dismantling?

At the end of each presentation time was allotted for discussion of the paper. Integral to the Topical Session was a facilitated plenary discussion on the topical issues identified above. The rapporteur briefly reviewed the main points at the end of the topical session.

The Topical Session is documented as follows. First a summary of the presentations is given along with the questions that were asked of each speaker; then follow a summary of the plenary discussions and the main points made. The extended abstracts or full papers supporting each presentation are given in Appendix 1.

As a follow-on to the Topical Session a Task Group has been constituted in order to propose to the WPDD a more detailed work programme in this area.

Acknowledgements

Special thanks are due to Richard Ferch for serving as rapporteur of this Topical Session, for liaising with the other WPDD members, and for providing the Secretariat with the relevant materials of this document.

II SUMMARY OF THE PRESENTATIONS

The presentations in this session were divided into five groups, as follows:

1 Experience and lessons learnt from VLLW and non-radioactive material management in Spain and Germany with special attention to recycling

In the first of two presentations in this group, Alejandro Rodriguez described the process used by ENRESA for triage and processing of waste arisings from the decommissioning of the Vandellos 1 reactor. In the second paper, Stefan Thierfeldt described German clearance requirements and lessons learned from their application to decommissioning of power reactors. In both cases, the great majority of the waste was in fact recycled, although the costs were dominated by the small fraction that had to be disposed of as radioactive waste. Among the lessons learned from the German experience were some revealing insights into the public perception of both clearance and of recycling of material from decommissioning of a nuclear installation.

2 Risk assessment of recycling and non-recycling: a CPD study

Here Shankar Menon described the work of a CPD Task Group on Recycling and Reuse, which demonstrated that the net risk of recycling is less than the net risk from disposal and replacement of the material disposed of. He also made several interesting and provocative observations on inconsistencies in regulation of various types of radioactive waste.

3 Waste acceptance issues within different national contexts

There were three papers presented in this group. Harald Maxeiner described clearance and waste conditioning requirements in Switzerland, and their impacts on decommissioning. Peter Lock described some particular cases which had given rise to difficult acceptance issues at NIREX, ranging from large size items to the impacts of chemicals used during decontamination on the mobility of radionuclides in a disposal facility. Eric Lanes described two facilities in France which will receive waste from decommissioning: Centre de l'Aube, for disposal of relatively short-lived low- and intermediate-level waste, and a proposed new facility or facilities to receive VLLW, mainly from D&D.

4 Radiological characterisation of materials during decommissioning, particularly difficult situations (large volumes, large items, α/β wastes, heterogeneous streams)

In the first of two papers in this group, Luis Valencia described in detail the technical aspects of radiological characterisation of materials arising during the decommissioning of research reactors. Scott Moore then described the dose-based radiological characterisation process used in the USA, and four current characterisation issues faced there. Both papers emphasized the importance of characterisation to control decommissioning hazards and costs.

5 Radiological characterisation of buildings as they stand, in order to allow conventional demolition

Two contrasting processes for demolition and clearance of buildings were described. The first was presented by Satoshi Yamagihara, describing the experience at the Japan Power Demonstration Reactor from a fairly conventional process of characterisation, decontamination, and conventional demolition of the remaining structure. The second presentation from Manfred Schrauben described a special-case process used by Belgoprocess to decommission a building where a more conventional process ran into difficulties: namely, the complete demolition of the building, crushing of the concrete, and clearance of the resulting rubble.

III SUMMARY OF THE DISCUSSIONS

One area of quite general agreement was the importance of characterisation, including reviews of historic site documentation, highlighting the importance of operational documentation for later use during decommissioning.

Another strong common element during the discussion was general agreement on the large costs of disposal of the radioactive component of waste arisings, and of radiological control generally, and therefore of the importance of actions which control or reduce those costs.

Another issue which also give rise to economic concerns, among others, was that of uniformity of limits to regulatory control, especially including clearance levels. This was an area where there seemed to be general agreement on the importance of the problem, but relatively little consensus on the way forward. It is also an area of much broader impact than decommissioning.

APPENDIX 1:
PAPERS AND EXTENDED ABSTRACTS

VANDELLÓS 1 NNP DECOMMISSIONING MATERIALS MANAGEMENT

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Materials management

The management of materials generated during dismantling is undoubtedly one of the major tasks to be undertaken by ENRESA. In this respect, a specific inter-departmental organisation model has been developed with a view to guaranteeing complete efficiency in the production, characterisation and treatment of the large volume of materials generated at the site.

Specifically, during Phase 2 of the Dismantling, 296,000 tons of materials will be produced, less than 1% of which (2,000 tons) will be managed as low and intermediate level radioactive wastes. For this reason, one of the essential points of the project is the exhaustive control of all the materials arising at the site, in order to segregate those considered clean from those others that have radiological implications.

In keeping with this operational logic, the materials managed are classified into two groups: those coming from conventional zones and those from active zones. Those belonging to the first group have never been in potentially contaminated zones and as a result have no radiological implication. Consequently, the legal standards in force in Catalonia are applied for their removal from the site and subsequent recycling or disposal at authorised tips.

For their part, materials from active zones are divided into declassifiable and radioactive. Declassifiable materials are those which are candidates for eventual management as conventional wastes, on the basis of a series of historical and operational parameters. This is accomplished through the so-called Declassification Process, which is required to demonstrate the absence of levels of activity in excess of those authorised by the CSN, the regulatory body. For their part, radioactive materials are meticulously characterised and conditioned for subsequent dispatch to the Low and Intermediate Level Waste Disposal Facility at El Cabril (Córdoba).

As the different levels of materials management are performed, the materials move around the site in containers along controlled routes, in all cases accompanied by their corresponding AHU (Authorised Handling Unit) docket. These dockets are completed in the different areas through which the containers circulate, and specify all the historical, radiometric and operating data that need to be known in subsequent phases to ensure optimum management.

In this respect, it should be pointed out that the management of the materials generated during the dismantling of Vandellós I implies joint efforts by all the different departments, integrated in actuation areas. Specifically, the Performance, Operations, Radiological Protection, Waste Management and Decontamination Services participate directly in the process, with their work coordinated by the Materials Control Service, dedicated exclusively to guaranteeing exhaustive control of all materials dismantled at the site.

All these departments are integrated in the Production, Declassification, Declassified/Conventional Materials Treatment and Radioactive Waste Treatment areas, such that – as shown in the table - the maximum degree of specialisation be maintained throughout the process.

The objective of the Production Area is the generation of homogeneous batches of materials susceptible to being measured with the available technology, as well as the preliminary classification of materials depending on the authorised contamination limit values. The Declassification Area confirms and accredits materials initially catalogued as de-classifiable and, if this is not applicable, assigns the corresponding category to them.

Finally, the Declassified/Conventional Materials Treatment and Radioactive Materials Treatment areas are in charge of the conditioning of the materials and their dispatch to authorised centres or to the El Cabril Low and Intermediate Level Waste Disposal Facility, respectively.

This complex process is controlled by means of the so-called Waste Management System (WMS), a computer-based corporate system that records all internal movements of the materials, from disassembly to dispatch. Furthermore, in the case of radioactive wastes, the surveillance included in the WMS continues during transport and covers up to final disposal at the El Cabril facility.

Process guarantees

In order to meet the objective of minimising the production of radioactive wastes from Level 2 Dismantling, a rigorous segregation and decontamination plan has to be put into place, and complete efficiency must be guaranteed throughout the process. In this respect, the site has five controls that are applied to all materials considered to be candidates for declassification, in other words those coming from active zones and to be removed from the site and sent to conventional destinations. Only such meticulous treatment can ensure that all the materials removed from the plant do not exceed the levels of activity imposed by the CSN for declassification as non-radioactive wastes.

The objective of the first two controls applied to a material, item of equipment or system is planning of the disassembly work and of the protection resources for the workers involved. These controls consist of historic knowledge of the operation of the equipment or system and analysis of the three-radiometric studies already performed on site.

Historic knowledge of the operation of the installation is a resource that is used as the main documentary means of determining which areas of the site are susceptible to including contaminated materials. This information may be used to perform an initial selection which, backed by analysis of the three radiometric studies, will allow time and resources to be optimised in the disassembly tasks.

For their part, the three radiometric studies, each more accurate than its predecessor, make up a real detailed radiological map of the site, since they were drawn up on the basis of more than 7,000 direct measurements performed prior to initiation of dismantling of the active parts of the plant.

The third control is aimed at checking on the spot, by means of direct measurements, which materials are radiological clean and which are contaminated. This *in situ* characterisation control is performed by the Radiological Protection technicians using portable measuring equipment and means the need to initially control all the materials in the working areas in which they are generated. Only after having obtained a positive result from these three initial controls may a material be considered to be presumably clean. In this case, it will be conditioned in its corresponding MDC (Measurement and Declassification Container) and subjected to the process of declassification. Otherwise, the material will be treated as a radioactive waste.

In order to undertake this declassification, which will allow non-contaminated materials to be managed as conventional waste, the fourth control is implemented, this being designed to certify the efficiency and quality of the process. The control consists of performing integrated measurement of the containers using a sophisticated device known as the Box Counter, which analyses the radiological charge of the material contained in the MDC by means of a gamma spectrometry measuring system. Only when the box counter has ratified once more that the material does not exceed the levels established by the CSN is the latter declassified by the Radiological Protection Service.

Finally, all the non-radioactive materials leaving the site are required to pass through a large gantry, now on the transport truck. This is located at the exit from the site and definitively checks that there are no radioactive components in the material prepared for dispatch.

Once these five controls have been performed with satisfactory results, conventional materials are given a permit to leave the site and be transported to their destination, either a recycling plant or an authorised tip. However, in keeping with the legal standards in force, all conventional wastes are required to have an Acceptance Docket subscribed between the producer, in this case ENRESA, and the company or organisation responsible for subsequent management. This acceptance docket must be submitted to the Waste Council of the Regional Government of Catalonia for validation, and must be associated with a Waste Tracking Sheet, the aim of which is to ensure that transport is carried out under suitable conditions.

