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The decommissioning of WWER type nuclear power plants

Final report of an IAEA Regional Technical Co-operation Project

INTERNATIONAL ATOMIC ENERGY AGENCY

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FOREWORD

Numerous WWER-440 nuclear power plants are in operation in central and eastern Europe and a small number have already been shut down. In addition to reactors already shut down, many other reactors will reach the end of their design lifetime in a few years and become candidates for decommissioning. It is unfortunate that little consideration was devoted to decommissioning of WWER-440 reactors at the plant design and construction stage, and little emphasis was placed on planning for decommissioning. It is within this context that the IAEA launched a regional technical co-operation project in 1994 with the aim of providing guidance on planning and management of decommissioning for WWERs.

The project, which had a duration of four years (1995–1998), included the organization of workshops and scientific visits to countries having WWERs and other countries where active decommissioning projects were under way. Eventually, participants suggested the consolidation of expert guidance and collective opinions into a TECDOC, which was drafted by both designated participants from project recipient countries and invited experts. The TECDOC has the aim of serving as a stimulus for all concerned parties in central and eastern European countries to initiate concrete decommissioning planning, including assessment of existing and required resources for the eventual implementation of decommissioning plans. In addition, the regional technical co-operation project has managed to bring together in this TECDOC a number of good practices that could be useful in WWER-440 decommissioning.

The IAEA wishes to express its thanks to all participants in the project and would like to take this opportunity to acknowledge the excellent co-operation and hospitality of the institutions which hosted the project workshops.

CONTENTS

1. BACKGROUND

The acronym WWER (water, water, energy, reactor) refers to Soviet design water cooled, water moderated, electricity generating reactors. The first commercial WWER was commissioned in the former USSR in 1963 at Novovoronezh. The first unit, known as WWER-210, was followed by a second prototype, a 365 MW(e) version that became operational in late 1969. From these prototypes, a standardized 440 MW(e) nuclear power plant, called WWER-440, was developed. The first WWER-440s use the standard plant design referred to as model 230. The first 230 unit was Novovoronezh Unit 3, which began power operation in 1971.

A later model of WWER-440 designated as model 213 with confinement and spray type steam suppression was commercially introduced in early 1980s in Rovno, Ukraine (Units 1 and 2). Model 213 with a nuclear steam supply system in a containment structure (of Westinghouse design) was used in Loviisa, Finland (Unit 1 operated since 1977-Unit 2 since 1980). In the earlier 1970s the former USSR designed and began to construct a 1000 MW(e) pressurized water reactor designated as WWER-1000. This reactor is a four-loop system, housed in a containment type structure and also with a spray type steam suppression. The WWER-1000 has an emergency core cooling system and is designed to withstand earthquakes. The first WWER-1000 became operational in 1980 at Novovoronezh. More details on WWER construction and operational features can be found in Refs [1–5].

According to recent statistics, 28 WWER-440 nuclear power reactors are operating in eight countries, of which 17 reactors belong to the model 213, 11 to the model 230. Five WWER units have been permanently shut down in the former German Democratic Republic (Greifswald) after Germany's reunification due to high costs associated with refurbishing the units to the required standards, and technical-political uncertainties. Two WWERs were shut down in Armenia, but one of these resumed operation a few years ago. Appendix 1 provides basic information on design characteristics of WWER-440 units as well as comprehensive lists of such reactors [6]. Although different from WWER-440 reactors, WWER-1000 reactors share with the latter a number of similar design and construction features. There are 19 operating WWER-1000 reactors in Bulgaria (2), Russia (6) and Ukraine (11). There are a few more WWERs under construction. In addition to reactors already shut down, some other reactors will reach the end of their design lifetime in a few years time, and become candidates for decommissioning. As WWER-440 reactors were put into operation before WWER-1000 reactors, it is assumed that the former will undergo decommissioning first. It is within this context, and taking into account Member States' interest in taking part in a regional project on planning and management of decommissioning for WWER nuclear power plants (NPPs), that the IAEA launched a regional technical co-operation project in 1994.

For the above reasons, this project has focused on WWER-440 reactors for which, in contrast with forthcoming needs, little consideration had been devoted to decommissioning at the plant design and construction stage, and little emphasis had been placed on planning for decommissioning until a few years ago. An additional complication is that information on some design, construction and operational aspects which will inevitably affect decommissioning is not readily available. The project was thus directly related to a major need in the region of central and eastern Europe (CEE). Although the project's emphasis is on WWER-440 reactors, Member States with prevailing experience on WWER-1000 reactors were also invited to share it with others. Much of the information from the project is applicable to WWER-1000 reactors as well.

2. TECHNICAL CO-OPERATION MECHANISMS FOR IMPLEMENTATION OF THE IAEA PROJECT

The project, which has had a duration of four years (1995–1998), included the organization of regional workshops and scientific visits to countries having WWERs and other countries where active decommissioning projects are underway. The project called upon the expertise already gathered in the countries which have accumulated experience in decommissioning reactors (particularly WWERs), to provide advice to other countries where this experience is lacking.

A mixed approach was initially envisaged for any given workshop, and is described below. It has been possible to implement the following mechanisms in parallel:

(a) **peer review of selected decommissioning projects/studies**. This mechanism was ideally implemented when the NPP subject to peer review or a given country in CEE was visited by the committee of experts under the regional project. To improve efficiency of the peer review process, the host country was requested to send relevant information to committee members well before the workshop. The scope and extent of the peer review process depended on the information provided by the host country and followed given indications. Although informally conducted, this mechanism managed to create an active debate among committee members and, at least, highlight solutions to common issues in CEE countries. Table I gives details of peer reviews conducted under the regional project.

(b) **progress reports**. Countries that are developing decommissioning projects and studies were invited to present their progress achieved at the project workshops.

(c) **discussion on specific topics**. Such topics were assigned to specified experts from and/or outside the region. A list of a few subjects dealt with at project workshops is provided below (see also Table II):

- (i) selection of a decommissioning strategy,
- (ii) pre-decommissioning radiological and physical characterization,
- (iii) decontamination technology,
- (iv) dismantling technology,
- (v) treatment, conditioning, transportation, storage and disposal of decommissioning waste, including clearance levels,
- (vi) cost assessment and funding,
- (vii) design, construction and operation to facilitate decommissioning, and
- (viii) drafting of decommissioning plans.

(d) **site visits**. Visits were arranged to sites where WWERs, or research and development facilities, are located, or sites where active decommissioning projects are underway. Table III provides more details on site visits conducted under the regional project.

Table I. Regional project peer reviews

Paks, Hungary, November 1996. Paks Units 1–2: Preliminary Decommissioning Plan
Bratislava, Slovakia, May 1997. Documentation for NPP-V1 Operation Termination (covering activities immediately before and after permanent shutdown)

Table II. Regional project topical subjects as discussed at project workshops

Workshop Subject	Vienna, IAEA, June 1995	Greifswald, Germany, March 1996	Paks, Hungary, November 1996	Bratislava, Slovakia, May 1997	Mol, Belgium Dec. 1997
Selection of a decommissioning strategy	B, C, G, R, H, S, U, FF, F, USA	J, USA, U, R	J	J	
National progress reports		B, C	C, H	H, C, S	
Pre-decommissioning characterization		B, H, S, U, F, C	R, G, U, USA		
Cost assessment and funding	USA	F, H, R, USA		UK	
Safe enclosure			R, B		
Clearance of solid materials/waste/site			G, USA, IAEA, BE		J
Regulatory infrastructures			S, U	U	
Pre-decommissioning activities			S, B		USA
Management of decommissioning waste			BE		
Development of decommissioning technology	B, FF, G, F, H, S, USA, R	J	J		
Decontamination technology				R, B, H, G, C, J	
Dismantling technology				BE, USA	J
Implementation of decommissioning: decommissioning plans, organizational aspects, QA, record-keeping, contractors				G, UK, USA, BE, R	
Design, construction and operational features to facilitate decommissioning				B	

Note: The letters in the table refer to presentations/reports given by experts from the following Member States: B (Bulgaria), BE (Belgium), C (Czech Republic), FF (Finland), F (France), G (Germany), H (Hungary), J (Japan), R (Russian Federation), S (Slovakia), U (Ukraine), UK (United Kingdom), USA (United States of America).

Table III. Regional project site visits

Greifswald, Germany, 25–29 March 1996 (Greifswald WWER-440/230/213 Decommissioning Project)
Balatonfüred, Hungary, 25–29 November 1996 (Paks WWER-440/213)
Mol, Belgium, 15–19 December 1997 (BR3, Belgoprocess Decommissioning Projects)
Tokai-Tokyo, Japan, 30 March–3 April 1998 (Visit to several D&D Research Centres)

For the purposes of this project, the IAEA played the role of international forum where information and views are shared and debated. Radioactive waste management, including decommissioning of nuclear installations, has a prominent role as an IAEA activity and has to be dynamic in nature to keep pace with the changing needs of Member States. Meeting these needs is a challenge in itself considering the diverse nature of waste management activities that are planned or underway in Member States. The main objective of IAEA's waste management and decommissioning programmes is to provide guidance on safe management of radioactive waste in accordance with IAEA's mandate to promote the safe and peaceful use of atomic energy. The following countries were primarily addressed by this project: Armenia, Bulgaria, Czech Republic, Hungary, Slovakia and Ukraine. Designated experts from these countries took part in the above-mentioned workshops and delivered the papers that are the primary source of information for this publication. Invited experts from Belgium, Finland, France, Germany, Japan, the Russian Federation, the United Kingdom and the United States of America provided information on decommissioning planning and management based on their national experience. It is felt that participation of experts from these countries has greatly contributed to technology and know-how transfer to project recipient countries. Although environmental and other conditions may differ from country to country and from site to site, similarities in technical factors may also be found.

Many IAEA publications pertain to the decommissioning of nuclear facilities (Table IV). Some IAEA Safety Series publications in the field of decommissioning are being prepared within the new RADWASS programme (e.g. [7]). Other existing Safety Series publications are applicable to specific aspects of decommissioning e.g. the regulatory process for decommissioning or pre-disposal management of radioactive waste. Technological aspects of decommissioning of nuclear facilities in general are covered by IAEA technical reports and TECDOCs (Table IV) [8, 9]. However, no IAEA publication specifically refers to WWER decommissioning. In the course of the project, it became apparent that consolidation of advice/guidance and practical experience provided by individual experts needed preparation of a document where all decommissioning aspects, in particular WWER-specific issues, were systematically dealt with and described. Current inadequacies in planning and management of WWER decommissioning were also highlighted and ways to upgrade existing resources and infrastructures identified. Recommendations were also provided to develop information and data needed for timely and successful planning of decommissioning. Indeed, planning for decommissioning is the main objective of this TECDOC. Project meetings held at Mol (December 1997), Tokai (March 1998) and Vienna (November 1998) were substantially devoted to the TECDOC preparation.

Table IV. List of IAEA publications published since 1980 on decommissioning and decontamination of nuclear facilities

Safety Series (SS)	
Factors Relevant to the Decommissioning of Land-Based Nuclear Reactor Plants	SS No. 52 (1980)
Safety in Decommissioning of Research Reactors	SS No. 74 (1986)
The Regulatory Process for the Decommissioning of Nuclear Facilities	SS No. 105 (1990)
Technical Report Series (TRS)	
Decommissioning of Nuclear Facilities: Decontamination, Disassembly and Waste Management	TRS No. 230 (1983)
Decontamination of Nuclear Facilities to Permit Operation, Inspection, Maintenance, Modification or Plant Decommissioning	TRS No. 249 (1985)
Methodology and Technology of Decommissioning Nuclear Facilities	TRS No. 267 (1986)
Methods for Reducing Occupational Exposures During the Decommissioning of Nuclear Facilities	TRS No. 278 (1987)
Decontamination and Demolition of Concrete and Metal Structures During the Decommissioning of Nuclear Facilities	TRS No. 286 (1987)
Factors Relevant to the Recycling or Reuse of Components Arising from the Decommissioning and Refurbishment of Nuclear Facilities	TRS No. 293 (1988)
Monitoring Programmes for Unrestricted Release Related to Decommissioning of Nuclear Facilities	TRS No. 334 (1992)
Cleanup and Decommissioning of a Nuclear Reactor After a Severe Accident	TRS No. 346 (1992)
Application of Remotely Operated Handling Equipment in the Decommissioning of Nuclear Facilities	TRS No. 348 (1993)
Planning and Management for the Decommissioning of Research Reactors and Other Small Nuclear Facilities	TRS No. 351 (1993)
Decontamination of Water Cooled Reactors	TRS No. 365 (1994)
Decommissioning Techniques for Research Reactors	TRS No. 373 (1994)
Safe Enclosure of Shut Down Nuclear Installations	TRS No. 375 (1995)
Design and Construction of Nuclear Power Plants to Facilitate Decommissioning	TRS No. 382 (1997)
Decommissioning of Nuclear Facilities Other than Reactors	TRS No. 386 (1998)
Radiological Characterization of Shut down Nuclear Reactors for Decommissioning Purposes	TRS No. 389 (1998)
TECDOCs	
Decontamination of Operational Nuclear Power Plants	IAEA-TECDOC-248 (1981)
Decontamination and Decommissioning of Nuclear Facilities	IAEA-TECDOC-511 (1989)
Decontamination of Transport Casks and of Spent Fuel Storage Facilities	IAEA-TECDOC-556 (1990)
Factors Relevant to the Sealing of Nuclear Facilities	IAEA-TECDOC-603 (1991)
Considerations in the Safety Assessment of Sealed Nuclear Facilities	IAEA-TECDOC-606 (1991)
National Policies and Regulations for Decommissioning Nuclear Facilities	IAEA-TECDOC-714 (1993)
Decontamination and Decommissioning of Nuclear Facilities	IAEA-TECDOC-716 (1993)
New Methods and Techniques for Decontamination in Maintenance or Decommissioning Operations	IAEA-TECDOC-1022 (1998)
Technologies for Gas Cooled Reactor Decommissioning, Fuel Storage and Waste Disposal	IAEA-TECDOC-1043 (1998)
Nuclear Data Series (NDS)	
Nuclear Data Requirements for Fission Reactor Decommissioning	INDC (NDS)-269 (1993)
International Benchmark Calculations of Radioactive Inventory for Fission Reactor Decommissioning	INDC (NDS)-355 (1996)

3. OBJECTIVES AND SCOPE

This TECDOC is expected to serve for general orientation and guidance of decision makers, regulators and operators of WWERs in CEE. Because the TECDOC has no ambition of being a decommissioning handbook, the guidance provided herein should be complemented by practical details available from other sources. It is expected that the IAEA project, and in particular the project outcome collected in this TECDOC, will attract Member States' attention on timely planning for decommissioning. In some Member States of CEE, in particular those with older WWERs, this is expected to lead to the formulation of specific decommissioning plans. Competent authorities, in particular those involved in decision-making, play a major role in the process. The TECDOC provides Member States with an overview of existing/proposed infrastructure and provisions, including legislation and regulations, and information/guidance on planning and management aspects of decommissioning, and on relevant technologies. This is intended to offer the directors and managers of utilities and the regulatory bodies the opportunity to estimate financial and other impacts from the decommissioning of their installations, so that decommissioning plans may be drafted with no undue delay. Aspects such as fuel and waste management and provision of other technical, administrative and financial resources need timely preparation. It is expected that the end result of the project would increase regional co-operation among CEE Member States which share a number of social, economical and technological aspects/issues. The project may also have a positive impact on developing countries outside Europe.

In more general terms, the project will contribute to enhance Member States' organizational capabilities at large. As decommissioning is a multi-disciplinary process, the project will stimulate Member States to develop an integrated approach to decommissioning by making use of resources available both domestically and internationally.

This TECDOC covers all decommissioning activities beginning with the permanent shutdown of a reactor and ending with site release/reuse. Although defuelling and removal of spent fuel from the reactor building/site are not considered in many Member States to be part of decommissioning, these activities have been addressed in this publication taking into account their importance to determine strategy, timing and scheduling of decommissioning. Similar considerations apply to other pre-decommissioning activities such as removal of operational wastes from the site. Preparatory activities such as evaluating social aspects, establishing infrastructures, conducting radiological characterization or decommissioning planning are also important parts of this publication.

Following introductory sections on background and mechanisms of the regional project (Sections 1 and 2), and objectives and scope (Section 3), this TECDOC describes the decommissioning process and provides generic guidance on factors important for the decommissioning of nuclear reactors (Section 4). Section 5 gives guidance on the planning and implementation of decommissioning projects, with a focus on organizational aspects. Section 6 deals with factors relevant to the selection of a decommissioning strategy for WWERs and gives examples of specific strategies as selected in CEE countries. Section 7 gives an overview of typical radioactive inventories in WWERs, including activation and contamination levels. Section 8 provides information on decontamination, dismantling and waste management technologies for WWERs and related available operating experience. Section 9 draws conclusions and Section 10 gives recommendations for future work. The TECDOC is complemented by appendices giving basic information on WWER design and construction features, decommissioning practices and comprehensive lists of WWER-440 units in CEE. These appendices provide details on national policies and schemes relevant to decommissioning infrastructures, planning and other important factors.

4. THE DECOMMISSIONING PROCESS

4.1. INTRODUCTION

The term “decommissioning” is used to describe the set of actions to be taken at the end of the useful life of a facility, in order to retire the facility from service while, simultaneously, ensuring proper protection of the workers, the general public including future generations and the environment. The ultimate goal of decommissioning is unrestricted release or use of the site. The time period to achieve this goal may range from a few to several hundred years [10].

General guidance provided in the following subsections is extracted from various IAEA publications (see Table IV) [7]. A significant role in the decommissioning process may be played by the newly issued Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management [11]. More details and applications of IAEA guidance to WWERs, including encountered or expected issues, are given in Sections 5–8. An outline of the decommissioning process is given in Fig. 1.

4.2. REGULATORY PROCESS

In principle, the regulatory process for a decommissioning undertaking is the same as for other types of activities involving the handling of radioactive materials. The national legislation defines the process in detail, taking into consideration the specific national situation. The main factors which influence the process are the following:

(a) The constitutional and legal system of the Member State;

(b) The distribution of authority and jurisdiction among various government agencies and departments;

(c) The ownership, organization and structure of the nuclear industry,

(d) The technical, personnel and financial resources available to all parties involved,

(e) The actions that may have to be performed immediately after termination of operation under the provisions of the operating licence, such as those delineated in Section 4.5.

4.3. SELECTION OF A DECOMMISSIONING STRATEGY

When a decision has been made not to return a facility to service, there are two basic decommissioning options. These are:

(a) Immediate dismantling (defined in former literature as IAEA Stage 3 decommissioning); or

(b) Deferred dismantling following one or more periods of safe enclosure (defined in former literature as IAEA Stage 1/Stage 2, followed by Stage 3).

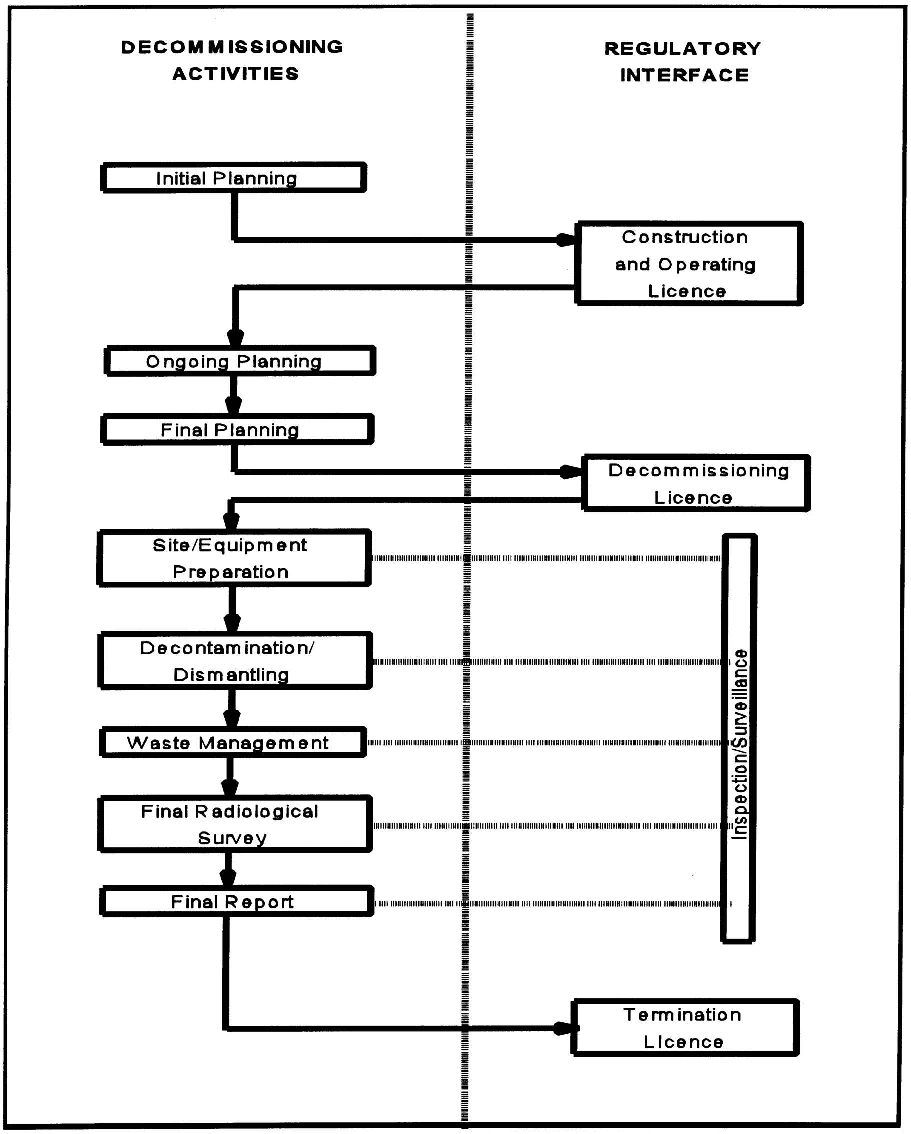


Fig. 1. The decommissioning process.

The generic expression "safe enclosure" refers to the phases of the decommissioning process during which a nuclear facility (or parts of it) is kept under surveillance, and monitored and maintained as considered appropriate from the technical and safety points of view. In situ entombment of a nuclear power reactor is not in use except in special cases. The most important factors relevant to the selection of a decommissioning strategy are:

- Legislative and regulatory requirements
- Decommissioning cost and funding
- Spent fuel management strategies
- Radioactive waste management infrastructure
- Criteria for removal of materials from regulatory control

- Social and public acceptance aspects
- Condition of the plant and ageing processes
- Owner's interest, including planned use of the site
- Availability of resources
- Radiological aspects.

Each factor effecting the selection of a decommissioning strategy cannot be examined without proper consideration of the conditions specific to the facility under consideration to arrive at a satisfactory decommissioning plan. Each of the above mentioned factors is briefly discussed in the Sections 4.3.1–4.3.9 and 4.7 in a general manner. Applications of this general guidance to WWERs are discussed in more detail in Sections 5–10.

4.3.1. Legislative and regulatory requirements

Member State legislative and regulatory requirements may to some extent dictate the strategy to be followed and may take into account any or all of the factors considered below. These requirements may prohibit certain strategies from being considered. They may also impose certain conditions such as time limits on safe enclosure phases.

4.3.2. Decommissioning cost and funding

The funds required to carry out decommissioning can be split into five categories:

(a) post-operational costs;

(b) care and maintenance costs;

(c) dismantling costs, including cleanup of contaminated soils;

(d) waste management costs; and

(e) overheads, including management, licensing, documentation, quality assurance and site development.

The first category may include, inter alia, the cost of treatment of operational waste, defuelling and further fuel management (see Section 4.3.3). If a facility is held in a state of safe enclosure, there is an on-going requirement for equipment and resources to maintain the plant in a safe state, resulting in additional care and maintenance costs. Waste management costs include the costs of waste conditioning and packaging, transport and disposal. Some of these costs may increase significantly with time during a safe enclosure period. On the other hand some components of dismantling and waste management costs may also be higher in immediate dismantling, as radioactive decay tends to simplify those activities in a deferred dismantling scenario. Whatever choices and decisions are made, it is the responsibility of the licensee of the plant to make financial provisions to cover the decommissioning costs in accordance with pertinent national legislation and funding requirements.

A financial case will be made for various decommissioning options. These estimates will be periodically reviewed, approved and revised as needed, and may affect the selection of the decommissioning strategy. Eventually detailed costs and the rate and time-scale of expenditures will be estimated for the selected option. The detailed cost estimate would be done as a joint activity between the technical management staff, who define the work packages and resources required, and those staff who have the cost estimating expertise.

4.3.3. Spent fuel management strategies

In some Member States, spent fuel management is not considered part of the decommissioning process, since it is assumed that the removal of fuel from the facility is a prerequisite for the implementation of major dismantling activities. However, experience shows that spent fuel management may strongly affect the selection of a decommissioning strategy. In particular, facilities to store, dispose of or reprocess spent fuel may not be readily available and the fuel must remain in the reactor facility. In several Member States contracts were negotiated with other Member States to transfer spent fuel, but now, for various reasons, this has become difficult or even impossible. The lack of a transfer route for spent fuel may force a licensee into a safe enclosure strategy with spent fuel in the facility. In general, it is desirable to remove spent fuel off-site or to a facility independent of the power plant as soon as possible.

4.3.4. Radioactive waste management infrastructure

The generation of radioactive wastes in various categories is a direct result of the decommissioning of radioactive facilities. The extent of waste arising in various categories will be influenced by the timing of decommissioning operations. Deferred dismantling may reduce the amounts of intermediate level waste (ILW) and increase amounts of low level waste (LLW) while some waste may reduce in activity to categories possibly exempt from regulatory control. This will influence disposal arrangements and costs. When dismantling contaminated structures in nuclear power reactors, very large LLW volumes can be generated. Management of operational waste may add a considerable burden to the decommissioning management.

If no suitable disposal facilities are available for the amounts and categories of waste, then the following options exist:

(a) maintain the whole nuclear facility or parts of it in a safe enclosure mode; or

(b) condition the waste and store in appropriate interim waste stores or containers.

Waste storage arrangements for conditioned waste may also be difficult to put in place. These considerations therefore will influence the timing of final dismantling and the period of safe enclosure. One important point is the need for conditioned wastes to fulfil the acceptance criteria of the disposal facility, as repackaging would result in unnecessary doses and additional cost. Particularly important in the context of decommissioning is the one-piece removal and disposal of large components.

Safe enclosure may not be considered to be an alternative long term strategy to the identification and qualification of a disposal site for decommissioning waste.

4.3.5. Criteria for removal of materials from regulatory control

Large amounts of materials resulting from decommissioning exhibit very low levels of activity or could be readily decontaminated to achieve such levels. Assuming that these materials should be all managed and disposed of as radioactive wastes would result in unnecessary penalties in terms of operational difficulties and significant extra costs. It is generally possible to establish radiological criteria and associated activity levels according to which materials can be released from regulatory control.

Three general classes of ways of removing solid materials/wastes from the facility can be identified as follows:

(c) Clearance for unrestricted reuse or disposal;

(b) Authorized release/reuse within the nuclear industry or in the public domain; or

(c) Storage /disposal under radiologically controlled and monitored conditions.

Criteria to be met for these classes vary between countries. Sometimes the criteria are based on nationally applicable regulations, while in other practices, they are based on a case-by-case evaluation. International recommendations are under preparation, and an interim report was issued [12]. The availability of suitable disposal facilities can be limited. The strategy for waste management can vary; however, consideration to achieving clearance or authorized release is important to reduce the volumes of radioactive waste for storage/disposal.

The guiding radiological criteria for clearance are expressed in terms of dose and cannot be used directly for establishing clearance levels. Hence it is necessary to derive practical quantities, which could be activity concentration (Bq/g), surface contamination (Bq/cm^2), total activity per unit time (Bq/year) and total mass per unit time (tonnes/year).

Release criteria are the significant factor in determining whether recycle and reuse practices can be applied on a large scale. The availability of national policies and long term strategies in support of recycle and reuse principles may have a profound impact on the efficiency and extent of such practices and determine the overall decommissioning strategy.

4.3.6. Social and public acceptance aspects

The process of deciding between the different decommissioning strategies may take into consideration the possible effects of factors such as:

(a) employment problems;

(b) environmental concerns (e.g. the value of the neighbouring land and the appearance of the site);

(c) impact on local infrastructures, in particular those more directly related to the plant’s activities (maintenance services, public transportation, catering, etc.);

(d) the public's understanding of the hazard and their perception of the related risks, whether the facility is maintained in a safe enclosure condition or is dismantled.

Public opinion about the proposed choice is usually taken into account in the procedure whereby the decommissioning strategy is submitted for the approval of the relevant authorities; public participation mechanisms vary from country to country.

In a situation where the regional industrial structure is poorly developed employment problems may heavily affect the selection of a decommissioning strategy. In general, the major societal impact occurs prior to active decommissioning at the shutdown of the plant, which reduces the work force nearly at once, as electricity generation ceases and safety requirements are reduced. The overall reduction takes place over a minimum period of a few years and this should tend to mitigate the adverse impact of job losses and reduced income to the regional community. Decommissioning activities may mitigate such impacts. In the case of safe enclosure, the work force may decrease to a small security and maintenance crew for the period of continuing care. Prudent planning should ensure that reduction of the number of employees will not be sudden or without warning, so people will have time to find other employment.

Tax revenues could also be lost to the local communities and to the state, but here again, the impact will be spread over a period of time. As employment reduces and people leave the area, public services will also reduce.

4.3.7. Condition of the plant and ageing processes

The condition of a shut down nuclear facility influences decisions concerning decommissioning from two viewpoints: the characteristics of the residual radionuclide inventory within the facility, and integrity and maintainability of the facility and its systems.

The quantities, the physical and chemical forms, and the half-lives and types of radiation characteristic of the radionuclides present in a shut down nuclear facility can influence decisions on when and how to decommission the facility. In general, reduction of radiation fields due to radioactive decay during safe enclosure periods, will simplify dismantling and the management of any generated waste. However the reduction of activity may make detection more difficult. (See Section 4.7). More details on radiological characterization of WWERs are given in Section 7.

As existing nuclear installations were designed for relatively short working lives, during a prolonged enclosure period there is a potential for gradual deterioration of the structures, systems (e.g. ventilation) and components designed to act as barriers between the contamination and the environment unless they are adequately maintained. This deterioration, which will be determined by the ageing processes, may also apply to systems (e.g. handling systems) that will be necessary during plant dismantling.

To implement safe enclosure, new systems may have to be installed. The long term integrity of existing and new systems is, in turn, a safety issue. Ensuring long term integrity may result in significant costs and be a factor against prolonged periods of safe enclosure.

4.3.8. Owner's interest, including planned use of the site

The choice of a decommissioning strategy may also depend on the following considerations:

(a) the owner may have a shortage of sites for new nuclear or non-nuclear installations;

(b) if the plant to be decommissioned is co-located with other operating facilities that will continue to be in service, safe enclosure or sharing of some facilities may be the preferred choice until all installations are shut down;

(c) as a factor in selecting to proceed to safe enclosure, the owner may wish to consider the re-use of some of the plant facilities for purposes other than those for which they were originally intended or as part of a new or modified plant;

(d) if several decommissioning strategies are available, the owner may wish to optimise expenditures, depending on the economic situation.

4.3.9. Availability of resources

An advantage of immediate dismantling is the retention and utilization of plant expertise on the site during the actual dismantling. This expertise could lessen the potential for unexpected problems and would reduce doses associated with retraining of personnel.

In a deferred dismantling strategy, expertise in the layout, maintenance and operation of the reactor lessens during the safe enclosure period as personnel leave the facility so that at the time of dismantling there may be no one with personal experience of the facility. This expertise will have to be re-acquired at the time of dismantling, with a possible corresponding penalty in costs, occupational exposure and other factors.

It is necessary to maintain full records of the construction, operation and initial decommissioning of the plant through any period of safe enclosure in a manner that ensures their availability at any future time. This is desirable in any event because even if dismantling starts soon after final shutdown, it will continue over sufficient time for experienced staff to have left the project before completion.

Basic technology for the accomplishment of decommissioning is reasonably well known and tested. Availability of technologies is essential for successful completion of decommissioning. These may be already available to the licensee at the plant shutdown time, or have to be procured from the market, including the use of contractors. Deferring dismantling may have the advantage that it is expected that more effective technologies will be available. A more detailed description of decontamination, dismantling and waste management technologies for WWERs is given in Section 8.

4.4. PLANNING FOR DECOMMISSIONING

The licensee is responsible for planning safe, timely and cost effective decommissioning of the facility. Preliminary plans for decommissioning are done at an early stage during facility's lifetime. These generic plans demonstrate the feasibility to decommission the facility. One important aspect is to ensure a financing system for the decommissioning. This includes overall cost estimates and mechanisms to establish decommissioning funds. At this stage there is no need for a detailed analysis of how decommissioning activities should be carried out.

The preliminary planning described above is reviewed periodically during operation and associated funding is revised as needed. It is also important that all the structural and equipment changes be documented. Records are kept of any spills or spread of contamination, including details of the radionuclides and their activity. All these actions are important to ensure that the decommissioning staff can have access to a full set of drawings and other documentation of the real situation in the facility. This database is reflected in the decommissioning plan.

Before or after permanent shutdown, depending on national legislation, an application for decommissioning is prepared and submitted to the regulatory body. A decommissioning plan to support the application is included. It is imperative that this decommissioning plan be prepared well before final shutdown occurs. Depending on what decommissioning options are chosen, some parts of the plan need to be elaborated in more detail. This decommissioning plan typically includes the following:

(a) Description of the plant, site, and operational history relevant to decommissioning.

(b) Applicable laws, regulations and other standards, including a review of existing operational requirements and their applicability during decommissioning. Particularly important is the availability of clearance criteria for waste or other materials.

(c) Results of surveys, laboratory measurements and calculations to determine the inventory and distribution of radionuclides in the facility.

(d) Choice of the decommissioning strategy, with a description of the steps involved in effecting its completion. Generally, the rationale for the choice of a given strategy should be elaborated, with allowance for such factors as radiological aspects, the availability of waste disposal or storage facilities and the long term integrity of buildings and structures.

(e) Description of techniques, tools and procedures to be used for the decommissioning. This should also include a classification of systems e.g. (1) those operational during decommissioning phase; (2) kept in a standby mode; or (3) to be utilized during particular decommissioning activities.

(f) Time schedule for major decommissioning activities.

(g) Analysis of radiological and non-radiological risks for workers.

(h) Environmental impact analysis, including estimation of maximum individual doses to the public and collective dose commitment from the airborne and liquid releases during decommissioning, as well as from steps such as transport of radioactive materials, recycling and reuse of components and other materials, waste management and residual activity left on the site.

(i) Safety assessment and evaluation of the radiological consequences of postulated accidents during decommissioning.

(j) Description of the anticipated inventory (activity, volume, mass, location, physical and chemical form) of radioactive, hazardous and other wastes/materials arising from the decommissioning and the manner of disposing of wastes or recovering materials, including means for handling, transporting and storage.

(k) Management structure, staff qualifications, training and certification procedures, and quality assurance programmes.

(l) Description of the measures to ensure radiation protection, health and safety of the workforce and the public.

