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FOREWORD

On-site disposal is not a novel decommissioning strategy in the history of the nuclear industry. Several projects based on this strategy have been implemented. Moreover, a number of studies and proposals have explored variations within the strategy, ranging from in situ disposal of entire facilities or portions thereof to disposal within the site boundary of major components such as the reactor pressure vessel or steam generators. Regardless of these initiatives, and despite a significant potential for dose, radioactive waste and cost reduction, on-site disposal has often been disregarded as a viable decommissioning strategy, generally as the result of environmental and other public concerns.

Little attention has been given to on-site disposal in previous IAEA publications in the field of decommissioning. The objective of this report is to establish an awareness of technical factors that may or may not favour the adoption of on-site disposal as a decommissioning strategy. In addition, this report presents an overview of relevant national experiences, studies and proposals. The expected end result is to show that, subject to safety and environmental protection assessment, on-site disposal can be a viable decommissioning option and should be taken into consideration in decision making.

This report was first drafted by a group of consultants in October 1997, then reviewed at an Advisory Group meeting in May 1998. At this meeting, a large amount of national experience, studies and proposals were made available by national nominees. The report was finalized by G.A. Brown in February 1999. The IAEA officer responsible for this work was M. Laraia of the Division of Nuclear Fuel Cycle and Waste Technology.

CONTENTS

1. INTRODUCTION

According to the IAEA definition [1], the ultimate goal of decommissioning is unrestricted release or use of the site. The definition recognizes that the time period to achieve this goal may range from a few to several hundred years. Subject to national legal and regulatory requirements, a nuclear facility or its remaining components may also be considered decommissioned if incorporated into a new or existing nuclear facility, and the site on which it is located remains under regulatory control. The purpose of the latter approach is to provide reasonable flexibility and implies that circumstances such as cost or practicality may impede achievement of unrestricted site release. Under these circumstances, only restricted site release may be possible because of the presence of significant amounts of residual radioactivity. Restrictions on the use of the site may range from a minimum (e.g. administrative acts forbidding residential or industrial uses) to prescribed surveillance and maintenance of protective barriers intended to confine residual radioactivity. The other aspect to be taken into account is the duration of restricted use, i.e. temporary or permanent. In the temporary case, the expression storage is used in IAEA terminology, while disposal is used for final and permanent conditions (see Glossary).

In most cases, decommissioning of nuclear facilities is accomplished under two basic strategies, namely: (1) immediate dismantling, or (2) safe enclosure followed by deferred dismantling. Both strategies are intended to lead eventually to unrestricted release of the site and imply removal of radioactive waste to an off-site repository. There is however a third strategy called on-site disposal, which consists of disposing of the nuclear facility on the same site where it had operated. Variations exist, ranging from local disposal of some waste to disposal of complete nuclear facilities such as reactor plants and fuel cycle facilities. In many cases dismantling may be minimal. The on-site disposal strategy has been studied and a few projects have been implemented in some IAEA Member States.

Technical literature has referred to the above described approach variously as in situ disposal, on-site decommissioning, or entombment. In this publication the term 'on-site disposal' is used and defined as decommissioning activities which encompass final disposal of the nuclear facilities or portions thereof within the nuclear site boundary.

On-site disposal is being accomplished in Finland for operational wastes and is being planned for decommissioning wastes and very low activity wastes are being disposed of on-site in Japan. On-site disposal however as a decommissioning strategy has not been broadly accepted by Member States although some reactors and other facilities have been disposed of successfully in some countries (e.g. the USA). However, several studies and proposals have been made more recently which consider on-site disposal on grounds of cost reduction, simplicity of operation, reduction of occupational radiation exposure, and sometimes technical expediency. The option is currently under review by the US Nuclear Regulatory Commission.

Technical reports and other publications published by the IAEA either do not mention, or only deal marginally with, on-site disposal as a practical decommissioning alternative [2]. More attention has been given to entombment of reactors damaged by a severe accident where decontamination and dismantling activities would be precluded by high radiation fields, and where confinement of the residual contamination would require extensive use of protective barriers [3, 4].