EXPERIENCE AND LESSONS LEARNT FROM CLEARANCE IN GERMANY

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Introduction

Clearance is an important corner-stone of waste management in nuclear installations in Germany. It has been practised successfully for around two decades. The importance of clearance can also be judged from the fact that it has been included detailedly in the new German Radiation Protection Ordinance (section 0).

Currently, there are 19 NPPs and a number of fuel cycle installations in operation in Germany. The main waste quantities, however, arise from the dismantling of NPPs and fuel cycle installations which are currently in decommissioning (section 0). As most of the decommissioning projects are targeted for early dismantling instead of safe enclosure, there is already considerable experience with the application of clearance procedures and verification of clearance levels (section 0).

The German waste management strategy is governed by two options:

Clearance of the material (after decontamination and release measurement). After clearance, the material is no longer regarded as radioactive in a legal sense.

Final disposal of the material as radioactive waste in a deep geological repository (no near-surface disposal facilities exist or are planned in Germany). A deep geological repository is planned to be operable around 2030.

Clearance Levels in the German Radiation Protection Ordinance

In Germany, clearance of material originating from the operational or decommissioning phase of nuclear installations as well as from other authorised use of radioactive material is regulated in the Radiation Protection Ordinance (RPO) of July 2001 0 which is based on the Atomic Encergy Act 0. The RPO transforms the EURATOM Basic Safety Standards (BSS) 0 into national legislation.

The clearance regulations in the new RPO are characterised as follows: One set of clearance levels (CL) exists for each of the clearance options which are described in the following. Each set contains CL for about 300 nuclides (those for which exemption levels exist in the BSS). All sets of CL are derived on the basis of 10 µSv/a individual dose. - A number of recommendations of the SSK (German Commission on Radiological Protection) preceded these regulations of clearance in the RPO, the first having been issued as early as 1988.

The following clearance options are contained in the German RPO:

I. Unconditional clearance

1. of all solid materials for reuse, recycling or disposal including building rubble of less than 1000 Mg per year,
2. of building rubble and soil of more than 1000 Mg per year,
3. of buildings for reuse or demolition,

4. of nuclear sites (after removal of the buildings);

II. Clearance

1. of solid materials for disposal on landfills or for incineration,
2. of buildings for demolition only,
3. of metals for melting only.

The clearance options listed under category no. 1 are termed “unconditional clearance” because no restrictions concerning the destiny of the material exist after clearance (i.e. the material need not be traced to a final destination). Those options, however, would not be termed “clearance” according to IAEA terminology because they are related to specific materials and may have mass limits attached. (According to IAEA, the term “clearance” should be used only for the release of any kind of material without any conditions on material type or quantity 0 which renders the qualifier “unconditional” superfluous.)

The clearance options listed under category no. 2 could be called “conditional clearance” because certain conditions must be fulfilled before clearance can be achieved. However, it is more appropriate to speak of “clearance of material for a certain purpose”.

The clearance regulations in Germany have a complex structure. They are, however, one of the most advanced and comprehensive worldwide. Their applicability has been successfully demonstrated in various licensing procedures, especially in decommissioning projects of nuclear power plants, research reactors and fuel cycle installations. Brenk Systemplanung has been involved in deriving all sets of CL on behalf of the German Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (*Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit, BMU*).

The basis for the derivation of all sets of clearance levels has been compiled in 0 and shall not be repeated here. The following table contains some examples of clearance levels for a few relevant radionuclides.

Table 1: Examples of clearance levels for a few relevant radionuclides (German Radiation Protection Ordinance)

Clearance option	H 3	C 14	Fe 55	Co 60	Cs 137	I 131	U 234	Pu 242	Am 241	unit
I.1 unconditional	1000	80	200	0.1	0.5	2	0.4	0.04	0.05	Bq/g
I.2 building rubble	60	10	200	0,09	0.4	0.6	0.4	0.04	0.05	Bq/g
I.3 buildings, reuse	1000	1000	1000	0.4	2	10	1	0.1	0.1	Bq/cm²
I.4 nuclear sites	3	0.04	6	0.03	0.06	0.2	-	0.04	0.06	Bq/g
II.1 disposal	1000	2000	10,000	4	10	20	9	1	1	Bq/g
II.2 buildings f. demolition	4000	6000	20,000	3	10	600	10	2	3	Bq/cm²
II.3 metal scrap f. recycling	1000	80	10,000	0.6	0.6	2	2	0.3	0.3	Bq/g

Waste Management Experience

Nuclear Power Plants

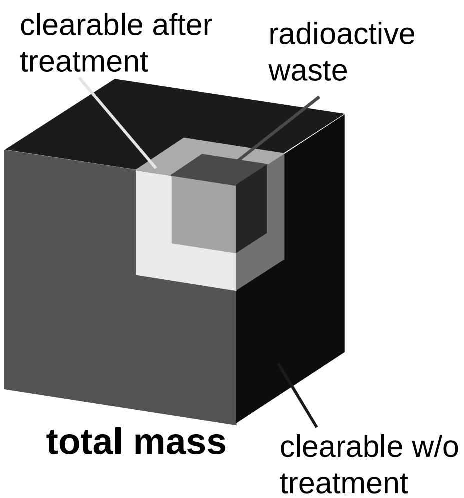
Currently there are 19 nuclear power plants (NPP) in operation in Germany. The list of larger NPP decommissioning projects contains smaller NPPs like

- VAK Kahl (BWR, 16 MWe, operation 1961 - 1985),
- KRB-A Gundremmingen (BWR, 250 MWe, operation 1966 - 1977),
- KKN Niederaichbach(GCHWR, 100 MWe, operation 1972 - 1974, completely dismantled) as well as larger NPPs like
- KGR Greifswald (5 block VVER, 440 MWe each, operation 1973/89 - 1990),
- KWW Würgassen, (BWR, 670 MWe, operation 1971 - 1995).

The total waste masses differ according to reactor size and reactor type. However, one common feature can be found amongst all of these decommissioning projects: the radioactive waste which has to be finally disposed of (in a deep geological repository) amounts to only a few percent (typically 3 to 5%) of the entire mass of the controlled area. The rest of the material is decontaminated (if necessary) and will be clearable.

For a large NPP of modern design (total mass of controlled area 150,000 - 200,000 Mg) it is estimated that between around 3,000 and 10,000 Mg will have to be disposed of as radioactive waste. This estimate is supported by experience gained from the above mentioned decommissioning projects. The following figure illustrates the relationship between material for clearance and radioactive waste.

Figure 1: Illustration for percentages of clearable material vs. radioactive waste from decommissioning of nuclear installations in Germany



Fuel Cycle Installations

A number of German fuel cycle installations are currently undergoing decommissioning, e.g. some fuel fabrication plants at Hanau, the WAK pilot reprocessing plant at Karlsruhe, hot cells etc. The waste quantities most of these plants are considerably smaller than from NPPs. However, also for fuel cycle installations only a few percent of the material of the controlled area have to be disposed of as radioactive waste while the rest is cleared.

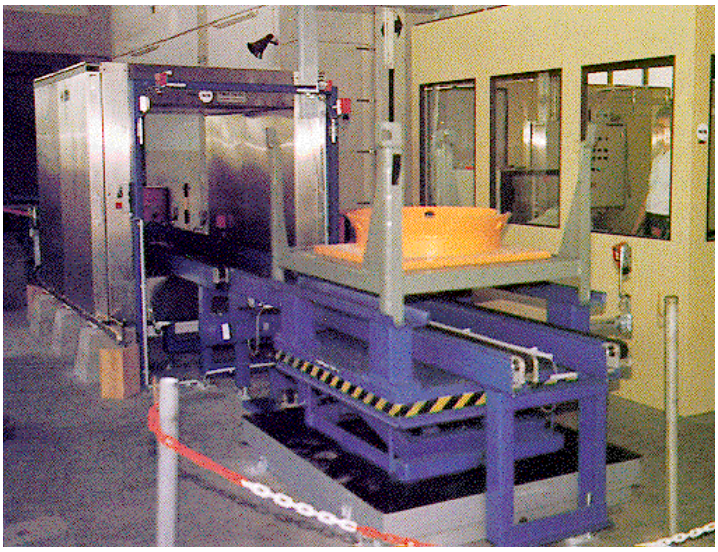
Lessons Learnt

Applicability of Clearance Levels and Clearance Measurements

It has been found that the German clearance levels presented in section 0 can be measured for each clearance option. A number of measurement techniques have been developed for this over the last decades. The most important measurement device is the so-called release measurement facility (rmf) where the material (typically up to a few 100 kg or even 1 Mg) is surrounded in the measurement chamber by large-area detectors (usually 24) on all 6 sides. The gamma measurement is fast and accurate allowing a daily throughput of up to 10 Mg.

A number of other measurement techniques for surface and mass specific activity exist. Comparatively new developments are the in-situ-gamma spectrometry (for building surfaces and sites) and the cobalt coincidence monitor which is especially sensitive to Co 60, the key nuclides in most NPPs. Correlation of nuclides which are hard to measure with easy-to-measure nuclides is also a common approach.

Figure 2: Example of a release measurement facility (rmf)



It can be concluded that the clearance levels have proven to be applicable for all types of decommissioning projects. Measurement techniques are available.

Costs for Clearance

The costs for clearance are significantly lower than for final disposal of radioactive waste. It is not easy to provide a really comprehensive and reliable cost estimate but the following figures may illustrate the situation:

In order to achieve unconditional clearance of metal scrap, total costs of 9 to 15 EUR/kg may be estimated. This includes decontamination, release measurement and personnel.

Costs for reuse of recycling in the nuclear field (i.e. melting under radiological control) may amount to around 12 to 15 EUR/kg.

Costs for final disposal can be estimated in the range of 50 to 250 EUR/kg including conditioning, waste package, interim storage and final storage. The costs depend to a large extent on the activity contents and the type of waste container chosen.

Clearance is therefore the cheapest option and can be applied for nearly the entire mass of the plants.

Public Opinion

All recommendations and regulations on clearance as well as all supporting documents describing their derivation have always been published. However, it seems that for many years the fact that clearance was taking place (even on a large scale) has not been widely known in the general public. It has even caused concern within the steel industry only to such an extent as the residual radioactivity contents in the material would be detectable in the entrance monitors (scrap which has been cleared on the basis of unconditional clearance levels, e.g. 0.1 Bq/g Co 60, will however not be detected).