(m) Emergency planning during decommissioning.

(n) Description of the methods proposed for the final radiometric measurements to ensure compliance with limits for clearance or authorized release/reuse of equipment, materials, buildings and the site.

(o) The need for a final decommissioning report at the completion of the planned decommissioning.

(p) Estimation of decommissioning costs and funding mechanisms.

(q) Interfaces with the regulatory bodies and competent authorities throughout the decommissioning process.

More details on the planning process are given in Section 5. A simple view of the decommissioning process including planning is given in Fig. 1.

4.5. SHORT TERM POST-OPERATIONAL ACTIVITIES

After final plant shutdown, typically the following work will be performed to prepare the facility for either immediate dismantling or safe enclosure. All of the activities will be performed under applicable regulatory provisions.

(a) Removal of the spent fuel from the unit and possibly from the site;

(b) Draining the coolant and process fluids;

(c) Cleaning or fixing of contamination in selected areas;

(d) Treatment, conditioning, storage and/or disposal of operational waste;

(e) Establishing a comprehensive radiological inventory including radiation surveys and dose mapping of contaminated work areas;

(f) Performing various housekeeping activities;

(g) Compiling and maintaining a comprehensive set of operational records relevant to decommissioning including radioactive spills and other incidents;

(h) Compiling a full inventory of equipment and structures including as-built drawings and plant modifications;

(i) Evaluating the need for operating or standby systems and establishing maintenance and surveillance provisions.

Any other activities which do not conflict with existing regulatory provisions can be performed during this period and have the advantage that experienced personnel and intact systems are available. For example, if it is planned to perform a full system decontamination, it is typically undertaken during this preparatory period. The decision to decontaminate the plant at this stage can have a strong impact on occupational exposures if immediate dismantling is performed.

4.6. IMPLEMENTATION OF DECOMMISSIONING

There are many activities performed during the implementation of the decommissioning process. These activities will involve a large number of workers, large amounts of money and a variety of different tasks. In order to perform these activities in a cost effective and efficient manner, good project management arrangements must be in place. A key to good management begins with the proper organization. It is important that the organization be complete to handle all aspects of project management, but not too large as to cause a financial burden on the project. The team normally comprises a proper mix of operational personnel and others specialized in decommissioning activities.

Another important factor in the management of the implementation of the project is to ensure that any regulatory or other approvals are received in a timely manner. These may include the approval of major work procedures and other documentation. A system for documentation control and for documenting facility changes will also be in place.

During decommissioning activities, large amounts of material will be generated not all of which will be waste. This material flow can be managed with the appropriate number of personnel being assigned to the task and having appropriate treatment, conditioning and monitoring equipment available. To achieve this it is important that the material management strategy and facilities be available before beginning the actual dismantling activities. More details on organizational aspects of decommissioning WWERs are given in Section 5.

4.7. RADIOLOGICAL ASPECTS

The radiological inventory during dismantling is much smaller than during operation, because the spent fuel has been removed. However the usual precautionary measures for radiation protection still need to be applied, and in particular doses to personnel must be kept as low as reasonably achievable. Doses may be reduced through a strategy of deferred dismantling to allow for radioactive decay of short-lived radionuclides and by decontamination of systems prior to dismantling. However, the dose burden during the safe enclosure phase or decontamination activities is not necessarily negligible. Alpha contamination can be an issue due to fuel leakages or other incidents. It should also be recognized that with large volumes of LLW being removed from the site, the risk involved in transport and potential diversion of the waste may be significant.

Radiological advantages of deferred dismantling of a facility include:

(a) lower dose rates in the facility;

(b) less radioactive waste to be processed;

(c) lower classification of waste by category;

(d) reduction of the consequences of a possible accident occurring during the subsequent dismantling work.

However, radiological disadvantages of deferred dismantling of a facility include the following:

(a) there is a need for long term surveillance of a facility which contains radioactive materials;

(b) persons with direct knowledge of the facility may not be available for the dismantling (this may make the work more difficult and potentially more dose-consuming);

(c) more difficult determination of radionuclides;

(d) possibility of stricter clearance regulations in the future.

4.8. COMPLETION OF DECOMMISSIONING AND FINAL SURVEY

The final step in decommissioning is to terminate the licence. To support such an application, the licensee needs to demonstrate to the authorities that the requirements for residual radioactive materials have been complied with. To do so, systematic monitoring of the site is necessary. It is the responsibility of the licensee to perform such monitoring. In most cases, the regulatory body or an independent consultant may perform additional monitoring to verify the licensee’s results.

When the facility has been totally decommissioned there will be no remaining radiological risk associated with the facility. All radioactive waste will have been conditioned and placed in proper interim stores or repositories.

The acceptable level of remaining radioactive materials and the procedures to apply for terminating a licence are established during the decommissioning planning phase and endorsed by the regulatory body unless they are already stated in a regulatory document.

5. PLANNING AND ORGANIZATION OF WWER DECOMMISSIONING

This section describes those activities that are performed before and during decommissioning to optimize this final stage in the life of a WWER power plant. It examines the planning needed to be ready for decommissioning and the project planning during decommissioning. These topics are further developed in sections concerned with organization issues, management systems to implement the goals of the project and the effective costing and management of work leading to a safe, economic project completion.

5.1. PLANNING

Generic issues associated with planning for decommissioning were discussed in Section 4.4. Operators of WWER power plant have already made some preparation for the future decommissioning of their plants as is indicated by the content of other sections and annexes of this publication. However, only at Greifswald in Germany has actual decommissioning of WWER plant begun and experience been gained beyond pre-shutdown planning.

The decommissioning of a WWER plant will take place through several phases. Not all phases will occur on each plant due to variations in strategy. Typical phases include:

- pre-shutdown planning
- post-operational phase including fuel removal and treatment of operational waste
- dismantling
- a safe enclosure period
- a deferred dismantling phase after a safe enclosure period
- a period after plant removal until the site is cleared for restricted or unrestricted reuse.

The approach to planning and organization will vary in detail between phases as illustrated in later Sections.

The aim here is to present information on pre-shutdown planning being undertaken by WWER operating countries together with some additional more generic material on planning during decommissioning to supplement Section 4.4 and supporting references.

Most countries operating WWER have established work programmes to prepare for the future decommissioning of their plants. The products of this work are already apparent elsewhere in this publication and its annexes. From the annexes an overall view of progress can be summarized as follows.

In all countries operating WWER their Atomic Acts identify the decommissioning of a nuclear installation as a practice related to nuclear energy utilization. Currently the development of regulations specific to decommissioning is being completed in Armenia, Bulgaria, the Czech Republic, Finland, Hungary, the Russian Federation and Slovakia. In Germany there is no specific law covering the decommissioning of nuclear facilities, instead, the legal provisions are regulated within the German Atomic Law (AtG), which requires a licence for the shutdown of a nuclear facility. The AtG regulates that the operating utility is responsible for the licence application, the decommissioning of its own nuclear facility including conditioning, packaging and interim storage of nuclear fuel, operational and decommissioning waste.

Detailed planning of the implementation of decommissioning is basically similar to that required to undertake any large scale project: appropriate organization, management systems and work management. With a nuclear plant there is the added problem of working in areas with radiological hazards. Generally radiological risks are not higher during decommissioning than during maintenance periods, but other non-radiological risks (asbestos, falling, electrocution, etc.) can be higher if not well managed. A strong safety culture is required even with the lower manning levels typical of decommissioning. This latter point is of particular importance as although the safety case for the defuelled plant may show hazards to be low, there will be some apparent loss of defence in depth represented by the fall in staff numbers and greater reliance on administrative safety controls than in operation. Shift staffing may need to be maintained and a minimum plant staff identified, at least until reduced emergency arrangements are agreed. Such restrictions will impact on the flexibility to perform decommissioning and will need detailed planning as the project progresses.

The early negotiation of staffing arrangements, in the context of regulatory and social factors, will be an important part of the WWER licensee pre-shutdown planning. This also takes into account the actual circumstances in the relevant country which may vary between today and the actual time of decommissioning and thus the whole decommissioning plan needs to be reviewed from time to time.

5.2. ORGANIZATION

A key element in achieving effective decommissioning is having a relevant organizational approach at each stage, in the context of the utility and country concerned. Section 5.2.1. discusses the range of organizational approaches being applied today by WWER operators preparing for decommissioning. Further generic guidance is provided in an IAEA report under preparation[1] and it is not intended to repeat that here.

5.2.1. Organization during preparation for decommissioning

The organization in the period well in advance of final shutdown does not necessarily need to be large. Assistance will be needed from those with detailed knowledge of the plant, technical experts and planning system specialists. It will also be vital for this team to learn from experience elsewhere so as to be able to consider the full range of options.

The number of people employed in decommissioning preparations by WWER operators ranges up to about 10 people. In each case there is a team leader for decommissioning but additional support is provided by plant personnel either as an extension of their normal duties or by temporary secondment. Such a structure corresponds with generic advice in the literature. Typically the work undertaken covers selection of decommissioning strategy, preparation of decommissioning documents, implementation of legislative requirements, data collection for decommissioning, as well as planning and management of research and development activities and other decommissioning topics.

WWER operators' decommissioning organizations evolve as work progresses, and the demands change. Furthermore, the organization of multi-unit sites, where operation and decommissioning of units may happen in parallel, is addressed by the operators as well as for units in isolation.

5.2.2. Organization during project execution

The aim of decommissioning is to take the plant from a complex operating state to a much simpler state thereby reducing the hazard and the costs associated with maintaining the remaining systems and site. During operation there is a recurrent cycle, one or two years in period, where similar sequences of events occur through the years. On a decommissioning plant, although there are still repetitive activities, there is a net change taking place that requires an appropriate project organization. A project manager is essential. The organization at the commencement of decommissioning will inevitably be that which ended the operational phase of the plant's life. The project manager needs to define and evolve the organization to suit the job. Part of the overall planning is to define how the organization, staffing, contractor usage, usage of mobile teams, etc. will be modified and controlled throughout the project.

The organization at a particular decommissioning project will be dependent on local constraints, whilst meeting the overall need for separation of project and operational

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Organization and Management for the Decommissioning of Large Nuclear Facilities (in press).

responsibilities. For example, at Greifswald the structure is based on the pre-existing operations structure but with modifications to recognise new roles in project management and dismantling, and extended roles in areas such as work planning. The structure at Greifswald is given in Fig. 2. The project manager's department (about 140 persons) plans, specifies and controls the dismantling work to be performed by the Plant Manager's staff. The latter's staff currently numbers about 600 plus up to 400 in the new dismantling department. As work progresses the numbers involved on plant operations, maintenance and engineering will fall whilst those in the project management area will remain the same for longer.

For other WWER operators there are no detailed organizations drafted as yet. However, an essential part of costing is to estimate the person-days to be employed and this is done in the context of an organizational model. A cost model for Bulgaria [13] includes a conceptual organization recognising the separation of project and operation derived costs.

5.2.3. Organization during safe enclosure

The safe enclosure phase is characterized by a plant of relative engineering simplicity which may require some operation, inspection, maintenance and monitoring but where the hazards are well defined and controlled.

As the safe enclosure phase is one of stability, there should be no need for any project staff. It is presumed that sufficient work has been performed to allow relaxation of the need for emergency arrangements. The necessary security measures may be able to be reduced and off site environmental monitoring may also be able to be scaled down.

The project phase is implemented to yield a cost optimized safe enclosure phase. The safe enclosure phase may last many decades so that any overmanning could lead to large overall costs. The policies of the country and the operator with respect to the discounting of future costs will have a major impact on what project work is judged economic to perform in order to lower surveillance and maintenance costs.

The size of the safe enclosure phase organization will depend on the level of dismantling and treatment of wastes that has been undertaken. It may even be appropriate to utilise multi-site mobile teams to perform much of the remaining work. A proposed structure is outlined in Fig. 3 and a significant degree of multi-skilling may be achievable in this organization.

Note the role identified related to record maintenance, vital to the successful deferred dismantling (see also Section 5.3.4).

5.2.4. Site clearance and completion organization

The final release of the site will require a simple organization focused on meeting regulator requirements to confirm, by measurement and evaluation as necessary, that the site is fit for controlled or uncontrolled re-use. This will be dependent on arrangements in a country and some guidance on this is provided in the appendices.

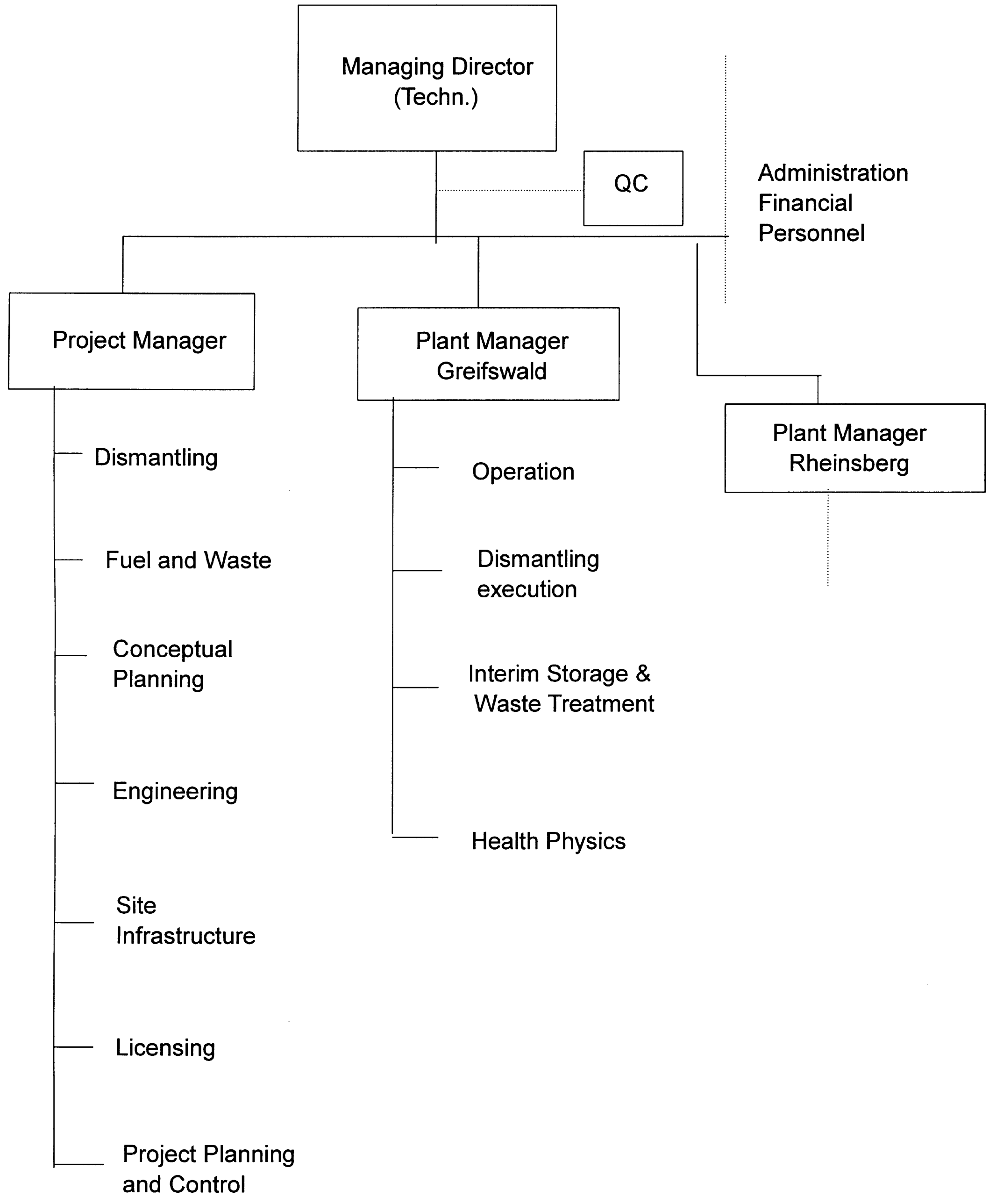


FIG. 2. Organizational structure during Greifswald decommissioning project.

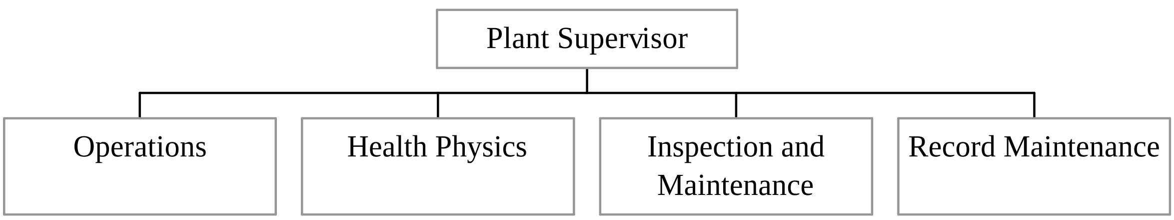


Fig. 3. Proposed organizational structure during a safe enclosure period.

5.3. MANAGEMENT SYSTEMS

The successful achievement of decommissioning depends on the organization being able to work using systems that ensure the right work is done in the right way in accordance with an appropriate quality assurance programme. It is particularly important that appropriate people carry out the work and that they are well selected, qualified and trained.

Effective decommissioning ultimately relies on term records of the site as maintained throughout the design, construction, operation and decommissioning phases. In particular, where a site will be left in a safe enclosure state of low manning for some decades, the existence and quality of records will be the only “site memory”.

5.3.1. Quality assurance programme

A quality assurance programme (QAP) will continue to be needed to control all activities. This includes the ongoing arrangements for the management of operations, safety and the decommissioning projects. It is normally developed from that used during operation, with increased coverage of those aspects associated with the conduct of projects. As such, it needs to evolve as the organization and manning evolves throughout the decommissioning programme. When the reactor is being defuelled or the primary circuit decontaminated, hazards and activities are similar to operation, with similar management controls being required. During dismantling and demolition the management of projects predominates and the QAP needs to reflect this.

If the site goes into safe enclosure with low staff numbers then a significantly reduced QAP is all that is necessary aimed primarily at ensuring stable site conditions and providing adequate record keeping. Thus the QAP needs to be optimized to deliver management of the site suited to the phase of the project.

In order to be able to free release as much material as possible, the QAP needs to provide procedures for the management of materials arising from dismantling. These need to ensure traceability of the materials from dismantling, through any treatment or decontamination and through the surveys allowing clearance of materials.

Most WWER power plant locations are likely to go through a period when one or more units on a site are operating whilst another is undergoing decommissioning. Where units share resources this will require the site organization and QAP to be adapted to cover both the operating and decommissioning units’ requirements.

A reduction in costs, staff and administration generally can be realized by a progressive simplification of the QAP as decommissioning proceeds. This ultimately comes about from modification of the safety case to reflect the lower nuclear hazards as work progresses. This allows a relaxation in the rules for the availability of plant, in maintenance requirements and their implementing documents. Plant can be progressively released from service and decommissioned. The overall effect is to reduce the need for operations and maintenance staff.

It is likely that some contractors will be used. These could range from one or two specialist contracts to the use of turnkey contracts for decommissioning whole areas of the site. The extent of contractor usage will be dependent on the policy on staff retention, and the cost and availability of suitable contractors.

Even with the turnkey type of contract, the site manager is likely to remain accountable for safety and licensing on site. As such he will have systems in his QAP to satisfy himself that the contractor personnel are suitably qualified and experienced, understand the hazards on the site and are adequately supervised.

5.3.2. Plant staffing

Staff numbers are likely to remain high until fuel is removed from the facility. The number of operational staff needed will then fall. The skills needed from the people on site may be different to those in operation as the focus shifts to plant dismantling. A range of approaches to resourcing projects can be followed to suit the plant and its social environment.

There are a number of basic decisions to be made:

- To what extent will operating staff be used to undertake decommissioning project tasks and in which roles?
- Can units share key resources on the site?
- What work will be placed with contractors?
- What operational staff reduction profile is preferred and possible?

The staff reduction profile will be dependent on the work to be performed, recognising that for every person actively involved in a hands-on decommissioning task there may be perhaps six others providing support (from health physics to welfare facilities). Having established such a profile, then commitments can be given to staff as to the length of their remaining employment on the site and progress on demanning can be monitored against the planned profile. In some countries with WWER reactors there will be local reasons to maintain employment at the site, in others less pressure to do so — this will influence the work to be done, the staff reduction profile and the use of contractors.

For example at Greifswald the original operating staff numbered about 6000. Following shutdown, removal of fuel and commencement of dismantling this number progressively reduced to 1300, even though there is a commitment, as far as possible, to use operating staff to perform the decommissioning tasks. Construction of new facilities such as interim waste stores has been performed by contractors.

The desire to maintain a “site memory”, that is people with immediate personal knowledge of the plant and its history, may encourage a higher retention of operating phase personnel. Emergency arrangements may also provide a similar pressure for staff retention.

It will be important to provide appropriate incentives to staff (and contractors) to work effectively and in a manner that delivers the decommissioning programme safely and to time and cost. These incentives may differ from situation to situation. One of the most important management tasks is to give the remaining workforce a clear perspective of the future as one means of motivation. Only in this way is it possible to execute a safe and cost effective project.

The management of staff reduction, identification of alternative employment, retraining, key skill retention and incentivization to complete the work will all be significant Human Resources (HR) challenges, likely to be specific to each site. Barriers to changing staff roles may require new contracts, pay profiles and the dealing with demarcation issues, including between operational staff and contractors. Decommissioning needs to be approached with well developed HR plans as much as engineering plans.

5.3.3. Qualification and training

If the operational staff are to undertake decommissioning tasks then they will require retraining in new skills and reorientation of attitudes towards a project completion outlook.

The pressure to reduce costs by reducing manning will tend to promote a degree of multi-skilling of tasks or skills that may have been seen as a protected craft in the operational phase. As such, HR issues on demarcation may arise as mentioned in Section 5.3.2. above.

Where operating staff are retrained in decommissioning tasks then this can be provided under contract from specialists who may go on to supervise the operating staff work.

Finally it is important to recognise that supervisors and managers may also need retraining if they are to operate as effective project managers providing appropriate team leadership. This was an important process at Greifswald in the transition from operation to decommissioning. In addition at Greifswald, more practical retraining has been necessary to allow the operating staff to undertake dismantling e.g. welding or specialist cutting.

5.3.4. Records

Records are primarily needed to establish and maintain an accurate inventory of the radioactive and other hazardous materials on the site and how they are contained or controlled within the as-built design. There may be other statutory and advisory reasons to keep records.

An adequate record management system is required from design conception onwards. Where limitations are identified in the records of the plant then these should be rectified during operation. There needs to be a system for identifying what records to keep and for how long.

Experience at Greifswald suggests that the following records have proved particularly useful in decommissioning that plant:

- Plant drawings and material specifications
- Plant equipment data sheets
- Waste inventories
- Radioactive history of the plant including incident reports, information on hidden contamination (e.g. under repainted surfaces) and contaminated ground
- Dose surveys of the plant

This list is not exhaustive.

Experience of WWER operators to date (e.g. in the Czech Republic, Bulgaria and Slovakia) is that in general adequate records exist for decommissioning and where there are uncertainties these are being removed by surveys and other campaigns. Work is also in hand to store records in an optimum medium for the long term.

The characterization of records could include the following:

- Requirement for the record (e.g. which legislation)
- Record type (i.e. what is recorded)
- Retention period
- Producer of record
- Responsible holder of record
- Storage medium
- Storage location(s)

Having established the records to be kept and the duration there is the matter of the medium. Four media are primarily in use today:

- Paper
- Microfilm
- Editable electronic (magnetic disks/tapes)
- Read only electronic (CD-ROM image)

Table V summarises the advantages and disadvantages of each.

For an older plant the solution will probably be a mixture of media to suit the record type, in all cases with reasonable backing up of key records.

Table V. Advantages and disadvantages of record-keeping media

Medium	**Paper**	**Microfilm**	**Editable electronic**	**Read only electronic**
Advantages	Readily available. Storage understood. Legally acceptable.	Compact. Standard technology Legally acceptable.	Can be updated. Compact. Accessible. Easy Storage.	Compact. Accessible. Easy storage.
Disadvantages	Bulky. Requires controlled storage conditions.	Awkward to access. Requires controlled storage conditions.	Need to keep hardware & software available. Only useful for new records. Corruptible. Legality unclear.	Need to keep hardware & software available. Legality unclear.

5.4. WORK MANAGEMENT

This section considers the project and contract management necessary to safely deliver the decommissioning project to the time, cost, and quality assumed in the strategy.

5.4.1. Preparations for decommissioning

The planning, costing and implementation of decommissioning all require a work breakdown structure to identify the work to be done and the resources which will have to be brought to bear to achieve it (examples of approaches are presented in [13, 14].

The first call for a work breakdown structure is in estimating the likely costs of decommissioning in the pre-decommissioning planning phase. At this stage an overly precise approach will mislead in terms of the real costs and so the approach taken is to define a basic work breakdown structure, cost it reasonably and then assign uncertainty and risk factors.

An overall cost methodology is used in accordance with a specific cost breakdown structure and this will be related to the work breakdown structure used in planning of the work itself. The individual costs are made up of a number of elements to form the base cost to which are added a contingency allowance and risk margin. Calculated costs are brought to a common price ruling date by the application of cost escalation factors. Because the decommissioning activities are performed over long periods of time future costs may also be discounted.

As decommissioning approaches and the project manager is appointed, he can begin his more detailed work planning based on this early work and in the context of his project and contract management strategies.

5.4.2. Decommissioning project management

Although the project manager has the benefit of the work performed before shutdown he will need to test the appropriateness of details of this as work progresses, in order to benefit from a flexible approach as opportunities and threats arise.

For example, management of the site will need to include decision making on particular sub-projects to undertake within the context of the overall decommissioning strategy. Particular decisions can be aided by guidance on whether sub-projects should be undertaken now or deferred to later. For example, where a job is necessary for safety it shall always be done, where necessary to deliver an environmental benefit it should be done wherever practicable. If a job would lower the overall decommissioning cost then it would be expected to be done and where it would be valuable to perform the job for some other reason to the benefit of the operator then it may be performed if finances are released. Such transparent decision making systems aid the project manager to manage the project in the most effective way.

The implementation of the project and the time scales in which it will be performed will be constrained by the relationship with various regulators who will have an interest in what is happening at the plant. These are likely to include the nuclear safety regulator, environment protection regulators, regulators of industrial safety and also perhaps, local planning authorities and the holders of the funds which must be released in order for the work to progress.

A project risk assessment would usually be performed and may include tasks such as:

- decide the scope of the assessment;
- identify the risks to project costs and time scales, their likelihood of occurrence, their consequences and thus identify those that are significant;

- perform sensitivity studies to test the identification of significant risks;
- put in place actions to manage these risks.
- review and revise as work progresses.

Where contractors are to be used the contract strategy must be defined. Issues to address include: are partnerships to be sought with a small group of contractors; to what extent will risk be laid off on the contractor; will internal and external suppliers be in competition with each other?

These issues arise in many situations outside nuclear plant decommissioning, however, the nature of national laws, licensing and regulations puts an extra constraint on what can be delegated to contractors. For example, the dismantling of an uncontaminated building may be readily contracted on a competitive basis; the decontamination of components to free release levels may be appropriately undertaken by contractors but some risk management by joint pilot studies may be appropriate; the safety case for a waste management activity may be best retained in-house, even if some contractor resource is used, because of the need for the operator to demonstrate ownership and responsiveness to the regulator.

Finally a key part of project management will be the monitoring of progress through meetings, regular reports, performance statistics, audits of activities, post-task and post-project reviews and by benchmarking performance against similar projects elsewhere.

Key performance indicators may include:

Safety: Injury frequency rate per number of person-hours worked

Actual radiation doses compared to pre-job assessments

Events reportable to the regulators.

Project: Project lifetime decommissioning costs (cash and discounted)

Material and waste volume quantities

Surveillance costs

Site infrastructure costs

Staffing numbers

Expenditure profiles– planned and actual earned value

Estimated cost of sub projects to completion versus sanctioned sums

Milestones achieved.

There may be other useful performance indicators (e.g. for safety, dose commitment per unit material dismantled).

5.4.3. Experience feedback

In many plants the lessons learnt on that plant will be applicable to the same plant later in decommissioning, to other plants in the country or to similar plants in other countries. Thus, there is value in collecting experience not only to improve project implementation as it progresses, but also to aid work on later projects.

This experience will be varied in nature. One key area is the actual cost of work against the estimates underpinning the funds provided for decommissioning. As these funds will have been based on costs plus uncertainty and risk factors there will often be the opportunity to identify reductions in the funds necessary for later work — a real saving as money can be released for other uses.

Reports are written at the completion of each phase describing the work done, the problems met, how these were overcome including what did not work well, the resources used and the costs borne.

6. DECOMMISSIONING STRATEGIES FOR WWER TYPE NPPs

There are a number of factors that require attention when determining a decommissioning strategy for any nuclear power plant. These factors are generally discussed in Section 4.3. Factors discussed below currently influence the preferred national strategies for decommissioning of WWER NPPs.

6.1. SELECTION OF A DECOMMISSIONING STRATEGY

Because most WWER type NPPs are only in mid-operation, and a possibility exists to extend their operational lifetime, planning of decommissioning is in conceptual stage for most WWERs. Two countries where the final decision on decommissioning strategy for WWERs has been made are Germany and Finland. In these countries, the selected early dismantling strategy was mainly the result of a cost evaluation including uncertainties of this strategy versus safe enclosure and deferred dismantling. Early dismantling was also justified by availability of funds, waste disposal and interim spent fuel storage facilities, and the expectation of no significant excess of radiation exposure for the staff. The socio-economic needs of the region in particular due to the simultaneous shutdown of five reactor units in Greifswald as well as the need for optimum use of existing personnel and their historical memory of events were also important factors for the selection of early dismantling.

The early dismantling strategy can also include the deferred dismantling of the reactor vessel, internals and biological shielding. The immediate dismantling of the reactor and internals can be disadvantageous as it requires the following provisions:

- extensive use of remote techniques;
- heavy shielding of dismantled parts for storage, transport, or disposal (e.g. shielded casks);
- long term interim storage of dismantled conditioned/unconditioned parts before disposal, and possibly repackaging as a consequence of changes in disposal acceptance criteria; and
- higher costs as a consequence of the above mentioned items.

Specific problems of countries with WWERs may affect the planning and implementation of decommissioning activities. Radioactive waste management problem areas include waste conditioning technologies, the lack of adequate storage and disposal facilities, and insufficient legislative support. The current decommissioning strategy for most WWERs in CEE countries based on safe enclosure of several decades after final shutdown and deferred dismantling is due to the fact that only recently have great efforts been given to legislation and to assuring the financial sources for decommissioning activities are available when needed.

It was concluded in decommissioning studies [15–19] prepared for Bulgaria, the Czech Republic, Hungary and Slovakia, that decommissioning could be facilitated by safe enclosure in various ways such as [20]:

- giving time to set up waste management strategy and to provide adequate facilities for conditioning, storage and disposal of decommissioning waste;
- taking advantage of radioactive decay;
- seeking improvement in the economics of decommissioning, including building up funds;
- taking advantage of technical development; and
- establishing standards for recycling, clearance of materials and very low level waste management.

Due to uncertainties prevailing at the time when preliminary decommissioning studies were prepared, the evaluation of decommissioning strategies for WWER-440 NPPs was based mainly on the decrease of radioactivity and dose rates during safe enclosure. Reactor dismantling was considered as the most significant technical aspect for the decision on optimal duration of safe enclosure. In these preliminary decommissioning plans, the radioactive decay of ^{60}Co was the only radiological factor taken into account to determine the time for deferred dismantling. Based on these evaluations, the selected duration of safe enclosure was 50–70 years. A more detailed evaluation of several variants of reactor dismantling [21] demonstrated the possibility to shorten the duration of safe enclosure in accordance with the recent world-wide trend for NPP decommissioning [22].

Factors against extended periods of safe enclosure include [20, 22]:

- unfamiliarity of new staff with the plant when safe enclosure period ends;
- costs connected with long term maintenance of NPP data
- general deterioration of the facility and its key equipment;
- public acceptance and the space availability;
- difficulties with cost predictability over a long period of time;
- more stringent legislation (ALARA, clearance levels);
- increase of waste management costs much more rapidly than inflation.

Currently preliminary decommissioning studies for WWERs are being updated including the optimization of the safe enclosure duration, scope and variants in accordance with the full set of parameters. An overview of decommissioning strategies in countries having WWERs, including factors relevant to the selection of the strategy are given in Table VI.

Table VI. Decommissioning strategy and current status (1998) in countries with WWERs

Country	Currently preferred strategy	Plan	Fund	Interim spent fuel storage	Waste disposal facility	Clearance criteria
Armenia	No strategy	-	-	+/*	-	-
Bulgaria	Safe enclosure for 70 years	+	+	+/**	/**	-
Czech Republic	Safe enclosure for 50 years	+	+	+/	+	+
Finland	Early dismantling	+	+	+	+	+
Germany	Early dismantling	+	+	+	+	+
Hungary	Safe enclosure for 70 years	+	+	+/*	/**	+
Russian Federation	Safe enclosure 30–100 years	partial	+	+	-	+
Slovakia	Safe enclosure for 70 years	+	+	+/*	/*	+
Ukraine	Safe enclosure under development	-	-	+/*	/**	+

+ available
- not available
/ additional capacity is needed
*under construction or licensing
** concept or site selection.

The re-evaluation could result in different type of safe enclosure such as applied in the USA [23]:

- minimum of cleanup and decontamination initially, followed by a period of continuing care with the active protection systems (e.g. ventilation) kept in service throughout the storage period (“custodial SAFSTOR”). On-site surveillance by staff, radiation monitoring, maintenance, intrusion prevention.
- more comprehensive cleanup and decontamination initially, sufficient to permit deactivation of the active protective system during the continuing care period (“passive SAFSTOR”). The security, electronic surveillance to detect intrusion, periodic monitoring and maintenance of the structures integrity.
- comprehensive cleanup and decontamination, construction of barriers around the areas containing significant part of radioactivity which make intrusion impossible or extremely difficult (“hardened SAFSTOR”). Detection of attack upon the barriers, maintenance the structure integrity, infrequent monitoring.

6.2. LEGISLATIVE AND REGULATORY REQUIREMENTS

In the 1960s and 1970s when the WWER type reactors were commissioned, NPPs were state-owned in most countries in central and eastern Europe, and the policy at the time was that the state should keep responsibility throughout the plant’s lifetime from construction through decommissioning including the provision of the necessary funding.

Waste management regulations were originally prepared in countries owning WWERs in accordance with the foreseen activities (radioactive waste sorting, treatment and storage). No specific regulations were issued for decommissioning in the initial period of commissioning and operation, nor were conditions for termination of the operating licence defined.

After the economic and political changes at the end of 1980s the countries owning WWERs faced problems relating to legal background, financing and technical preparation for decommissioning. The eventual strategies selected will depend on circumstances and conditions which vary from country to country.

Over the last five years, new atomic energy acts and regulations were issued nearly in all countries where WWERs are operated with the objective of defining operator's responsibilities and regulatory requirements. These legislative acts concerning nuclear and radiation safety were intended to cover all nuclear activities including NPP decommissioning. The operator responsible for the nuclear facility has the primary responsibility for all technical and financial measures necessary for decommissioning and safe handling. There may be a different approach for the responsibility for the disposal of radioactive waste under the supervision of national regulatory bodies.