Clearance has become an issue in the general public only when the draft of the Radiation Protection Ordinance has been presented for public discussion in 1999/2000. Some environmental groups have developed and published a number of scientifically absolutely unjustified scenarios calculating horrifyingly large doses. Recent endeavours of the German Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (*Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit, BMU*) to bring the discussion back to a scientific basis have, however, been fruitful. A working group which is dedicated to clearance has been installed under the German Commission on Radiological Protection (*Strahlenschutzkommission, SSK*) in September 2000. This working group (as well as other discussion groups) provide fora for information exchange among all stakeholders.

Conclusions

Clearance is an important part of waste management in Germany. The regulatory framework for clearance is well advanced and is in agreement with recommendations of the European Commission. Clerance of the material is much less expensive than disposal as radioactive waste. Decision for clearance is therefore mainly driven by cost considerations. The clearance procedure (including decontamination, measurements etc.) is meanwhile very advanced so that 95% or more of the total mass of decommissioning projects can be cleared.

Literature

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RISK ASSESSMENT OF RECYCLING AND NON-RECYCLING

S. Menon

Quite early during the operation of the NEA Co-operative Programme on Decommissioning, it became obvious that the management of the large volumes of contaminated materials arising from the decommissioning of nuclear facilities represents one of the most substantial cost fractions of such projects. Consequently, the minimisation of the volumes that have to be disposed of as radioactive waste is a high priority goal for decommissioners. It was also noted that much of this redundant material was valuable, e.g. stainless and other high quality steels. The recycling of such material (or its reuse or disposal) without radiological restrictions could be a significant means of achieving the aim of waste minimisation. At the same time, such an approach would help conserve resources and have environmental advantages. So, in 1992, the Programme set up a task group to study the recycling and reuse of redundant material from the decommissioning of nuclear facilities.

The Co-operative Programme's Task Group on Recycling and Reuse made a survey of the current practices and national regulations in this area, studied the technologies associated with recycling and analysed the proposed international recommendations and proposals for release criteria. A report of the work of the Task Group was published in 1996.

Unlike other international organisations like the IAEA and the EC recommendations that consider only the radiological risks associated with the release of material, the Task Group assessed the total health risks, comparing the radiological risks associated with the recycling of material with the risk of disposing the material instead as radioactive waste and replacing it with new material. The results of this comparison showed that

The hypothetical radiological risks associated with both alternatives are very small in comparison with the non-radiological industrial safety risks, these non-radiological risks are much lower for recycling because product manufacture starts from scrap metal. The risks associated with mining and refining of metal are avoided.

Since the publication of the report, the activities of the Task Group, with the Technical Secretary in particular, have more or less been concentrated on promulgating the views expressed in the report and on the issue of technologically enhanced naturally occurring radioactive material (TENORM) that has emerged during the last decade, which can have a very significant impact on clearance regulations. Studies have shown that TENORM can be of the same activity levels as low level waste and is very similar to the candidate material for exemption and clearance in the nuclear industry, but occurs in many non-nuclear industries in quantities that are huge in comparison (2-3 orders of magnitude larger quantities than those used in European studies on nuclear recycling).

The appearance of these huge quantities of TENORM on the scene made it impossible to justify the use of the current regulatory approach of "exemption" (for smaller quantities) and "clearance" (for large quantities), in connection with the exemption/release of material from radiological regulation.

Both the EC and the IAEA seem, at present, to be proposing double standards of 10 μSv/year individual dose criterion for release of material from the nuclear industry and 300 μSv/year for the orders of magnitude larger quantities of material from the non-nuclear industries. In doing this, a message is being sent to the public that nuclear radioactivity is up to 30 times as dangerous as TENORM radioactivity, which is directly in conflict with the US National Academy of Sciences, which has stated that there is no plausible rationale for any difference in risks from naturally occurring or any other radionuclides.

The glaring inconsistency in the regulatory treatment of radioactivity (and the consequent doses to the public) in the nuclear industry and the non-nuclear industries is illustrated clearly in the proposed clearance levels for Ra 226 in the EC document Radiation Protection 122. In Part I, which covers “practices” (i.e. the nuclear industry), the prescribed clearance level in 0.01 Bq/g. In Part II, which deals with “work activities” (i.e. TENORM industries. This is still a draft.), the proposed clearance level is 0.5 Bq/g, a value that is 50 times higher.

This inconsistency has very significant commercial implications as the nuclear industry is living in a world where electricity is being deregulated and competition between various sources of power production is fierce. The double standards for clearance being proposed by the EC for material from the nuclear industries and for TENORM takes on a special significance when it is noted that two of the largest sources of TENORM are the coal and the oil & gas industries, both major producers of electricity.

The EC proposal of the 300 µSv/year criterion for TENORM has been supported in their guidance document by a number of comparisons/justifications. On closer scrutiny, these justifications appear to be equally valid and relevant for material from the nuclear industry. It can also be noted that even the 300 µSv/year criterion is 2 to 3 orders of magnitude lower than the doses taken for generations by tens of thousands of people living in the high background dose areas of the world, without showing noticeable effects on cancer mortality, life expectancy, chromosome aberrations or immune function.

It is therefore suggested that the proposed EC dose criterion for exemption/clearance of TENORM should also apply for material from the nuclear industry. It is time to do away with inconsistencies and have one unique dose criterion for all types of exposure to ionising radiation, regardless of its source.

Finally, it should be noted that the growing diversity of concepts and levels (exemption, clearance, intervention exemption) has been acknowledged to be “a real cause for confusion for policy makers, users and the public” by the international radiation protection establishment. Both the ICRP and the IAEA have signalled possible changes in the regulatory approach in this area, that is of such importance to the nuclear decommissioning world.

WASTE ACCEPTANCE ISSUES WITHIN DIFFERENT NATIONAL CONTEXTS

E.Lanes and **M. Dutzer,** France

Here is the radioactive waste classification, as it is defined in France.

In the French radioactive waste classification, waste is classified according to its activity level, and the half-life of the main radionuclides that it contains.

This classification clearly shows 4 types of waste. Each kind corresponds to an existing disposal installation, or almost existing, or to an installation under study. D&D will produce mainly ILW-LLW and VLLW.

The first facility that can receive D&D waste already exists. It is dedicated to the intermediate and low-level waste, and is called Centre de l'Aube disposal center. This facility has opened in 1991, to replace the Centre de la Manche, the first facility closed after 25 years of operation, and that after more of 500 000 m³ of waste packages have been disposed of.

The Centre de l'Aube capacity is 1 million of cubic meters. Today, a little bit more of ten percent of this capacity has been used.

The acceptance criteria allow to receive waste from facilities in operating (such as water filters, ion exchange resins, and so on…) of the most active waste from decommissioning.

Actually, the activity of the waste package received is very lower than the specified limits, at least regarding to the short-lived radionuclides.

The D&D implementers can use various kinds of container, according to the activity level, physical nature or the size of waste. The largest part of D&D waste can be packaged in these containers. Of course, a good container for a D&D operator is as large as possible, because it allows to not cut out waste in all small pieces, and thus to save time, money, and dose for operating agents. But, unfortunately, the large packages are difficult to measure, and even sometimes impossible. It is then necessary to find sophisticated measurement systems, when it's possible, and it can be more interesting to make smaller packages and thus easier to measure.

For very large waste, when the standard containers are not suitable, it is always possible to find a specific conditioning and disposal system. For instance, two examples of very large waste at the Centre de l'Aube : the spent fuel storage racks and the reactor vessel heads.

In one of the French NPP, 10 storage racks were deformed by the action of alkaline water on the aluminum which composes neutron absorber material, so that the fuel assemblies could not be introduced any more into the racks because of the deformation.

The racks were too big to be treated like the other waste usually received on the disposal center. Once the packages filled with the injection grout, they were too heavy to be able to be handled. So they were packaged one by one in a specific metallic box. At their arrival on the disposal center, they were directly unloaded on the disposal unit, and the grouting was carried out directly on the vault, as shown on the photos.

Another example: the reactor vessel heads. In 1991, EDF, the utility in France, has detected some cracks on the penetration pipes of a head (pipes which guide the control rods through the vessel head). The investigations have shown that all NPPs could be touched by the defaults. Therefore, EDF decided to replace 55 vessel heads. Each one weighs more than one hundred tons.

Andra has just been licensed by the Safety Authority to receive the vessel heads on the Centre de l'Aube. As for the racks, they cannot be handled once grouted. They also will be grouted directly into their disposal units. Special vaults, with particular dimensions, will be built for the vessel heads.

Some photos to show how the vessel heads were packaged : a first cap upon the penetration pipes, a second one bolted directly on the vessel head, and then a wrap only used for the transportation, and removed for the disposal.

The second disposal center which can receive D&D waste is the Very Low Level Waste disposal facility. It will be built near just besides the Centre de l'Aube, at the end of 2003. It is especially dedicated to D&D waste. Its capacity is 650 000 m^3, and the total amount of activity expected in waste is less than the lower limit for a Basis Nuclear Installation. So that Andra can only ask for an ICPE, which is easier regarding the licensing process.

The waste acceptance criteria are much simpler than those of the Centre de l'Aube, because in this concept, the package doesn't have to be a barrier against the radionuclides migration. The activity limits are high enough to be able to accept a large part of D&D waste.

Once arrived at the disposal center, waste will go directly to the disposal vaults, if it can. It can if its density and its compressive strength are sufficient to avoid collapses on the long-term. If not, first to its disposal, it will go through a treatment building where it will be compacted, or stabilized. The metallic waste will be compacted in blocks by a specific press. Other compressible waste will be compacted in bales. The powdery waste, such as mud, or ashes, will be melted with a hydraulic binder and poured into big-bags for disposal.

The suitable containers will depend on the category of waste: directly disposable or not. For the directly disposable waste, it can be delivered without container if it can be handled and labelled individually, and without risk of contamination for the operating agents. For small waste, or for waste with external loose contamination, a lost container is required. This lost container can be a big-bag, for rubble for example, or a metallic box for waste which can cut the woven material of a big-bag. Several kinds of container may be used for waste which requires a treatment before disposal.

In France, there is one, and soon 2 facilities able to receive D&D waste. One of them has been especially designed for D&D waste. The waste production forecasts shows that, if the disposal capacity available for Low and Intermediate Level Waste seems sufficient, we will need to increase the capacity for VLLW in a few years, when the D&D of the pressurized waters reactors will begin to be carry out.

ACCEPTANCE ISSUES FOR LARGE ITEMS AND DIFFICULT WASTE

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Introduction

The UK strategy for intermediate level and certain low level radioactive waste disposal is based on production of cementitious wasteforms packaged in a standard range of containers as follows:

- 500 litre Drum - the normal container for most operational ILW (0.8m diameter x 1.2m high);
- 3m^3 Box – a larger container for solid wastes (1.72m x 1.72m plan x 1.2m high);
- 3m^3 Drum – a larger container for in-drum mixing and immobilisation of sludge wasteforms (1.72m diameter x 1.2m high);
- 4m Box - for large items of waste, especially from decommissioning (4.0m x 2.4m plan x 2.2m high);
- 2m LLW Box - for higher-density wastes (2.0m x 2.4m plan x 2.2m high).
- In addition the majority of LLW is packaged by supercompaction followed by grouting in modified ISO freight containers (6m x 2.5m x 2.5m).
- Some wastes do not fit easily into this strategy. These wastes include:
- very large items, (too big for the 4m box) which, if dealt with whole, pose transport and disposal problems. These items are discussed further in Section 2;
- waste whose characteristics make packaging difficult. Such wastes are described in more detail in Section 3.

Large Items

The dismantling of nuclear facilities in the UK may lead to large items of radioactive equipment that require disposal. Such items could be transported and disposed of in one piece or further size reduced for packaging and disposal in standard Nirex waste containers.

Size reduction of large items exposes operators to doses which could be avoided or minimised by transporting and disposing of these items whole. Comparative safety studies will determine which option is preferred.

However disposal of large items is only feasible if the constraints imposed by transport and disposal are met. Considerations of these issues form the remainder of this section.

Disposal Constraints

Perhaps the first question to be asked when determining the appropriate strategy is “how will the item be disposed of”? Clearly the answer is dependent on the type of facility needed, ie are the levels of long-lived hazardous species such that necessitates long-term isolation in deep facilities or are the hazards low enough (or diminish sufficiently in time) to allow disposal to near surface or shallow facilities. Each imposes different restrictions.

Deep disposal

Using the UK Nirex phased disposal concept as an example of deep geological disposal the following constraints may apply:

Physical Constraints

- maximum width of around 2.5 metres. Limited by the size of the access drift or shaft;
- maximum length of around 6 metres. Limited by the size of the access drift;
- maximum weight of around 65 tonnes Limited by lifting capacity of the drift loco or hoisting technology.

Constraints on Contents

- need for good inventories to show compliance with transport regulations, operational safety cases and disposal authorisations. Often information is needed on the amounts and form of key radionuclides for transport and disposal which are above and beyond that required for existing on-site operations. Uncertainties in inventories can eat into disposal authorisations limiting site utilisation.
- physical and chemical form are important factors influencing long-term handling performance and ultimate mobility of the waste following disposal. Often robust wasteforms with appropriate chemical environments are needed.

The above factors can require D&D operators to make provisions for improving characterisation of the wastes, seek conditioning medium which provide long-term durable wasteforms which limit the long-term mobility of activity and in some cases include pre-treatment processes which restrict mobility (see Section 3).

Near Surface and Shallow Disposal

Using the UK example of Drigg where modified ISO steel transport containers are stacked in concrete lined trenches. The following constraints may apply:

Physical Constraints

- maximum size (routinely handled) is constrained by the ISO containers which are. around 6m x 2.5m x 2.5m [1]. Special arrangements are possible for larger sizes;
- maximum weight (routinely handled) is again based on maximum ISO containers weight e.g. around 35te [1]. Special lifting facilities could be procured for heavier lifts.

Constraints on Contents

These are similar to those for deep disposal. Issues of particular significance include: showing compliance with limits on contents, criteria for uniformity or demonstrating the activity is fixed can prove constraining where the package is both a transport package and a waste container; near-surface disposal often requires not only showing compliance with limits on short lived species, but also the absence of long-lived species to very low levels. This constraint can lead to wastes having to be restricted from near-surface facilities because of inaccuracies in inventory determinations.

Having established whether the waste is capable of near-surface or deep disposal and any constraints this imposes, it is necessary to facilitate transport to any facility.

Transport Infrastructure constraints

The UK transport system imposes the following logistical constraints:

- maximum height of 2.2 metres – constrained by the UK rail gauge and wagon design considerations;
- maximum width of 2.3 metres – constrained by the UK rail gauge and wagon design considerations;
- maximum weight package for rail transport of about 65t – constrained by the max UK axle weight of 22.5 tonnes;
- maximum weight for road packages transport up to 150 t are routinely used for moving loads in the UK;
- special permits are required for loads exceeding 150t.

Constraints on weights are likely to be imposed by bridges. Constraints on size are likely to be imposed by road side furniture and buildings. Road infrastructures may have to be modified for such transports.

Difficult Waste for Future management

The previous section focused on generic constraints set by the type of waste receipt facility. The following represents some particular waste types which can raise issues for the waste receiving body.

Highly Heterogeneous Wastes With Divergent Properties

Waste streams in this category are primary those associated with facilities that have historically accepted wastes from a number of different processes and sources. These streams are therefore highly heterogeneous and contain items with divergent properties.

Such wastes require facilities to undertake characterisation, and where necessary segregation, to enable production of packages with properties consistent with future long-term management.

There are a number of plants, which are making provision for characterisation of waste materials and segregation of problematic items or materials, thus facilitating the onward processing of the bulk of the waste streams. The problematic wastes are intercepted and treated off-line before being returned for packaging. Such design philosophies acknowledge that some components of the waste present different hazards and will require alternative additional treatment. The extra characterisation and segregation necessary to remove unsuitable material can dramatically impact on plant throughput and hence the time for D&D operations. Therefore this flexibility needs to be designed in to avoid bottlenecks in the processing.

Materials Where Effective Immobilisation Is Difficult

There are some waste types for which the achievement of immobilisation has been found to be difficult. These include soft low density and/or absorbent wastes such as plastics/cellulosics; wastes with restricted access and/or small porosity such as HEPA filters, filter beds and ion exchange columns; and wastes which are containerised or wrapped such as drummed vault wastes, bagged waste items and super-compacted hard wastes.

Further discussion of progress on these particular wastes is summarised below.

Plastics/cellulosics

A number of plants are recognising the difficulties of direct immobilisation of uncontrolled quantities of plastic and absorbent materials. As a result they are either segregating such materials and controlling their loadings to within pre-established acceptable limits and/or are placing such materials into drums for subsequent high force compaction.

Filter/ion exchange materials

Conditioning of filter and ion exchange media by use of in-drum mixing systems have been successfully implemented for a number of wastestreams where provision for removal of the materials is integral within the plant design. For items where the media are not readily accessible due to the design of the filter assembly, work is in-hand to investigate novel encapsulation matrices for immobilising ion exchange cartridges or filters.

Supercompacted and Containerised wastes

Supercompaction has historically been proposed for many wastestreams. Supercompaction has been endorsed as an appropriate conditioning technology for long-term management where the supercompacted waste has been found to produce a wasteform which provides effective immobilisation of the wastes. This includes predominantly soft or compressible wastes. Hard, non-compactible items are more suitable for processing via alternative concepts such as direct cement encapsulation which offers benefits in both immobilisation of potential mobile activity. For containerised wastes, facilities are being incorporated in a number of plants to open the containerised and/or wrapped wastes, to allow inspection, monitoring, sorting and effective grout infiltration.

Materials with Inherent Hazards

Examples of materials with inherent hazards include wastes containing accessible Wigner Energy such as low temperature irradiated graphite; reactive metals; wastes containing pyrophoric materials such as uranium hydride, finely divided metals, sodium metal etc; and wastes with high fissile contents.

Further discussion of particular wastes is summarised below.

Low temperature Irradiated Graphite

Irradiated graphite can contain stored energy (Wigner energy) which, if released, can cause significant disruption to the system. In particular graphites from low temperature reactor systems have the potential to undergo significant energy releases at temperatures expected during transport, storage and disposal. Plans to release the low temperature Wigner peak in a controlled manner, prior to immobilisation by heating the graphite blocks in an oven, are being investigated.

Reactive Metals

Reactive metals include magnesium, uranium, sodium and aluminium and are present in many wastes and their corrosion/reactions can lead to significant wasteform disruption. This can result in high releases under normal and accident conditions. In the UK, substantial development has been undertaken on magnesium to support the development of suitable wasteforms where evolution over time can be predicted. Similar development work has also been carried out for uranium metal. For sodium, plants to treat the sodium associated with solid waste items are under development and work to confirm tolerance to any residual material after processing is in place.

For aluminium bearing streams work is on-going to develop acceptable wasteforms. This is proving challenging within existing cement systems, and in some cases require excessive restrictions on package contents leading to significantly increased package numbers.

Pyrophoric materials

Metallic uranium is known to have the potential to form pyrophoric uranium hydride (UH_3). Work on encapsulation of metallic uranium has been able to identify conditions capable of avoiding hydride formation and this has been used in support of encapsulation of uranium residues in plants. For uranium hydride already present in wastes, methods for identifying and or passivating the hydride are being progressed.

High Fissile Wastes

A number of waste streams contain potentially high inventories of fissile materials and therefore have the potential for more reactive configurations if any neutron absorbers within the packages are removed or extensive degradation of arrays of packages occur. (This results in an increased potential for a criticality event).

A number of plants are controlling the inventories of fissile materials within packages to meet both short-term criticality constraints and additional constraints that may apply due to longer term processes such as loss of neutron absorption or reconfiguration.

Plants being designed need to be capable of achieving the necessary precision and accuracy in determining fissile inventories and significant work is in progress. This is most challenging for projects which process wastes of different fuel types and enrichments. This is frequently further exacerbated where the fissile materials are intimately mixed with a wide range of other materials.