In the previous legal framework of countries operating WWERs, there were no requirements concerning initial and ongoing decommissioning planning for nuclear facilities. This documentation was therefore missing. The new atomic energy acts codify these requirements and consequently regulations concerning scope and contents of the decommissioning documentation based on IAEA recommendations [7] were issued. For example, initial and ongoing planning documentation for decommissioning is being completed for WWER units under operation (Bulgaria, Czech Republic, Finland, Hungary, Slovakia) as well as final detailed decommissioning plans for WWERs close to permanent shutdown (Russian Federation). The only country where the decommissioning of WWERs has already started on the basis of the final plan approved by the regulatory body is Germany, where the premature final shutdown was not based on the technical performance of the units, but was largely determined by the high costs associated with refurbishing the units to the required standards and by technical and political uncertainties.

According to legislation being enacted, the regulatory body should also approve the final survey report validating completion of the decommissioning activities. Based on the evaluation of conditions for clearance or authorized use, the regulatory body decides on issuing the license termination, or in the case of authorized use, permission for the use of the site for other nuclear purposes.

CEE countries which are expected to join the European Union (EU) in the near term will experience a changing economic and legal environment. Because of the need for harmonization with EU legislation and policies, changes in conditions relevant to decommissioning may arise which could lead to strategies different from those that have been worked out during the last 5–10 years period. For example at the moment factors like labour costs and site reuse options may have lower importance than is likely after EU membership is achieved.

6.3. DECOMMISSIONING COST AND FUNDING

Decommissioning costs are influenced by many factors and vary with decommissioning strategies and time. Dismantling costs may tend to decrease with time because of radioactive decay. Surveillance, maintenance, labour and waste disposal costs generally increase with time. If a long period of safe enclosure is envisaged, the forecasting of funding requirements may be uncertain. For example, currently waste disposal costs are escalating in some countries to such an extent that they are becoming the major decommissioning cost factor.

After recent economic changes in the countries operating WWERs, the first step to tackle decommissioning was to perform preliminary cost evaluations for different decommissioning options and on that basis to determine the preferred strategy selection. In Bulgaria, Czech Republic, Hungary and Slovakia the preliminary decommissioning study was prepared and partially reviewed by the same independent company.

As one example of cost estimates, for the Greifswald decommissioning project (early dismantling), 28% of costs are represented by post operational costs, 9% by care and maintenance costs, 43% by dismantling and waste management costs and 20% by overheads as defined in Section 4.3.2 (excluding spent fuel management costs).

In comparison, for the decommissioning project of NPP Jaslovske Bohunice 3 and 4 (safe enclosure of 70 years and deferred dismantling), 6.7% of cost is represented by post operational cost, 17% by care and maintenance, 72% by dismantling, demolition and waste management and 6.1% by overheads (some of the post operational costs as well as overheads costs are included into waste management costs). The cash flow over the years will roughly be some 3% during pre and post shutdown period (6 years), 72% during the next 10–11 years for decontamination, dismantling and demolition of the parts outside the safe enclosure structures, 17% for the next 70 years (safe enclosure) and 8% for dismantling and demolition of the safe enclosure structures during the next 5 years [16].

Because of recent economic changes in most countries where WWERs are operated, the cost estimates vary from initial estimates of several tens of millions US dollars to hundreds of million US dollars which is comparable to western PWR decommissioning costs depending on the range of involved works. Total costs for decommissioning of two WWER-440 units vary [24]. Some projected costs are given in Tables VII and VIII.

In order to provide the necessary confidence that resources will be available to maintain radiological and environmental protection during decommissioning, it is vital that provisions for allocating resources be established early in the nuclear power plant's design planning. It is required that the mechanism be sufficiently robust to provide for decommissioning needs, even in the event of premature shutdown of the reactor facility. It is also required that responsibilities, timing and costs be clear, as well as how the fund is handled and for what purposes and when it can be used. International recommendations can be found in the literature e.g. [7, 25, 26].

Table VII. Decommissioning costs for WWER reactors (immediate dismantling) [24]

Country	Reactor	Cost in mUS$ at 01/01/97
Bulgaria	WWER-440/230 (Kozloduy 1, 2)	193 (2 units)
Finland	WWER-440/213 (Loviisa)	113 (1 unit)
Germany	WWER-440/230 (Greifswald)	300 (1 unit)
Russian Fed.	WWER-440/230	52–69 (1 unit)
Slovakia	WWER-440/230 (Bohunice 1, 2)	353 (2 units)
	WWER-440/213 (Bohunice 3, 4)	305 (2 units)

Table VIII. Decommissioning costs for WWER reactors (long term safe enclosure) [24]

Country	Reactor	Cost in MUS$ at 01/01/97
Bulgaria	WWER-440/230 (Kozloduy 1, 2)	257 (2 units)
Czech Republic	WWER-440/213 (Dukovany))	461 (4 units)
Germany	WWER-440/230 (Greifswald)	330 (1 unit)
Russian Fed.	WWER-440/230	212 (1 unit)
Slovakia	WWER-440/230 (Bohunice 1, 2)	366 (2 units)
	WWER-440/213 (Bohunice 3, 4)	296 (2 units)

No funds for meeting future NPP decommissioning costs were set up in CEE countries, where WWERs are operated, until economic and political changes occurred at the end of 1980s. As previously stated, it was assumed that the state/ government was responsible for funding decommissioning activities at a later stage.

Only in Finland has the liability and the fund contribution been defined for each nuclear waste producer on a yearly basis. Non-recurrent costs (decommissioning costs and investments for disposal facilities) are spread over the first 25 years of operation. The fund invests the assets through loans and the interest collected is credited to the account of the waste producer.

After the above-mentioned recent changes, new acts on peaceful atomic energy uses were issued or are in the process of being issued in CEE countries. These acts require the licensees to create their own financial reserves to cover decommissioning costs and the government to take steps aimed at the overall financing of the fuel cycle back end. However, in CEE countries a practical solution to legislative requirements remains difficult. For example, it is recognized that responsibility for the decommissioning of some older facilities will remain with the state.

In Germany, the State/government took responsibility for funding all decommissioning costs for the NPP Greifswald, because the decision for prompt final shutdown and early dismantling was taken without previous funding mechanisms being in place.

In the Russian Federation, decrees issued in 1992 allocated reserves to cover NPP decommissioning and related R&D expenses. For individual units, the operators were made responsible for the collection and handling of such funds. The collection mechanism is based on 1.3% of the electricity revenues.

In Slovakia the act creating a state fund for decommissioning and disposal investment was issued in 1994, including conditions for NPP contribution (10% of the electric energy price) and use of the fund.

In the Czech Republic the licensee is required to create his own financial reserve (act issued in 1997) for decommissioning at the stipulated time and based on foreseen activities. Moreover, each waste producer contributes to the nuclear fund, which covers spent fuel and radioactive waste disposal costs.

Funds for decommissioning and for radioactive waste storage and disposal were created in Bulgaria. The payment into these funds started on 1 January 1999.

Before the new Act on Atomic Energy entered into force in 1997 a reserve fund for future costs of nuclear waste management and NPP decommissioning was not established in

Hungary. In 1998 a separate state fund exclusively earmarked for financing the costs of nuclear back-end, radioactive waste disposal and decommissioning of nuclear facilities was set up. The approach to long term provisions for financial liabilities currently includes a levy of 0.5 Ft on each kWh of electricity produced by NPP collected in the fund. The payments into the fund started on 1 January 1998.

In Ukraine, the laws which were issued in 1995 created a decommissioning fund as well as a waste management fund. The calculated cost of decommissioning is to be taken into account when pricing electricity and this will cover the payments into the fund. The waste management fund is to be raised by contributions from waste producers based on the volume of the radioactive waste.

Although decommissioning funds are or will soon be available in CEE countries, the high expected costs for waste disposal and spent fuel storage/disposal facilities can affect the timely availability of funds for NPP decommissioning and consequently the decommissioning strategy.

6.4. SPENT FUEL MANAGEMENT STRATEGIES

Based on bilateral agreements, the original approach for all of the WWER owners was to transport the spent fuel to the former USSR following several years (typically 3 to 5 years) of spent fuel cooling in ponds located close to the reactors. It was not foreseen that any of the reprocessing wastes be shipped back to the countries where the spent fuel originated. Following USSR dissolution at the end of 1980s, it became more and more difficult to transfer spent fuel to Russian Federation and ultimately almost impossible by the mid-1990s. Russian and partly Ukrainian spent fuel from WWER-440 reactors is still reprocessed by the enterprise MAYAK, where storage capacity for reprocessing waste and spent fuel before reprocessing is sufficient. The other WWER owners had to increase their own storage capacities through building additional spent fuel interim storage facilities, developing technologies for spent fuel conditioning and preparing for disposal.

How spent fuel storage and disposal (after reprocessing or not) is managed can heavily influence the decommissioning strategy. In the case of wet or dry storage, the spent fuel is stored for a long time usually at the NPP site and this site cannot be released until the spent fuel is disposed of, and the decommissioning of the storage facility is finished. On the contrary, in the case of dry storage in casks (e.g. CASTOR type), the spent fuel can be easily transported to another site/location and the constraints on the decommissioning strategy are of less importance.

There are several alternatives on how to increase the spent fuel interim storage capacity. Some WWER owners increased their ponds’ capacities up to 5–6 years’ production with more compacted rack geometry (e.g. Paks, Dukovany, Mochovce) and later built on-site interim storage facilities of different types. The present wet interim stores can be extended with additional pools or more geometrical and material changes. Also, dry stores of the cask or vault types are planned or built.

The Czech approach is to use dry storage in CASTOR casks at the Dukovany interim spent fuel storage facility, which was commissioned in November 1995 with an expected design life of 40–50 years. As the storage capacity is limited, construction of additional stores is being prepared.

The Modular Vault Dry Store at Paks NPP was commissioned in 1997 with the design life of 50 years. Provision is made for further extension of the facility to accommodate a total Paks NPP spent fuel production of 30 years‘ reactor operation**.**

A combination of wet interim store and dry storage in CASTOR casks is used in Greifswald NPP, however the complete transfer of spent fuel from wet store to CASTOR casks is planned.

In Slovakia, Soviet design wet interim spent fuel storage with a total capacity of 10 years' production from the former Czechoslovakia was commissioned in 1987. A similar wet storage facility was commissioned in Bulgaria in 1990. Beginning in 1995, spent fuel produced by Czech NPP and stored in Slovak interim storage was transferred from Slovakia back to the Czech Republic. In 1997, work started to increase the Slovak wet interim storage facility capacity by geometry and material changes. The objective is to achieve sufficient capacity to store the lifetime production of Jaslovske Bohunice NPP. Seismic upgrading will enable the facility to store the spent fuel for 40–50 years.

Responsibility for spent fuel storage rests with either a state organization (e.g. Paks) or with the operator of NPP (e.g. Greifswald, Dukovany, Jaslovske Bohunice, Mochovce) while the spent fuel disposal is to be the responsibility of the state.

The facts presented above, and the world-wide scarcity of near term reprocessing and disposal options, give evidence that the problem of long term spent fuel presence on-site after final shutdown and resulting constraints will probably not be solved in the near term. These factors, including related spent fuel management costs, can strongly influence the decommissioning strategy and the eventual release/reuse of the site.

6.5. RADIOACTIVE WASTE MANAGEMENT INFRASTRUCTURES

The original Soviet concept for the management of waste from the WWER NPPs assumed the collection, pre-treatment (by evaporation of liquid and by compaction of solid waste) and interim storage of all radioactive waste produced at the site during the entire operational period. This concept postponed final decisions on conditioning and disposal of operational waste to the decommissioning stage. Radioactive waste from both operational and decommissioning period could then be handled together. Therefore, radioactive waste treatment, conditioning technologies and disposal facilities were missing in the countries having Soviet designed reactors. Also, capacities for radioactive waste storage were often not sufficient [27].

The approach to waste management in countries where WWERs are operated changed in 1980s, including development of conditioning technologies (focused especially on operational waste) and construction of disposal facilities. Evaporation and drying processes are used for conditioning of concentrates, resins and sludges in Germany where the disposal acceptance criteria allowed the disposal of unsolidified waste in high integrity containers. Bituminization is used in the Czech Republic and Slovakia for solidification of concentrates. Sludges and resins are not yet conditioned in either of these countries. Concentrates cementation is also used in Russian Federation and the Ukraine.

Low-pressure compaction is common practice at all WWERs. Super-compaction of solid waste is available in Germany and Bulgaria, and it should be available by 1999 in Slovakia.

Compactible radioactive wastes were treated in the Czech Republic using a mobile super-compactor of British origin as a service. Incineration is used in Slovakia for low level radioactive waste. A new incinerator should be commissioned in Slovakia by 1999.

A facility for the treatment of dismantled material through cutting, decontamination, measurements for clearance or authorized recycling/release, volume reduction and compaction for disposal is available in Greifswald. A facility with a similar purpose is planned in Jaslovske Bohunice which will initially be used for metallic decommissioning waste from NPP A1 (a prototype HWGCR being decommissioned) and refurbishment materials from WWERs, and then later for WWER decommissioning waste. The construction of an on-site melting plant for metal waste is being discussed in Jaslovske Bohunice. In Hungary efforts have been made for intensive volume reduction of radioactive waste by using new technologies (e.g. radionuclide removal, incineration).

Currently, disposal facilities planned for low and intermediate level radioactive waste from NPP decommissioning, are available in the Czech Republic and Finland. In Germany and Slovakia such facilities are under licensing. In Hungary site selection and conceptual design for repository is under progress. A conceptual design for a disposal facility exists in Bulgaria. The disposal capacity is not sufficient for all decommissioning waste from WWERs in some of the above-mentioned countries, but it is planned to increase the capacity of existing repositories. Preparation for the disposal of long lived and high level radioactive waste are being made by studying the potential host geological formations (e.g. Czech Republic, Finland, Germany, Hungary, Slovakia).

Transportation of radioactive waste, particularly off site, should conform to national regulations. International recommendations on transportation are available [28].

The decommissioning plans for WWERs should address whether existing waste management systems are capable of coping with the anticipated decommissioning waste, including waste resulting from unplanned events or incidents during decontamination, dismantling and demolition. If they cannot handle these waste volumes or categories, new facilities may have to be provided.

The issue of the disposal of radioactive waste that is not acceptable for near surface repositories is not fully resolved. This type of waste may include highly activated/contaminated components and alpha bearing waste including plutonium and americium sources used in WWERs as boron concentration detectors. The extended presence of such waste on-site can strongly influence the decommissioning strategy and release/reuse of the site. One of the solutions for waste unacceptable for near surface disposal is to build an interim/buffer storage facility and to dispose of the waste later when a disposal facility is available or the waste is acceptable for near surface disposal. If waste acceptance criteria for disposal are not known at the time of waste packaging for storage, then later it will require additional costs to meet these criteria by e.g. repackaging. Another solution is safe enclosure of the reactor systems containing these materials.

Insufficient storage and disposal capacities together with cost estimates have been the strongest arguments in favour of a decommissioning strategy based on deferred dismantling after 50 to 70 years of safe enclosure of the whole reactor building or parts thereof.

6.6. CRITERIA FOR REMOVAL OF MATERIALS FROM REGULATORY CONTROL

The timing of dismantling will influence the extent of waste arising in various activity categories. In general, a large portion of the decommissioning waste may be eligible for clearance and authorized release that should comply with national criteria based on individual dose or derived clearance levels [12]. Clearance requires that all reasonably possible exposure routes are evaluated in the derivation of the clearance levels. Alternatively, removal from regulatory control may be constrained, usually because the fate of the material being considered for release is known, so that only a limited number of reasonably possible exposure scenarios have to be evaluated. Authorized release may then be granted with certain conditions (for example re-melting, disposal together with municipal waste, etc.). If the clearance criteria and criteria for authorized release are excessively restrictive, large quantities of material will require disposal as radioactive waste. In addition, many derived clearance levels are close to, or below, current limits of detection for field instrumentation.

The availability of national policies and long term strategies in support of recycle and reuse principles may have a profound impact on the efficiency and extent of recycle and reuse practices and can influence decommissioning strategy and schedule [9]. Also economic penalties e.g. radioactive waste treatment and disposal costs, may encourage clearance, recycle and reuse practices. A fund created for radioactive waste treatment and disposal by waste producers with no regard to the volume of generated and disposed waste brings little or no incentive for recycle and reuse.

Clearance criteria based on individual annual doses of 10 μSv are commonly used in the CEE (50 μSv in Ukraine). Recommendations for derived clearance nuclide specific levels were issued by IAEA [12] and European Commission [29].

In some countries, where WWERs are located, derived clearance levels were codified in national legislation for mass specific activities and for surface specific activities (see Table IX). In some cases limits are ten times higher for smaller contaminated areas i.e. "hot spots" or individual pieces. Additionally, some countries specify separate total limits for alpha and beta-gamma emitters, while others maintain nuclide specific limits. In addition, in some countries further restrictions have been applied in terms of total activity, total mass or total volume.

Table IX. Derived clearance levels

Country	Surface specific activity (Bq/cm^2)		Mass specific activity (Bq/g)	Averaged area (cm^2)	Averaged weight (kg)
	Beta,gamma	Alpha			
Germany	5;0.5	0.05	0.1	100; 10 000****	100****
Czech Rep.	0.3–3000*		0.3–3000*	100	1
Finland	0.4	0.04	1(0.1 alpha)	1000	1000
Russian Fed			0.3–10*		
Ukraine	5	0.05	1 (0.1 alpha)		1000
Slovakia***	0.3–(3**)	0.03–(0.3**)	0.1–(1**)	300 (10**)	1000 (1 piece**)

* nuclide specific levels with higher values for less harmful nuclides
** hot spots
*** only for scrap material
**** applicable for measurement chamber.

Special rules are in force in Germany and Slovakia for authorized release of scrap metal with the mass specific activity up to 1 Bq/g for recycling through re-melting.

A general tendency in some countries where WWER are located is to recycle as much material as possible (e.g. scrap metal from pond reconstruction and replaced components- more than 300 t of steel, aluminium and brass were released from the NPP Dukovany).

6.7. SOCIAL AND PUBLIC ACCEPTANCE ASPECTS

Socio-political and other non-technical issues of NPP decommissioning and waste management are covered in an environmental impact assessment in most countries where WWERs are located. These documents required by relevant acts assess the direct and indirect impacts resulting from the changes connected with NPP decommissioning to urban structure, health, living conditions and well being including reactor staff. The environmental impact assessment process includes hearings of citizens in local and neighbouring municipalities, local initiatives and actions by public institutes. Local authorities, individual citizens, and public institutions may express their comments and opinions in public hearings and as written statements. A positive statement from the safety authorities has been a binding prerequisite for the acceptance of decision by the responsible authority.

WWER NPPs used to have many more personnel than western PWRs because of the maintenance and support (refuelling, inspection) activities performed by their own staff. The approach to retain as many operational personnel as possible for decommissioning purposes is often the case of WWERs. The social impact of decommissioning depends on the specific NPP and local conditions and infrastructures. The impact is higher when the decommissioned NPP is isolated from other NPPs or is sited in an area with a low density of population and industry. This is typical of the Russian Federation. In fact, it was the policy of the former USSR to establish strategically placed economic and industrial developments bound to become population centres at a later stage.

Closure of several units in a short time can significantly affect the local conditions and infrastructures (Greifswald) and the need for additional investments to compensate for missing financial contributions to the area, retraining programmes for plant personnel and other population groups, and possibly funds for reconstruction of the local power grid.

The additional costs for the first 10 years' period of decommissioning including preparatory stages and achievement of Stage 1 in the Russian Federation could be up to a hundred million US$ [30]. The estimated loss of the gross national income due to cessation of electricity generation for one WWER unit, including losses of the supporting industry, is in the order of a billion US$ [30]. The socio-economic costs of retiring Russian reactors are staggering, many times more than the cost of labour and equipment for the physical decommissioning itself. In practice the Russian Federation cannot afford to decommission its NPPs without immediately building new units to replace the decommissioned units and using the site infrastructures.

The existence of several NPPs in the same area mitigates the social and economic consequences for population and personnel. In such a case the major societal impacts are connected with the end of operation and occur prior to decommissioning. The initial reduction in workforce will take place over a period of approximately five years. For example, the final shutdown of the Slovak NPP Jaslovske Bohunice (units 1and 2) will result in 65 redundant working positions during the first 5 year period, under the assumption that all new required positions/jobs will be taken by operational personnel and taking into account the personnel reduction due to retirement for all four NPP units. To maintain good social conditions, a detailed analysis of personnel need (new needed jobs including vacant positions versus cancelled jobs) is planned, resulting in timely staff re-qualification and motivation of specialists to prevent their premature exit from key work positions. The additional costs resulting from these requirements are supposed to exceed a million US$ [15, 31].

6.8. CONDITION OF THE PLANT

Some design, construction and operational features may either facilitate or complicate decommissioning. The features facilitating decommissioning are those reducing the radiation source, dismantling time and/or the volume of radioactive waste. All of them will, generally, result in the reduction of decommissioning costs. More details specific to WWERs are given in Section 7.2.1.

6.9. OWNERS' INTERESTS INCLUDING PLANNED USE OF THE SITE

It is typical for WWERs to be built in two-unit modules (two units in one building). So decommissioning of two units in parallel are expected to be planned. In some countries, several such modules are located on one site (e.g. Greifswald, Dukovany) or some auxiliary systems are common to several units which may be shutdown at different times (e.g. Jaslovske Bohunice). These facts influence the selection of decommissioning strategy. The resources of remaining operating facilities at the site could provide the necessary security, monitoring, surveillance and maintenance for the shutdown facility. Also the use of some parts or equipment from shutdown units for use in operating unit is possible [32]. In the case of several co-located NPPs where shutdown is foreseen at different times, a sequential deferred dismantling could be planned.

The strategy to reuse the NPP sites for new NPPs in the Russian Federation was discussed in Section 6.7. There are no other WWER specific considerations that need to be highlighted, generic factors described in Section 4.3.8 are also valid for WWERs.

6.10. AVAILABILITY OF RESOURCES

Practical experience in actual WWER decommissioning project exists only at NPP Greifswald, which is only a partial sample of prevailing conditions in CEE.

In terms of plant expertise, it may be critical in many cases to retain key personnel to partly compensate for lack of as-built drawings and historical records. On the other hand, in a sequential decommissioning strategy (most WWERs are located at multi-unit sites) personnel from shutdown units might be re-assigned to operating units on-site. Depending on conditions prevailing in CEE when the dismantling of any unit takes place, it is possible that decontamination and decommissioning expertise might be available and be recruited from other projects in that country or in the region.

Decontamination and decommissioning technologies are generally available in countries where WWER are located as it is described in the Section 8 but should be optimized to the needs of individual countries/facilities. As decommissioning activities are not expected to start in most of these countries for at least ten years, further development of the decommissioning technologies can be expected. Applications of and R&D on these technologies in various countries e.g. in the Russian Federation are extensively described in Ref. [8].

6.11. RADIOLOGICAL ASPECTS

Immediately after final shutdown, radioactive inventory and dose rates in WWER reactor structures, primary circuit and other systems activated and contaminated during reactor's operation will be most influenced by relatively short-lived corrosion products as it is shown in Section 7.3.

Safe enclosure for periods of several decades is therefore expected to produce some benefit from the dose rate reduction point of the view. Subsequently, however, the total activity and dose rates will be mainly determined by long-lived radionuclides ^{59}Ni and ^{63}Ni, so that additional dose rate reduction from further continuation of safe enclosure would be marginal.

A theoretical study of Loviisa 1, 2 NPP decommissioning points out that a 30-year safe enclosure period results in a reduction of a factor 10 when comparing occupational exposures from deferred to immediate dismantling [20].

7. CHARACTERIZATION OF WWERs

The activation or contamination of construction materials and equipment makes it necessary to protect workers during decontamination, dismantling, and transport, storage and disposal of radioactive components resulting from decommissioning. Major factors to be taken into account in deciding the method and extent of decommissioning are physical and radiological conditions of the NPP.

7.1. GENERAL

Characterization activities for the purposes of decommissioning of nuclear power plants include:

- physical description of the NPP site
- verification of existing design drawings
- estimation of remaining life time of buildings and equipment as related to various decommissioning options
- determination of material inventory
- determination of the plant radiological situation after shutdown
- determination of waste streams (radioactive and non-radioactive) and availability of appropriate processing technology
- implementation of additional radiological measuring systems and methods
- criteria for authorized use or clearance of materials
- development of methods for determination of difficult to measure nuclides.

During the decommissioning planning stage, the amount of material and radioactive waste (including its categories) arising from the dismantling activities have to be estimated on the basis of material inventory and radiological measurements.

Protection measures for the dismantling work can be planned in detail on the basis of the radiological characterization of the plant. In order to decide on the methods and extent of decommissioning required, it is important to determine the location and estimate the quantities of radionuclides present and the nature of their physical and chemical form.

7.2. PHYSICAL CHARACTERIZATION

Detailed and complete physical characterization include:

- description of the current status of the plant (number of units, type of reactor, connections to existing infrastructure, operational history)
- description of the main technological systems and equipment
- material inventory (masses, material types, shape, location)
- description of structures, including protective barriers and an estimation of their residual life).

Basic technical parameters of WWER-440 reactors are given in Appendices 1 and 2. A plan of a multi-unit site is given in Fig. 4.

7.2.1. WWER construction and operational features relevant to decommissioning

The following paragraphs indicate features of WWER-440 reactors, arising from design or operating history, that can influence the approach to decommissioning [33] (see also Table X).

Table X. WWER features related to decommissioning based on Greifswald experience

Facilitating	Complicating
Design & construction	
• on-site independent fuel storage normally available	• liquid and solid waste storages not designed for retrievability
• enough space in controlled area for easy dismantling	• twin-unit and multi-unit sites with several systems in common, i.e. dismantling of one unit difficult
• primary loop components exchangeable	• tube storage module for activated components in reactor hall
• low-Co and low-Ni steels used	• building construction very simple (concrete slabs, etc.)
Operation	
• low contamination due to water chemistry	• SG leakage has often led to contamination of secondary loop
• low level of fuel leakage	• often fuel pool leakages
	• borate precipitation in liquid waste storage tanks

(a) All WWER-440 NPPs have been constructed as so called twin units. The main feature of this type of construction is the structural arrangement of two reactors in one building with a common reactor hall.

(b) Most WWER-440 sites are multi-unit sites, i.e. at NPP site there are several twin units. Some of the technological systems serve more than one twin unit building.

(c) A majority of the WWER-440 sites have sufficient space for further development (nuclear or non nuclear). Thus there is little pressure to complete decommissioning rapidly to enable such development.

(d) Due to the nature of building constructions, i.e. prefabricated slabs and no containment, the boundary of the potential safe enclosure structure is not evident. A detailed study is necessary in order to define the optimum boundary.

(e) Any leakages into secondary circuits (for example steam generator tube leaks) will require more thorough radiological investigations and radiological classification of the equipment and systems in the turbine hall.

(f) The ventilation systems of WWER-440 plants have to be modified for decommissioning purposes. Due to the lack of pressure regulated control systems and filtering in certain important areas (e.g. reactor hall), special care has to be taken to avoid airborne contamination and release of radioactivity to the environment.

(g) The spent fuel is stored in the storage pools located in the vicinity of the reactor within the reactor hall. From the nuclear safety point of view this is a factor limiting the operation termination activities and early stages of decommissioning. For safety reasons a storage of at least 3 years is required before the fuel assemblies can be removed from the reactor hall. Any damaged or leaking fuel assemblies will complicate this process.

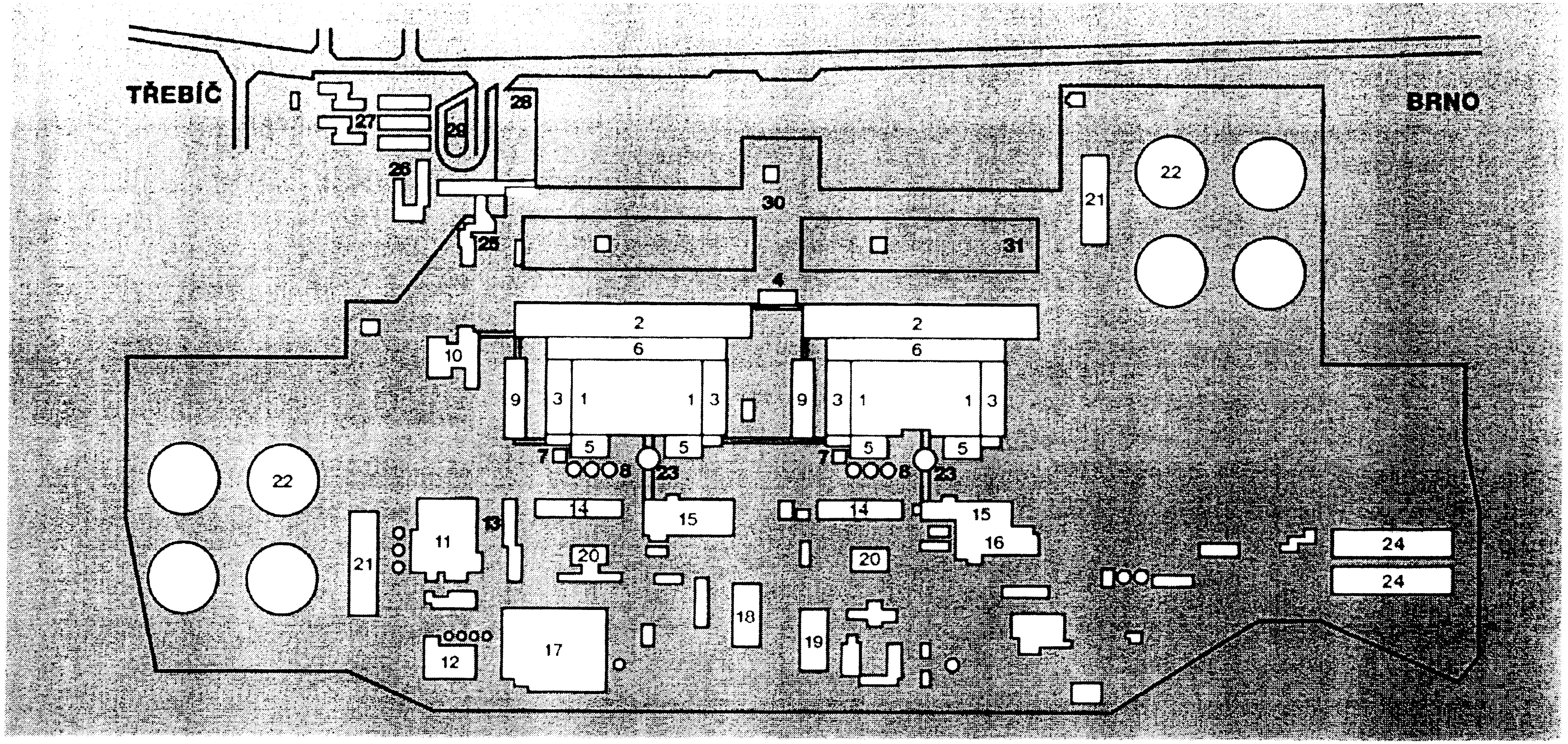
(h) Radioactive waste processing technology was not included in the original design of the plant. Therefore, operating NPPs have accumulated large volumes of liquid (concentrates, sludges, spent ion exchange resins, contaminated oil) and solid radioactive waste. Waste is stored at the NPP either unconditioned or partially processed (evaporated, compacted) to meet the available storage capacity. Problems can be expected during removal of solid waste from storage vaults where the waste may not have been placed in an orderly fashion. This material may be difficult to remove due to potential biological degradation, gas generation and waste volume changes.

(i) Storage tanks contain precipitated borates. Chemical regime, high concentration of salts in evaporator concentrate and boric acid content are the main factors resulting in precipitation of salts. Removal of borates from the storage tanks requires the application of specific techniques.

(j) Activated parts are stored in the reactor hall pits. Removal of these activated parts will require special tools and transport containers.

(k) Leakage of contaminated liquid (e.g. from the storage pools) has occurred in some cases so that contaminated concrete must be removed under radiological controls.

(l) Stability of barriers can be another issue. According to current WWER440 decommissioning studies (Paks, Jaslovske Bohunice, Kozloduy) it is intended to use existing hermetic compartments for long term protection of activated and contaminated components. Stability of these barriers or components will have to be demonstrated for this purpose.



1 Reactor building
2 Turbine hall
3 Transversal building
4 NPP electrical despatching
5 Bubbler tower
6 Longitudinal building
7 Emergency feedwater pumps building
8 Emergency feedwater tanks
9 Service building
10 NPP management building
11 Chemical plant
12 Chemical plant
13 Fire brigade
14 Diesel-generator station
15 Auxiliary building
16 Low level waste treatment
17 Machine maintenance workshop
18 Structural maintenance building
19 Refrigerator building
20 Diesel fuel tanks
21 Central pumping station
22 Cooling tower
23 Ventilation stack
24 Low level waste depository
25 Entrance hall
26 Information centre building
27 Administrative building
28 Car park
29 Bus station
30 Waste cleaning station
31 Switch yard

Fig. 4. General layout of WWER-440 plant (4 units).

7.2.2. Material inventory

Aspects of WWER-440 design and construction (no containment, 6 circulation loops, 2 turbines per reactor) mean that the material inventory of WWER differs from western PWRs. Additional differences result from local conditions (e.g. absence of cooling towers, size of site). Almost all activated and contaminated materials are situated in the controlled zone of the reactor building and the auxiliary building. Only a small part of the sanitary building belongs to the controlled zone. The turbine hall is classified as a monitored area. Auxiliary buildings and sanitary buildings are connected to the reactor building by corridors (ee Fig. 4). Characteristic data of structures (NPP Paks, V-213 type) [18] are given in Appendix 2.

The total volume of materials arising from the dismantling of a double unit WWER-440 V-213 type technological equipment (detail parameters of the main components are shown in Appendix 3), auxiliary and sanitary buildings as estimated for the NPP Paks are summarized in Table XI [18].

The total quantity of material to be dismantled at the NPP Greifswald units 1–8 have been estimated as 1.8 million tonnes. From the total decommissioning mass, the non-contaminated plant parts (mainly turbine hall as well as the non-contaminated building structures and remaining materials) represent 1.2 million tonnes. The remaining material quantity of approximately 0.5 million tonnes and their classification are described in [34].

Table XI. Material balance calculated for NPP Paks (Unit I, II) [18]

	Stainless steel (t)	Carbon steel (t)	Insulation (m^3)	Basement concrete (m^3)	Structural concrete (t)	Steel structures (t)	Others (t)
Main building (see Appendix 2)							
Activated	700	300	-	-	200	-	-
Contaminated	7 450	4 690	-	**ND**	**ND**	-	-
Non-contaminated	900	2 370	550	7 800	130 000	29 400	3 080
Material outside of controlled zone	80	15 290	520	4 200	48 800	6 800	2 340
Total	**9 130**	**22 920**	**1070**	**12 000**	**1 78 800**	**36 200**	**5 420**
Auxiliary building							
Contaminated	760	271	-	**ND**	**ND**	-	-
Non-contaminated	90	131	25	1200	11 700	580	306
Total	**850**	**402**	**25**	**1 200**	**11 700**	**580**	**306**
Sanitary building							
Contaminated	50	16	-	**ND**	**ND**	-	10
Non-contaminated	2	66	1100	-	13 350	1 100	25
Total	**52**	**82**	**1100**	-	**13 350**	**1 100**	**35**

ND: no data available
-: no material.