Materials which enhance radionuclide mobility

Discussion of particular wastes and derived wasteforms that fall into this category is summarised below.

Superplasticisers

Superplasticisers have been proposed in some cases as additions to existing immobilisation grouts to improve workability and to facilitate infiltration of waste items without needing to apply mechanical energy (vibration). Such additives have been found to have the potential to adversely affect longer-term waste management as they can provide a source of complexants that can enhance radionuclide mobility in a disposal environment.

Nirex advises on a number of alternative methods for aiding grout infiltration that may be capable of satisfying short-term processing needs without compromising long-term waste management needs. Application of such alternatives can most effectively be achieved when considered at the design stage of the project.

Effluent Treatment

Some methods for reducing discharges involve chemical additions to complex key radionuclides, for example use of tetraphenyl phosphonium ions for trapping Tc-99. Such additions can give uncertain long term effects on radionuclide mobility necessitating major research programmes to establish behaviour and demonstrate acceptability. Alternatives that do not involve chemical additions or utilise chemicals with only limited potential to enhance long-term mobility need to be considered at the outset so operational factors can be balanced against the longer-term management issues at the outset of the project to minimise project risks.

Conclusions

There are a number of issues and constraints associated with the disposal of large items and problem wastes.

For large items the amount of size reduction required (and hence additional dose uptake) is ultimately dependent on transport infrastructure constraints and the final disposal option. The amount of waste characterisation required is also strongly dependent on the disposal option and in particular the disposal authorisation.

In the UK, a number of issues have been identified with particular wastes or processes. These include:

- achieving adequate characterisation of highly heterogeneous waste;
- ensuring adequate immobilisation of particular wastes;

- ensuring additives which adversely effect radionuclide mobility are avoided or controlled and;
- ensuring wastes with inherent hazards are pre-treated to remove the hazard or made safe by their packaging.

UK experience has been that through effective dialogue between the D&D operators and the waste receiving body, particularly at the outset of the project, then effective management of the issues of size reduction, waste characterisation, waste treatment and waste conditioning can be achieved.

References

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WASTE ACCEPTANCE AND IMPACT ON D&D IN SWITZERLAND

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Five commercial reactors are in operation in Switzerland, with a combined output of 3.2 GW(e). The following Table provides key data on the five reactors:

NPP	Reactor type	Output	Operating since
KKB-I: Beznau I	PWR (Westinghouse)	365 MW(e)	1969
KKB-II: Beznau II	PWR (Westinghouse)	365 MW(e)	1971
KKG: Gösgen	PWR (KWU)	970 MW(e)	1979
KKL: Leibstadt	BWR (General Electric)	1145 MW(e)	1984
KKM: Mühleberg	BWR (General Electric)	355 MW(e)	1972

Although decommissioning of the first (oldest) reactor will not take place until 2009 at the earliest (hypothetical operating lifetime of 40 years), detailed decommissioning studies have to be carried out today, in order to demonstrate the feasibility of the technologies to be used and to determine anticipated costs (for the purpose of calculating financial contributions to a decommissioning fund). The studies are based on waste acceptance criteria and guidelines that apply to waste already in existence. The focus is on preparing inventories of activated and contaminated components and conditioning of these components.

The basis for present and future conditioning of radioactive wastes, as well as for their interim storage and final disposal, is provided by the official guideline HSK R-14. According to this guideline, raw waste requires to be solidified (inter alia with cement) and the resulting waste product must

- remain intact until final disposal
- not be readily dispersible
- be resistant to aqueous media
- not be readily combustible
- not contain any unnecessary voids
- contain as little organic material as possible

The waste package containing the waste product must

- constitute a further barrier to dispersion
- outlast (at least) interim storage
- be documented with details of manufacturing, composition, properties
- be designed to resist corrosion using suitable materials
- be characterised by a quality assurance program for
  - raw waste
  - waste product
  - waste package

The only possible reasons for interim storage of waste without solidification are :

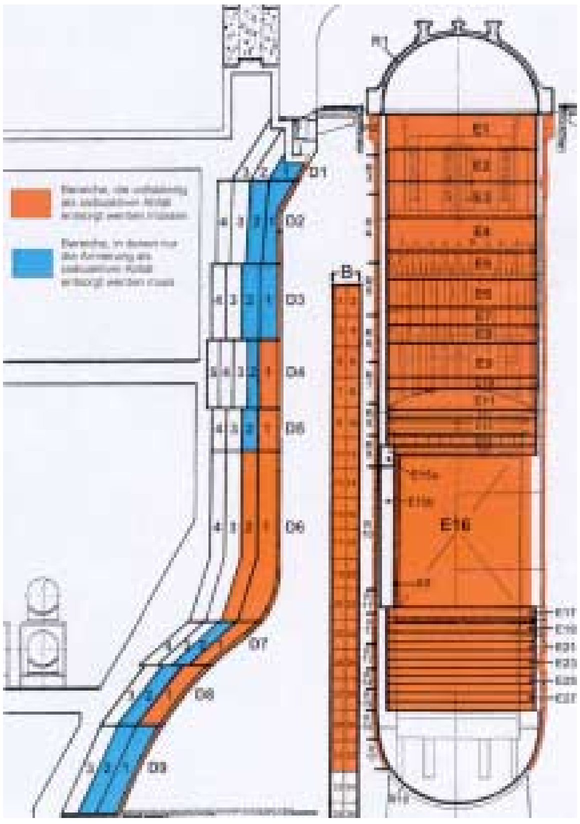
- decay storage followed by conventional waste management
- if waste packages fulfil acceptance criteria for the final repository without further treatment
- if, in the foreseeable future, an alternative conditioning method can be expected

The guidelines and acceptance criteria mentioned set strict requirements in terms of characterisation and documentation of waste packages. Complete inventories of safety-relevant nuclides and of contained materials have to be calculated for raw wastes. For waste products solidified with cement, compressive strength, water resistance and leach rate have to be measured. Information is also required on. dose rates and surface contamination for waste packages

Within the framework of the decommissioning studies, particular importance has been attached to inventorying all activated and contaminated components (reactor internals, reactor pressure vessel, bioshield, drywell). For this purpose, each reactor was broken down into partial volumes (see Figure), for which

- the thermal, epithermal and fast neutron flux was first calculated
- the complete inventories of safety-relevant nuclides were then calculated; an activation program specially developed for making calculations outside the reactor core was used for this purpose.

Standardised containers are foreseen for packaging the waste; the highly activated reactor internals will first be packaged in MOSAIK-containers (which will then be cemented into standardised containers). Incineration of waste will also be carried out in the plasma oven at ZWILAG. The resulting glass will be cemented into 200-l drums, which will also be placed in standardised containers.



RADIOLOGICAL CHARACTERIZATION AND CHALLENGES AT DECOMMISSIONING SITES

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Introduction

The License Termination Rule (LTR), Subpart E to 10 CFR Part 20, provides the dose-based criteria that the U.S. Nuclear Regulatory Commission (NRC) uses as the basis for regulating cleanup at material and reactor sites. The LTR permits the release of sites for unrestricted use, if the radioactivity that is distinguishable from background results in a total effective dose equivalent to an average member of a critical group that does not exceed 0.25 milliSievert per year (mSv/yr) (25 millirem/year) and the residual radioactivity has been reduced to levels that are as low as reasonably achievable. Additionally, the LTR establishes criteria for license termination with restrictions on future land use, which allow for a dose to the critical group of 0.25 mSv/yr (25 millirem/year) with restrictions in place, and 1 mSv/yr (100 millirem/year) if the restrictions fail. In certain circumstances as outlined in Subpart E, a dose as high as 5 mSv/yr (500 millirem/year) is permitted if restrictions fail. The LTR is available on the internet at: http://www.access.gpo.gov/nara/cfr/waisidx_01/10cfr20_01.html

Following issuance of the dose-based LTR in 1997, NRC staff developed the Standard Review Plan for Decommissioning Plans (NUREG-1727). NUREG-1727 is a guidance document that describes the methods that NRC has determined are acceptable for implementing the LTR and other decommissioning regulations. While NUREG-1727 is focused on the review of decommissioning plans for nuclear material sites, it provides general guidance that in many cases is applicable to reactor sites (e.g., review criteria for dose-modeling and radiological surveys). In addition to NUREG-1727, staff developed the Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans (NUREG-1700) as specific guidance for reactor decommissioning. NUREG-1700 provides the review methodology that NRC staff follows when reviewing License Termination Plans submitted by decommissioning reactors. Both NUREG-1700 and NUREG-1727 are available on NRC's public web site: www.nrc.gov (when the redesigned site becomes available).

Characterization Guidance

The Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM), implemented in 1997, is NRC's primary guidance document for radiological surveys. MARSSIM was developed by a committee of multiple U.S. Government agencies [including NRC, the Environmental Protection Agency (EPA), the Department of Energy, and the Department of Defense] with the objective of developing a consistent approach for planning, performing, and assessing radiological surveys to meet dose- or risk-based cleanup criteria. The MARSSIM process is focused on final status surveys, but also provides a methodology for conducting site characterization. The MARSSIM framework consists of an historical site assessment, followed by area classification, biased scoping survey(s), and then a characterization survey. The goal of this process as detailed in NUREG-1700, is to determine the radiological condition of the property well enough to: 1) conduct an effective remediation project; 2) not endanger the workers; 3) provide adequate assurance that it is unlikely that significant quantities of radiation have gone undetected;

and 4) provide sufficient information to plan the final status survey. MARSSIM is available on the EPA's website at: http://www.epa.gov/radiation/marssim/.

The complement to MARSSIM, which sets a standard methodology for laboratory analysis, is the Multi-Agency Radiological Laboratory Analytical Protocols Manual (MARLAP). MARLAP was developed by a multi-agency committee with the goal of standardizing analytical methods so that radioanalytical data used for regulatory decision-making is of a known quality that is appropriate for its intended use. MARLAP is available at: http://www.eml.doe.gov/marlap/; MARLAP has not been finalized and is currently available in draft form for review and comment.