7.3. RADIOLOGICAL CHARACTERIZATION OF NON-ACTIVATED PARTS

Knowledge of the radioactive inventory is a key prerequisite for the decommissioning process, being needed for the initial planning, control of radioactive waste flow during dismantling and clearance of the site at the completion of the decommissioning programme. There is considerable similarity of estimation methods in use at various facilities, although the codes used at each step in the calculation tend to be chosen on the basis of local history and preference.

Contamination modelling is considerably less developed than that of activation modelling. The route to improved modelling will depend on whether the most cost effective way of achieving the required level of knowledge is perceived as direct measurement or development of a model supported by direct measurements.

Characterization of WWER contamination comprises the following two main areas:

- estimation of NPP contamination levels (contamination map)
- measuring dose rates (dose rate inventory).

7.3.1. Contamination levels

The main purposes for determination of contamination levels are:

- estimation of total activity content for licensing purposes
- identification of personnel risks during decommissioning activities
- quantification of possible releases to the environment
- detailed classification of materials for final processing (disposal, decontamination, melting, clearance).

A majority of the material in contact with primary circuit water (containing boric acid 0–12 g/L, pH maintained mainly by KOH) is austenitic stainless steel stabilized with titanium, and the fuel cladding made of zirconium alloy with 1% niobium. The surfaces of this material react with the primary coolant and become contaminated with activated corrosion products or fission products. The contact surface for a typical WWER-440 is about 16 500 m^2 (not including the surface of fuel elements) [35]. The volume of the primary circuit together with the volume of pressurizer (volume control system) is about 240 m^3.

Data on surface contamination available at operational NPPs cover only a limited part of the plant systems. Operational records and expert estimation have been used in preparing decommissioning studies for the WWER-440 Plants at Paks, Dukovany, and Jaslovske Bohunice. Data resulting from a conservative approach used for classification of contaminated technological systems at NPP Paks [18] are shown in Appendix 4.

Comparison of data collected at operational WWER-440 NPPs [36, 37] indicates that the contamination levels of the main parts of the primary circuit (piping, main circulation pump housing, main isolation valve, steam generator collectors) vary in the range of 10^4–10^5 Bq/cm^2. Similar results were obtained during radiological investigations of the Armenian NPP Unit I [38]**.** Some results of this investigation are shown in Table XII. The level of total surface activity of different primary loop pipes changed only slightly, with a major portion of the surface activity resulting from ^{54}Mn, ^{60}Co, ^{110m}Ag and ^{137}Cs. The contribution of ^{60}Co to total surface activity varied from 20 to 80%. Significant differences can be found in ^{60}Co and ^{58}Co distribution at some operational NPPs (Table XIII [36, 37, 39]). This is due to the variation of Co content in construction materials produced by different manufacturers.

Table XII. Averaged surface activity in inner surface of primary loop pipelines at different sections (Armenian NPP Unit 1 after 12 years of operation -1992) [38]

Loop section	**(kBq/cm^2)**
Reactor–main isolation valve	6.1×10^1
Main isolation valve–steam generator	4.1×10^1
Steam generator–main circulation pump	3.8×10^1
Main circulation pump–main isolation valve	5.3×10^1
Main isolation valve-reactor	2.8×10^1

Table XIII. Surface contamination levels at various WWER-440s [36–39]

Composition of surface contamination (kBq/cm^2)								
	^{51}Cr	^{54}Mn	^{58}Co	^{59}Fe	^{60}Co	^{110m}Ag	^{124}Sb	NPP
Main circulation pump - housing		0.52	1.1	0.2	5.6	0.3		Paks
Main circulation piping - cold leg	1.6	1.9	13.7	1.6	3.6		0.4	Dukovany
Main circulation piping - hot leg		4.32	27.8	1.1	6.1	0.4	1.2	Dukovany
Main isolation valve		18.2	17.8	3	52	1.9		Paks
Main isolation valve		2.1	16.9	2.4	18.4	0.7		Paks
Steam generator cold collector		6.4	10.4	0.8	4.9	5.7	7	Dukovany
Steam generator hot collector		11.4	39.8	1.8	12.9	0.4	4.8	Dukovany

Distribution of radionuclides in systems where a thermal gradient occurs (heat exchangers, steam generator) may differ from other parts of the same circuit. Levels of contamination in auxiliary systems such as drainage systems, valves or filters also need special attention because of the presence of hot spots. A high level of ^{110m}Ag is a specific feature of systems connected with the spent fuel storage pools [40].

An extensive radiological survey performed in the framework of the NPP Greifswald decommissioning project has provided a valuable set of data covering all areas of the NPP including the controlled zone, monitored area, secondary circuit systems, concrete structures and activated components. A complete radiochemical analysis including all relevant nuclides requires considerable radiochemical analytical effort that cost time and money. These data were used for the definition of the nuclide vector. A nuclide vector is a representative mathematical model for estimation of all relevant nuclides based on the measurement of a limited number of key nuclides. This approach can only be used if the radionuclide content of the various systems has been properly analyzed. This approach is very useful during dismantling and decontamination phases of the decommissioning project. Key nuclides are easily measurable nuclides (gamma emitters) such as ^{60}Co and ^{137}Cs (Ba). The nuclide vectors determined for the Greifswald Unit 5 (controlled area), Units 1–4 (controlled area) and turbine halls Units 1–4 are shown in Table XIV [41].

7.3.2. Dose rates

Dose rates measured on major components of the WWER reactor coolant system (Table XV [42–44]) are up to an order of magnitude lower than for most western PWRs. This is due to different composition of construction materials (lower nickel and cobalt content) in the WWERs (see Appendix 5). The dose rate from major components in the primary loop at the NPP Greifswald was comparatively low although some hot spots up to 10 mSv/h were found [34]. Detailed radiological characterization of the NPP Kozloduy Unit 3 (Table XV) was performed at specific points of primary system equipment before the full system decontamination in 1994 [42]. Lower reported occupational exposures in operating WWERs as compared to Western PWRs (Table XVI [45]) are generally due to lower dose rates and may entail somehow lower decommissioning doses, although this assumption is not proven.

Table XIV. Nuclide vectors (NPP Greifswald 1995) [41]

area	**^{60}Co (%)**	**^{137}Cs (%)**	**^{55}Fe (%)**	**^{63}Ni (%)**
turbine hall monitored area	18	2	60	20
Unit 5 controlled area	28	3	65	4
Unit 1–4 controlled area	17	2	71	10

Note: Unit 5: only 70 days of operation in 1989

Table XV. Equivalent gamma-dose rates of some primary system equipment

Measuring point	**Equivalent gamma-dose rate (mSv/h)**	**NPP**
Outer surfaces		
Steam generator outer surface	0.003–0.08	Greifswald
Main circulation pump outer surface	0.15–0.45	Greifswald
Main isolation valve outer surface	0.05–0.24	Greifswald
Pressurizer outer surface	0.02–0.05 (max.7)	Greifswald
Heat exchanger	0.5–10.2	Greifswald
Inner surfaces		
Steam generator — collector inner surface	12–45	Kozloduy
Steam generator — tube bundle outer surface	3–11	Kozloduy
Pressurizer inner surface	0.6–8.9	Kozloduy
Primary coolant blow down heat exchanger	10–20	Kozloduy
Primary circuit pipeline	up to 7	Kozloduy
PWR data [44]		
Steam generator - channel head	100 (typical value)	Westinghouse and KWU NPPs, 1992
Primary circuit pipeline	1–5	KWU NPPs, 1992

Table XVI. Occupational exposures of WWERs, elaboration from [45]

Country	Annual collective dose, average per reactor (man Sv)	
	1986–1989	**1990–1996**
Czech Republic	0.31	0.37
Finland	0.93	1.14
Hungary	0.56	0.60
Slovakia	0.62	0.58
World, PWR	2.51	1.70

7.4. RADIOLOGICAL CHARACTERIZATION OF ACTIVATED PARTS

Activated materials are located in and near the core. The reactor core is the most activated part of the reactor structure. The portion of the reactor exposed to relatively low neutron fluxes is essentially the biological shield, made of concrete with steel reinforcements.

Where the neutron flux, composition of material and history of operation are known, the radioactive inventory can be calculated with reasonable accuracy [46]. However in some cases, where the composition is less certain, for example the biological shield, it is necessary to support calculation by taking statistically significant samples, and by measuring the actual activity. The activation levels depend on:

- chemical composition of the material (see Appendix 5)
- neutron energy
- irradiation time length
- power plant history
- decay time after final shut down.

At present there is insufficient data to enable to exact calculation and verification of the induced radioactive inventory of particular WWER-440 reactors. Regulatory requirements for reactor operators to provide the radioactive inventory in the framework of decommissioning studies have resulted in several attempts to quantify main radiological parameters of the reactor pressure vessel (RPV), RPV internals and other activated materials.

The first comprehensive calculation for estimating radionuclide inventories of activated decommissioning waste from WWER-440 (NPP Loviisa) were carried out in 1988–1989 at VTT Energy (Nuclear Engineering Laboratory of the Technical Research Center of Finland) [47]. During the next few years, some further analyses, including measurement of material compositions were performed**.** Neutron flux distribution and spectra in and around the Loviisa reactor core were calculated with the REPVICS program system, which was originally developed to estimate fast neutron fluences in the reactor pressure vessel of the Loviisa reactors. Radionuclide inventories of decommissioning waste were estimated (see Table XVII) with the ORIGEN-S code, using flux distributions and spectra from the one-dimension

ANISN code calculations as input data [46].Calculations were made for the 30 year period with the original core and with 80% plant utilization.

The dummy elements, reactor internals, reactor vessel, thermal shielding and biological protection tank which will arise from the dismantling of the NPP Kola reactor Units I and II have been classified as high level radioactive wastes. Their total activity was estimated to be 47.7 TBq (results for the year 2010) with a mass of 717 tonnes [48]. Mass activities of the main reactor components calculated for different WWER-440 reactors are shown in Table XVIII [3, 47–49].

The activity of dummy elements is a major contributor largely due to their high activity when compared to the total inventory. On the other hand, higher activation of the reactor pressure vessel construction material and RPV cladding can be expected where the dummy elements were not used.

Only minor differences were found between WWER core basket mass specific activities which are all of the order of 10^6 GBq/t [3]. The dose rate of core basket outer surfaces were all approximately 10^2 Sv/h and the reactor pressure vessel outer surfaces were of the order of 0.1–1 Sv/h. These data will play an important role in designing storage containers, choosing reactor dismantling technology and estimating the collective dose (see Appendix 6).

Data calculated for WWER-440 units [49] indicate that in spite of rapid decrease of dose rate the radioactivity of WWER activated parts remain relatively high in the long term due to Ni63 (proposed safe enclosure periods for WWER reactors are 50–70 years). See also Table XIX.

Table XVII Mean specific activity of some radionuclides calculated for reactor components [47]

		[GBq/t]			
Isotope	Dummy elements	Core basket	Shaft	RPV cladding	RPV
^{3}H	6.15×10^{-2}	8.34×10^{-1}			
^{14}C	2.74×10^{2}	2.04×10^{2}	6.36×10^{1}	2.94×10^{1}	2.63×10^{-1}
^{32}P	4.55×10^{4}	3.46×10^{3}	1.68×10^{3}		
^{35}S	4.05×10^{2}	4.28×10^{1}	1.06×10^{1}		
^{45}Ca	3.09×10^{1}	1.53	5.63×10^{-1}		
^{46}Sc	3.69×10^{3}	1.82×10^{2}	7.15×10^{1}		
^{51}Cr	1.95×10^{7}	2.01×10^{6}	9.52×10^{5}	9.18×10^{4}	1.45×10^{3}
^{54}Mn	1.55×10^{6}	7.80×10^{4}	3.06×10^{4}	1.32×10^{3}	8.86×10^{2}
^{55}Fe	1.00×10^{7}	1.42×10^{6}	4.18×10^{5}	4.91×10^{4}	9.06×10^{3}
^{59}Fe	4.68×10^{5}	4.62×10^{4}	1.43×10^{4}	1.36×10^{3}	3.31×10^{3}
^{58}Co	2.69×10^{6}	2.01×10^{5}	8.20×10^{4}	4.55×10^{3}	3.53×10^{1}
^{60}Co	1.53×10^{6}	3.04×10^{5}	1.12×10^{5}	9.21×10^{3}	1.20×10^{3}
^{59}Ni	1.76×10^{3}	1.40×10^{3}	4.68×10^{2}	7.74×10^{1}	1.60×10^{-1}
^{63}Ni	2.50×10^{5}	1.72×10^{5}	5.41×10^{4}	8.64×10^{3}	1.83×10^{1}

Table XVIII. Mass activity of the main reactor components calculated for some NPPs (GBq/t)

Component	**NPP Loviisa** (1)	**NPP Jaslovske Bohunice** (2)	**NPP Kola** (3)	**NPP Greifswald** (4)
Dummy elements	7.7×10^7		1.6×10^6	
Protecting tube system				1.9×10^6
Core basket	1.4×10^7	1.8×10^6	0.3×10^6	7.4×10^5
Partition ring	3.2×10^7			
Annular water tank				2.0×10^2
Shaft	5.1×10^6			
Reactor vessel	3.5×10^5		8.1×10^3	2.3×10^3
Thermal shielding	2.2×10^4		9.6×10^3	
Bottom of core basket	6.5×10^6			
Pressure vessel cladding		2.7×10^5		
Shaft bottom	5.7×10^4			
Reference	[47]	[49]	[48]	[3]
Reference date	after 30 years of operation	5 years after final shutdown (30 years of normal operation)	data related to 2010	data related to July 1999

(1) maximum surface activity
(2) average specific activity
(3) specific activity, no further details available
(4) maximum mass specific activity ^{60}Co.

Table XIX. Time dependence of core basket specific activity for some radionuclides [49]

Specific activity [GBq/t]						
	Time after shut down [years]					
Isotope	5	50	100	125	150	200
^{108}Ag	2.23×10^1	1.75×10^1	1.33×10^1	1.16×10^1	1.01×10^1	0.77×10^1
^{63}Ni	3.51×10^5	2.53×10^5	1.77×10^5	1.47×10^5	1.23×10^4	8.57×10^3
^{59}Ni	2.76×10^3	2.76×10^3	2.76×10^3	2.76×10^3	2.76×10^3	2.76×10^3
^{60}Co	1.58×10^5	4.28×10^2	6×10^{-1}	2×10^{-2}	8×10^{-4}	1.1×10^{-6}
^{55}Fe	1.23×10^6	1.1×10^1	3.2×10^{-4}	5.53×10^{-8}		
^{54}Mn	1.7×10^3					

7.5. NON-RADIOACTIVE HAZARDOUS MATERIAL

Non-radioactive hazardous materials have been defined as "the materials that represent actual or potential risk for human beings or the environment in the case of improper handling" [50]. Hazardous materials are classified and grouped according to their characteristics:

- toxic
- aggressive (acidic, caustic)
- oil
- inflammable or spontaneously inflammable
- explosive
- corrosive
- biologically active.

Significant volumes of these hazardous materials exist at WWER-440 NPP sites. These materials were used during construction or operation and may represent a serious problem during decommissioning.

Typical categories of hazardous waste at WWER-440 NPPs are:

- asbestos, dust and fibres (fire protective materials, coatings, insulating materials)
- oil and oil sludge (auxiliary heating units, oil systems)
- spent solvent (maintenance, cleaning, degreasing)
- freon (refrigerating units, air conditioning)
- polychlorinated biphenyles — PCBs (electrical devices, condensers)
- PCB contaminated materials (transformer oil)
- materials contaminated by chlorinated hydrocarbons (soil, organic solvents)
- resins, adhesives
- accumulators (acids, caustics — Ni, Pb)
- mercury (electrical systems, illumination tubes, switchers)
- inorganic and organic waste contaminated by heavy metals (Ni, Mn, Cr, V, Cu, Ni, Hg, Cd)
- lead based paints.

It is important that consideration be given to the selection and implementation of decontamination, dismantling and waste management processes that minimize the generation of such waste. Hazardous materials present will be handled, processed and/or disposed of as soon as possible in accordance with national regulations. Delays and extra costs during decommissioning may result from the need of:

- additional licenses and approvals
- additional chemical analyses
- implementation of special devices and dismantling technologies

- installation of additional safety systems (ventilation, filters, containers, monitors)
- specialist contracts for hazardous waste processing (high temperature incineration, solidification processes, recycling)
- implementation of special regulations for packaging and transportation
- implementation of separate health and safety programmes for operations with hazardous waste.

It is imperative that production of waste that contains both radioactive and hazardous materials will be minimized because the disposal of this type of waste may require special consideration.

Although a complete inventory of hazardous waste arising on WWER-440 sites is not available, PCBs and asbestos are considered to be the most important hazardous waste. Planners and operators will keep in mind possible reclassification of some groups of materials in the framework of implementation of new environment protection principles.

8. DECOMMISSIONING TECHNOLOGIES FOR WWERs

8.1. INTRODUCTION

Over the past decades, experience has been gained on decommissioning large nuclear facilities. The overall spectrum of the existing technologies that are suitable for the performance of decommissioning is well described in the US Department of Energy “Decommissioning Handbook” [51] and in the European Commission “Handbook on Decommissioning of Nuclear Installations” [52]. Decontamination, waste treatment and dismantling technologies are to a large extent independent of plant type. Nevertheless, first major experiences in WWER decommissioning practice at the Greifswald NPP show that most commercially available equipment can be used with minor modifications to meet WWER design specifics, e.g. dimension of components, geometry, material composition, constructional arrangements. This section describes the applicability of existing technologies for use when performing decontamination, dismantling and waste management activities at WWERs.

8.2. DECONTAMINATION

An extensive description of the decontamination techniques is available in the technical literature [53]. General approaches are given in the EC Handbook [52], including a detailed description of the existing techniques. Various methods have been developed to decontaminate contaminated materials using chemical, physical, electrochemical and ultrasonic processes.

8.2.1. Operational decontamination experience

In most countries where WWERs are operating, the approach to reduce occupational radiation exposure includes regular decontamination of equipment, components, some reactor systems and subsystems. Chemical decontamination of isolated components, removable parts and auxiliary systems is very successful in reducing occupational radiation exposure [53]. These decontamination cycles also reduce the inventory of contaminants in the systems.

Physical, mechanical, chemical and electrochemical techniques have been developed and are used for routine decontamination of miscellaneous tools, removable equipment and individually isolated components of reactor systems and subsystems. These techniques range from non-aggressive physical techniques such as vacuum cleaning or water jetting which remove loosely held surface contamination to the more aggressive techniques where the material surface is partially removed along with the contaminants.

Regular decontamination at WWER plants is normally performed during the maintenance periods when the main parts of reactor cooling systems are checked and repaired. Typical components where routine decontamination is performed are the:

- main circulation pumps;
- main isolating valves; and
- steam generators.

High pressure water jetting, a modified chemical procedure (AP-Citrox) and electrochemical decontamination using a mixture of organic (oxalic and citric) or inorganic acids (phosphoric and sulphuric), are the basic decontamination technologies for these components. Decontamination factors in the range of 10–100 and 100–1000 are generally achieved for chemical and electrochemical decontamination, respectively [54].

Radiological surveys have shown that decontamination of large components such as steam generators or full system decontamination, is effective in lowering the overall radiation level in the plant. One steam generator represents 15% of the total surface of the primary loop and activity in the order of 10^{11}–10^{12} Bq is removed by decontamination.

Although the decontamination techniques for large volume components and removable components are well-known and in use, there is limited practical experience in full system decontamination in routine operation (Table XX).

Table XX. Examples of WWER full system and large volume component decontamination processes

Decontamination process	Decontamination object	Application plant
AP-Citrox	Full Primary circuit	NPP Rheinsberg, 1968–1975
AP-CE	Full Primary circuit	NPP Greifswald 1–4, 1978–1990
AP–CE	Full Primary circuit in decommissioning stage	NPP Novovoronesh-1,2 1981, 1984
AP-CE	Full Primary circuit in decommissioning stage	NPP Greifswald-5, 1994
CORD-UV	Full Primary circuit	Loviisa NPP-2, 1994
AP-CE	Full Primary circuit	Kozloduy 3 1994, 4-1996
AP-CITROX/AP-LOMI	Steam generator	Paks 3- 1993
AP-CITROX	Steam generators	J.Bohunice 1-1982, 2-1985
AP-CITROX	Steam generators	Dukovany I–IV,1994-1996
Electrolytical	Isolated Primary loop	NPP Greifswald-1,2 1996, 1997

Radiological conditions in WWER reactors do not require full system decontamination during normal operation. This method is used only in cases where there is an abnormal increase of dose rate, significant leakage through the protective barrier (failed fuel assemblies) or a large amount of maintenance work or modification has to be performed.

The full system chemical decontamination process developed at the All Russian Thermal Engineering Institute (VTI) was used under plant conditions at NPP Kozloduy Units III and IV. The two-step decontamination process was performed without fuel in the reactor. The first step was the oxidation phase which uses a low alkaline solution of $KMnO_4$ at 135°C. The second step dissolves the corrosion layer using a mixture of citric acid and $EDTANa_2$ at 160°C. The initial dose rates of 7–57 mSv/h were measured using more than 100 measuring points. These dose rates were reduced following decontamination with average decontamination factors of 1.2–28. The total radioactivity removed during decontamination was estimated as 9.3×10^{12} Bq and 1.4×10^{13} Bq, respectively for each unit.

During 1993–1996 more than 20 steam generators were decontaminated in the Dukovany NPP using the AP-CITROX process [36]. Approximately 1.2×10^{12}–3.5×10^{12} Bq of radioactivity were removed during the decontamination process. Decontamination factors in the range 2.7–10 were estimated for the tube bundle.

At Loviisa II the dose rate at the surface of primary circuit increased markedly in few years prior to decontamination (mean values: hot legs — 0.79 mSv/h, cold legs — 2.01 mSv/h, regenerative heat-exchangers 14.0 mSv/h). Large inspection and repair tasks scheduled for outage resulted in a decision to perform the full system decontamination using the CORD/UV process in 1994 [55]. After nine days and four CORD/UV decontamination cycles, the Loviisa full system decontamination resulted in the removal of about 291 kg of corrosion products (Fe, Cr, Ni). Approx. 4.1×10^{13} Bq were removed from the primary system and 3.6×10^{13} Bq were retained on ion exchange resins. Remaining contamination 2×10^{3}–5×10^{4} Bq/cm² was found on the surface of the main circulation piping. Average dose rate reduction factors of 14–153 were estimated on the basis of dose rate measurements.

The application of concentrated decontamination processes (AP–AC, AP–ACE, AP–CITROX) is a typical feature of WWERs decontamination activities. This is significantly different from western type PWRs where diluted decontamination processes (AP–LOMI, CAN–DECON) are generally used. Comparison of both processes carried out in [52] shows several advantages of diluted processes:

- lower production of liquid waste (system is filled up once, chemicals are added into systems and removed with use of ion exchange filters)
- lower corrosion of the base material
- process is better controlled and does not require drainage of the system
- radioactive waste is largely produced in the form of a small volume of the high active resins suitable for direct disposal or processing.

Despite these advantages, highly concentrated processes are still used at WWER plants. The main reasons are:

- relatively low cost,
- absence of suitable technology for processing of very high active ion exchange resins, and
- significant refurbishment costs should dilute processes be used.

8.2.2. Pre-dismantling decontamination

All operating WWER plants have a common basic design. In practice, however, the various units differ with respect to the materials used for major components. For example, the reactor pressure vessel and pressurizers in some of the units have stainless steel cladding (later design) while older models are fabricated without cladding. Plant auxiliary systems may also vary in design and construction. Therefore, there are variations between individual units as far as the composition of the oxide layers is concerned.

It has been shown that complete WWER primary circuit system decontamination is beneficial prior to starting decommissioning activities, for the following reasons:

- With a full system decontamination the in-plant radiation is reduced considerably at the beginning of the decommissioning process. This facilitates subsequent cutting work due to a reduced radiation level and reduced generation of radioactive aerosols.
- Full system decontamination prepares the entire primary circuit including the RPV and most of the primary auxiliary systems for dismantling in one single step.
- Carrying out in-situ decontamination at the very beginning of the decommissioning activities makes it possible to use existing plant systems, experienced personnel and available infrastructure to the maximum extent.

Practical demonstrations of full system decontamination were undertaken at the Greifswald NPP unit 5 with AP-CE in 1994.

Despite significant benefits that can be achieved using full system decontamination technologies, the following issues have also to be taken into account:

- production of liquid and/or solid waste,
- possible redistribution of radioactivity and hot spot generation,
- need for complete flushing and draining of system parts,
- costs,
- dose commitment in performing the decontamination.

8.2.3. Post-dismantling decontamination methods

The huge volumes of contaminated material generated from the equipment dismantling activities can be significantly reduced with the use of appropriate post-dismantling decontamination methods. The main criteria that should be taken into account when selecting a decontamination process include:

- existing clearance criteria,
- decontamination efficiency,
- volume of the secondary waste produced,
- operational safety and staff experience,
- licensing,
- cost,
- dose commitment.

Processes suitable for post-dismantling decontamination include the following basic groups of methods:

- hard chemical decontamination processes,
- ultrasonic cleaning,
- electrochemical methods,
- mechanical-abrasive methods, and
- special methods (foams, gels, strippable films, etc.).

Although these methods are common for all decommissioning projects, they are not widely used at the operational WWER NPPs. Table XXI provides references of the post-dismantling decontaminations performed at operational plants.

Table XXI. Examples of decontamination methods of dismantled components

NPP	**Part**	**Method**	**Reference**
Kozloduy	pressurizer surge lines	chemical, mechanical	[61]
Novovoronesh	heat exchangers, steam generators	hard chemical	[62]
Dukovany	spent fuel storage pool racks	electrochemical	[63]

The following is intended to provide one example of a decontamination process as applied to shutdown WWERs. The Strong Ozone Decontamination Process (SODP) is a one step, room temperature process based on nitric acid, cerium and ozone. The secondary waste does not contain any chelates. It was developed in the late 1980s by Studsvik RadWaste AB and tested on stainless steel material from the Greifswald plant [56]. Four samples from steam generator tubes were exposed to SODP at the Studsvik laboratories. The result indicates generally that activity levels below 1 Bq/g can be achieved after 72 h of exposure. The conclusion of the tests performed with WWER SG tubes is that the SODP method is suitable for the WWER SG recycling by decontamination and melting.

8.2.4. Decontamination of concrete structures

Decontamination of buildings and concrete structures is a common task for all NPP decommissioning projects before releasing the structures. Although there is no relevant experience yet from WWERs on these activities, several techniques and tools are well known and available. Scabblers, shavers, hydraulic hammers, cutting wires and similar methods have been successfully used in other decommissioning projects (e.g. JPDR [57] and Eurochemic [58]).

8.3. DISMANTLING AND DEMOLITION TECHNIQUES

8.3.1. General

A detailed survey of the dismantling techniques used for activated parts of RPVs, internals, piping, tanks, and components and for demolition of concrete are given in [51, 52]. General guidance for the dismantling projects are where possible:

- keep the techniques and tools as simple as possible,
- use proven industrial equipment, and
- use mock-ups for potentially hazardous areas (remote dismantling) to test the equipment, tools and working (cutting) procedures.

This guidance is recognized and is being implemented in the Greifswald NPP decommissioning project. Numerous new and adapted industrial equipment have been introduced in this project for the dismantling of all kinds of material.

8.3.2. Dismantling techniques in WWER practice

The first complete dismantling of a complete WWER NPP station started in 1995 being the Greifswald NPP decommissioning project. Complete removal of radioactive material from the site is expected by 2009. The main experiences and results from the Greifswald project are given in Appendix 7.

8.4. ON-SITE WASTE MANAGEMENT INCLUDING MINIMIZATION TECHNIQUES

8.4.1. Introduction

Most of the operating WWER units were designed in the late 1960s and were based on the standard Soviet regulations valid at that time. It was usual practice at that time to adopt the specifications of the supplier because limited experience in this area was available in the former COMECON countries. This concept had a major influence on the general waste management policy in all the COMECOM countries, in which WWER reactors had been commissioned and operated. This resulted in waste management systems being based on the regulatory requirements of the mid-1970s [27].

The concept developed for the management of waste at WWERs was to store the waste on-site and to postpone decisions on its conditioning and disposal until the decommissioning stage. This meant that wastes from operation and dismantling would be handled together. Waste collection and storage systems were developed to accommodate ten years generation of the partially pre-treated operational waste with possible extension of storage capacities. The waste management technology was not sufficiently developed in the design of the systems and further treatment of solid wastes and concentrates was anticipated at the decommissioning stage. Sorting and removal of solid wastes from the storage vaults for treatment was not considered.

Modifications of the original waste management system design occurred, mostly in some waste stream segregation (laundry and shower waste), using of thermally resistant and long life ion exchange resins, and solid waste compacting.

Because a great quantity of information and references are already available in the open literature about various aspects of waste management [27, 51, 52, 59, 60], only methodological issues will be introduced in the following sections.

According to the IAEA definition [10] waste management encompasses all activities, administrative and operational, that are involved in the handling, pre-treatment (collection, segregation, decontamination), treatment (volume reduction, removal of radionuclides from waste, change of composition), conditioning, transportation, storage and disposal of waste from a nuclear facility. Minimization is a concept which embodies the reduction of waste with regard to its quantity and activity to a level as low as reasonably achievable. Minimization as a practice includes source reduction, recycling and reuse as well as waste management optimization.

8.4.2. Radioactive waste inventory

As a heritage of the above mentioned original concept for waste management at WWERs, the following types of radioactive waste can be expected for treatment at a NPP site after final shutdown:

(a) operational waste and

(b) waste from post shutdown activities (see Section 7). Details are given here for operational waste only.

Operational radioactive waste can be classified using various criteria, but most commonly wastes produced at the WWER-440 NPP sites are divided into the five categories discussed below.

Radioactive concentrates

A specific feature of the WWER waste processing system is evaporation of the waste water and thus significant production of liquid radioactive waste concentrate. All water in the controlled area from the technological processes is collected in an active drainage system and consequently processed in the evaporator plants. Evaporated condensate is cleaned further with ion exchange filters and can then be reused at the plant. The concentrate is routed to the storage tanks where it is temporarily stored. The average composition of evaporator concentrates and currently stored volumes are shown in Table XXII [61–63].

Due to the design of the drainage system the concentrate contains significant amounts of boric acid and borates, the solubility of which is the limiting factor for the concentrates regarding further storage and treatment [64].

Evaporation, cementation and bituminization of concentrates are the basic technologies proposed or used at the WWER sites. Attempts to reduce volume and salt contents of concentrate by recovery of the boric acid have been investigated but not implemented.

Table XXII. Average composition of evaporator concentrates 1997

	PAKS	KOZLODUY Unit 1, 2	DUKOVANY Unit 1–4	BOHUNICE V–1	NOVOVORONESH
Total volume (m^3)	2800	2180	4010	3900	4000
pH	12	8–9	11.1–11.6	11.3–13	8.7–9.1
H_3BO_3 (g/L)	181–209	18.6–39.5	54.4–86.6	75–100	27–105
Dry residue (g/L)	269–335	300–450	101–185	150–390	300–400
Sodium (g/L)	74–85.8	8–18	33–53	42–100	18–84
Potassium (g/L)	5.4–15.7		4.1–8.6	9.4–21	15–20

Contaminated oil

Contaminated oil comes from in-leakage of the primary coolant into the auxiliary systems of the primary circuit components such as the main circulation pumps. The generation of contaminated oil is in the order of $0.5 m^3/a$/reactor unit with a specific activity of 10^4–10^5 Bq/L.

Sludges

The total volume and the main characteristics of sludges remain as problems to be addressed at most WWER plants. Considerable quantities of sludge are located in the waste water collecting system and in tanks (sedimentation tank, overflowing tank, waste water tanks, storage tanks). Due to limited accessibility, the lack of a sampling system and inability to agitate the material, characterization of sludges will require further development. The total volume of sludges at the NPP Greifswald is estimated to be 95 m^3 with an average specific activity of 4×10^{10} Bq/m^3.

Ion exchange resins

Spent ion exchange resins are stored in the storage tanks located in the auxiliary buildings. The transport of sorption materials to the storage tanks is performed hydraulically with flushing water. Due to limited production of ion exchange resins (<15 m^3 /unit/year), capacity of the storage tanks seems to be sufficient for the entire NPP lifetime. Main parameters of the resins being stored differ from site to site. The total volume of the spent ion exchange resins will reach approximately 250–350 m^3 throughout the NPP lifetime with a specific activity of approximately 10^{10} Bq/m^3.

There are low active ion exchange resins (300 m^3) and high active resins originating from the reactor coolant treatment unit (35 m^3). Both types of resin are stored in resin storage tanks. Certain complications can arise from the presence of radioactive carbon and sludges in the ion exchange resins storage tanks and from the presence of spent ion exchange resins throughout the waste water treatment system.

Solid Waste

The composition and the amount of waste varies with time. The majority of the solid waste is generated during the outage periods when the composition also differs from that generated during normal operation. Annual production of solid wastes varies from 30 to 400 m^3/unit depending on the site. This volume is dependent on the proposed or installed technology, segregation of non-radioactive waste, pre-compaction and sorting into combustible or compactible categories. Segregation of the metallic waste simplifies further processing and makes possible decontamination and/or recycling. Data collected during outages show that the compactible waste represents approximately 70–80% of the total.

The operational solid waste generated in the controlled zone is sorted, collected in plastic bags, and transported to the auxiliary building where the waste is stored in concrete pits. These pits were considered as final disposal and not filled in an orderly manner. Certain improvements can be found in sites where the waste is stored on metallic pallets or in 200 litre drums. As the storage capacities are almost exhausted, volume reduction is being performed and/or additional storage capacities are being built.

More than 20 000 samples were taken before the high pressure compacting of approximately 1500 m^3 of waste in 1996 at the NPP Dukovany. Samples were divided into 10 batches, analyzed and the results were used for estimation of total activity. The activity of all relevant nuclides, including long lived ones, and α, β emitters (^{41}Ca, ^{90}Sr, ^{129}I, ^{137}Cs, ^{94}Nb, ^{239}Pu, ^{59}Ni, ^{241}Am, ^{63}Ni, ^{99}Tc), had to be determined in order to meet the disposal site acceptance criteria. Estimation of total disposed activity was based on the results of the radiochemical analyses taking into account the weight and gamma doses of the compacted drums. A total of 311 tonnes of material was compacted and disposed of with an activity of 5.6×10^{11} Bq.

Dose rate distribution of non-processed waste estimated during compaction was as follows:

<0.1 mGy/h–65%,

0.1 mGy/h to <1.0 mGy/h–24%,

1.0 mGy/h to <10 mGy/h–6%, and

>10 mGy/h–5%.

8.4.3. Waste management processes

The management of materials generated during the decommissioning is basically an optimization between the cost for conditioning and final disposal or interim storage on the one hand and the cost for achieving reuse or disposal as non-radioactive material on the other modified by any specific legal requirements. In particular, costs for proving compliance with release criteria need to be considered. Criteria for decisions, definitions and working steps during the treatment of all waste streams need to be made very thoroughly so that decisions related to waste management can be made appropriately and cost effectively.

A very important prerequisite for optimal waste management is good characterization, i.e. radionuclide concentrations, concentration of chemical elements and physical parameters. These are important to enable effective waste segregation into streams, evaluation of possible methods of treatment, and selection and planning of appropriate processes so that packages will be acceptable to the disposal facility.

The choice of an appropriate process for the conditioning of certain waste streams depends on the physical and chemical properties of the waste itself. The activity as well as the radionuclide contents and chemical composition are the key factors in designing the process and equipment for treatment and for radiation protection. A survey of methods recommended for treatment of various waste streams treatment is presented in Table XXIII [53, 59].

8.4.4. Principles of a waste minimization strategy

The objective of waste minimization is to reduce the volume of waste requiring storage and/or disposal and consequently reduce the environmental impact, as well as the total costs associated with waste management. Waste minimization is an inherent part of waste management.

The factors that should be considered when preparing the strategic and technical decisions related to waste minimization strategy can be summarized as:

- source reduction, and
- recycle and reuse.

Table XXIII. Methods recommended for waste treatment

Waste	Processing
1. Solid 1.1 metallic	• size reduction cold cutting methods: sawing, cutting, hydraulic shears, water jet, abrasive cutting wheels hot cutting methods: oxy-acetylene, plasma arc • decontamination • high pressure compaction • melting
1.2 from civil construction	• crushing - for segregation/release • grouting
1.3 soil	• washing/leaching • sandwich disposal
1.4 burnable	• shredding • incineration • low force compaction
1.5 non burnable	• shredding • low force compaction
1.6 sludges	• thermoplastics • cementation • bituminization
1.7 ion exchangers	• thermoplastics • cementation • bituminization
2. Liquid	• evaporation • cementation • bituminization • filtration • treatment with specific sorbents
3. Gaseous	• filtration (close to place of origin) • wet scrubbing (e.g. during incineration)

Source reduction

There are technical and organizational measures that are useful in reducing waste at the source where it is generated. The following ways were identified:

- re-evaluation of existing operational practices with the aim of limiting the waste streams at the source, detailed and reasonable segregation of wastes and clear specification of the responsibilities of all the personnel participating in the decommissioning activities;
- reducing the amount of material allowed to enter a controlled area (wrapping materials, plastics, application of multiple usable materials);

- minimizing the spread of radioactivity as much as possible by using extensive radiological controls, local ventilation and working area protection (tent);
- minimizing the generation of secondary waste by using appropriate techniques for dismantling and waste handling; and
- considering radioactive decay through the timing of dismantling operations.

Recycle and reuse

The prerequisite for recycle and reuse of materials from decommissioning is establishing relevant criteria by national regulatory authorities as well as developing procedures for their implementation. At present only a few countries have issued firm criteria for recycling and reuse of materials, although it is an attractive alternative to radioactive waste disposal. Recommendations on clearance criteria are currently under preparation by the IAEA, EC and OECD/NEA [12, 29, 60].

The following materials are usually considered for reuse or recycling:

- steel and stainless steel, nickel,
- non-ferrous metals (copper, aluminium); and
- non-metallic (concrete, sand).

Materials for reuse or recycling can typically belong to the following categories:

- clearance for unconditional release/reuse;
- authorized reuse within nuclear industry (manufacturing of shielding, containers for radioactive waste and other products); or
- authorized release e.g. to a specified melter.

Reuse can be achieved by:

(1) decontamination

(2) separation and segregation of those parts that cannot be easily decontaminated

(3) other treatment (e.g. removal of plastics from cables), or

(4) melting.

An important and at times rather complex procedure, is the monitoring of a material’s radioactivity for compliance with release criteria.

9. CONCLUSIONS

1. Initially, Member States that had WWERs did not have approved policies concerning decommissioning. Recently, attention has been given to this activity and requirements have been or are being developed.

2. Spent fuel management is not considered part of decommissioning; however, it can strongly affect the decommissioning strategy.

3. The characterization survey, i.e. a comprehensive engineering and radiological survey, of the WWER is an important part of the decommissioning process. This survey will provide data for all other planning activities.

4. The organization will change during the various phases of decommissioning. The number and types of individuals needed to support each phase will depend on the activities to be performed. The operational organization is not generally considered to be appropriate for decommissioning.

5. Technologies are currently available to safely decommission WWERs. New technologies are expected to be limited in effect to the decrease of costs and the exposure to workers. Many decommissioning techniques used for other types of reactors are also applicable to the decommissioning of WWERs.

10. RECOMMENDATIONS

1. It is important that planning for the decommissioning of WWERs begins prior to the reactor final shutdown. This planning includes an environmental impact assessment and consideration of the social-economic aspects.
2. Developing a spent fuel and waste management strategy is a high priority with the regulatory authorities of the Member States that have WWERs. Inception of dismantling is subject to the availability of spent fuel and waste management policies.
3. Establishing release criteria with regulatory authorities in the early stages of planning is very important to a successful decommissioning project. This criteria should include those for clearance, recycling, reuse and authorized discharge.
4. It is important to determine the cost to decommission a facility early in the planning process. Methods for funding this cost are best identified during the operating life of the plant. If funds are not available, the alternative may be to place the facility into long term storage. A strategy for minimizing decommissioning costs while ensuring safe and timely decommissioning is an essential consideration by planners.

APPENDICES 1–7

Appendix 1

WWER DESIGN AND OPERATIONAL FEATURES

The acronym WWER (Water, Water, Energy, Reactor) refers to Soviet design water cooled, water moderated, electricity generating reactors. The acronym WWER-440 designates the WWER type having a designed net electrical output of 440 MW(e).

All WWER-440 plants have six loops, with isolation valves and a main circulating pump on each loop and horizontal steam generators (SG). All use 220 MW(e) steam turbines. The reactor core is composed of hexagonal fuel assemblies with 126 fuel rod positions each. Control rod assemblies are combination of fuel assembly and an absorbing extension. The WWER-440 uses a rack and pinion drive mechanism to move the control rods.

The reactor pressure vessel and the key mechanical and fluid system components are of a standardized design and are produced with standardized manufacturing procedures. However some reactor pressure vessels have a stainless steel cladding others do not. The primary coolant system including steam generator heat transfer tubes is invariably in stainless steel. The secondary side including SG shells is in carbon steel. Although the basic WWER design has undergone changes in engineering and institutional judgements, there has been little change in the basic component and system design. The changes have tended to be material changes and the addition to later plants of safety-related features and equipment. The necessity for rail transport limits the diameter of the reactor pressure vessel.

For the original design of WWER-440/230 the design basis accident (DBA) is a rupture of a 100 mm pipe from the primary circuit with a diaphragm with 32 mm diameter for flow restriction with simultaneous loss of electrical supply. A few modifications of the safety systems have been implemented in recent years.

The pressure compartment system is a closed compartment system with a net volume of about 14 000 m^3 enclosing as a confinement the main components of the primary circuit and is designed for an overpressure of 1 bar. The compartment system is connected to the environment by exhaust ventilation flaps. Opening of the flaps protects the pressure compartment system in case of a rupture of a pipe up to 200 mm nominal diameter.

The safety design of the WWER-440/213 differs essentially from the older model 230 by a higher safety standard. The installed active and passive core cooling systems cover a large spectrum of accident scenarios up to the DBA what is the guillotine pipe break of the main cooling circuit pipe with a diameter of 500 mm with simultaneous loss of electrical supply. The safety systems of the WWER-440/213 are provided with higher capacities and redundantly designed as $3 \times 100\%$ systems. They are physically separated from the operational system.

The WWER-440/213 has a confinement room system with an integrated condensation bubble tower designed for a pressure up to 2.5 bar and thus, is able to maintain its localization function also in case of DBA. The pressure vessel of the WWER-440/213 is cladded with stainless steel. Tables 1-1 and 1-2 provide comprehensive lists of WWER-440 units. An outline of the major features of models 213 and 230, are given in Figs 1-1 and 1-2, respectively.

Table 1-1 [6]

WWER-440 MODEL 230 REACTORS			
Russian Federation		**First power**	**Design lifetime final shutdown year***
Novovoronezh	Unit 3	1971	2001
	Unit 4	1972	2002
Kola	Unit 1	1973	2003
	Unit 2	1974	2004
Armenia			
Armenia	Unit 1	1979	-
		(permanent shutdown in 1989)	
	Unit 2	1980	2015**
Bulgaria			
Kozloduy	Unit 1	1974	2004
	Unit 2	1975	2005
	Unit 3	1980	2010
	Unit 4	1982	2012
Slovakia			
Bohunice	Unit 1	1978	2003
	Unit 2	1981	2006
Germany			
Greifswald	Unit 1	1973	-
	Unit 2	1974	-
	Unit 3	1978	-
	Unit 4	1979	-
		(permanent shutdown of all units in 1990)	

* The design lifetime concept does not imply that a unit should be closed down necessarily at the established time if a case can be made that it might continue operation safely and efficiently. In some countries, the design lifetime is calculated from the startup rather than from the commercial operation date.

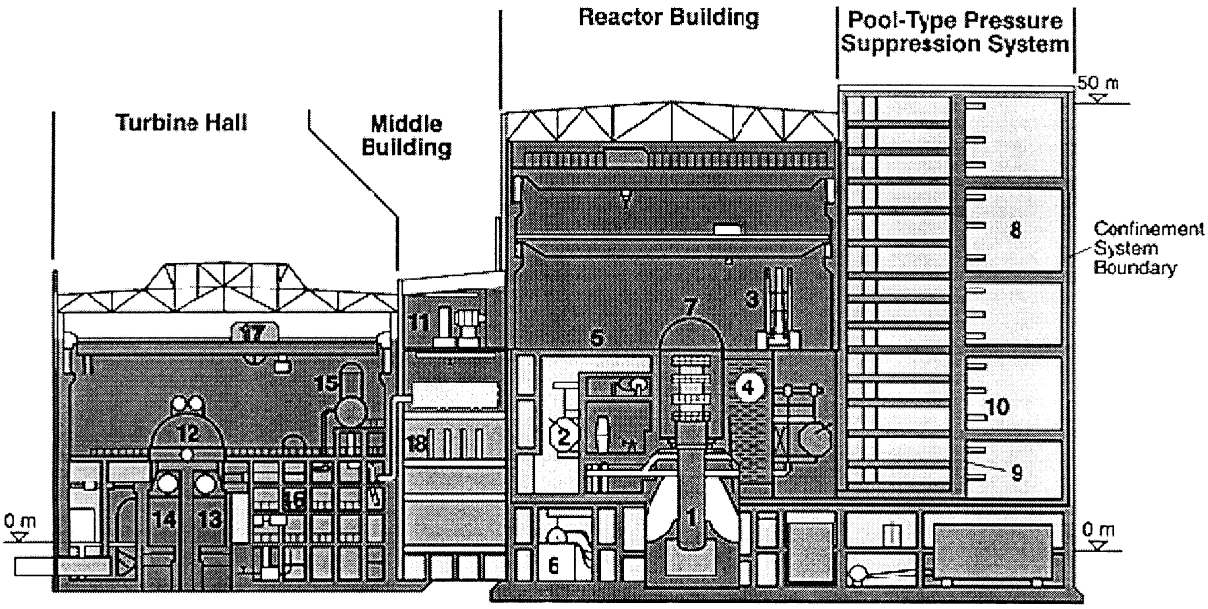
** Including 5 years of cold shutdown.

Table 1-2 [6]

WWER-440 MODEL 213 REACTORS			
Russian Federation		**First power**	**Design lifetime final shutdown year***
Kola	Unit 3	1981	2011
	Unit 4	1984	2014
Slovakia			
Bohunice	Unit 3	1984	2014
	Unit 4	1985	2015
Mochovce	Unit 1	1998	2028
	Unit 2	1999 (expected)	2029
Czech Republic			
Dukovany	Unit 1	1985	2015
	Unit 2	1986	2016
	Unit 3	1986	2016
	Unit 4	1987	2017
Finland			
Loviisa	Unit 1	1977	n.a.
	Unit 2	1981	n.a.
Germany			
Greifswald	Unit 5	1988 (permanent shutdown in 1990)	-
Hungary			
Paks	Unit 1	1983	2013
	Unit 2	1984	2014
	Unit 3	1986	2016
	Unit 4	1987	2017
Ukraine			
Rovno	Unit 1	1980	2010
	Unit 2	1981	2011

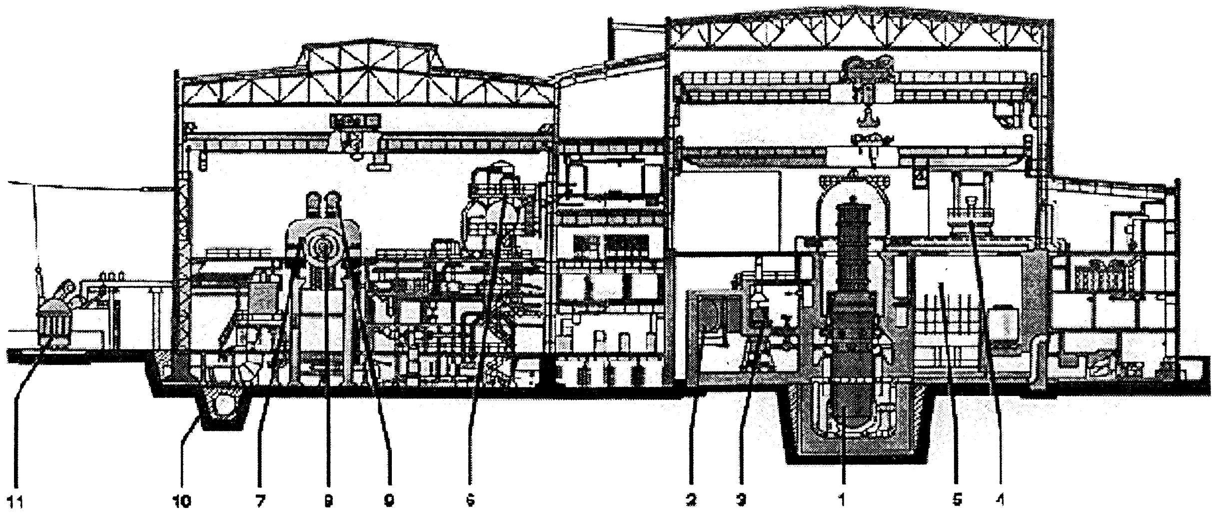
n.a. = information not available.

* = The design lifetime concept does not imply that a unit should be closed down necessarily at the established time if a case can be made that it might continue operation safely and efficiently. In some countries, the design lifetime is calculated from the startup rather than from the commercial operation date.



Legend: 1. Reactor pressure vessel, 2. Steam generator, 3. Refuelling machine, 4. Spent fuel pond, 5. Confinement system, 6. ECCS compartment, 7 Protective cover, 8. Confinement system, 9. Sparging system, 10. Check valves, 11. Intake air unit, 12. Turbine, 13. Condenser, 14. Turbine block, 15. Feedwater tank with degasifier, 16 Preheater, 17. Turbine hall crane, 18 Electrical instrumentation and control compartments.

FIG. 1-1. Important features of WWER-440/213.



Legend: 1. Reactor, 2. Steam generator, 3. Main circulation pump, 4. Refuelling machine, 5. Cooling pond, 6. Feedwater storage tank, 7. Turbine, 8. Generator, 9. Steam piping, 10. Cooling water piping, 11. Transformer.

Fig. 1-2. Important features of WWER-440/230.

Appendix 2

CHARACTERISTIC DATA OF STRUCTURES (NPP PAKS V-213 TYPE) [18]

Main building

The main building includes the reactor building, the turbine hall and the electrical galleries. Main construction parameters of these elements are given below.

Reactor building

The reactor building includes the reactor hall and covers the main part of controlled zone with only a small part within the monitored area. Its structure below +18.9 m level consists of thick reinforced concrete walls and slabs, supported by a reinforced concrete ground slab. Above this level the structure is made mostly of steel with a concrete panel shell.

Characteristic data:

Max. depth of the foundation	–9.5 m
Height of the building	50.6 m
Max. thickness of the basement floor	3.0 m
Max. slab thickness	1.5 m
Area of the basement	11 058 m^2
Total area of all levels	64 162 m^2
Building volume	6 48 744 m^3

Turbine hall and the galleries

The turbine hall and the galleries consists of steel structures with concrete panel shell and poured concrete floors. Inner partitioning walls are made of bricks.

Characteristic data:

Max. depth of the foundation	–6.5 m
Height of the machine hall	29.4 m
Cross/and longitudinal gallery	24.0 m
Area of the basement	16 640 m^2
Total area of all levels	69 512 m^2
Building volume	6 29 342 m^3

Auxiliary building

The walls and the slabs consist of a reinforced concrete structure.

Characteristic data:

Max. depth of the foundation:	–2.5 m
Building height	23.5 m
Area of the basement	9 803 m^2
Building volume	63 199 m^3

Sanitary building

Part of building belongs to the controlled zone and consists of mounted reinforced concrete framework with partition brick walls.

Characteristic data:

Max. depth of the foundation:	–6.2 m
Building height	30.0 m
Area of the basement	2 967 m^2
Basic area of all levels	16 527 m^2
Building volume	86 498 m^3

Appendix 3

MAIN CHARACTERISTIC DATA OF PRIMARY CIRCUIT COMPONENTS [3]

Reactor		
	V-230	V-213
Total height of reactor	23770 mm	23 682 mm
Volume of reactor	110 m^3	110 m^3
Weight of RPV Weight of sealing parts	199870 kg	2 09 470 kg 39 600 kg
Total height of RPV	11800 mm	11 800 mm
Weight of upper unit	210 000 kg	140 335 kg
Height of upper unit	11970 mm	11 882 mm
Diameter of ring water tank	4100/6100 mm	(not applicable)
Weight of metal construction of dry biological protection	(not applicable)	15 490 kg
Weight of serpentinite concrete filling	(not applicable)	66 000 kg
Material of reactor internals	Austenite	Austenite
Material of RPV	C-steel, Austenite cladding	C-steel, Austenite cladding

Reactor coolant pipes	
Length per loop	ca 27 500 mm
Diameter	560 mm
Wall thickness	32 mm
Material	Austenite
Number	6

Main circulation pumps		
	V-230	V-213
Height	6 630 mm	9 300 mm
Diameter	3 675 mm	3 400 mm
Weight	51 t	48.9 t
Material	Austenite	Austenite, C-steel
Number	6	6

Steam generators		
	V-230	V-213
Diameter	3 474 mm	3 480 mm
Length	11 500 mm	12 260 mm
Volume (secondary)	70 m^3	73 m^3
Weight	156 t	163 t
Material	C-steel, Austenite	C-steel, Austenite
Number	6	6

Pressurizer		
	V-230	V-213
Height	10 615 mm	12 000 mm
Diameter	2 400 mm	2 718 mm
Weight	101 t	130 t
Material	C-steel, Austenite	C-steel, Austenite
Number	1	1

Main isolation valves	
Height	2 750 mm
Weight	8 t
Material	C-steel, Austenitic
Number	12

Appendix 4

LIST OF CONTAMINATED TECHNOLOGICAL SYSTEMS IN THE MAIN AND AUXILIARY BUILDINGS [18]

Turbine hall	
Name of system	**[kBq/cm^2]**
Transportation technology equipment	0.5–20
Ventilation systems	0.01–0.1
Auxiliary condensate system	0.01–2
Steam generator blowdown system	0.005–0.01
Lubrication system of the MCPs	20–40
Regeneration and treatment of the boric acid solutions	500–5000
ECCS and hydroaccumulators	20–40
Spent fuel storage pool water cleaning system	20–200
Continuous primary coolant purification system	500–5000
Intermediate cooling circuits for the reactor building	<0.0004
Spent fuel storage pool cooling system	20–40
Low pressure safety injection system	20–40
High pressure safety injection system	20–40
Chemical and volume control system	50–100
Liquid waste treatment system	20–40
Spray system	20–40
Boron recycling and storage system	50–100
Gaseous waste treatment system, Hydrogen recombination and gas off purification system	20–200
Decontamination system	0.01–2
Sampling system	00.4–500
The organized leakage system	500–1000
Drains and vents and leakages collection system	50–100
Radioactive drain system	<20
Steel linings	<0.01
Bubbler/condenser tower system	<0.0004
Main primary cooling circuit	50–100
Autonomous MCP cooling system	20–75
Pressurized water reactor systems	50–100
Volume compensation system	50–100
Auxiliary building	
Name of system	**[kBq/cm^2]**
Ventilation systems	0.01–0.1
Steam generator blowdown system	0.05–0.005
Chemical and boron solution system	<0.004
Chemical and volume control system	0.01–0.02
Liquid waste treatment system	20–200
Gaseous waste treatment system,	20–200
Sampling system	0.04–500
Liquid waste storage system	20–1000
Radioactive drain system	<20
Steel linings	<0.01

Appendix 5

COMPOSITION OF MATERIALS BY WEIGHT (%) USED BY SKODA PLZEN [49]

Nuclide	I.	II.	III.	IV.	V.	VI.	VII.	VIII.	IX.
Fe	68.8451	67.13	59.99	94.68	99.7	98.17	67.345	4.867	3.796
C	0.08	0.1	0.09	0.155	0.2	0.2	0.08		
Mn	1.5	2.	1.5	0.45		0.5	2.	0.094	
Si	0.8	1.	0.85	0.27		0.35	1.	14.405	31.714
P	0.0349	0.03	0.03	0.025		0.04	0.045		
S	0.02	0.02	0.02	0.025	0.05	0.04	0.03	0.319	0.571
Cr	18.	19.	24.5	2.75	0.05	0.3	18.		
Ni	10.	9.75	13.	0.4		0.2	11.		
Ti	0.7						0.5	0.202	
Co	0.02	0.02	0.02	0.015					
Nb		0.95							
Mo				0.7					
Y				0.3					
Cu				0.15		0.2			
As				0.08					
Al								1.332	3.429
Ca								11.684	7.021
Mg								14.407	0.857
Ba									
Na								0.108	
K								0.244	
O								50.444	51.796
H								1.894	0.816
Density g/cm^3	7.9	7.9	7.85	7.8	7.9	7.9	7.9	2.133	2.45

Legend:

I:	08Ch18N10T	Core basket, shaft
II:	Sv-08Ch19N10G2V	Cladding
III:	07Ch25N13	Transition layer
IV:	13Ch2MF	Pressure vessel
V:	11375.1	Inside serpentinite jacket
VI:	11416.1	Insulation wrapping
VII:		Thermal insulation
VIII:		Serpentinite concrete
IX:		Ordinary concrete

Appendix 6

LIST OF ACTIVATED REACTOR COMPONENTS (NPP GREIFSWALD UNIT I) [3]

Protecting tube system	
Total weight	approximately 27 250 kg
Maximum gamma dose rate	2.1×10^{2} mSv/h in 0.5 m distance
Maximum mass specific activity ^{60}Co	1.9×10^{8} Bq/g
Total activity ^{60}Co	4.7×10^{14} Bq
Maximum contamination ^{60}Co	1.9×10^{3} Bq/cm^2 outside the core 1.9×10^{5} Bq/cm^2 inside the core
Core basket	
Total weight	approximately 22 420 kg
Maximum gamma dose rate	1.3×10^{5} mSv/h in 0.5 m distance
Maximum mass specific activity ^{60}Co	7.4×10^{8} Bq/g
total activity ^{60}Co	5.0×10^{15} Bq
maximum contamination ^{60}Co	7.4×10^{5} Bq/cm2 inside the core
Reactor cavity with cavity bottom	
total weight	approximately 57 770 kg
maximum gamma dose rate	2.3×10^{3} mSv/h in 0.5m distance
maximum mass specific activity ^{60}Co	1.7×10^{8} Bq/g
total activity ^{60}Co	1.0×10^{15} Bq
maximum contamination ^{60}Co	1.9×10^{3} Bq/cm^2 outside the core 1.7×10^{5} Bq/cm^2 inside the core
Reactor pressure vessel	
total weight	approximately 200 000 kg
maximum gamma dose rate	1.8×10^{2} mSv/h in 1m distance
maximum mass specific activity ^{60}Co	2.3×10^{6} Bq/g
total activity ^{60}Co	5×10^{13} Bq
maximum contamination ^{60}Co	1.9×10^{4} Bq/cm^2 outside the core 2.3×10^{4} Bq/cm^2 inside the core
Insulation	
total weight	approximately 17 000 kg
maximum gamma dose rate	approximately 6 mSv in 1m distance, inside
maximum mass specific activity ^{60}Co	4.1×10^{4} Bq/g
total activity ^{60}Co	approximately 2×10^{10} Bq
Annular water tank	
total weight	approximately 62 000 kg
maximum gamma dose rate	3 mSv/h in 0.5 m distance, outside
maximum mass specific activity ^{60}Co	2.0×10^{5} Bq/g
total activity ^{60}Co	3.5×10^{12} Bq

Appendix 7

DISMANTLING TECHNIQUES IN WWER PRACTICE (NPP GREIFSWALD)

The first total dismantling of a complete WWER NPP station started in 1995 being the Greifswald NPP decommissioning project. Complete removal of radioactive material from the site is expected by 2009. The main experiences and results from the Greifswald project are given below:

- The planning of dismantling is performed on a system basis. During planning, all preparatory measures for the plant part to be dismantled are identified. This includes: a radiological and hazard assessment (local dose rates, contamination levels, pollutant identification), the isolation of the part to be dismantled, the need for other systems (mechanical, electrical and I&C interfaces), pre-dismantling decontamination measures, and adaptations of the operation manual. The dismantling itself is performed on a room by room basis, the detailed dismantling job description, which also includes the mass and inventory data, and cost estimates. All these activities are performed by the licensee under close supervision by the regulatory body. In this way it is assured that all necessary conditions are fulfilled before the dismantling work of a plant part starts, i.e.:
  - the plant part is shut down and isolated from post operational and neighbouring systems;
  - operational media are removed, and the plant part is decontaminated, rinsed and dried, as needed;
  - electrical drives and measuring devices are disconnected from their power supply;
  - all necessary auxiliary equipment and material are available (local ventilation, lifting devices, transport containers, etc.);
  - the plant part which has to be dismantled is identified and the documentation shows the current status of the plant systems; and
  - approved dismantling instructions are available.

- After dismantling of about 12 000 tonnes of different kinds of plant parts and components, the overall average dismantling rate is 42 kg/person·hour, whereby for the controlled areas of unit 5, it is 39 kg/person·hour and for the monitored area (turbine hall) 47 kg/person·hour. These results were achieved by using proven industrial cutting tools and equipment. The following tools were identified to be very useful:
  - circular cutting machine for carbon and stainless steel for pipes with diameter 152–330 mm and 457–622 mm
  - pipe miller for big pipe diameters (315–1515 mm)
  - plasma arc cutting for austenitic materials in not easily accessible areas (1–75 mm material thickness)
  - guillotine pipe saw for carbon and stainless steel pipes with diameters of 101–323 mm
  - manual bandsaws up to pipe diameters of 120 mm
  - nibblers for cutting different materials as steel, aluminium, non-ferrous metal, plastics

- cable cutter for non-deformation cutting of all commonly used cables (Cu-cables up to 800 mm^2 and Al-cables up to 1000 $mm^{2)}$
- cable isolation removal machine
- heavy metal shears
- diamond cutting techniques including diamond wire saw, core drilling equipment, wall sawing machine
- rollers for heavy loads for 10 tonnes and 80 tonnes load lifting per unit
- mobile and motor driven wire ropes up to 3 tonnes traction force (lifting power)
- lifting platforms of various types.

• In some rooms of the WWER controlled areas there are no existing exits for large component parts. During the construction phase some equipment was positioned in the rooms and only after that, ceilings or walls were completed. In such cases the reinforced concrete structures must first be demolished in order to be able to remove the equipment. The enlargement of existing small hatches and doors is also useful to facilitate material transport. Thus, much better working conditions are established on the site, and this finally results in an higher dismantling rate.

• The material flow limits the dismantling rate. The sorting of different materials, the organization of suitable buffer stores on site, the creation of transport routes, the performance of packaging, weighing, and radiological measurements beside the dismantling work require an integrated organization. Thus, for a continuous material flow, the number of dismantling workers need not exceed 40–50 people for the controlled area of a WWER double unit.

• For some WWER equipment it was necessary to obtain some special dismantling equipment (see Table 7-1).

Remote techniques to be used during the Greifswald WWER reactor dismantling

The remote dismantling activities at the Greifswald site offer a good example of how various conventional cutting and handling techniques can be applied to this reactor type.

The optimised concept was defined as follows:

As two reactors are arranged in one building, the strategy is to install one wet and one dry new cutting caisson for 2 reactors. The decision not to use the existing reactor pool and the spent fuel pond connected to it was based mainly on the integrity problems of the liner. Also some missing documentation on the construction of these structures could possibly have caused problems in the licensing procedure. The limited space of the existing pools also had an impact for this decision.

Table 7-1. Adapted tools and technologies for WWER decommissioning

WWER component	Special tool, technology	Remark
V-230 Reactor protection cover	Carbide band saw in connection with jack hammer	Extremely high binding power between the outer reinforced concrete layer and the carbon steel cover
Reactor internals	two power manipulators for packaging three master slave manipulators, for handling different cutting tools	maximum lifting capacity of 0.5 tonnes CAMC (Contact Arc Metal Cutting), Plasma oxygen burner
Reactor pressure vessel	special load lifting facility	
V-230 Annular water tank	Band saw with special adapter	After filling the tank with concrete and the installation of an inner radiation protection wall
Steam generator (SG) (removal as a whole)	Collector head traverse Transport rollers with guide rails	Precondition for dismounting the SG support constructions. For horizontal cross-movement of the SGs.
Steam generator (dismantling)	Ice saw Water abrasive cutting	Performed outside the unit in a special caisson of the interim storage facility. For low contaminated SGs.
Works on concrete removal (walls, ceilings, hatches)	Diamond wire	Loose metal particles in cut area of the concrete led to serious problems for most types of diamond wires.
Room liners	Laser device	Development in progress.
Transformer	Special disassembly concept	Reuse of iron core.

The remote dismantling equipment and installation work were split into 11 tender packages:

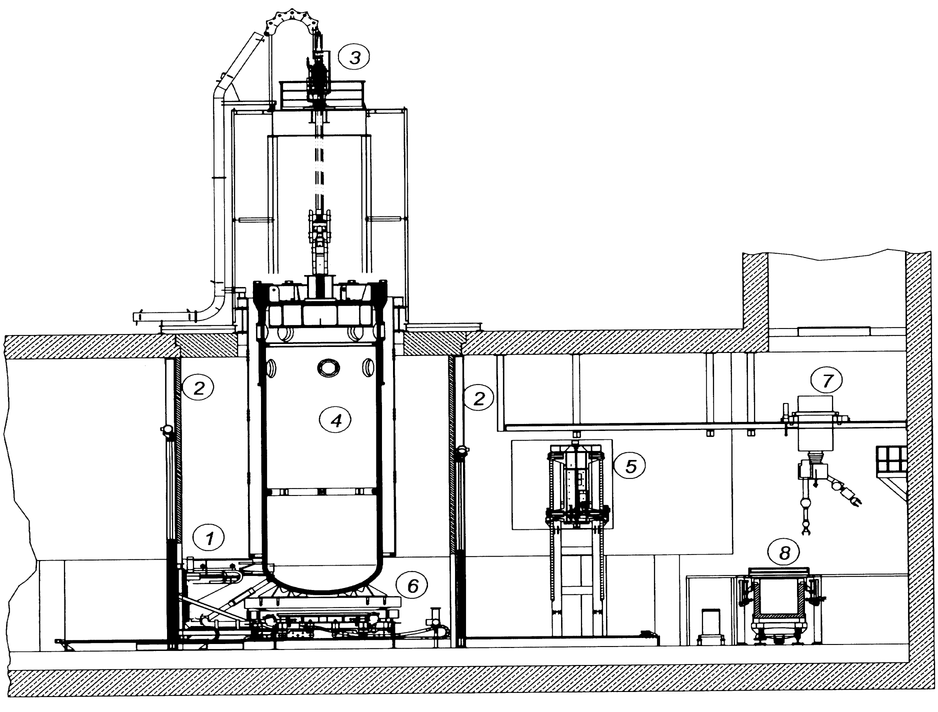
- load lifting in the hatch area of RPV,
- dry cutting station,
- wet cutting station,
- water treatment plant,
- packing station,
- ventilation systems,
- reactor milling facility,
- handling equipment,
- container manufacturing,
- dismantling core components, and
- cutting station for annular water tank.

The remote dismantling equipment will be installed into the empty steam generator box below the SG openings. Some instrumentation and auxiliary equipment will be installed in the reactor hall. The reactor internals and the pressure vessel will be removed with the reactor hall crane and lowered into the cutting boxes.

In the area of the dry cutting caisson (Fig. 7-1), the RPV, parts of the reactor cavity and the protecting tube system will be dismantled. The components will be placed on a rotating table with a carriage and cut horizontally. Each segment is transported to the post cutting area where it will be cut vertically into pieces to place into storage containers.

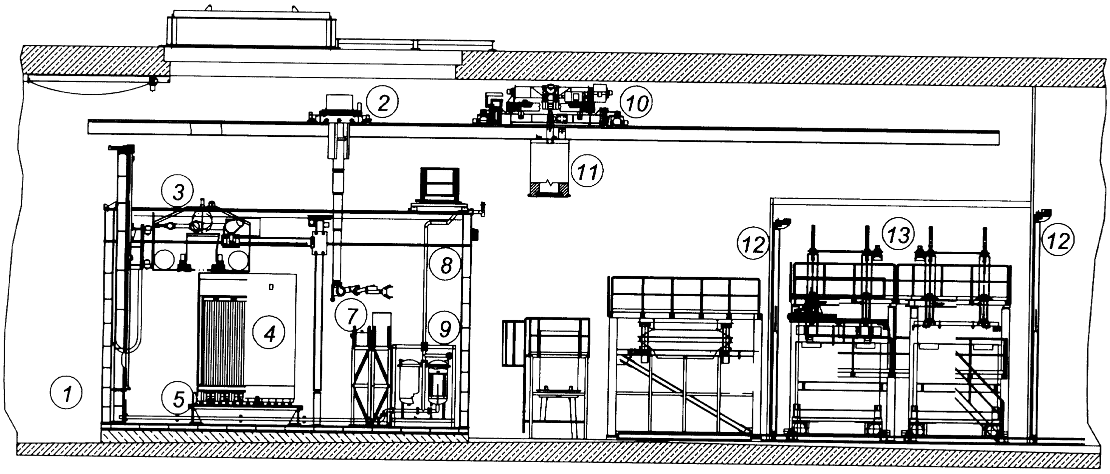
In the area of the wet cutting caisson (Fig. 7-2), the highest activated components as core basket, parts of reactor cavity, parts of protection tube unit and reactor cavity bottom will be dismantled. In this caisson, the components are also placed on a rotating table, but under water. The cutting, however, will be made horizontally and vertically. For the handling of the parts, 2 power manipulators are foreseen. The cut pieces are placed in wire baskets and transported with an overhead crane.