Challenging Issues

Reactor Vessel Decommissioning

A challenging area in reactor decommissioning is the problem of how to ultimately dispose of reactor vessels. NRC regulations (10 CFR 61) recognize three different classes of Low-Level Waste (LLW), which in ascending order of hazard are termed Class A, Class B, and Class C. For each waste class, NRC regulations set concentration limits (in the form of formulas) for short- and long-lived radionuclides, reflecting the half-lives and hazards of the radionuclides in each class. A rule-of-thumb is that Class A waste is intended to be relatively safe after 100 years, Class B after 300 years, and Class C after 500 years. Waste exceeding the criteria of Class C is termed "Greater Than Class C" (GTCC) waste and cannot be disposed of in LLW disposal facilities. Because reactor vessels typically contain waste that could be classified as GTCC, disposal at a LLW facility is typically not an option. This has led to the practice of segmenting the GTCC reactor internals before disposing of the vessel in a LLW facility. The remaining internals are then stored on site in the spent fuel pool or in dry storage awaiting ultimate disposal with high-level waste. In two cases, however, the Trojan Nuclear Plant and the Saxton Facility, the reactor vessels were characterized such that they were able to be removed intact and disposed of at LLW disposal sites.

Partial Site Release

Another challenging issue in decommissioning is "partial site release." Partial site release involves a scenario whereby a licensee wishes to release a clean portion of a site for unrestricted release, while maintaining a license and control of the remainder of the site. This released land can be used in economically beneficial ways (e.g., in a recent example, the owner of a decommissioning reactor wishes to site a gas turbine power plant in the parking area of the previous nuclear plant, which would lead to a revitalized tax base for the local community). The challenge this presents to NRC, as regulators, is how to release the land, while maintaining the integrity of the decommissioning process. NRC has established procedures for accomplishing these partial site releases, and has initiated a rulemaking to integrate the approach into the regulations. The current process allows for licensees to apply for partial release of non-radiologically impacted areas of their sites by submitting a letter request. The licensee should utilize the MARSSIM approach (or equivalent) to demonstrate that the portion of the site is non-impacted. If the licensee successfully demonstrates that the portion of the site is non-impacted, then the site can be released with minimal surveys (if any) being conducted as part of NRC's inspection process. For radiologically impacted areas of the site, NRC requires that the licensee apply for a license amendment to release the portion of the site. As part of the amendment process the licensee must demonstrate (based on final status surveys) that the portion of the site they wish to release meets NRC's criteria for unrestricted release.

Information relating to NRC's partial site release process and current rulemaking effort is available at http://ruleforum.llnl.gov/cgi-bin/rulemake?source=Unrestricted_Use.

Entombment and Rubblization

Two other areas that present challenges, in the decommissioning area, are entombment and rubblization.

Entombment is a decommissioning alternative whereby radioactive structures are encased in concrete, rather than being decontaminated and removed. The entombed structure is then maintained, and continued surveillance is conducted until the radioactivity decays to a level that permits termination of the license. Despite the cost and dose savings implicit in this methodology, entombment may not be feasible under the current NRC regulations, because most reactors have radionuclides in concentrations exceeding the limits for unrestricted release even after 100 years. However, this option might be acceptable for reactor facilities that can demonstrate that radionuclide levels will decay to levels that will allow restricted use release of the sites. Currently no NRC licensees have proposed the entombment option, and only three demonstration reactors have been entombed.

Rubblization is another unique decommissioning technique whereby above grade structures are partially decontaminated and then demolished and disposed in the below-grade portions of the structures. While rubblization may present a low cost alternative that could potentially meet NRC's dose-based cleanup standards, it presents many challenges. These include: how to measure the contamination, how to model expected doses from the rubblized material, and policy issues inherent with leaving the contaminated material on site.

Summary

NRC has developed and implemented a dose-based cleanup standard. In addition, a comprehensive suite of guidance documents has been developed, enabling application of the new regulations to the characterization and cleanup of radioactively contaminated sites. As NRC and its licensees gain experience, we look forward to applying these new performance-based standards and implementing methodologies to their fullest potential in current and yet to be discovered decommissioning challenges.

RADIOLOGICAL CHARACTERIZATION OF BUILDINGS AS THEY STAND, IN ORDER TO ALLOW CONVENTIONAL DEMOLITION

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Introduction

Since a nuclear facility that terminated its operation has still radioactivity, radiological characterization is necessary for safe and economical dismantling and waste management. For effective waste management in decommissioning nuclear facilities, a regulatory framework has been studied in such fields of clearance levels and waste categorization in Japan. Though clearance levels have been decided to be included in the regulatory frame, a practical procedure in compliance with its regulation is under development considering future decommissioning of nuclear facilities. So far, the concept of classifying non-radioactive materials, which was studied by the Nuclear Safety Commission, is only applicable way to allow conventional demolition of buildings. The classification of non-radioactive materials was actually conducted in the Japan Power Demonstration Reactor (JPDR) dismantling project based on the concept.

This paper deals with the procedure of radiological characterization of buildings and experience in the JPDR Dismantling Project.

Radiological Characterization

The Nuclear Safety Commission established the concept for classifying non-radioactive materials in 1992. It was first applied to the JPDR dismantling project, and then to replacement of steam generators in some nuclear power plants. The basic procedure for radiological characterization of buildings is as follows:

Rooms or zones are designated as contaminated areas as long as there is possibility of contamination by examining plant operation records. For example, a room where radioactive components are located is classified as a contaminated area.

Radioactivity is roughly measured to characterize contamination areas with a simple method.

Radioactivity is then measured in detail to characterize contamination in width and depth referring to the simple measurement and operation records.

Contaminated parts are removed according to the radiological characterization.

Confirmatory measurement is conducted to determine that there is no significant radioactivity. The criterion is applied to be less than natural background level (Average plus 3σ)

Experience in JPDR Dismantling Project

After removing all components in the facilities, building surfaces were decontaminated and their radioactivity was measured for cancellation of controlled areas in the several steps. At the first step, facility operation history was surveyed to identify contamination areas. Next, the contamination on building surfaces was characterized in detail by sampling and measurement. Samples were taken in 2 by 2 m block base; contamination depth and nuclides were checked in the process. Contamination maps were then drew on the basis of the measurement. The building surfaces were decontaminated; the surface layer in 2 mm depth was removed even in non-contaminated areas, the contaminated layer was segregates with twice of measured contamination depth in surface contaminated areas, and the contamination layer was removed with 5 mm margin of its depth in deep contamination areas.

The final radiological survey was conducted by two steps; first, direct measurement by handy type detectors, second, sampling and measurement of gamma-ray spectra. The direct measurement was performed on every 0.8 by 0.8 m block moving a survey meter and the highest counting rate was identified in each block to measure its counting rate. The counting rate was compared with the background level to be confirmed that there was no significant radioactivity in the block. After the direct measurement, samples were taken from each room for confirmation of that radioactivity level is less than 3 Bq/kg. Figure 1 shows the basic flow for identifying non-radioactive materials. The total number of measurement on building surfaces resulted in more than 50 thousand in total.

Data Collection

In radiological characterization process of the JPDR dismantling project, various data were collected on contamination width and depth in room-by-room base. Nuclides and contaminated materials were also identified in the process. These data were used for designing dismantling activities as follows.

- Selection of machines and tools to be applied
- Design of dismantling and decontamination procedures and time schedule
- Prediction of wastes volume in radioactive and non-radioactive considering radioactivity levels and materials
- Prediction of number of radioactive waste containers
- Design of a temporary waste storage yard and transportation to storage facilities
- Prediction of manpower needs, worker exposure and cost

In the decontamination and demolition activities, data were also collected and recoded on radioactive inventory, waste materials and volume in container base for the following treatment and disposal activities.

Concluding Remarks

A procedure for radiological characterization was applied to the JPDR facilities, based on the concept made by the nuclear safety commission. It was found that identification of non-radioactivity was a time-consumable work if criterion of background level plus 3□ was applied to. Also, special consideration was necessary for removal of embedded pipes in building structures if radioactivity of embedded pipes was not measurable. Detailed radiological characterization of building is useful to reduce radioactive wastes generation in demolition of buildings. It results in the reduction of cost for decommissioning waste management. However the detailed measurement of radioactivity itself costs a great deal. It might be necessary to consider a comprehensive waste management plan for its optimization, which enables the nuclear facility decommissioning more economical.