The packing area is common to the both cutting boxes. Above the packing area, there is an opening into the reactor hall so that the transport to and from the area can be made by the reactor hall crane.



1. cutting device pre-cutting place | 2. shielding door | 3 lifting facility
4. reactor component | 5. cutting device post-cutting place | 6. transport vehicle
7. power manipulator for packaging | 8. packing station

Fig. 7-1. Greifswald decommissioning project, cutting and packaging of activated reactor components dry cutting area.



1. working and intervention area
2. power manipulator for packaging
3. cutting device
4. reactor component
5. rotating table
6. tool support system
7. rack to insert baskets
8. cutting pond
9. water cleaning system
10. overhead crane
11. shielding cast
12. shielding door
13. packing station

Fig. 7.2. Griefswald Decommissioning Project, cutting and packaging of activated reactor components wet cutting area.

Cutting techniques (wet and dry)

Mechanical techniques; dry cutting area

For the pre-cutting station, a high power band saw will be used. With this equipment, the pressure vessel as well as the upper part of the protecting tube system and reactor cavity will be cut horizontally into sections.

Mechanical techniques; wet cutting area

For the wet cutting area there are a number of underwater techniques that will be used which include:

- high power vertical band saw
- abrasive cutting machines
- shears
- milling cutter.

Thermal techniques

The following procedures are used for underwater cutting:

- CAMC (Contact Arc Metal Cutting)
- plasma cutting

Reactor components of austenitic material will be cut by the plasma cutting device or by arc oxygen impulse cutting. Components of ferritic material will be cut by gas cutting, autogenous flame gouging and arc-oxygen impulse cutting.

These techniques, except the CAMC technique, can also be mounted in the dry cutting caisson and also on a band saw frame of the pre-and post-cutting places.

Support and transport systems

For material handling, the following existing crane facilities and load lifting devices will be used:

- bridge crane 250 tonnes (reactor hall crane),
- bridge crane 32 tonnes (reactor hall crane), and
- grabtool with reactor protection container 60 tonnes.

The following load lifting devices which have to be manufactured are foreseen:

- transport traverse for the RPV, lifting capacity 250 tonnes including five screwed bolts M 140×6 which connect the shielding and transport traverse and the RPV;
- coupling piece for the transport traverse RPV with connecting bolt (lifting capacity 250 tonnes);
- traverse for the horizontal transport of assemblies, lifting capacity 250 tonnes;
- lifting and lowering device dry cutting caisson;
- transport crane wet cutting caisson (16 tonnes); and
- traverse for the waste containers.

Ventilation and water cleaning

Additional ventilation systems

The task of the additional ventilation systems is to exhaust and filter dust, metal fumes, carbon black aerosols and fuel gas produced during the remote dismantling of the reactor components in the steam generator room. The systems will also transport the cleaned exhaust air to the central exhaust air system for controlled release to the environment. This exhaust air system generates a pressure difference between the cutting locations and the accessible areas. The accessible areas are directly supplied from the external air system (20 000m^3/h).

The additional exhaust air system generates a directed air stream from the accessible part to the cutting locations and also a direct exhaust from the cutting locations. The direct exhaust air from the cutting locations is lead through a cyclone separator (5000m^3/h) which separates the heavy particles from the air. The airborne particles will finally be separated by HEPA particle filters. There are three HEPA filters of which two are always operational and the third is in a stand by position or being regenerated. From the accessible areas (rooms), the cleaning of the exhaust air streams works generally the same way except pre-cleaning by cyclone separator is not necessary.

Water cleaning system

The cutting pond has connections for water supply and discharge. The designed minimum filling volume for the water cleaning system is about 270 m^3 with the maximum volume of 390m^3. During the cutting activities, the water can be circulated by a jet pump (80m^3/h) and cleaned with coarse and fine multiple tube filters. The water cleaning system is necessary to maintain the required water quality in the cutting pond and to reduce the activity concentration of the pond water. The floating particles (material from cutting) and coarse particles depositing on the bottom will be separated from the liquid by multiple tube filters. The filter cleaning levels are 3 mm for pre-filtration, 150 μm, 10 μm and 3 μm for fine filtration. These filters can be switched on or off depending of the dimension of the particles.

Due to radiation protection reasons, the whole water cleaning system is mounted in a steel framework and the complete module is placed into the pond. The filter exchange will be performed mainly under water.

Mock-up dismantling

Due to the novelty of the remote dismantling, it was decided at an early stage in the project to perform mock-up tests. On-site there are all of the major components from the unfinished reactors 7 and 8, i.e. also pressure vessels and all reactor internals. Thus, inactive original components are available and very realistic tests can be made. The aim of these tests can be summarised as:

- test performance of systems and procedures;
- test handling operations and maintenance actions;
- train personnel;
- estimate secondary waste production; and
- optimise equipment, tools and procedures.

By involving the regulatory body in these tests, it is expected that a smoother licensing procedure for operation in active zones will be achieved.

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ANNEXES 1–9

Annex 1

ARMENIA

1. DECOMMISSIONING INFRASTRUCTURES

Armenia has one NPP with two WWER-440, V-270 type reactors (V-230 with seismic resistance upgrading) which is located near Metsamor and managed by Armenian Nuclear Power Company (ANPP).

Unit 1 and Unit 2 were connected to the grid in 1976 and 1979 respectively. Both units were shut down on 25 February and on 18 March 1989 respectively. Before shutdown, the ANPP was generating about 27% of the total energy generated in Armenia.

Unit 2 was restarted on 5 November 1995 (and after resumption of operation it has generated more than 5 billion kWh of electricity). At the moment about 40% of the power used in Armenia is generated by ANPP. The design lifetime of unit 2 will end in 2015, but depending on the commissioning of replacement electricity generating capacities, earlier decommissioning would be possible.

Up to now, there is in Armenia no infrastructure specifically devoted to decommissioning.

1.1. Regulations and legal provisions

The Parliament of the Republic of Armenia approved on 24 August 1993 Armenia's accession to the Vienna Convention of 1963. On 14 March 1993 the Russian-Armenian intergovernmental agreement was concluded, according to which all laws and regulatory acts of the Russian Federation regarding nuclear power use are effective within the territory of Armenia.

The Article 6 of the constitution of the Republic of Armenia confirms priority of international treaties, ratified by the Armenian Parliament, over the domestic laws.

The Parliament of the Armenian Republic accepted The Law on Peaceful Uses of Atomic Energy in the first version. This law has the following objectives:

- to define government policy regarding the peaceful uses of nuclear energy and the basis and principles of nuclear legislation;
- to ensure the protection of people, property and the environment against the harmful effects of ionizing radiation;
- to ensure the fulfillment of commitments under international agreements to which Armenia is a party; and
- to prevent unauthorized export, import, transportation, use and disposal of nuclear and radioactive waste, special materials, equipment and technologies.

Supplementary to this law, the regulatory documents are the statutes of Armenian Nuclear Power Plant Company, and of Regulatory Organization, and "Statement concerning the Regulatory Authority" by Armenian Government.

The control on nuclear and radiation safety is carried out by the Armenian Nuclear Regulatory Authority (Armgosatomnadsor). This authority is responsible for regulating and supervising all uses of nuclear energy within Armenia. The utilization of atomic energy must be performed safely, to ensure the health of the public and of the personnel working at the nuclear power stations and to protect the environment. The Regulatory Authority is directly responsible to the Prime Minister and is independent of other governmental organizations and licensees.

For emergency planning and management, the Armenian Government has established the Emergency Management Administration, which has the overall internal co-ordinating responsibility and the responsibility for international co-operation and assistance in case of a nuclear or radiation accident. The responsibility for the notification of nuclear accidents is assigned to the Armenian Nuclear Regulatory Authority.

The Guide "The norms of radiation safety (NRB-96)" establishes the system of dose limits and principles of their application.

Main dose limits in total external and internal exposure mSv per calendar year are:

a) Permissible dose for category A (workforce) 50 mSv

b) Dose limit for category B — limited part of the population (members of so called critical group) — 5 mSv.

The Guide "Main sanitary rules(OSP-72/87)" defines the rules for activities with radioactive materials and other ionising radiation sources.

1.1.1. Clearance criteria

Since there is no experience of material recycling/reuse, Russian clearance criteria, regulations and standards are in force.

1.2. Decommissioning funding

At the moment neither the Government of the Republic of Armenia, nor the operating organization have the funds necessary for safe solution of NPP decommissioning.

However, in order to exclude the perpetuation of such situation in the future, according to the above-mentioned Law on the Peaceful Use of Nuclear Energy, one necessary condition for obtaining the license for construction of any installation which uses, generates, or stores radioactive materials is that a decommissioning project be available for the given installation.

1.3. Spent fuel management

Now the construction of the interim storage for spent fuel, which was performed jointly with a French company (Framatome**)** has been terminated. The first part of this facility, located on the NPP site, consists of: one transfer cask and eleven canisters, allowing to place 56 non-leaking fuel assemblies, with cooling time not less than 3 years each. The design time for safe storage is 50 years.

1.4. Radioactive waste management

For activated parts storage purposes, the central hall of the reactor building has a waste storage pit volume of 78.34m^3. It is foreseen to store HLRWs arising during the entire ANPP operation. At the moment the reactor hall waste storage pits are filled only 2.5%.

ANPP operates a mobile decontamination assembly using alkaline and acid solutions. Besides, electrochemical, steam ejection and chemical-mechanical decontamination methods are in use. After application of all these decontamination methods during Scheduled Preventive Maintenance, more than 90% of activity is removed.

1.5. Social aspects

According to the article 11 of Peaceful Use of Atomic Energy Low any juridical or private person has right to receive information related to safety of any nuclear facility being under construction, operation or decommissioning, if this information is not a state secret. This information should be given by the authorized state body free of charge.

Any juridical or private person has right to get information related to individual dose records.

2. STRATEGY

The decommissioning strategy for NPP has not been defined yet. The first activities on decommissioning plan development were done in 1991 together with VNIIAES. This work should be the basis for development of decommissioning strategy and decommissioning plan.

Annex 2

BULGARIA

1. DECOMMISSIONING INFRASTRUCTURES

The Bulgarian nuclear power generating units are situated at the site of Kozloduy. All of them are in operation. Units 1 to 4 are of WWER-440 (mod.230) type and units 5 & 6 are of WWER-1000 (mod.320) type. Each unit has a design life-time of thirty years. Units with WWER-440 reactors were commissioned in 1974, 1975, 1980 and 1982, respectively. A three-stage programme for upgrading their safety is near its successful finish. A realistic schedule for decommissioning WWER-440 units is under preparation and will be adopted at the beginning of 1999. Units with WWER-1000 reactors were commissioned in 1987 and 1991 respectively. A programme for their modernization has started and it is foreseen to finish in 2005. The NPP Kozloduy is owned and operated by the National Electrical Company.

1.1. Regulations and legal provisions

The following fundamental acts of the Bulgarian legislation are currently applicable in the matter of safe utilization of nuclear energy and in respect of nuclear material procurement, accountability, storage and transport:

(a) Act on the Use of Atomic Energy for Peaceful Purposes;

(b) Regulations for the Application of the Act on the Use of Atomic Energy for Peaceful Purposes;

(c) Ordinance No.2 on the Cases and the Procedure for Giving Notice to the Committee on the Use of Atomic Energy for Peaceful Purposes of Any Operation Modifications, Developments and Emergency Situations Relating to Nuclear Safety and Radiation Protection;

(d) Ordinance No.3 on Safety Guarantees on the Design, Construction and Operation of Nuclear Power Plants;

(e) Ordinance No.4 on Nuclear Material Accountability, Storage and Transport ;

(f) Ordinance No.5 on the Licence Issuance Procedure for Utilization of Atomic Energy;

(g) Ordinance No.6 on the Criteria and Requirements for Training, Qualifications and Capacity of Personnel Employed in Nuclear Energy Engineering;

(h) Ordinance No.7 on Collection, Temporary Storage, Reprocessing, Storage and Disposal of Radioactive Waste on the Territory of the Republic of Bulgaria;

(i) Ordinance No.8 on the Physical Protection of Nuclear Facility Sites and Nuclear Material.

The Act on the Use of Atomic Energy for Peaceful Purposes defines the responsibilities for decommissioning financial assurance.

Currently there are no regulations specific to decommissioning.

As part of the convergence process of the Bulgarian nuclear legislation to that of the European Union, documents will be elaborated to cover decommissioning of the nuclear facilities.

IAEA recommendations have been applied in all Bulgarian Feasibility Studies on decommissioning NPP Kozloduy units.

All the Conventions and Agreements in force will be considered, where applicable, in planning and decommissioning process itself.

1.1.1. Clearance criteria

In the Ordinance No. 7 issued by Committee on the Use of Atomic Energy for Peaceful Purposes in 1992, it is stated that radioactive waste, produced at NPP, is collected and remains on its site (during NPP operation). In the same ordinance, classification of the wastes is defined according to their activity. After checking the radioactive content and/or surface contamination, the waste with activity below the threshold of the content and/or surface contamination can be released unrestrictedly. There are no derived clearance criteria developed yet.

1.2. Decommissioning funding

Article 6 of the Bulgarian Act on the Use of Atomic Energy for Peaceful Purposes establishes two financing funds. Corporate Bodies and Bodies which produce radioactive waste as a result of their activities have to contribute to the Fund for Safe Storage of Radioactive Waste and Corporate Bodies which operate nuclear facilities — to the Fund for Decommissioning of Nuclear Facilities as well. The Council of Ministers defines in an Ordinance the amount of the contributions, conditions and order for granting from each fund. The funds start their functioning on 1.01.1999. Funds will be managed by the Committee of Energy through two relevant Steering Committees.

1.3. Spent fuel management

According to the initial contracts, spent fuel was shipped back to Russia. Currently it is stored on the site. There is a pool in each unit, close to the reactor, where the spent fuel is stored for at least three years after the discharge from the core. Afterwards it can be transported to the pools of the interim spent fuel on-site storage facility. Construction of an interim dry storage facility is planned.

1.4. Waste management

There are no special arrangements made for the waste produced by decommissioning activities. It is expected that operational facilities will be used for treatment of the waste from the initial decontamination during decommissioning.

Operational waste storage facilities had been erected together with the NPP — one auxiliary building per two units — and they are still operable. New facilities for treatment of low and intermediate level waste are under construction at the NPP Kozloduy site (compacting facility has already been in operation for sometime). A special location on the site has been defined for temporary storage of the treated radioactive waste (RAW). Activated metal waste will be treated together with decommissioning waste. No disposal facilities are available either for operational waste or for the future decommissioning waste, but there is a concept for establishment of a national repository for final RAW disposal. A National Institution for safe radioactive waste management is envisaged to be established.

1.5. Social and public acceptability aspects

Local municipality authorities have the right to hold referendums on major problems related to the local population.

2. STRATEGY

Until now, only preliminary studies have been developed.

3. PLANS FOR DECOMMISSIONING

In the preliminary studies for decommissioning five options were assessed:

- Direct dismantling without safe storage;
- Deferred dismantling — safe storage of part of the reactor building (so called hermetic area);
- Deferred dismantling — safe storage of reactor shafts;
- Deferred dismantling — safe storage of the reactor building;
- Deferred dismantling — safe enclosure with surveillance of the active buildings.

The characteristic quantitative data application permitted a comparison of the reference options and a well-based selection of the option, most appropriate for the given conditions. Based upon a multi-criterial analysis, the option Safe enclosure with surveillance of the active buildings had been selected as the preferred one. A tender for development of "A Technical Project for the First Stage of Decommissioning Units 1&2 at NPP Kozloduy — Final Shutdown of Units" is in progress.

4. RESPONSIBILITIES

The operator will carry out the initial steps of decommissioning till all the units on the site reach the end of their life-time (design or resource dependent).

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Annex 3

CZECH REPUBLIC

1. DECOMMISSIONING INFRASTRUCTURES

In the Czech Republic about 20% of total electricity generation is produced by the Dukovany Nuclear Power Plant.

At present four units of the Dukovany Nuclear Power Plant (WWER-440/213) are in operation. The first unit was put into operation in 1985. The last one in 1987.

There are two other units (WWER-1000/320) under the final stage of construction at the Temelín site. The first unit is expected to start operation in 2000.

Both the nuclear power plants are owned and operated by the Czech Power Company (CEZ).

The decommissioning of the NPP Dukovany is expected to start after the year 2015, since its conservative design life is assumed to be 30 years. The decommissioning of the Dukovany NPP may be delayed if life extension is implemented.

1.1. Regulations and legal provisions

In 1997 Parliament passed Act No. 18/1997 On The Peaceful Utilization of Nuclear Energy and Ionizing Radiation (the Atomic Act) and on Alterations and Amendments to Related Acts.

The Atomic Act identifies the decommissioning of a nuclear installation as a practice related to nuclear energy utilization. This means that for decommissioning practices the following shall be ensured [3-1]:

- Nuclear safety;
- Radiation protection;
- Physical protection;
- Emergency preparedness**.**

The Atomic Act defines decommissioning as activities carried out after the termination of operation of nuclear installations which are intended for their utilization for other purposes, or are aimed at exemption from this Act.

Other legal regulations related to the Atomic Act

Published decrees of the State Office for Nuclear Safety, which are relevant to decommissioning:

(1) No. 214/1997 on quality assurance during activities related to the utilization of nuclear energy and activities resulting in irradiation.

(2) No. 184/1997 on requirements for securing radiation protection.

(3) No. 144/1997 on physical protection of nuclear materials and nuclear installations.

(4) No. 146/1997 on setting out activities directly affecting nuclear safety and activities particularly important from the radiation protection viewpoint, and on requirements for qualifications and professional training. This decree also describes procedures to validate professional qualifications and grant authorizations to selected personnel and on the documentation to be approved to permit training of selected personnel.

(5) No. 143/1997 on transfer and transportation of nuclear materials and radionuclide sources.

(6) No. 219/1997 on details of the emergency preparedness of nuclear installations and sources of ionizing radiation and on requirements for an internal emergency plan and emergency regulation in transport.

(7) No. 142/1997 on approval of packages for transportation, storage or disposal of radionuclide sources and nuclear materials.

(8) Draft decree on decommissioning of nuclear installations and workplaces with ionizing radiation sources.

1.1.1. Clearance criteria

Decree No. 184/1997, issued by the State Office for Nuclear Safety, defines the clearance procedure, specifies radiation doses and provides criteria for defining materials as radioactive. It applies to premises with sources of ionising radiation, as well as discharges into water courses or the atmosphere, disposal in a scrap-yard or release into the environment in other ways [2]:

(a) Without a licence issued by the Office, materials, substances and objects containing radionuclides or being contaminated to any degree, are subjected to the following conditions:

    (i) in any calendar year, the average effective dose for a critical group of the population does not exceed 0.01 mSv and at the same time, the overall collective dose to the public does not exceed 1 man-Sv;

    (ii) on releasing solid substances and objects into the environment, the sum of the ratios of individual radionuclides mass activities and the mass activity release levels of the respective radionuclides given by law (ranging from 0.3 to 3 000 kBq/kg), is not greater than unity and the sum of the ratios of individual radionuclides surface mass activities and the surface activity release levels of the respective radionuclides given by law (ranging from 3 to 30 kBq/m^2) is not greater than unity.

(b) With a licence issued by the Office materials, substances and objects containing radionuclides or being contaminated by them to any degree, the average effective dose for the critical group of the population must not exceed 0.25 mSv in any calendar year.

1.2. Decommissioning funding

In accordance with the Atomic Act, the licensee (operator of facility) shall gradually create a financial reserve for the decommissioning of the given nuclear facility so that financial resources are available for preparation and implementation of decommissioning, at the required time and in the required amount, in line with the decommissioning strategy approved by the State Office for Nuclear Safety.

Based on the Atomic Act the Ministry of Industry and Trade established in 1997 the **R**adio**A**ctive **W**aste **R**epository **A**uthority (RAWRA) as a state organization, which will monitor the financial reserves of licensees for decommissioning of their installations. This agency also verifies the cost estimate for decommissioning.

1.3. Spent fuel management

Only the part of spent fuel management, that starts from the removal of the spent fuel from the reactor, its cooling in a storage pond up to its preparation for the transport to the storage facility, is classified as a decommissioning activity. The remaining steps of the fuel management are solved separately and are not linked with the decommissioning of the nuclear power plant.

The strategy "away from reactor and dry type storage of spent fuel" has been adopted for both the Czech sites. It is assumed that the spent fuel, after discharge from the reactor, will be cooled in storage ponds for a few years. After this period it will be loaded into storage casks.

A spent fuel storage facility has been built at the Dukovany site and commissioned in November 1995. The planned lifetime of the facility is 40–50 years. As the storage capacity is limited the construction of additional capacity is being prepared.

1.4. Waste management

Decree No. 184/1997, issued by the State Office for Nuclear Safety, defines the requirements for radioactive waste management. The limits and conditions for the safe management of radioactive waste during the individual stages of the decommissioning of a nuclear facility shall be subject to approval by the State Office for Nuclear Safety.

All the activities associated with radioactive waste disposal shall be administered by RAWRA. The waste producer shall provide RAWRA with data on short-term and long-term generation of the radioactive waste. With regard to these estimates, RAWRA is responsible for planning of disposal facilities. It is assumed that a suitable final disposal facility, for various amounts and categories of generated waste, will be available and temporary storage is not considered an alternative strategy to the disposal of decommissioning wastes.

A strategy of waste minimization has been adopted by CEZ (e.g. reuse and recycling of material will be utilized during the decommissioning process where practical).

In the Czech Republic the categories of radioactive wastes, e.g., very low level waste (VLLW), LLW, ILW, etc., are not defined by any law (decree). However, in practice handling with radioactive waste, is classified with respect to its basic physical properties, i.e., as solid, liquid, and gaseous, and with regard to its origin and treatment as low-level, medium-level, and high-level waste. Low-level waste represents most of the waste produced in the course of the NPP operation. Spent resins from primary coolant decontamination could be classified as intermediate-level waste. Spent fuel could be classified as high-level waste.

1.4.1. Waste acceptance criteria

The waste produced during the operation of the current nuclear power plant is being disposed of in the shallow land type repository at Dukovany, situated within the site of the power plant. This repository is calculated to cover the needs of the Dukovany NPP, and also of the Temelin NPP in future.

The repository at Dukovany is dedicated to the waste from the operation of the nuclear power plants and serves for the disposal of LLW and ILW.

The safety documentation for the repository, approved by the State Office for Nuclear Safety, comprises data concerning the maximum inventory of wastes with the exclusion of certain specified radionuclides (e.g. transuranium elements).

The Office also authorizes the so-called Limits and Conditions for Repository Operation (they constitute a part of the safety report), in which the maximum surface contamination of drums, dose rates on the surface of drums, limit volume activities of the radionuclide under surveillance and their total contents in the pits, limit values of the radionuclide release from the final form of waste, etc., are defined. Pyrophoric, explosive and toxic substances as well as free liquids are excluded from the disposal in the repositories.

There are no special acceptance criteria for radioactive waste from decommissioning.

1.5. Social-political aspects

In the process of selection between direct dismantling and deferred dismantling the licensee should take these factors into account:

- Impact on the environment;
- Impact on employment;
- Impact on public opinion.

As experiences from other parts of the world show, public opinion vis a vis nuclear power installations is a very important factor in the selection of government and other related bodies' policies. For this reason, the strategy of conversion of the facility into a repository has not been considered by CEZ.

According to the Atomic Act the applicant for a licence for the individual stages of decommissioning of a nuclear plant must prepare the relevant documentation, which should include a description of modifications to the local environment due to nuclear installation operation.

An environmental impact assessment is required in order to obtain a decommissioning licence. Based on the Act No. 244/1992 the Ministry of Environmental is responsible for environmental impact assessment.

2. STRATEGY

In the Czech Republic the Atomic Act stipulates, that for the decommissioning of a nuclear installation a licence is required. The Office shall issue a licence either separately for each stage of the decommissioning process or for the overall process.

There are no legislative and regulatory requirements, which would dictate the strategy to be followed.

Regulatory arrangements allow the licensee to propose a strategy — deferred dismantling, direct dismantling, etc.

The licensee shall ensure adequate nuclear safety and radiation protection.

3. PLANS FOR DECOMMISSIONING

A conceptual study on the decommissioning of the Dukovany Nuclear Power Plant was completed in June 1997. In this study three basic options were assessed [3-3]:

- Deferred dismantling — safe enclosure of the active buildings for 50 years;
- Deferred dismantling — safe enclosure of reactor shafts for 50 years;
- Direct dismantling without safe enclosure.

For every option the following parameters were evaluated:

(1) Total working capacity (man-days)

(2) Required number of personnel

(3) Amount of radioactive waste for final disposal

(4) Total collective dose equivalent (man-Sv)

(5) Time schedule

(6) Impact on the environment

(7) Amount of non-radioactive waste for disposal

(8) Amount of non-radioactive waste generated for reuse (concrete and metal)

(9) Cost.

All the proposed options assumed that the plant would ultimately be decommissioned (released) without any restriction from the regulatory body and from the radiological protection point of view, no further surveillance, inspection or test would be necessary.

The preferred strategy for decommissioning is one of deferred dismantling — safe enclosure, which consists of the following stages:

- Reactor shutdown (dispatch of spent fuel from the storage pond);
- Preparation of the plant for safe enclosure period;
- Operation of safe enclosure (care and maintenance);
- Dismantling.

Reasons for the selection of this strategy include:

- Lower dose rates in the plant;
- Lower radiation exposure of personnel;
- Less radioactive waste for treatment;
- Less radioactive waste for final disposal;
- Reduction of the consequence of a possible accident occurring during reactor dismantling;
- Sufficient financial reserve to shift to other possible strategies, if necessary.

In 1998 the State Office for Nuclear Safety approved the strategy of safe enclosure of the active buildings for 50 years.

4. RESPONSIBILITIES

The basic requirements for the planning and management of decommissioning are stipulated in the Atomic Act. Under the terms of the Atomic Act [3-1]:

- An applicant for a licence for individual activities (i.e. for the siting, the construction of the nuclear power plant, the first loading of the nuclear fuel into the reactor, the operation of the nuclear power plant, or individual stages of decommissioning) shall prepare relevant documentation dealing with decommissioning.
- Administration and supervision of nuclear energy and ionizing radiation uses and in the field of radiation protection are performed by the State Office for Nuclear Safety, which shall issue licences for discharge of radionuclides into the environment, radioactive waste management etc. The Office approves the proposed decommissioning method, which shall be submitted by the utilities.
- The owners of radioactive waste (generators) shall bear all the costs associated with its management, from generation to disposal, including monitoring of radioactive waste repositories after their closure, and including the necessary research and development activities.
- The generators of radioactive waste establish financial provisions to cover expenses for disposal of radioactive waste, which have been arising or will arise. These financial resources shall be accumulated in “the Nuclear Account” in the form of levies. The “Nuclear Account” is managed by the Ministry of Finance through the Czech National Bank.

The state guarantees the safe disposal of all radioactive waste, including monitoring and supervision of repositories after their closure.

REFERENCES TO ANNEX 3

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Annex 4

FINLAND

1. DECOMMISSIONING INFRASTRUCTURES

There are three major nuclear sites in Finland: two nuclear power plants, each with two reactors, and a research reactor. Two WWER-440 reactors are located at Loviisa and are managed by Imatran Voima Oy (IVO) Company. Unit 1 started operation in 1977, Unit 2 in 1981. According to the present plans, the nuclear power plants will be operated until about 2020 or later. The research reactor may be shutdown earlier. As highlighted below and described in detail in the quoted literature, the most important decommissioning infrastructures are in place in Finland or at the planning stage.

1.1. Regulations and legal provisions

The most important legislation regulating nuclear activities in Finland is the Nuclear Energy Act and Decree (renewed in 1988) and the Radiation Protection Act and Decree (renewed in 1992). The licensees are responsible for the implementation of decommissioning. The regulatory authorities are the Ministry of Trade and Industry which is responsible for decommissioning policy and the Radiation and Nuclear Safety Authority (STUK) which covers the regulation of safety [4-1].

The Nuclear Energy Act and Decree define the responsibilities and the principles for decommissioning financial provisions. The licensing procedures for decommissioning are not yet defined in detail. The safety related regulations are issued either by the Government (general rules) or by STUK (detailed rules). Currently there are no regulations specific to decommissioning. However, the regulations concerning clearance from regulatory control of nuclear waste are also applicable to decommissioning [2] (see Section 1.1.1.).

1.1.1. Clearance criteria

Guide YVL 8.2 [4-2] issued by the STUK specifies radiation dose and activity constraints, defines the clearance procedure and provides criteria for determination of the waste activities. The following dose criteria apply:

- An effective dose of 10 μSv in a year to the most exposed individuals (members of the so called critical group), or
- a collective dose commitment of 1 manSv per year of practice.

The following activity constraints are applicable to unrestricted clearance:

- The total activity concentration, averaged over a maximum amount of 1000 kg of waste, shall not exceed 1 kBq/kg of beta or gamma activity or 100 Bq/kg of alpha activity. In addition, no single item or waste package weighing less than 100 kg may contain more than 100 kBq of beta and gamma activity or 10 kBq of alpha activity.

- The total surface contamination of non-fixed radioactive substances, averaged over a maximum area of 0.1 m^2 for accessible surfaces, shall not exceed 4 kBq/m^2 of beta and gamma activity or 400 Bq/m^2 of alpha activity.

For restricted clearance, activity constraints based on a case-by-case approval by the STUK are applied which, however, shall not exceed those included in Section 10, points 1 and 2 of the Nuclear Energy Decree, namely:

- The average activity concentration in the waste is less than 10 kBq/kg.
- The total activity of exempted waste received by a transferee in one year is less than 1 GBq and the alpha activity less than 10 MBq.

1.2. Decommissioning funding

In Finland, holders of licences for nuclear technology activities are responsible for the management of all types of waste (spent fuel, operational low and intermediate level waste and decommissioning waste), as well as all management measures (R&D, handling, conditioning, storage, transportation, decommissioning and final disposal).

If the costs of the waste management are estimated to exceed FIM 200 000 (about US $40 000), assets must be collected in advance in a separate fund. The Ministry of Trade and Industry determines yearly the amount of the liability and the fund contribution. The liability and the fund contribution are defined for each nuclear waste producer Non-recurrent costs, such as decommissioning costs and investments for disposal facilities, can be distributed over no more than the first 25 operating years of the power plants. Therefore the fund assets will not match the liability during the early years of operation. To compensate for this, the waste producer shall furnish securities to cover the outstanding liabilities The assets, plus the securities, must cover the total liability.

The fund invests the assets through loans and the interest is credited to the account of the waste producer. The waste producer has the right to borrow up to 75% of his own contribution backed by normal commercial guarantees. The State may borrow the remaining 25%. If the rights are not exercised, the fund lends on the open market. According to this system, which is continuously updated, all waste management measures and activities are covered and paid for by the originator of the liability [4-3].

1.3. Spent fuel management

In the past, spent fuel from the Loviisa NPP was transported to the Russian Federation with no return of reprocessing wastes. However, plans for spent fuel management have been revised based on the amendment to the Nuclear Energy Act prohibiting the export of spent fuel beyond 1996 [4-4].

As a consequence, a new company, Posiva Ltd, was founded by the two utilities TVO and IVO. Posiva will take care of the planning and future implementation of the disposal of spent fuel from the Finnish NPPs. The Loviisa NPP site is one of the four sites subject to siting investigations for the spent fuel repository. TVO and IVO will retain the responsibility for the spent fuel and other nuclear waste in accordance with the Nuclear Energy Act. The mission of Posiva Ltd neither includes the work for decommissioning nor management or disposal of low and intermediate level wastes.

1.4. Waste management

Intermediate and low level operational wastes are conditioned and stored at the Loviisa site. Spent ion-exchange resins and evaporation concentrates are stored temporarily without conditioning in a tank storage facility. IVO has plans for a cementation-based solidification plant.

Operational waste produced at the Loviisa NPP will be disposed of on-site in a repository. The construction of the repository started in 1993 and the operating license was granted in 1998. The repository, located at a depth of about 110 m, comprises a transport tunnel, two disposal tunnels for LLW, a disposal cavern for solidified ILW, and two shafts for stairways and ventilation [4-5].

It is foreseen that decommissioning waste will be disposed of at a repository co-located with the one for operational waste. According to the current decommissioning and disposal plans for the Loviisa NPP, the reactor vessels (each about 215 tons) would be transferred intact into that repository. The reactor vessel internals would be emplaced inside the vessels. The vessels are filled with concrete and surrounded with appropriate engineered barriers. Steam generators and pressurizers would also be disposed of as intact. Other decommissioning waste would be cut into smaller pieces and emplaced in concrete containers [4-4–4-6].

1.5. Social aspects

Consideration of socio-political issues related nuclear waste management is based on the Environmental Impact Assessment Act and the Nuclear Energy Act.

The Environmental Impact Assessment Act requires assessment of direct and indirect impacts of major nuclear waste management activities on:

- health, living conditions and satisfaction of people;
- soil, water, air, climate, flora, fauna and interaction between them as well as biodiversity,
- urban structure, buildings, landscape, townscape, cultural heritage, and
- use of natural resources.

According to the Nuclear Energy Act, the municipalities also have an absolute veto-right in locating nuclear waste management and disposal facility in their area. Both the Environmental Impact Assessment Act and the Nuclear Energy Act call for hearings of citizens in local and neighbouring municipalities and authorities and other groups whose interests may be affected by the project [4-7].

2. STRATEGY

According to the policy decision made by the Finnish government in 1983, the licensees are required to update their decommissioning plans every five years. The latest update was published at the end of 1998. There are no prescriptive policy decisions on the strategy and scheduling of decommissioning, which are left to the licensees.

3. DECOMMISSIONING PLANS

The decommissioning plans aim at ensuring that decommissioning can be appropriately performed when needed and that the estimates for decommissioning costs are realistic. The decommissioning plans, prepared by the utilities, are quite detailed and they are based on different strategies. The decommissioning plan for the Loviisa NPP is based on approximately

30 years of operation and immediate dismantling after a preparatory stage of two years after shutdown. This strategy is justified by the likely future use of the site. Only structures and components exceeding the clearance limits would be dismantled. On-site disposal of decommissioning waste is envisaged [4-1, 4-6]. The first version of the decommissioning plan was published in 1987 and is available in English [4-8]. A subsequent version, focussed on workforce dose reduction, was made available in 1994 [4-9]. The latest version, published at the end of 1998, contains inter alia updated activity inventory calculations [4-10] and a detailed safety assessment for the disposal of decommissioning waste [4-11].

4. RESPONSIBILITIES

According to the nuclear energy legislation, the licensee has full responsibility for the financing and implementation, with related R&D, for decommissioning. The licensee’s responsibility terminates after completion of all decommissioning and waste disposal activities. In the event that the licensee is unable of implementing decommissioning, the state has backup responsibility. Utilities are also required to arrange financing to cover decommissioning and waste management in the event of premature shutdown of the plant.

The Government is the licensing authority for nuclear installations. The Ministry of Trade and Industry approves the decommissioning plans and administrates the waste management fund. STUK is responsible for the review of decommissioning plans and for the control of the safety of decommissioning activities.

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Annex 5

GERMANY

1. DECOMMISSIONING INFRASTRUCTURE

In Germany, fourteen NPPs, including six Soviet WWER type reactors of the former German Democratic Republic have been finally shut down:

NPP	Type	MW_e	Decommissioning stage (in accordance with IAEA definition)	Operation
HDR Großwelzheim (Kahl)	BWR	25	undergoing to stage 3	1969–1971
KKN Niederaichbach	HWR	106	3	1972–1974
KRB-A Gundremmingen	BWR	250	undergoing to stage 3	1966–1977
KWL Lingen	BWR	52	1	1968–1977
MZFR Karlsruhe	HWR	58	undergoing to stage 3	1965–1984
VAK Kahl	BWR	16	undergoing to stage 3	1960–1985
THTR 300 Uentrop	HTR	307	1	1983–1988
AVR Jülich	HTR	15	undergoing to stage 3	1966–1988
KKR Rheinsberg	WWER	70	undergoing to stage 3	1966–1990
KGR Greifswald 1	WWER	440	undergoing to stage 3	1973–1990
KGR Greifswald 2	WWER	440	undergoing to stage 3	1974–1990
KGR Greifswald 3	WWER	440	undergoing to stage 3	1978–1990
KGR Greifswald 4	WWER	440	undergoing to stage 3	1979–1990
KGR Greifswald 5	WWER	440	undergoing to stage 3	1988–1990
KNK II Karlsruhe	FBR	20	undergoing to stage 2	1977–1990
KWW Würgassen	PWR	670	undergoing to stage 3	1971–1994

The following sites in deep geological formations will be or are available for final disposal of radioactive waste in Germany:

- for non-heat generating waste a former iron-ore mine (Konrad);
- for heat-generating waste the salt dome at Gorleben.

The salt dome disposal site at Morsleben (ERAM) was closed in 1998.

1.1 Regulations and legal provisions

In Germany, the legal basis for the use of nuclear energy, radiological protection and related activities is the Atomic Energy Act (AtG) 1959 (last amended 1998) and the Radiation Protection Ordinance (StrlSchV), issued in a new version in 1997. There is no specific law in Germany covering decommissioning of nuclear facilities. Instead, the legal provisions are regulated in paragraph 7 (3) of the German Atomic Law (AtG), which requires a licence for the shutdown of a nuclear facility. The AtG dictates that the operating utility is responsible for licence application, and implementation of decommissioning of its own nuclear facility including conditioning, packaging and interim storage of the nuclear fuel, operational and decommissioning waste.

The decommissioning licence can be achieved within the framework of the existing regulations although only few of them refer specifically to decommissioning. The development of more specific regulations will proceed according to the practical needs and is supported by the experience already gained from a number of successfully completed decommissioning projects.

1.1.1. Clearance criteria

Considerable experience is available on clearance and reuse of materials in Germany. In the past, decisions has been made on a case-by-case basis. The available experiences and the relevant recommendations on clearance criteria and reuse of radioactive residues published by the German Radiation Protection Commission [5-1–5-5] have made it possible to develop a general approach to this matter. This general approach is based on the 10µSv concept of the IAEA [5-6] and mass specific clearance levels for about 800 radionuclides published by the Federal Office for Radiation Protection (BfS) [5-7]. An atomic energy law ordinance on residual materials and waste (AtRAV) is being prepared at ministerial level. Drafts currently under discussion contain split sets of standards, one set for unrestricted release, and one set for restricted release.

1.2. Decommissioning funding

Cost assessments for decommissioning were first developed in Germany in the early seventies. The estimated financial sum is accumulated by the power supply companies over a 19 years operational period. The expected costs were determined in decommissioning plans for a reference BWR (Brunsbuettel) and PWR (Biblis-A) respectively. The plans are reviewed regularly and adapted if necessary. The so called Stillko-2 calculation method used for this purpose has an approved status and can be also adapted to other plant types than the two reference NPPs. For these reference plans, costs of almost 590 and 630 million DM are estimated for the PWR, and 700 and 697 million DM for the BWR, in case of immediate or deferred dismantling respectively. These sums do not include final disposal charges.

The dismantling cost assessments performed on the WWER reactors at the Greifswald and Rheinsberg sites with a WWER modified Stillko-2 procedure [5-8] resulted in costs of approx. 400 million DM per unit, excluding conventional demolition. This figure is comparable to the values given above, taking into account cost savings due to serial effect and the availability of the Interim Storage Facility North (ZLN) on site.

1.3 Spent fuel management

Up to September 1998 the German NPP operators were sending their fuel to Cogema or BNFL reprocessing plants. All wastes arising from these reprocessing activities have to be taken back to Germany. Vitrified high level waste has to be stored in storage flasks at the central interim storage facility Gorleben for about 30 years. Finally it will be disposed of in the Gorleben salt-dome (operation not expected before 2010).

For the WWER fuel from Greifswald and Rheinsberg [5-9], a decision was taken for the direct interim storage option for at least 40 years at the newly erected Interim Storage Facility North (ZLN) on site (ZLN hall No 8). For the dry storage of the hexagonal shaped WWER fuel elements, CASTOR casks had to be internally adapted. This CASTOR cask type 440/84 can accommodate 84 of these fuel elements and has a weight of 116 t after loading. Presently at the Greifswald site there are 4547 fuel elements stored in the wet fuel storage (ZAB). Furthermore there are 252 fuel elements in CASTOR casks and three defect elements in one cooling pond. Because of limited licence validity date, regulated in the §57a of Atomic Law, all fuel elements have to be loaded into CASTOR casks for interim storage. Additional to the fuel elements, various adsorbers, shielding protective cassets and in-core measurement parts must be removed from the reactor units and stored in CASTOR casks. In Rheinsberg there are 220 fuel elements loaded in 3 CASTOR casks and 26 modified fuel elements which will be loaded into internally adapted CASTOR casks.

1.4 Waste management

The waste management strategy for the WWER Greifswald site will be shortly described in the following. The already produced operational waste and wastes generated during post shutdown operation was disposed of in the final storage in Morsleben (ERAM) until it was closed in 1998. Up to the closure date, 5875 m^3 of waste could be disposed of there. Included here are activated and highly contaminated solid waste, low-active and medium-active resins as well as dried evaporator concentrates, sludges and sludge-resins mixtures, solid radioactive mixed wastes from storage bunkers as well as from temporary buffer storages. The major part of this operational waste will be treated by common conditioning techniques.

The large amounts of decommissioning waste coming from the running dismantling work in the five reactors requires additional interim storage capacity, in order to be independent from the licensing of further final disposal sites in Germany. ZLN facility can accumulate these waste streams. It contains also 5 caissons for waste treatment and conditioning, e.g. cutting, volume reduction, high pressure compaction, concentrating of liquid waste, drying and packaging. In this way, the logistic security for continuous dismantling is guaranteed. After dismantling, the plant parts will be sorted and packed in appropriate packages for further treatment and storage. Immediately after the package is ready, it will be provided with a routing card, containing all important data, for registration and tracking of the packages over the whole treatment process until the corresponding disposal goal has been achieved. This system is supported by developed software routines.The main stations of the material flow consist of radiological instrumentation for material clearance; hot workshop with machine tools; baths for chemical decontamination; facilities for decontamination by abrasive blasting, high pressure water jetting as well as electropolishing; the 5 ZLN treatment caissons; and various buffer stores in dedicated areas inside units 5 and 6, and elsewhere on site.

1.5 Social aspects

When the ultimate decision for decommissioning a NPP has been taken, the employment of the former operational personnel becomes extremely important together with safety and technical aspects. This is especially true in Eastern European plants, due to large operational crews. Decommissioning experience nowadays shows a trend for immediate dismantling up to IAEA stage 2 or 3 without any intermediate safe enclosure time. One of the reasons for this strategy is the possibility to involve the former operational crew in order to achieve a cost effective decommissioning project. With the increasing number of decommissioning projects world wide, conversion of former operational staff to specialized decommissioners should be given attention. First experiences in this process are on the way in the Greifswald and Würgassen decommissioning projects.

2. STRATEGY

In Germany two decommissioning variants are generally possible, i.e.:

- complete dismantling after a safe enclosure time

- immediate total dismantling.

Each of these variants is proceeded by the so called post-operation phase, in which normally under operation licence the fuel and the operational waste are removed. The actual decommissioning work requires a decommissioning licence. The application for this licence and the preparation of all necessary licensing documents should be performed before and partly during the post-operational phase. In this way the licence can be issued before the post-operation phase has been completed, so minimizing any time delays.

Which decommissioning variant is chosen depends on project planning and the prevailing boundary conditions, e.g. single or multiple unit site, availability of final disposal facilities, continued employment of the operational personnel, possibilities for reusing the site, cash-flow, radioactive waste produced, dose commitment etc. Finally the variant presenting lower cost and project risks will be selected.

3. DECOMMISSIONING PLANS

In the safety criteria for nuclear power plants of the Ministry of the Interior [5-10], the following criterion for the preparation of decommissioning has been determined:

criterion 2.10: decommissioning and removal of nuclear power plants

“Nuclear Power Plants have to be constructed in such a way that they can be decommissioned in compliance with radiation protection regulations. A concept for decommissioning after shutdown in compliance with the radiation protection regulations must be available”.

Item 2.14 of the safety requirements for nuclear fuel utilities is similar. Thus, an analysis of the decommissioning is required long before the termination of operation. In the operational licence for nuclear power plants, a periodical revision of the decommissioning concept is prescribed. The technical documentation of the plant, the systems, components, buildings and materials as well as data relevant for radiation protection (dose rate, contamination inventory) and implications of incidents relevant to decommissioning are therefore of great importance. All maintenance measures, also mentioned under criterion 2.14 of the safety critera of the Federal Ministry of the Interior, can be used for the decommissioning planning. Recently a guidance report for decommissioning has been issued [5-11].

4. RESPONSIBILITIES

The Federal Ministry of the Environment, Nature and Nuclear Safety (BMU) is responsible for nuclear safety and radiological protection. The BMU issues acts and ordinances as well as rules, guidelines and criteria, supervises the States (Länder), which act on behalf of the Federal Government as the licensing authorities for construction, operation and decommissioning. The licensee of the nuclear facility is responsible for decommissioning. The BMU can give directives to the states to ensure a legally consistent regulatory framework. The BMU receives advice on all issues concerning nuclear safety and radiation protection from the Reactor Safety Commission (RSK) and from the Commission of Radiological Protection (SSK). The licensing authorities consult expert organizations for assessment of the Safety Analysis Report and independent evaluations of all safety related issues arising during construction, operation and decommissioning.

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Annex 6

HUNGARY

1. DECOMMISSIONING INFRASTRUCTURE

At present there is one nuclear power plant in Hungary (with four VVER-440 Model V213 reactors), located at Paks, which is operated by Hungarian Electricity Works Co. The plant′s four units have a capacity of 1840 MW(e), generating about 40% of the country′s domestic electricity production. The other main institution of nuclear activity is the research reactor at Budapest (operating at a maximal thermal power of 10 MW). No nuclear facility in Hungary has reached the stage of decommissioning. Decommissioning of the first NPP unit is expected to start around 2012. Timing of termination of the reactor operation could be decisively influenced by the need of extending the plant life.

1.1. Regulations and provisions

The legal regime applicable to nuclear activities in Hungary was previously set down in the Atomic Energy Act of 1980. On 10 December 1996 the Hungarian Parliament adopted the new Atomic Energy Act, which replaced the 1980 Act. As with the 1980 Act, the different Ministries are responsible for implementing the Act in their respective fields of jurisdiction by means of legal regulations. Until new regulations are brought into effect the existing regulations continue to apply, with some exceptions where new regulations came into force on the same day as the Act itself.

The powers to implement the Government′s responsibility under the Act for the control and supervision of the safe utilization of nuclear energy are vested in the Hungarian Atomic Energy Commission (HAEC), and the Hungarian Atomic Energy Authority (HAEA), as well as Ministers concerned.

The responsibilities of the HAEA and HAEC have been specified in Government Decree No 87/1997. (V. 28). The Act leaves many of the details of the regulatory scheme to be governed by separate regulations. Currently no regulations specific to decommissioning exist.

1.1.1. Clearance criteria

Decree No.23/199.(VII.18) issued by the Ministry for Public Welfare defines the exemption levels in terms of activity and activity concentration. It also defines clearance procedure.

The dose criteria set down are as follows:

Practice may be exempted where appropriate without further consideration, in accordance with the basic criteria, provided that the following criteria are met in all feasible circumstances:

- the effective dose expected to be incurred by any member of the public due to exempted practice is of the order of 10 μSv or less in a year and

- either the collective effective dose committed during one year of performance of the practice is no more than about 1 man Sv or an assessment of the optimization of protection shows that exemption is the optimum option.

Clearance levels are defined as values, established by national competent authorities, and expressed in terms of activity concentrations and/or total activity, at or below which radioactive substances or materials may be released from the requirements of the Decree.

- In cases of mixtures of more than one nuclide, the requirement for reporting may be waived if the sum of the ratios for each nuclide of the total amount present divided by the values listed in the Decree is less than or equal to 1. The summation rule also applies to activity concentrations where the various nuclides concerned are contained in the same matrix.
  - The values laid down in the Decree apply to the total inventory of radioactive substances held by a person or undertaking as part of a specific practice at any point in time.
  - For radionuclides not listed in the Decree, the competent authority shall assign appropriate values for the quantities and concentrations of activity per unit mass.

For conditioned clearance, activity constraints will be determined on a case-by-case approval by the Health Authority.

1.2. Decommissioning funding

No reserve fund for the future costs of nuclear waste management and decommissioning of nuclear facilities was available in Hungary till 1998. However the new Act provided for the Government to take steps aimed at a financing system to implement a coherent and comprehensive solution for — among other things — decommissioning of nuclear facilities.

The licensee (or, in the case of budget funded organizations, the budget) is liable to cover the costs of the final disposal of radioactive waste, as well as the interim storage and final disposal of spent fuel. For this purpose the Central Nuclear Financial Fund was established (as of 1 January 1998). The Fund is managed by the HAEA and is a separate State fund pursuant to Act XXXVIII of 1992 on Public Finance, exclusively earmarked for financing the construction and operation of facilities for the final disposal of radioactive waste, as well as for the interim storage and final disposal of spent fuel, and the decommissioning of nuclear facilities. Payments into the Fund by licensees of nuclear facilities will be determined in such a way that the Fund fully covers all the costs arising from the waste management, both from the operation of the facility and its decommissioning. In the case of a nuclear power plant, payments made by the licensees to the Fund should be taken into account when pricing electricity.

1.3. Spent fuel management

Fuel for Hungarian NPP just as for all other East European WWERs has in the past been supplied by the ex-Soviet Union. As part of Hungary’s agreement on fuel supply the Soviet Union was bound to take back and dispose of all spent fuel. In the past, Hungary was glad to have the problem of disposal dealt with in this way. At present, although contracts for fuel supply and return are still open issues, Hungary is less certain that this "comfortable arrangement" will last any longer in the future.

The likely interruption of the former spent fuel disposal route forced Hungary to take steps. In 1993 decision was made to construct at Paks a Modular Vault Dry Store (MVDS) based on the GEC Alsthom design. The basic function of the facility is to store the spent fuel assemblies discharged from the reactors of Paks NPP for an interim period of 50 years. In February, 1997 the operational licence of the first modules of the MVDS was issued by

HAEA. In 1997 the loading of the first module of the MVDS started. By the end of December 1997 450 spent fuel assemblies had been loaded.

The capacity of Phase 1 construction still licensed is 4950 fuel assemblies. This number equates to the amount of the spent fuel generated during 10 years of operations of the NPP. The storage capacity can eventually be increased up to 14 850 storage positions by building further vaults. This number is sufficient for storing all of the spent fuel generated by the power plant during 30-year of operation. The interim storage facility will in the future be extended as need arises.

1.4. Waste management

Low and intermediate level operational radwastes are collected and stored at Paks site. Compactible solid LLW are compacted in 200-litre steel drums. Activated metal components (interior parts of the reactor etc.) are placed in storage holes in reactor halls. Activated metal wastes will be treated together with decommissioning wastes. Waters of high salinity are treated by evaporation, the steam produced is condensed and cleaned through ion-exchange resins. Evaporator concentrates are transported by pipeline to the liquid waste tanks. Conditioning of the evaporation concentrate has not started yet. Following the boric acid recovery and Caesium filtration, the residue is planned to be cemented. The spent resins will be subject to cement solidification in the future. In Hungary the disposal capacity currently available at the near surface disposal facility (Püspökszilágy) ensures disposal of institutional wastes for many decades, but for low and intermediate level waste (LILW) of NPP origin a new facility should be built. Disposal of operational and decommissioning LLW and short-lived ILW is planned at the same site and in the same depth. Long-lived ILW is anticipated to be disposed of together with the HLW. Since 1992 a systematic work has been going with the aim to select a site to dispose of LILW of Paks NPP origin. Currently the preferred option is a tunnel-type repository in a granitic site. The new repository is due to be operational around 2003.

1.5. Social aspects

According to the Act on Atomic Energy a licence for the application of atomic energy will only be granted if the safe interim storage or final disposal of radioactive waste generated can be assured in accordance with the most recent scientific knowledge. Fundamental scientific, technical, knowledge and other information — including risks — related to the application of atomic energy shall be disseminated to the public through public news services and through public education. In order to regularly provide information to the population of communities in the vicinity of nuclear facilities; the licensee of a nuclear power plant as well as that of a radioactive waste disposal facility shall promote the establishment of a public control and information association and can grant assistance to its activities. Considerations of social-political issues concerning radioactive waste management are given in the Act on Environmental Protection (1995. LIII.). The Act requires assessment of impacts of — among others — major waste management activities in the form of Environmental Impact Assessment. (EIA). The EIA calls for hearings of citizens in local and neighbouring municipalities and other interested groups.

2. STRATEGY

The nuclear safety licensing required for — among other things — decommissioning of nuclear facilities falls into the scope of HAEA. Currently there are neither legislative and regulatory requirements which would dictate the decommissioning strategy to be followed, nor prescriptive policy decision and scheduling. Although licensees have so far not been legally required to prepare decommissioning plans, Paks NPP has already compiled the first version of this plan, which later has been updated.

3. DECOMMISSIONING PLAN

The very first preliminary decommissioning study was completed only at the end of 1993 with the close collaboration of the Slovakian DECOM company. Later (1997) it was revised.

In principle the following options seem to be possible for Paks NPP:

(a) Complete dismantling of the plant

(b) Decommissioning of the twin units by safe enclosure including:

  (i) common isolation of the twin unit together with its own facilities either with- or without decreasing the height of the building;

  (ii) independent isolation of both units with their own facilities, either with decreasing the height of the building or leaving it as it is;

  (iii) isolated closing of the two reactor pits with the reactors in them;

(c) Store with surveillance the twin units in its original state.

By making proposals for the Paks NPP 1st and 2nd Unit decommissioning, the main considerations were:

- to get the optimal version from the options presented,
- the versions compared should differ to such extent that one could estimate the influence of the proceedings and terms on decommissioning characteristics, costs, the technology, safety, meeting the requirement of waste disposal areas,
- as there is a big difference between the options to be chosen in terms of: implementation time, volume of radioactive waste generated, and technical tools and cost, to select the most acceptable solution needs a comparative evaluation taking into account all the conditions (technical, financial, safety).

In case of the 3rd version of decommissioning of Paks NPP, three periods were taken into account for safe enclosure period: 70-year was the target value, but 50 and 100 years have been also estimated.

4. RESPONSIBILITIES

The Hungarian Atomic Energy Authority — under the control of the Government — co-ordinate or accomplish the regulatory tasks in connection with the safe application of atomic energy. The performance of tasks related to final disposal of radioactive waste, as well as to the interim storage and final disposal of spent fuel, and to the decommissioning of nuclear facility is the responsibility of the organization (Agency) designated by the Government for their solution is in the national interest. The governmental decree establishing this organization entered into force on the 1, June 1997. The licensees are obliged to cover the costs of the — among others — decommissioning (demolishing) of nuclear facilities by contributing to the Nuclear Fund.

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Annex 7

RUSSIAN FEDERATION

1. DECOMMISSIONING INFRASTRUCTURES

There are nine major nuclear power plant sites in the Russian Federation (Table 7.1.) of which there are four reactor types currently operating. All of these reactors were designed in the former USSR.

The first type of reactor is the WWER. This is a pressurised water reactor with water as the moderator and coolant. There are 13 units at four NPP sites with a total capacity of 9594 MW(e). There are three generations of WWER's being operated in the Russian Federation. These are the WWER-440 type V-230; WWER-440 type V-213; and theWWER-1000 type V-320.

Table 7.1. Nuclear Power Plants in the Russian Federation

	NPP site	# of units	Reactor type, installed capacity, (MW)	Year of start up	Year of designed lifetime expiration	Remaining operating time, years (at 1997)
1	Beloyarsk	1 2 3	AMB- 100 AMB- 200 BN — 600	1964 1967 1980	1983 1989 2010	shut down shut down 13
2	Balakovo	1 2 3	WWER-1000 WWER-1000 WWER-1000	1985 1987 1988	2015 2017 2018	18 20 21
3	Bilibino	1 2 3 4	EGP-6 (12 MW(e)) EGP-6 (12 MW(e)) EGP-6 (12 MW(e)) EGP-6 (12 MW(e))	1974 1974 1975 1976	2004 2004 2005 2006	7 7 8 9
4	Kalinin	1 2	WWER-1000 WWER-1000	1984 1986	2014 2016	17 19
5	Kursk	1 2 3 4	RBMK-1000 RBMK-1000 RBMK-1000 RBMK-1000	1976 1979 1983 1985	2006 2009 2013 2015	9 12 16 18
6	Kola	1 2 3 4	WWER-440 WWER-440 WWER-440 WWER-440	1973 1974 1981 1984	2003 2004 2011 2014	6 7 14 17
7	Leningrad	1 2 3 4	RBMK-1000 RBMK-1000 RBMK-1000 RBMK-1000	1973 1975 1979 1981	2003 2005 2009 2011	6 8 12 14
8	Novovoronezh	1 2 3 4 5	WWER-210 WWER-365 WWER-440 (417 MW(e)) WWER-440 (417 MW(e)) WWER-1000	1964 1969 1971 1972 1980	1984 1990 2001 2002 2010	shut down shut down 4 5 13
9	Smolensk	1 2 3	RBMK-1000 RBMK-1000 RBMK-1000	1982 1985 1990	2012 2015 2020	15 18 23

The second type is the RBMK-1000 reactor. This is a heterogeneous, channel-type, thermal neutron power reactor which uses graphite as its moderator. There are 11units located at three NPP sites with a total capacity of 11 000 MW(e). This type of reactor has two generations.

The third type is the BN-600 located at the Beloyarsk NPP site. This is a 600 MW(e) fast breeder reactor. And the final type of operating reactor is the EGP-6. There are four units located at the Bilidino NPP site. This reactor is a small pressure tube, uranium — graphite reactor with a capacity of 12 MW(e) for each.

All nuclear power plants are managed and operated by Rosenergoatom Concern, except for the reactors at the Leningrad NPP.

At present, there are four commercial power reactors that have been shut down. These units had a total capacity of 875 MW(e). These are:

- Novovoronezh Unit 1 with a WWER-210,
- Novovoronezh Unit 2 with a WWER-365,
- Beloyarsk Unit 1 with an AMB-100 reactor, and
- Beloyarsk Unit 2 with an AMB-200 reactor.

Both of the AMB reactors at Beloyarsk are uranium-graphite pressure tube reactors which were a prototype of the RBMK reactor.

By 2020, there are 28 reactors that will have reached their designed lifetime. These units have a total capacity of 20 242 MW. It is currently planned that they will be decommissioned when they reach their designed life; however, there is the possibility of operational life extension for each plant.

1.1. Regulations and legal provisions

The system of legal and normative documents which regulates the activities in the field of the application of atomic energy, has a multilevel hierarchical structure as shown:

I. Laws

II. The normative legal acts of President and The Government of Russian Federation

III. Federal norms and rules in the field application of atomic energy

IV. The normative documents of State Inspectorate of Russia on nuclear and radiation safety (Gosatomnadzor)

V. Standards, construction norms and rules, normative documents of state organizations connected with using of nuclear energy

Currently decommissioning of NPP’s in the Russian Federation are regulated by the following documents.

The first level is the Federal law "About the Application of Atomic Energy" issued in October 1995. In the Federal law, the powers and responsibility of the State bodies, state management bodies of atomic energy usage, state surveillance and safety bodies, and operational organizations which are carrying out activities in the field of atomic energy usage are presented. The Law also identifies the rights of both the citizens and public organizations regarding radiation safety. The law defines principles of the legal regulations which include:

- Safe use of atomic energy;
- Availability of the information;
- Participation of the citizens and organizations in discussions of state policy in the field of use of atomic energy; and
- Necessity of licensing any activity in the field of atomic energy usage.

The second level is the decree of The Government of Russian Federation issued in August 1992. "About the adopting of the rule about the composition of the outlay during the manufacturing and the realization of the production and about the order of forming of financial results allowed for taxation of profit".

The third level of documents are the following:

- Common conditions providing nuclear safety for designing, construction and operating of NPP, (OPB-97).
- Radiation Safety Norms (NRB-96). Hygienic standards, 1996.
- Main sanitary rules (OSP-72/87), 1987
- Provisional regulations 30-86.07.02.96 on the management of metal radioactive wastes contaminated with radioactive substances and their reuse after recycling.

The fourth level is "The Requirements to structure and contents of the documents proving maintenance of nuclear and radiating safety of nuclear installation, item of storage, radiating source and/or of declared activity (for nuclear power plants)". The Document establishes rules about licensing activity with reference to nuclear power plants.

Currently there are no additional regulations developed by Gosatomnadzor specific to decommissioning. Several documents are under development. There are documents and standards authorized by the ministries of the Russian Federation which are at the same level as regulations and include the requirements for a decommissioning procedure. These are:

- Requirements of the quality assurance programme for NPPs, 1991.
- The standard content of the NPP safety report (TS TOB AS — 85),1985
- The standard content of the reactor installation safety report (TS TOB RU — 87), 1987.
- Regulations for nuclear safety for reactor installation and NPP (PBY RU AS-89), 1990.
- Sanitary rules of design and operation of NPP (SP-AS-88/93)
- "Concept of NPP Decommissioning after lifetime expiration ", 1991.

The Guide entitled "The Norms of Radiation Safety (NRB-96)" establishes a system of dose limits and the principles of their application. The main dose limits for total external and internal exposure in mSv per calendar year are:

a) Permissible dose for category A (workforce) — 50 mSv

b) Dose limit for category B (limited part of the population called the critical group) — 5 mSv.

1.1.1. Clearance criteria

The Guide entitled "Main sanitary rules (OSP-72/87)" defines the rules for the activities involved with radioactive materials and other ionizing irradiation sources. Provisional regulation 30-86.07.02.96 is concerned with the management of metal radioactive wastes contaminated with radioactive substances and their reuse after reprocessing. Table 7.2 provides a classification of radioactive wastes based on this Guide.

Table 7.2. Classification table of solid and liquid radioactive wastes

Group of waste	Solid waste			Liquid waste
	gamma-emission μSv/h	beta-activity kBq/kg	alpha-activity kBq/kg	kBq/L
Group I: low level	<0.3	7.4×10^{1}–3.7×10^{3}	7.4–3.7×10^{2}	$<3.7 \times 10^{2}$
Group II: intermediate level	0.3–10	3.7×10^{3}–3.7×10^{6}	3.7×10^{2}–3.7×10^{5}	3.7×10^{2}–3.7×10^{7}
Group III–high level	>10	$>3.7 \times 10^{6}$	3.7×10^{5}	$>3.7 \times 10^{7}$

The Guide states:

(a) Metal waste of the first group of radioactive waste (OSP-72/87) can be reprocessed for reuse in industry.

(b) After reprocessing, metal can be divided into three categories:

(i) Gamma emissions of less than 0.2 μSv/h and a specific residual activity less than or equal to 0.37 kBq/kg for alpha emitters and 3.7 kBq/kg for beta emitters. This category has unrestricted use in industry.

(ii) Gamma emissions of 0.2–1.0 μSv/h and specific residual activity of 0.37–1.85 kBq/kg for alpha emitters and from 3.7–18.5 kBq/kg for beta emitters. This category has restricted use in the nuclear and nuclear power industry.

(iii) Gamma emissions of 1.0–2.0 μSv/h and specific residual activity of 1.85–3.7 kBq/kg for alpha emitters and from 18.5–37 kBq/kg for beta emitters. This category of material can be placed in an interim storage facility.

An interim Guide entitled "Acceptable levels of metal radioactivity for its restricted and unrestricted reuse in the Minatom system" has been prepared and is in use. The Guide is based on the collective and individual dose and determines the limiting concentration of radionuclides in metal scrap. These values are shown in Table 7.3.

Table 7.3. Clearance levels in the Russian Federation

Radionuclides	Limit concentration for unrestricted reuse, Bq/g	Limit concentration for restricted reuse, Bq/g
Mn-54	0.90	7.3
Co-58	0.82	6.2
Fe-59	0.71	5.3
Co-60	0.36	2.6
Zn-65	1.44	10.9
Zr-95 + Nb-95m	0.90	6.9
Ru-103 + Rh-103m	1.55	11.7
Ru-106 + Rh-106m	3.84	29.0
Ag-110m	0.29	2.1
Sb-124	0.46	3.5
Sb-125 + Te-125m	1.38	10.2
Cs-134	0.46	3.8
Cs-137 + Ba-137m	1.38	10.4
Ce-141	10.16	76.2
Ce-144 + Pr-144m + Pr-144	16.60	124.5
Eu-154	0.66	5.1

1.2. Decommissioning funding

The Ministry of Atomic Energy of the Russian Federation issued in 1992 a Decree concerning "adopting a rule about the composition of the outlay during the manufacturing and the realization of the production and about the order of forming of financial results allowed for taxation of profit". In 1993 a document was developed and adopted that stated the "Peculiarity of the composition of outlay will be included into the cost of the NPP production". In this document it states, that the cost of NPP production is included by "…allocation of reserves to cover NPP decommissioning expenses (for individual units) at the level of 1.3% of actualised production revenues".

1.3. Spent fuel management

Since the beginning of NPP operation to the present time, more than 22 000 t of spent fuel has been unloaded from the reactors. This includes 9200 t from the RBMK-1000 reactors. The conception of a closed fuel cycle was adopted as the basis for nuclear power development in Russia, but until now this concept has only been fully implemented for spent fuel from WWER-440 and BN-600 reactors. The irradiated fuel from the other reactors is stored in at-reactor storage facilities and in the Federal storage facility.

In accordance with the accepted program of nuclear power development in Russia, the following strategy for spent fuel management is used:

(a) WWER-440 and BN-600 spent fuel is removed from the NPP site and reprocessed using the recovered uranium for the production of fresh fuel for RBMK-1000 reactors;

(b) WWER-1000 spent fuel is stored in a Federal storage facility; and

(c) RBMK-1000, AMB and EGP-6 spent fuel is placed into interim storage in the at-reactor storage facility.

WWER spent fuel is unloaded from the core and placed into unit cooling ponds once a year. Safe storage conditions are ensured during the period when the residual heat is decreasing to levels which permit the transportation of the spent fuel from NPP site (about 3 years).

At NPPs with WWER-440, the rate of spent fuel accumulation corresponds to the rate of its removal from the NPP site for subsequent reprocessing. The reprocessing of WWER-440 and BN-600 spent fuel is carried out at the RT-1 plant ("Mayak"). The RT-1 plant was commissioned in 1977 and has a design capacity of 440 t spent fuel per year. Altogether it has reprocessed more than 7700 t of WWER-440 fuel and 1900 t of BN-600 spent fuel from Russian NPPs.

1.4. Waste management

During NPP decommissioning, the quantity of radioactive waste which will require processing comes from:

- Operational liquid (solidified) and solid radioactive waste are accumulated during the operational lifetime of the NPP. Total accumulated amounts of waste for one NPP unit are shown in Table 7.4. The amount of solid waste accumulated during NPP unit operation can be divided into groups as follows: low — level waste — 80.5%; intermediate level waste — 18.0%; high-level waste — 0.5%;

- The operational wastes arising annually are shown in Table 7.5.

- Operational liquid and solid radioactive waste arising during NPP post operational period and using of installations on-site for processing (these amounts are shown in Table 7.6).

- Solid waste from the dismantling of NPP equipment, buildings and structures. These quantities are shown in Table 7.7.

Management of radioactive waste is one of the most important aspects of decommissioning a NPP. The engineering provisions for transportation, storage and disposal of waste define the most preferable way for plant decommissioning.

Implementation of a multi-barrier conception for environmental protection, while managing liquid and solid radioactive waste, is planned to be carried out in six stages.

The first stage — collection and segregation. The liquid radioactive waste is separated based on activity level, salinity, and content of surface-active agents. The solid radioactive waste is segregated based on activity level as well as combustiblity, compactability, ability to be decontaminated and other groups depending of the decision taken for their eventual processing or storage.

The second stage — temporary storage. This is the storage of waste at the plants where it is generated without processing. This may be caused by the absence of processing facilities or the necessity to allow the radioactivity of the waste to decrease with the decay of short-living radionuclides.

The third stage — waste conditioning. This is the transformation of liquid and solid waste into a suitable conditioning for storage, transportation and disposal. The criteria used to choose this condition are: chemical, thermal and radioactive stability, explosion-proof, fire safety, mechanical strength, etc. The methods for liquid waste conditioning include solidification and packing of solidified waste into drums or other containers. Solid waste conditioning foresees incineration, pressing, decontamination, protective coating, and packing into containers.

The fourth stage — interim storage of conditioned waste. Interim storage may be considered to allow short lived isotopes to decay or if, for other wastes, there is a lack of a regional repository for its disposal.

The fifth stage — transportation is subject to the availability of a regional repository. Different containers manufactured of concrete, metal or their composition are used to transport the waste. These containers can be used not only for transportation but also for interim storage and disposal of the waste.

The sixth stage — radioactive waste disposal. This stage is intended to remove the waste away from the area of human activities. Waste can be disposed in near-surface-, buried- or deep geological formations depending of its activity level and availability of long-living radionuclides.