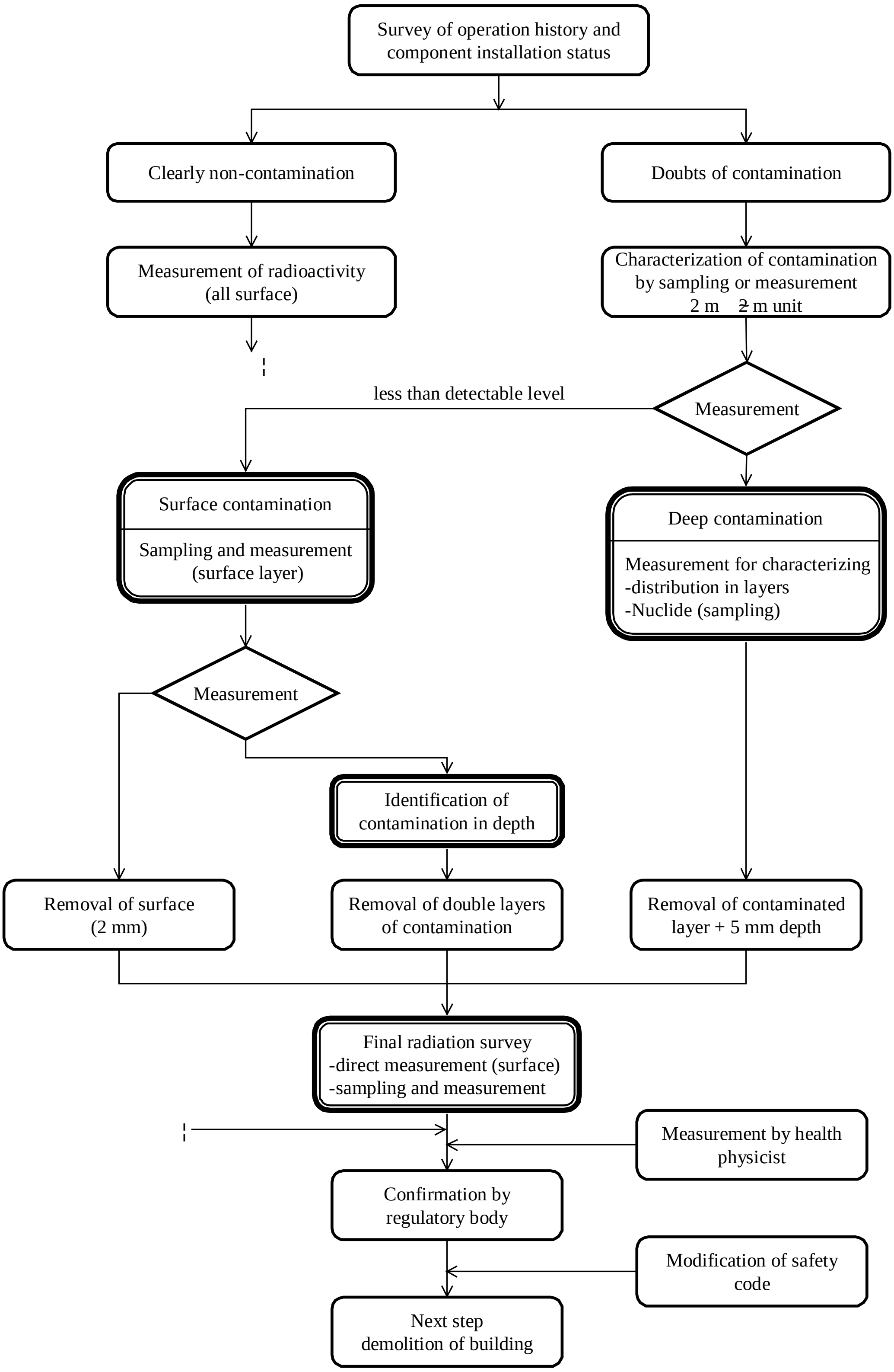

Fig.1 Basic flow for identifying non-radioactive materials

CONCRETE CRUSHING AND SAMPLING, A METHODOLOGY AND TECHNOLOGY FOR THE UNCONDITIONAL RELEASE OF CONCRETE MATERIAL FROM DECOMMISSIONING

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Abstract

Belgoprocess started the industrial decommissioning of the main process building of the former Eurochemic reprocessing plant in 1990, after completion of a pilot project. Two small storage buildings for final products from reprocessing were dismantled to verify the assumptions made in a previous paper study on decommissioning, to demonstrate and develop dismantling techniques and to train personnel. Both buildings were emptied and decontaminated to background levels. They were demolished and the remaining concrete debris was disposed of as industrial waste and green field conditions restored.

Currently, the decommissioning operations carried out at the main building have made substantial progress. They are executed on an industrial scale and will continue till the end of 2005. In view of the final demolition of the building, a clearance methodology has to be proposed. Application of the methodology applied for the storage buildings of the pilot project is complicated for several reasons. Although this methodology is not rejected as such, an alternative has been studied thoroughly. It considers at least one complete measurement of all concrete structures and the removal of all detected residual radioactivity. This monitoring sequence is followed by a controlled demolition of the concrete structures and crushing of the resulting concrete parts to smaller particles. During the crushing operations, metal parts are separated from the concrete and representative concrete samples are taken. The frequency of sampling meets the prevailing standards. In a further step, the concrete samples are milled, homogenised, and a smaller fraction is sent to the laboratory for analyses. The paper describes the developed concrete crushing and sampling methodology.

Introduction

Belgoprocess started the industrial decommissioning of the main process building of the former Eurochemic reprocessing plant in 1990, after completion of a pilot project. Two small storage buildings for final products from reprocessing were dismantled to verify the assumptions made in a previous paper study on decommissioning, to demonstrate and develop dismantling techniques and to train personnel. Both buildings were emptied and decontaminated to background levels. They were demolished and the remaining concrete debris was disposed of as industrial waste and green field conditions restored [1]. The main conclusions of this pilot decommissioning project denoted that emphasis had to be put on:

The automation of concrete decontamination, and The decontamination of metal components.

The main process building is a large rectangular construction, about 80 m long, 27 m wide and 30 m high [2]. About 106 cell structures have to be dismantled, involving the removal and decontamination of equipment from each cell, the decontamination of the cell walls, ceilings and floors, the dismantling of the ventilation system. These activities are followed by a complete monitoring to allow for unconditional release of the remaining structures. As such, about 1,500 Mg of metal structures, and 12,500 m³ of concrete with 55,000 m² of concrete surfaces have to be removed and/or to be decontaminated. Most of the work involves hands-on operations under protective clothing tailored to each specific task. Tool automation and automatic positioning systems are successfully applied.

The specific Belgoprocess approach should be highlighted in which decommissioning activities are carried out on an industrial scale with special emphasis on cost minimisation, a commitment to results within an overall planning, and the use of technology on an industrial representative scale [3]. This approach includes specific actions to reduce standby costs. It takes great care to limit radioactive waste management costs, keeping the generation of radioactive waste to a minimum, minimising the spread of radioactivity as much as possible, and optimising the possibilities for recycling and reuse of valuable components from existing and potential waste streams. Extensive use of adequate decontamination techniques is made in order to allow dismantled components and materials to be unconditionally released, taking into account the limited availability of funding [4].

Today, the decommissioning operations carried out at the main process building of the former Eurochemic reprocessing plant have made substantial progress. When the removal and decontamination of equipment from each cell will have been completed, and cell walls, ceilings and floors will have been decontaminated and the ventilation systems removed, a monitoring programme will be required in order to obtain the unconditional release of the remaining structures. It is the aim to demolish these structures and to remove the remaining concrete debris for recycle or reuse in the non-nuclear industry.

Final Demolition Of Storage Buildings For End Products Of Reprocessing

As indicated before Belgoprocess started its decommissioning activities with the dismantling and decontamination of two small storage buildings for end products from reprocessing. Both buildings were emptied and after decontamination of the concrete structures down to a level of 0.04 Bq/cm² for alpha and of 0.4 Bq/cm² for beta-gamma emitters, two independent measurements of all building surfaces were carried out by the in-house Health Physics Department in order to confirm the above mentioned contamination levels. A third random control measurement was performed by an officially approved radiation protection control organisation. All three measurements gave the same results.

Core samples were taken on the previously most contaminated spots. The specific activities of these samples proved to be well below 1 Bq/g. Measurements and analyses on these samples only confirmed the presence of natural radioisotopes. Consequently, the buildings could be withdrawn from the controlled area.

The final steps in the pilot decommissioning project were the demolition of the two buildings, the removal of the demolition waste to an industrial dumping ground for inert wastes and restoration of the green field conditions. To enable this last step in the objectives of the pilot project, it was necessary to provide sufficient evidence to justify such action to the public opinion. In doing so, the actual absence of any Belgian regulation for unlimited reuse or uncontrolled dumping of suspected and/or decontaminated materials, other than the qualitative definition of radioactive waste, had to be taken into account.

The evidences to be supplied had to be evaluated in the context of a number of exceptive regulations, which were applied in other decommissioning projects, and in the context of available recommendations for recycling of materials resulting from the decommissioning of nuclear installations [5], by radiation protection experts of the European Community:

Removable surface contamination in alpha <= 0.04 Bq/cm², Removable surface contamination in beta-gamma <= 0.4 Bq/cm²,

Total specific beta-gamma activity <= 1 Bq/g, mean value over an arbitrary mass of 1 000 kg with an individual maximum of 10 Bq/g.

The results of the multiple 100 % surface measurements in the two buildings and the additional controls on selective core samples (gammaspectrometry and total alpha and beta measurements) showed that the requirements of the first two criteria were met, and that the third criterion, limited to the core samples taken, was also complied with. The only thing that remained to be demonstrated was, that also a possible alpha contamination was characterised by sufficiently low specific activities.

Based on experiences from the operational period of the plant (localised contamination) and on located contamination during decommissioning operations, it was decided not to take concrete core samples at random, but at the previously most contaminated spots. This should increase the probability on detection of possible remaining contamination.

By doing so, it was accepted in a conservative way that the collected core samples were representative for all concrete volumes, as if all of them had originally known the same history, and as if all of them were also brought to their actual characteristics in a similar manner. This also means that it was accepted that every part of both buildings was originally contaminated to the same degree as those parts where core samples were taken, and where before decontamination the highest degree of contamination was found, which, in reality was certainly not so.

This hypothesis, however, offered the possibility of extrapolating in a conservative way, the results of the analyses on the core samples, expressed in the mean value for the measurements and the standard deviation, to the whole of the building material. And for this hypothesis the core sampling carried out could be considered as random.

The total analysis of the core samples via gammaspectrometry, by means of a 305 x 102 mm NaI(Tl)-detector in a screened bunker with low background radiation, revealed no indication of artificial contamination via low level activity measurements. The resulting spectrum offered a qualitative view with high sensitivity. In the core samples only natural nuclides were found.

A gammaspectrometric monitoring (Ge-detector) of the individual core samples was carried out to obtain a quantitative measurement of the gamma emitters, and the results were compared to the results of similar measurements on a reference, non contaminated, core sample. This method has but little sensitivity. In fact a surface measurement is carried out which can only detect radioisotopes in the specimen at low depths. The detected radionuclides were:

Natural radionuclides:

^{40}K, ^{226}Ra and daughters, which can normally be found in elements of concrete and other construction material. The mean values of the detected activities for both nuclides were comparable to the values found in the non contaminated reference core sample (x = 0.4 mBq/g, s = 0.3 mBq/g and x = 0.27 Bq/g, s = 0.34 Bq/g respectively). Operational history of the installations, and the fact that in the spectrum no peaks for ^{238}U or ^{235}U were found, do not give reasons to consider that the installations had ever artificially been contaminated with ^{40}K or ^{226}Ra. Moreover, this argument would only have been important if the detected values for ^{40}K or ^{226}Ra would have been inexplicably high.

Nuclides due to artificial radioactivity:

^{137}Cs. Compared to the detected activity in the non contaminated reference core sample, the resulting total contamination in the core samples (2 mBq/g), for a homogeneous distribution of the specific activity, was not higher than 1 Bq/g, mean value over an arbitrary mass of 1 000 kg, taking into account the gamma penetration in concrete.

By means of the adopted gammaspectrometry, no ^{241}Am was found. Considering the operational history of the installations, this was an indication for the absence of alpha contamination by plutonium. If there had still been some remaining contamination in the decontaminated floors or in other surfaces of both buildings, then, if the remaining material of the construction was still the original one, parts close to the surface would normally have shown the highest contamination concentration. Therefore some cross-sections of the upper part of the core samples were analysed for alpha and beta activity control.

The analyses carried out on six cross-sections of the core samples taken resulted in a mean value x = 16.7 mBq/cm² and a standard deviation s = 8.2 mBq/cm².

With the basic hypothesis of a generally spread maximum amount of contamination in the remaining structures as indicated at the beginning of this paragraph, probabiliorism tells us, that for a random sampling of N values from a normal population with calculated mean value x and standard deviation s, the most probable value for the mean will be somewhere between $x + t \times s / N^{0.5}$ where t represents the Student-factor, depending on one single parameter, the degree of freedom of the sampling problem. For a degree of freedom N - 1 = 5, and for a confidence interval of 99 %, it could be stated with 99 % certainty that the most probable mean value for the measurements carried out, would not be higher than 30.2 mBq/cm². In practice, considering the adopted hypotheses it would even have been lower.

It was furthermore accepted that the each of the fission products ^{90}Sr, ^{90}Y and ^{137}Cs were detected with equal activity (resulting mass-absorption coefficient for the medium 0.026 cm²/mg) and a homogeneous distribution in a concrete layer of 1 mm thickness. It could then be calculated that the transmission due to self-absorption would be limited to 17.4 % [6]. In this way the mean specific beta activity of the analysed concrete layers proved to be 0.44 Bq/g and the maximum for the mean value 0.79 Bq/g.

The evaluation of alpha measurements only made sense in case also other than natural alpha activity had to be considered. Indeed, it could not be the aim to prove that natural radioactivity in the remaining concrete was by coincidence lower than somewhere else in other constructions or concrete elements. In the evaluation carried out, it had therefore to be taken into account that an important part of the considered alpha contamination was coming from the radionuclide ^{226}Ra, due to natural radioactivity, as stated before.

To detect possible remaining alpha contamination in the core samples, the same six cross-sections that were analysed before were at both sides analysed on alpha-radiation, resulting in a mean value x = 0.96 mBq/cm² and a standard deviation s = 0.54 mBq/cm².

With the hypotheses a generally spread maximum amount of contamination in the remaining structures as indicated at the beginning of this paragraph, in the same way probabiliorism tells us, that for a random sampling of N values from a normal population with calculated mean value x and standard deviation s, the most probable value for the mean will be somewhere between $x + t \times s / N^{0.5}$ where t again represents the Student-factor, again depending on only one single parameter, the degree of freedom of the sampling problem.

For a degree of freedom N - 1 = 11, and for a confidence interval of 99 %, it could again be stated with 99 % certainty that the most probable mean value for the measurements carried out, would not be higher than 1.44 mBq/cm². In case of a homogeneous distribution of the activity in a concrete layer with thickness equal to the range R, it could be calculated that the transmission due to self-absorption, for 2 Π - measurements, would be limited to 50 % [7]. With a value for R = 6.5 mg/cm² [8], the mean specific alpha activity in the analysed concrete layers proved to be 0.30 Bq/g, and the maximum for the mean value 0.44 Bq/g.

To check the indicated evaluations with reality, destructive analyses were carried out. By means of alpha-gamma spectrometry the total alpha-beta activity in a cross-section of the core sample with the highest detected activity was determined. The results of these direct measurements revealed respectively a beta activity of 0.78 Bq/g, and an alpha activity of 0.64 Bq/g. These practical analyses confirmed the evaluations carried out in calculating the specific values for possible alpha or beta activity in the core samples taken.

As such, by means of practical measurements, and by means of evaluations, confirmed by the results of destructive alpha and beta analyses, it was demonstrated in a conservative way that, for a possible alpha or beta-gamma contamination in the remaining structure of both buildings, sufficiently low surface activities and specific activities were obtained. Taking into account the hypotheses where it was accepted in a conservative way that all volumes originally showed the highest degree of contamination, despite the fact that an important part of the surfaces of the buildings originally showed no detectable contamination to the used detectors, the resulting specific activities are well below the proposed levels.

In this way, sufficient evidence was delivered to carry out the last part of the decommissioning project for the two buildings, i.e. the final demolition of the remaining structures and the removal of the demolition waste to an industrial dumping ground, as indicated in the pictures of figures 1 and 2.

Figure 1. Demolition of buildings 6A/6B

Figure 2. Green field conditions after demolition of buildings 6A/6B

Final Demolition Of The Main Process Building

The final demolition of the main process building (Fig. 3) requires a specific clearance methodology. It has been evaluated that application of the methodology applied for the pilot project is complicated for several reasons, among which important ones are:

The type and spread of contamination: at the end of the reprocessing activities, all cells were cleaned using the high-pressure water jet technique, which caused in-depth penetration of contamination.

The total surface is large, which will require extensive manpower if all surfaces have to be monitored twice in view of unconditional release.

Taking core samples at the previously most contaminated places will result in a large amount of samples to be taken and to be analysed, and it will be very difficult or impossible to prove that these samples are representative for the remaining structures of the building;

In view of the structural stability of the building, it will be impossible to remove all the pipe penetrations prior to the demolition of the building.

The fundamental question also arises whether the authorities will accept that a building is released before all the pipe penetrations have been removed. In the same sense, it may be questioned whether a controlled demolition of a building will be acceptable once it has been released, but without additional monitoring during breakdown.

Although the application of the methodology used for the two small buildings in the pilot project is not rejected as such, an alternative has been thoroughly studied. It considers at least one complete measurement of all concrete structures and the removal of all detected residual radioactivity. This monitoring sequence is followed by a controlled demolition of the concrete structures and crushing of the resulting concrete parts to smaller particles. The concrete blocks containing the remaining pipe penetrations are sent to a controlled area in order to separate the tubes from the concrete.

Figure 3. Main process building of the former Eurochemic reprocessing plant

During the crushing operations, metal parts are separated from the concrete and representative concrete samples are taken. The sampling frequency meets the prevailing standards. In next step, the concrete samples are milled and homogenised. A smaller fraction is sent to the laboratory for analyses.

Both methodologies, as mentioned above, were discussed with an independent radiation protection control organisation, prior to submitting one or both of them to the authorities.

In view of this proposal for the unconditional release of concrete material, a research and development programme was carried out in order to crush, mill, sample and monitor concrete dust similar to the procedure that is adopted for the melting of metal material. Discussions were organised with the independent radiation protection control organisation in order to install the adequate crushing and milling technology so that the resulting concrete material can be reused in road constructions. A final report was prepared and agreement was obtained from the technical as well as from the financial point of view. The licensing documents were prepared and approved.

The research and development programme resulted in a set of achievable goals, that had to be met during the technical design. The most important goals and the relating achievements were:

- The definition of a representative sampling technique, based on prevailing standards from the mineral processing industry. A specific sampling unit was developed taking approximately 75 partial samples of 2 kg per processed batch of 7 000 kg of concrete blocks, comprising a crusher to bring the granulate dimensions to the requested level for measurement, and a sample divider to split the total sample into a reduced sample and a reference sample.
- The definition of a crushing technique, in order to separate the reinforcement bars from the concrete parts, but also to provide the right granulate dimension to both the sampling unit and the concrete processor. A typical electrically powered jaw crusher was installed, with automated feed rate control. A remote controlled hammering unit can be activated in case of obstructions on the inside of the crusher.
- The definition of a technique for removing reinforcement bars, in order to prevent these bars to block either the sampling unit or the sample crusher.
- The definition of the transport devices to and from the various components in order to smooth the complete process. A tilting device is used to load the installation. Vibration systems and conveyor belts are used for the internal transportation of the material.
- The definition of a ventilation system in order to prevent the release of dust into the environment. The complete installation is encapsulated and extracted via self-cleaning pre-filters and absolute filters. An additional dust-sampling unit is provided in both the extraction circuits upstream and downstream of the crusher.

Figure 4. General view of the concrete crushing and sampling facility

The orders for the practical installation of the various parts of the equipment in an existing building were placed in the first trimester of 2000. The building was partially dismantled and the area for the equipment was prepared. Concrete works for the supporting structures were finalised in September 2000. The entire crushing installation, metal separator, transport and filter systems were delivered in the beginning of September 2000 and installation of all systems was finalised in the middle of December 2000 (Fig. 4). The complete installation is 48 m long, 10 m wide and 9 m high and represents an investment of about 2.5 million Euro. Its nominal capacity is set at 28 Mg per day. Operational and cold tests were carried out in January 2001, and training of the operators finalised.

The required operational risk evaluation was carried out as well as the worker risk evaluation. The required documentation file has been submitted to the respective safety authorities in order to get the start-up permit. The conventional and nuclear safety inspection before start-up was carried out in the second week of June, 2001. As a result, operations could be started at the end of June, 2001.

Conclusions

In view of the final demolition of the main process building of the former Eurochemic reprocessing plant, a clearance methodology has to be proposed. Application of the methodology applied for two storage buildings of a pilot decommissioning project is complicated for several reasons. Although this methodology is not rejected as such, an alternative has been studied thoroughly.

The alternative considers at least one complete measurement of all concrete structures and the removal of all detected residual radioactivity. This monitoring sequence is followed by a controlled demolition of the concrete structures and crushing of the resulting concrete parts to smaller particles. During the crushing operations, metal parts are separated from the concrete and representative concrete samples are taken. The frequency of sampling meets the prevailing standards. In a further step, the concrete samples are milled, homogenised, and a smaller fraction is sent to the laboratory for analyses.

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Practical installation of the various parts of the equipment in an existing building was finalised in the middle of December 2000. Operational and cold tests were carried out in January 2001, and training of the operators finalised.

Conventional and nuclear safety inspection before start-up was carried out in the second week of June 2001, and as a result, operations could be started at the end of June, 2001.

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