Table 7.4. Operational radioactive waste arising from one NPP unit during its 30 years of operation

Reactor Type	Solidified waste		Solid waste
	Bituminized, m^3	Cemented, m^3	m^3
WWER-440	3900	5200	6000
WWER-1000	7500	10000	9000

Table 7.5. Annual data of operational wastes arising during one NPP unit operation

Type of waste	WWER-440	WWER-1000
Evaporated concentrate		
Generation rate, m^3	120–170	220–300
Average salt content of the concentrate, g/L	300–400	300–400
Total amount of salts, tonnes	50	90
Specific activity of concentrate, MBq/L	18500	18500
Low active sorbents		
Generation rate, m^3	8.0	16.0
Specific activity, MBq/kg	111.0	111.0
High active sorbents		
Generation rate, m^3	3.0	5.3
Specific activity, MBq/kg	1850	1850
Perlite		
Generation rate, m^3	-	9.0
Specific activity, MBq/kg	-	74.0
Solid radioactive waste		
Generation rate, m^3	200	300

Table 7.6. Operational liquid and solid radioactive waste arising during NPP unit decommissioning and using of radioactive wastes processing installation

Reactor Type	Solidified waste		Solid waste
	Bituminized, m^3	Cemented, m^3	m^3
WWER-440	390	520	600
WWER-1000	750	1000	900

Table 7.7. Solid radioactive waste from dismantling of NPP-equipment, buildings and structures

Reactor type	Concrete, tonnes	Metal, tonnes	Equipment, tonnes
WWER-440	9000	500	4000
WWER-1000	12000	900	6000

The general situation concerning radioactive waste management in the Russian nuclear power industry is that only the first four stages of the above concept can be implemented today. Nevertheless the preparatory activities for decision on radioactive waste disposal have stated. A document on Technical and Economic Estimation of the Opportunity of Russian NPP Radioactive Wastes Disposal on the Novaya Zemlya Archipelago was produced in 1997 by Joint-Stock Company — The Small Power. Technical and cost estimations have shown that the creation of a radioactive waste transport and disposal system in the Novaya Zemlya will increase the cost of electricity of 0.02 cents and the cost of disposal of 1 cubic metre of conditioned wastes will be about $4000.

1.5. Social aspects

The Federal law “About radiation safety of the population” provides the fundamentals for maintenance of the population radiation safety with the purpose of protecting its health. The law includes principles of maintenance of radiation safety, measures for maintenance of radiation safety, installs the main hygienic specifications (allowable limits of doses) exposure, and also general (common) requests for the maintenance of radiation safety, including radiation failures.

The Federal Law “About a sanitary ... health of the population” establishes, that the sanitary rules, norms and hygienic specifications establishing standards of safety and/or harmlessness for the person of his habitable environmental factors and requirements to maintain favourable conditions of his habitability are mandatory for observance by all state organs, etc.

Evaluation have shown that in Russia, the socio-economic consequences from nuclear power plant decommissioning without construction of replacing capacities on the existing sites can result in heavy socio-economic losses. Such as:

- compensation of a social rehabilitation program for displaced NPP employees and population of near-by plant town;

- compensation to the regional budget, near-by plant town budget and other budgets effected by NPP decommissioning;

- sharp reduction or complete termination of use of a nuclear power engineering fuel base;
- non-use of the scientific, technical and industrial base for the development, construction and operation of nuclear power plants;
- loss of competent staff in the field of nuclear power engineering; and
- additional cost of possible reconstruction of regional electric power grid to increase its capacity.

During the planning for a NPP shutdown, it is necessary to consider costs not only to perform the decommissioning and safety activities but also costs to cover:

- indemnification of consequences, connected with change of power and economic situation in region of NPP accommodation, and
- liquidation of social consequences from realization of these activities.

Research shows that the maximum use of the NPP infrastructure for the construction of replacement electrical capacity permits not only collect of the cost for NPP decommissioning but can practically eliminate the social problems.

2. STRATEGY

The concept of NPP unit decommissioning after lifetime expiration in Russia includes the following principles:

- While planning NPP unit decommissioning it is necessary to proceed from the principle of restoring or replacing shutdown units with new, safer NPP units;
- Reuse the sites of decommissioned NPP's as much as it possible;
- To the maximum extent possible, reuse the NPP sites that will be decommissioned for other nuclear power needs;

In accordance with the decommissioning strategy, the operator has to perform a feasibility study for unit closure options five years before its scheduled shutdown date. The goal of the feasibility study is to analyze the technical and economic feasibility of options for extending the operational life time of the plant or shutting the unit down and performing decommissioning. If it is decided to decommission, in accordance with regulatory body requirements, the unit decommissioning program is prepared and sent to the regulatory body for review along with other documents necessary to gain a decommissioning licence.

After final shutdown of the unit, the period of preparing the NPP unit for decommissioning begins. In accordance with the regulatory body requirements, the unit undergoing preparation for decommissioning is still considered to be in operation with regards to the safety regulations. A special set of procedures is prepared and approved for this purpose. The period of preparation includes: removal of all fuel from the unit, standard decontamination of equipment, processing of the operational radioactive wastes, and the development of all necessary documentation for the decommissioning activities. The period of preparation can last from three to five years.

3. DECOMMISSIONING PLANS

Decommissioning is the final stage in the life cycle of a NPP unit. It consists of a set of activities after shut down, which includes the removal of all nuclear fuel, which ensures the unit will never again be used as a source of energy and provides for the radiation safety of personnel, population and environment.

To implement NPP decommissioning, a technical policy entitled “Concept of NPP Unit Decommissioning After Lifetime Expiration” was developed and adopted by Russia's Ministry of Atomic Energy in the form of a regulatory document in 1991. This concept specifies the implementation of the following stages for a NPP unit having completed its lifetime or its extended lifetime:

Stage 1 — preparation of the unit for safe storage under surveillance. The facility is placed into a dormant or moth balled state. The highly radioactive equipment (such as the reactor as an assembly) is localized in the reactor building rooms. The non-active and low contaminated equipment, buildings and structures are dismantled. The radioactive waste is processed to ensure it is in a form for safe storage at the NPP site. The length of this stage is from three to five years.

Stage 2 — safe storage under surveillance of localized (isolated) equipment. The total radioactivity reduction during a 20–30 year period of safe storage is estimated to be 60–100 times. The length of this stage varies according to the type of reactor, the service period of the construction elements in which the localized equipment is located, and the necessity of using the site for new construction. The length of this stage can vary from 30 to 100 years.

Stage 3 — complete dismantling with preparation of the site for new construction. During this stage the following activities are performed:

- complete dismantling of localized equipment;
- dismantling of those buildings and construction elements of the unit not to be further used;
- processing and removal all radioactive wastes to a regional storage facility or disposal;
- re-landscaping of the site for construction of a new NPP unit or its use for some other industrial re-use.

The length of this stage is about five years.

The decommissioning project should be developed stage by stage. At first, the conceptual decommissioning project should be prepared in which all stages are co-ordinated on the purposes and tasks. Next the detail project of the first stage has to be developed. After this is completion, the detail project of the second stage, and so on, has to be developed.

4. RESPONSIBILITIES

The organization of or the decommissioning process is based on:

- Documents and orders of Ministry of Atomic Energy and Rosenergoatom;

- The Guide “Allocation of main functions between organizations and enterprises at realization of work on the termination of operation and decommissioning of NPP”.

On the basis of the specified documents:

(a) The “Customer” of decommissioning is the operational organization, “Rosenergoatom” or the nuclear power station.

(b) The manufacturer of the NPP decommissioning activities:

  (i) For NPP with RBMK type reactor — a specialised decommissioning division which has been organized on Beloyarsk NPP

  (ii) For NPP with WWER type reactor — specialised decommissioning division which has been organized on Novovoronezh NPP

(c) The head scientific organization is VNIIAES;

(d) The designer of the NPP decommissioning project;

  (i) Parent organization on design-technological problems of decommissioning;

  (ii) Parent organization on development of methods and means for NPP dismantling;

(e) The scientific chiefs under waste management, transportation, storage and disposal of radioactive wastes, are specialised organizations.

Gosatomnadzor carries out the supervision of maintenance of the rules, legislation and norms on nuclear and radiation safety during decommissioning, issues the licenses to the enterprises on all kinds of activities connected with decommissioning, supervises the skill level and preparations of the staff, and co-ordinates the documents regulating safety during decommissioning activities. Gosatomnadzor is responsible for:

- informing state bodies and the population about changes in a condition of nuclear and radiating safety, and

- providing a definition of an examination procedure of realization and acceptance of the decisions by its results.

In order to obtain a license to perform the decommissioning activities, the licensee should present to Gosatomnadzor the following documents which should ensured the security of nuclear and radiation safety during the NPP decommissioning:

- The safety decommissioning report;

- The report of comprehensive inspection results;

- The decommissioning programme;

- Quality assurance programme for decommissioning;

- The plan of measures for staff protection in case of a design accident; and

- The instruction for the accountability and control of radioactive wastes generated during NPP decommissioning.

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ANNEX 8

SLOVAKIA

1. DECOMMISSIONING INFRASTRUCTURE

Two nuclear power plants with two WWER0-440 reactors each are currently operating in Jaslovske Bohunice and one NPP with HWGCR is under decommissioning on the same site. The second nuclear site in the Slovakia is Mochovce, where a nuclear power plant with four WWER reactors- 440 MW each is under construction, of which the first unit was commissioned in 1998. All nuclear power plants are owned and managed by the Slovenské elektrárne a.s. (Slovak Electricity Co. plc.).

1.1. Regulations and legal provisions

The most important legal provisions in Slovakia are the Atomic Energy Act, the Act on Protection of Population Health and the Decree on Nuclear Safety in Radioactive Waste Management.

This legislation defines the responsibilities, roles and authorities for all organizations involved in the design, manufacturing, operation, waste management and decommissioning of nuclear installations. The licensee is responsible for the implementation of decommissioning.

The policy of nuclear facilities decommissioning and the role of the operator, as well as that of the Nuclear Regulatory Authority of Slovakia (NRA SR) was codified in the Act on Peaceful Utilization of Nuclear Energy (Atomic Energy Act Act No.130/98 Coll.) approved by the Parliament in 1998. In the relevant part of this Act, decommissioning is defined as a safe removal of nuclear facilities from service and the reduction of residual radioactivity to a level permitting release of property for use as another nuclear facility or site release and termination of licence.

In accordance with the Atomic Energy Act, the Nuclear Regulatory Authority requires conceptual decommissioning plans. The frequency of their updating is 10 years. The NRA SR will assess the final decommissioning plans for each decommissioning phase and the final survey report verifying completion of the decommissioning activities. Based on evaluation of conditions to accomplish site release or reuse, NRA SR issues the licence termination or permission for the use of the site for another nuclear facility purpose. In 1998 the conceptual plans corresponding to NRA SR requirements and IAEA recommendations were completed. The regulations addressing the content and range of decommissioning documentation for NPPs decommissioned after the planned life time are also under preparation by NRASR. A safety guide on this subject has been issued. It is based on the IAEA documents prepared under the RADWASS programme (see [7] of the main text).

1.1.1. Clearance criteria

The Ministry of Health is the authority establishing criteria for site release and material clearance or authorized release (Act No. 272/1994 Coll. on Protection of Population Health, amendments No. 222/1996 Coll. and No. 290/1996 Coll.) in accordance with IAEA /NEA guidance (Safety Series No. 89) on exemption/clearance principles based on the limit of effective dose for the average member of the critical group which is 0.01 mSv/y from each exemption/clearance practice and the limit of collective effective dose of 1 man-Sv/y.

Clearance levels of 3 kBq/m^2 (beta, gamma), 0.3 kBq/m^2 (alpha) for surface activity of released metal materials and specific activity 0.1 kBq/kg (above the natural activity — background) were established. Because it is unlikely that unconditionally cleared materials will contain uniform levels of contamination, procedures were developed for determining the appropriate mass or material surface over which radionuclide quantities may be averaged. The probability of the release of material pieces with significantly higher specific activity than the recommended levels can be reduced by activity averaging over small mass portions and surface areas. Therefore the following conditions are required to be met for release:

- For each part of a surface corresponding to an area of 0.03 m^2, the applied average limit for surface activity is 3 kBq/m^2 (beta, gamma) and 0.3 kBq/m^2 (alpha). For spots with surface area below 0.001 m^2, the applied surface activity limit could be 10 times higher.

- Specific activity limit of 0,1 kBq/kg is applied for each metric ton of material. Specific activity of each individual piece must be lower than 1 kBq/kg.

The derived criteria for metal scrap remelting are also codified in this Act . The criteria is based on activity reduction and dilution by melting, and allows for remelting and authorized release to use metals with specific activity 10 times higher than for clearance. The conditions for averaging are the same as for clearance. The total activity of materials taken for remelting from one site should be less than or equal to 1 GBq/year. Because it is very complicated to find proper facilities for remelting of contaminated metals, a cost benefit evaluation to operate such small plant at the NPP site is now underway.

1.2. Decommissioning funding

Since 1995, the State fund for NPP decommissioning including spent fuel and radioactive waste handling and disposal is established. The owners of NPPs are obliged to contribute 10% of the market price of energy sold to the grid to this fund. A state grant is the another source of income of the fund. Existence of this fund opened the possibility to commence many activities connected with decommissioning of HWGCR A1 and preparation of documentation for WWER reactor decommissioning in the Slovakia.

1.3. Spent fuel management

The original intent was to transport the spent fuel from NPP sites to the former USSR after three years of cooling in the reactor pools. The advanced storage pools at NPP Mochovce will allow to store the spent fuel for a period of six years. No return of reprocessing waste back to Slovakia was foreseen.

For longer fuel storage before transport for reprocessing, wet interim spent fuel store with total capacity corresponding to about 10 years production was constructed and commissioned in 1987 in Jaslovské Bohunice according to a Soviet design. As the transfer of spent fuel to Russia was interupted in the early 1990s, the owner of all Slovak NPPs made a decision to reconstruct the existing interim store by compaction of storage racks so that it will provide for storage of the entire spent fuel production for the operation lifetime of NPP V-1 and V-2 for a time of 40 years as a minimum. It is expected that after 40 years the deep geological repository for high level waste will be available.

1.4. Radioactive waste management

Waste collection and storage systems were designed for 10 years production of pretreated liquid operational waste and non treated or by low pressure compacted solid waste, with possible increasing of storage capacities. According to design strategy practically all operational radioactive waste is now stored in tanks and storage vaults at NPP Jaslovské Bohunice. The originally designed storage volumes, as well as an additionally constructed new set of storage tanks, are almost full. Average year's production at the Bohunice site (4 units WWER-440) is about 400 m^3 of concentrates, 100–200 m^3 of solid waste and about 25 m^3 spent sorbents. Based on safety evaluation of NPP operation and in accordance with actual updated policy at other NPPs, the general policy in waste management in Slovakia has been gradually changed. The main effort was to provide NPPs with basic technologies routinely used for final conditioning of stored waste and to put into operation a repository for conditioned waste. Both of these measures have a direct influence on the decommissioning strategy. Final conditioning and removal of the operational waste make the start conditions for decommissioning more favourable and reduce the time necessary for preparation to safe enclosure.

The following waste treatment technologies at Bohunice site are under operation or construction:

- experimental incineration facility for low level waste. It is in operation since 1992 including a facility for cementation of incinerator ashes which is in operation since 1995;
- two bituminization facilities in operation;
- the Bohunice radioactive waste conditioning centre with cementation plant, incinerator and supercompactor, under licensing;
- another bituminization plant under licensing; and
- segmentation facilities under active testing (1999).

Temporary activity limits were approved by NRA SR for solidified waste (drums with bitumen and cement products) pending licensing of the repository to enable NPP to treat some amount of operational waste. Start of the active tests and commissioning of the repository is expected in 1999. It is foreseen that low and intermediate level conditioned waste from decommissioning will be disposed of at a repository located in Mochovce.

1.5 Social aspects

The Act of National Council of SR No 127/1994 concerning an environmental impact assessment for all new industrial activities, including proposed NPP decommissioning, establishes the responsibilities and authority of the licensee and all involved parties. The environmental impact assessment process includes preparation of an appropriate study by the licensee, hearings of citizens in local and neighbouring NPP municipalities and authorities as well as other stakeholders. The statements of Nuclear Regulatory Authority, Ministry of Health and further regulatory bodies are obligatory for the final stand point.

2. STRATEGY

Based on lessons learned during NPP A-1 decommissioning a thorough and timely approach has been chosen for selection and justification of the strategy for the NPP V-1 decommissioning preparation and development of appropriate documentation. This approach has been reflected on:

a) elaboration of two feasibility studies of NPP V-1 decommissioning

b) selection of preferred decommissioning option

c) preparation of documents for the transition period for the selected option.

The following five decommissioning options were analysed in detail in these feasibility studies:

1. Immediate total NPP dismantling after final shutdown

2. Safe enclosure of the so called "hermetic area" (part of the reactor building) for each unit separately

3. Safe enclosure of the reactor cavity with each reactor separately

4. Safe enclosure of the whole reactor building

5. NPP closing under surveillance (formerly stage 1 according to IAEA classification).

For every option important parameters necessary for decision had been determined. Detail results from the estimation of the above mentioned parameters can be found in [8-1, 8-2].

A multi-attribute analysis was used for complex assessment of the results and to select the preferred option under given conditions. The weights (scoring) of the important attributes were determined by experts. Based on this procedure, option No. 2 "Safe enclosure for the hermetic area for each unit separately" was selected.

Feasibility studies for NPP V-2 and for NPP Mochovce have been developed.

3. DECOMMISSIONING PLANS

Preparation of documents for pre-shutdown and immediate post-operational activities (under operational licence) and for all activities aimed at achieving safe enclosure conditions has been carried out for the selected option. Typical decommissioning operations [8-3] such as the health physics control, decontamination, radioactive waste (RAW) management, plant modifications, transportation and RAW disposal, have been dealt with in these documents.

Decommissioning for the NPP V-1 is planned to be carried out in the following phases:

1. final shutdown
2. lead time (preparation) to safe enclosure
3. safe enclosure

4. final dismantling.

4. RESPONSIBILITIES

The Government Decision No. 190/1994 determined basic strategy for radioactive waste management in the Slovakia. Based on this Decision a new organization named “Nuclear Installation Decommissioning, Radwaste and Spent Fuel Management” (acronym SE-VYZ) was set up within Slovenske elektrarne a.s. on 1 January, 1996 to take responsibility for radioactive waste conditioning (from operation and from decommissioning), disposal of this waste and nuclear facilities decommissioning.

In Atomic Energy Act a new organization responsible for radioactive waste disposal and independent on the radioactive waste producer is suggested to be established.

REFERENCES TO ANNEX 8

[1] HLADKÝ, E., et al., NPP V- 1 Feasibility Study, Parts I-IX, Trnava, Slovakia, November 1991.

[2] HLADKÝ, E., et al., NPP V- 1 Feasibility Study, Study of Options, Parts I-VII, Trnava, Slovakia, December 1992.

[3] US DEPARTMENT OF ENERGY, Decommissioning Handbook, DOE/EM-0142P (1994).

ANNEX 9

UKRAINE

1. DECOMMISSIONING INFRASTRUCTURES

There are five NPP sites in Ukraine. With the exception of Chernobyl, all other sites (Khmelnitski, Rovno, South Ukraine and Zaporozhe) have WWERs. The Rovno site is the only one to have WWER-440 reactors (two units), while all the others have WWER-1000 reactors.

1.1. Regulations and legal provisions

The legal provisions for the Ukrainian nuclear facilities decommissioning are based on following Laws of the Ukrainian Republic:

- “Use of Nuclear Power and Radiation Safety”, 08.02.1995;
- “Radioactive Waste Management”, 30.06.1995;
- “Human Protection Against Impact of Ionizing Radiation”, 14.01.1998.

On the basis of the above mentioned laws the following regulatory documents were issued:

- “General provisions on safety assurance of decommissioning of nuclear power plants and research reactors”, 23.01.1998;
- “Rules and order of exemption of radioactive waste and by-product radioactive materials from regulatory control”, 10.12.1997.

1.1.1. Clearance criteria

In accordance with “Rules and order of exemption of radioactive waste and by-product radioactive materials from regulatory control”(articles 3.3–3.5) the basic dose criteria that determine radioactive waste exemption from the regulatory control are the following:

- individual equivalent dose for critical group shall not exceed 0.05 mSv per year;
- collective dose incurred because of the exempted practice should not exceed 1 man/Sv per year.

Final clearance from the regulatory control is applicable to those waste and by-product materials with radioactive concentrations meeting the following exemption levels (when the radionuclide composition of wastes is unknown):

(a) total specific activity averaged over the waste batch, but not more than 1000 kg, does not exceed 1 kBq/kg for beta and gamma sources or 0.1 kBq/kg for alpha. Total weight of exempted radioactive waste shall not exceed 3000 tons per year;

(b) total activity of a single package containing less than 100 kg of waste is not more than 100 kBq for beta and gamma sources or 10 kBq for alpha;

(c) for decommissioning of nuclear facility the following clearance levels of surface contamination of construction materials are established:

  (i) alpha-emitters 0.05 Bq/cm^2 (exemption level of 5 kBq),

  (ii) beta- and gamma-emitters 5 Bq/cm^2 (exemption level of 5000 kBq),

  (iii) other radionuclides 0.5 Bq/cm^2 (5 kBq < exemption level < 5000 kBq).

Exemption levels for restricted release from regulatory control are established in each certain case on the basis of calculated individual and collective doses for credible scenarios.

1.2. Decommissioning funding

According to the Law of Ukraine on “Use of Nuclear Power and Radiation Safety” (article 39), financing of the nuclear installation decommissioning shall be provided by the owner. The owner transfers money into the decommissioning fund and includes decommissioning expenses in electricity charges.

Article 8.2 of the Regulatory Document “General provisions on safety assurance of decommissioning of nuclear power plants and research reactors” reads:

“The mechanism of accumulation of financial resources for decommissioning should be in force during the whole period of commercial operation of the facility”.

1.3. Spent fuel management

Article 3.2 and 3.5 of the Regulatory Document “General provisions on safety assurance of decommissioning of nuclear power plants and research reactors” reads respectively:

“Action of the decommissioning licence begins only after putting the facility into nuclear safe conditions, which means absence of nuclear fuel on its site or its location within this site only inside nuclear fuel storage facilities, which are aimed at long-term safe storage”.

“The main objective of this stage is transformation of the facility to a condition, which corresponds to absence of nuclear fuel on its site or location of the fuel within the site boundaries only inside nuclear fuel storage facilities, which are aimed at long-term safe storage”.

At the present time spent fuel is stored at the NPP sites. Each Unit of the NPP has a pool located close to the reactor facility where the spent fuel is stored at least for three years after discharge from the reactor core. Then the fuel can be transferred to the on-site pools of the intermediate spent fuel storage facilities. The problem of the construction of a repository for spent fuel final disposal is now under consideration.

1.4. Radioactive waste management

Article 6.2 of the Regulatory Document “General provisions on safety assurance of decommissioning of nuclear power plants and research reactors” reads:

“The infrastructure of radioactive waste management should be developed with taking into account the necessities of due implementation in future of all required operations on management of radioactive wastes, accumulated during operation and arising during facility decommissioning”.

As of today, the temporary storage and first conditioning are carried out at the NPP sites. The centralized radioactive waste processing plant is planned to be constructed according to State's programme. The operational radwastes from all NPPs will be processed at this plant in view of their disposal.

1.5. Social aspects

According to Ukraine legislation and tradition social aspects of NPP decommissioning are tackled by President of the Republic and Government specifically for each reactor to be decommissioned in agreement with local authorities.

2. STRATEGY

In accordance with "General provisions on safety assurance of decommissioning of nuclear power plants and research reactors" (article 4.7) decommissioning strategy is the sequence of facility decommissioning steps which is adopted in compliance with the legislation in force and taking into account consequences, duration and main activities of decommissioning phases as well as the state of the facility after each phase.

Selection of decommissioning strategy should be carried out by consideration of various decommissioning options and choice of the optimal one on the basis of the "cost-benefit" principle, implying the balanced assessment of the following major factors:

(a) safety requirements established by the legislation in force;

(b) results of assessments of possible dangerous impact on the staff, the public and the environment;

(c) radiological conditions of the facility and their foreseen changes in time;

(d) physical state of facility and its foreseen change in time;

(e) radioactive waste management issues, including final disposal;

(f) possibility of re-use of facility components or recycling of these materials;

(g) possibility of materials clearance;

(h) plans of further use of the facility site;

(i) availability of engineering means and technologies for decommissioning;

(j) availability of staff, in particular use of knowledge and experience of operational staff;

(k) required financial resources and their availability;

(l) available decommissioning experience, both domestic and international;

(m) social aspects.

3. DECOMMISSIONING PLANS

According to the Law of Ukraine on "Use of Nuclear Power and Radiation Safety" (article 39), procedures for decommissioning of nuclear installation have to be anticipated in plans in accordance with the norms, regulations and standards regarding use of nuclear power. In accordance with "General provisions on safety assurance of decommissioning of nuclear power plants and research reactors" (articles 3.4, 3.7) the following stages of decommissioning: final closure, preservation, long-term storage, dismantling are established.

Final closure is a decommissioning stage at which the facility is transformed to the state that excludes the possibility of its further use with the purposes, for which it was constructed. At this stage the following main activities shall be carried out:

- Dismantling of systems and components of facility, external to nuclear reactor, not relevant to safety and not necessary for further use;

- Maintaining and reinforcing of barriers to prevent the spreading of radioactivity into the environment;

- Preparing the inventory of contaminated and activated systems and components of the facility including their radiological conditions;

- Establishing administrative and organizational measures, appropriate to the changed state of the facility.

Preservation is a decommissioning stage at which the facility is transformed to the state aimed at safe enclosure of the radioactive sources located inside the facility for a defined period.

At this stage the following main activities should be carried out:

- Preparing a detailed inventory of contaminated and activated systems and components of facility including their radiological conditions;

- Ensuring reliable confinement of non-dismantled parts of facility (tight closure of pipes, tightening of connections, additional hermetic sealing of premises, etc.);

- Establishing maintenance procedures for interim storage of radioactive substances on the facility;

- Implementing collection and conditioning of radioactive waste arising at this stage and transferring that waste to specialized enterprises;

- Establishing administrative and organizational measures, appropriate to changed facility state.

Long-term storage is a decommissioning stage at which the facility stays in preserved state aimed at safe enclosure of radioactive substances located inside.

The main purpose of this stage is to benefit from radionuclide decay to reduce substantially the quantities of radioactive substances.

At this stage the following main activities should be carried out:

- Implementing operation of systems and components to ensure safe enclosure of radioactive substances at the preserved facility;

- Performing periodic inspections of the state of the preserved facility;

- Establishing administrative and organizational provisions, appropriate to the changed condition of facility.

The premises of the preserved facility can be reused for interim storage of solid radioactive waste arising during decommissioning activities of this facility.

Dismantling is a decommissioning stage at which the radioactive sources located inside the facility are removed.

Complete dismantling includes the following main activities:

- Dismantling and removing of all facility systems and components which are subject to control as sources of ionizing radiation;
- Collecting and conditioning of radioactive waste arising at this stage and transferring this waste to specialized enterprises;
- Surveying of the residual components which are not dismantled;
- Establishing procedures on termination of radiation monitoring previously required;
- Establishing administrative and organizational measures, appropriate to the changed facility state.

The necessity of each separate stage and the order of stage priorities should be defined during development of decommissioning strategy. The stages of facility decommissioning can be implemented either completely or partially for different parts of the facility according to the chosen strategy.

4. RESPONSIBILITIES

According to the Law of Ukraine on “Use of Nuclear Power and Radiation Safety” (article 28) a licence should be granted to carry out activities for nuclear installation decommissioning. The following is in compliance with “General provisions on safety assurance of decommissioning of nuclear power plants and research reactors” (articles 3.5, 3.6). The decommissioning licence comes into force only after putting the facility into nuclear safe conditions, which means absence of nuclear fuel on site or its location within on site nuclear fuel storage facilities, which are aimed at long-term safe storage. Since the decommissioning licence is in force the previous licence given for facility operation is cancelled and cannot be resumed. The licence for facility decommissioning includes reception from the Ukraine’s Ministry for Environmental Protection and Nuclear Safety of separate permissions to implement each decommissioning stage as determined in the above mentioned articles. To obtain permission to move each subsequent stage the operating organization should submit to State nuclear regulatory authority the following documents, as appropriate to each given stage:

- Decommissioning Stage Implementation Plan;
- Safety Analysis Report;
- Technological Requirements of Decommissioning.

ABBREVIATIONS

ALARA	as low as reasonably achievable
AP	alkaline permanganate
AP-AC	alkaline permanganate-ammonium citrate
CAMC	contact arc metal cutting
CEE	central and eastern Europe
COMECON	Council for Mutual Economic Assistance
CORD	chemical oxidation reduction decontamination
EC	European Commission
EDTA	ethylendiaminetetra-acetic acid
HEPA	high efficiency particulate air
HLW	high level waste
HWGCR	heavy water gas cooled reactor
ILW	intermediate level waste
JPDR	Japan Power Demonstration Reactor
LLW	low level waste
LOMI	low oxidation-state metal ion
OECD/NEA	OECD Nuclear Energy Agency
PCB	polychlorinated biphenyl
PWR	pressurized water reactor
R&D	research and development
RPV	reactor pressure vessel
SG	steam generator
UV	ultraviolet
WWER	water cooled, water moderated, electricity generating reactor, a Soviet-design PWR

GLOSSARY

clearance levels. A set of values, established by the regulatory body in a country or state, expressed in terms of activity concentrations and/or total activities, at or below which sources of radiation can be released from nuclear regulatory control.

contamination. The presence of radioactive substances in or on a material or in the human body or other place where they are undesirable or could be harmful.

decommissioning. Actions taken at the end of useful life of a nuclear facility in retiring it from service with adequate regard for the health and safety of workers and members of the public and protection of the environment. The ultimate goal of decommissioning is unrestricted release or use of the site. The time period to achieve this goal may range from a few to several hundred years. Subject to legal and regulatory requirements of a Member State, a nuclear facility or its remaining parts may also be considered decommissioned if it is incorporated into a new or existing facility, or even if the site in which it is located is still under regulatory or institutional control. This definition does not apply to nuclear facilities used for mining and milling of radioactive materials (closeout) or for the disposal of radioactive waste (closure).

decommissioning option. One of various decommissioning strategies which may be considered when decommissioning is being planned. A variety of factors, such as timing and the availability of technologies, will influence which decommissioning strategy is ultimately chosen.

decommissioning phase. Well defined and discrete parts of the decommissioning process or work.

decommissioning stages. (See decommissioning phase.) Previous IAEA documents have referred to three discrete stages of decommissioning (storage with surveillance, restricted release and unrestricted release). As a result of decommissioning experience, an increasing number of Member States now use different terminologies and approaches, and therefore this glossary no longer refers to the three states identified above.

decontamination. The removal or reduction of radioactive contamination by a physical and/or chemical process. (See also contamination).

dismantling. The disassembly and removal of any structure, system or component during decommissioning. Dismantling may be performed immediately after permanent retirement of a nuclear facility or may be deferred.

disposal. The emplacement of waste in an approved, specified facility (e.g. near surface or geological repository) without the intention of retrieval. Disposal may also include the approved direct discharge of effluents (e.g. liquid and gaseous wastes) into the environment with subsequent dispersion.

enclosure, safe (during decommissioning). A condition of a nuclear facility during the decommissioning process in which surveillance and maintenance of the facility takes place. The duration of safe enclosure can vary from a few years to the order of one hundred years.

licence. A formal, legally prescribed document issued to the applicant (i.e. operating organization) by the regulatory body to perform specified activities related to the siting, design, construction, commissioning, operation, decommissioning of a nuclear facility, closure of a disposal facility, closeout of a mining and mill tailings site, or institutional control.

licensee. The holder of a licence issued by the regulatory body to perform specific activities related to the siting, design, construction, commissioning, operation, decommissioning of a nuclear facility, closure of a disposal facility, closeout of a mining and mill tailings site, or institutional control. The applicant becomes the licensee after it receives a licence issued by the regulatory body.

minimization. A concept which embodies the reduction of waste with regard to its quantity and activity to a level as low as reasonably achievable. Waste minimization begins with nuclear facility design and ends with decommissioning. Minimization as a practice includes source reduction, recycling and reuse, and treatment with due consideration for secondary as well as primary waste materials.

operation. All activities performed to achieve the purpose for which the nuclear facility was constructed, including maintenance, refuelling, in-service inspection and other associated activities.

operational period (operating period). The period during which a nuclear facility is being used for its intended purpose until it is decommissioned or is submitted to permanent closure.

operator (or operating organization). In waste management, the organization (and its contractors) which performs activities to select and investigate the suitability of a site for a nuclear facility, and/or undertakes to design, construct, commission, operate and decommission such a facility. This term is preferred to ‘implementing organization’ which appeared in earlier literature.

regulatory body. An authority or a system of authorities designated by the government of a country or state as having legal authority for conducting the licensing process, for issuing licenses and thereby for regulating the siting, design, construction, commissioning, operation, closure, closeout, decommissioning and, if required, subsequent institutional control of the nuclear facilities (e.g. near surface repository) or specific aspects thereof. This authority could be a body (existing or to be established) in the field of nuclear related health and safety, mining safety or environmental protection vested or empowered with such legal authority.

restricted release or use. A designation, by the regulatory body in a country or state, to restrict the release or use of equipment, materials, buildings or the site because of its potential radiological hazards.

site. The area containing, or under investigation for its suitability to construct, a nuclear facility. It is defined by a boundary and is under effective control of the operating organization.

storage (interim). The placement of waste in a nuclear facility where isolation, environmental protection and human control (e.g. monitoring) are provided with the intent that the waste will be retrieved for exemption or processing and/or disposal at a later time.

unrestricted release or use. A designation, by the regulatory body in a country or state, that enables the release or use of equipment, materials, buildings or the site without radiological restriction.

waste, decommissioning. Radioactive waste from decommissioning activities.

waste, exempt. In the context of radioactive waste management, waste (from a nuclear facility) that is released from nuclear regulatory control in accordance with clearance levels because the associated radiological hazards are negligible. The designation should be in terms of activity concentration and/or total activity and may include a specification of the type, chemical/physical form, mass or volume of waste, and its potential use. (See also clearance levels).

waste, high level (HLW). a) The radioactive liquid containing most of the fission products and actinides originally present in spent fuel and forming the residue from the first solvent extraction cycle in reprocessing and some of the associated waste streams. b) Solidified high level waste from (a) above and spent fuel (if it is declared a waste). c) Any other waste with an activity level comparable to (a) or (b). High level waste in practice is considered long lived. One of the characteristics which distinguishes HLW from less active waste is its level of thermal power.

waste, low and intermediate level. Radioactive wastes in which the concentration of or quantity or radionuclides is above clearance levels established by the regulatory body, but with a radionuclide content and thermal power below those of high level waste. Low and intermediate level waste is often separated into short lived and long lived wastes. Short lived waste may be disposed of in near surface disposal facilities. Plans call for the disposal of long lived waste in geological repositories.

waste, radioactive. For legal and regulatory purposes, radioactive waste may be defined as material that contains or is contaminated with radionuclides at concentrations or activities greater than clearance levels as established by the regulatory body, and for which no use is foreseen. (It should be recognized that this definition is purely for regulatory purposes, and that material with activity concentrations equal to or less than clearance levels is radioactive from a physical viewpoint — although the associated radiological hazards are negligible).

waste acceptance criteria. Those criteria relevant to the acceptance of waste packages for handling, storage and disposal.

waste management, radioactive. All activities, administrative and operational, that are involved in the handling, pretreatment, treatment, conditioning, transportation, storage and disposal of waste from a nuclear facility.

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Workshops

Vienna, Austria: 19–23 June 1995, 9–13 November 1998
Greifswald, Germany: 25–29 March 1996
Balatonfüred, Hungary: 25–29 November 1996
Bratislava, Slovakia: 12–16 May 1997
Mol, Belgium: 15–19 December 1997
Tokai, Tokyo, Japan: 30 March–3 April 1998

Consultants Meeting

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