Some guidance and useful information on many aspects of near surface disposal can be gained from related studies reported in IAEA publications [5, 6]. These reports, in particular, deal with barriers, physical and geological aspects of burial sites and proposed monitoring and surveillance requirements. There is also an IAEA publication on the siting of near surface disposal facilities which may give some guidance [7] It is now considered appropriate to collect international experience and views because there could be sound reasons for choosing on-site disposal as the planned decommissioning strategy for a range of nuclear facilities Member States would then have the option of evaluating the extent to which on-site disposal fits their needs and the potential for licensing this approach in their countries It is recognized that radiological safety and environmental protection are of paramount importance in gaining acceptance of the on-site disposal option

Long lived radionuclides are a potential safety problem and the IAEA currently recommends that their concentration is minimized in near surface disposal sites The preference is for disposal of long lived radionuclides in geological repositories Safety assessment must take account of the environmental impact of any long lived radionuclides that are included in the waste for disposal

Plans for disposing of radioactive wastes give rise to a number of unique problems mainly due to the very long time-scales which have to be considered To assist in promoting discussion amongst international experts and in developing consensus on waste disposal the IAEA established a working group This group produced three TECDOCs on safety indicators in different time frames, waste disposal issues and regulatory decision making for long lived wastes [8–10] These are of interest and relevance to on-site disposal

2. SCOPE

This report covers the on-site disposal strategy for decommissioning of most types of nuclear installations, ranging from small research reactors to commercial nuclear power plants and nuclear fuel cycle facilities, with emphasis on disposal of the activated and contaminated materials The advantages and disadvantages of on-site disposal are mentioned in relation to other decommissioning strategies In this context, the report also reviews aspects which have not been fully covered previously in IAEA publications Facilities shut down under both normal planned conditions and as the result of serious accidents are addressed However, reactors shutdown because of serious accidents are addressed as special cases emphasizing that some principles of on-site disposal may be utilized to decommission such facilities or to provide secure interim storage Uranium and thorium mining and milling facilities and radioactive waste repositories are not considered here because of the unique features of these installations It should be pointed out that this report is not intended as a safety guide relating to disposal and does not imply that the IAEA shows any preference for on-site disposal as a decommissioning strategy

3. OBJECTIVE

The objective of this report is to give an overview of the factors relevant to the selection of on-site disposal as a decommissioning strategy and of the actual experience available The report is intended to provide policy makers, regulators, operators and other interested parties with information for considering this decommissioning strategy together with other decommissioning alternatives

4. CONSIDERATIONS INFLUENCING THE SELECTION OF ON-SITE DISPOSAL

Selection of on-site disposal as a decommissioning strategy is influenced by a number of considerations, which may be either technical or political, and sometimes both. Important technical considerations are discussed in Section 4.1 and the political issues and public concerns are discussed in Section 4.2.

4.1. TECHNICAL CONSIDERATIONS

The technical considerations important to deciding whether or not to proceed with on-site disposal at a specific facility are listed below, with more detailed discussions in subsequent subsections.

- cost of on-site disposal compared with alternative strategies
- occupational radiation dose from on-site disposal compared with alternative strategies
- value of independence from cost and availability constraints of off-site disposal facilities
- environmental impact of establishing a disposal site
- creation of additional disposal sites within a country as compared with centralized disposal facilities
- safety and environmental aspects of on-site disposal.

These factors are discussed further in subsequent paragraphs.

Cost

A major factor potentially favouring the selection of on-site disposal as a decommissioning strategy is lower cost. Compared to the decontamination and dismantling strategies, on-site disposal should entail less complex and therefore less costly operations except perhaps in the case of reprocessing facilities. Employing the strategy minimizes the need for decontamination for the shorter lived radionuclides because natural decay of radioactivity in disposed structures achieves the same result in the long term. Likewise, there is less need for segmentation of systems and structures, as many large components would be disposed of intact or simply left in place. Long term surveillance and monitoring costs must be included in the overall cost of this strategy because of remaining on-site hazards [11]. The overall cost of this strategy is anticipated to be less than that of other strategies because of the reduced implementation cost, especially where dismantling activities are minimized, and lower long term monitoring and surveillance costs (vs. a safe enclosure-deferred dismantling strategy). The total cost to a Member State could be higher however if many separate on-site disposal facilities need continuous monitoring and surveillance for long institutional control periods. In addition, if long lived radionuclides have to be separated or extracted and sent for geological disposal, then overall costs will increase. There is therefore a cost incentive to include long lived wastes in the safety assessment.

Occupational radiation dose

Because the level of decontamination and dismantling activities necessary in many types of facilities is reduced for on-site disposal, it is expected that radiation dose to the decommissioning workforce will be less for the on-site disposal strategy. However, it is important to note that where significant dismantling and decontamination of structures external to the primary envelope is done, the expected savings in dose commitment can be rapidly eroded. For example, according to an analysis of the decommissioning of a reference PWR in the USA [11], radiation doses from immediate dismantling and from on-site disposal would be similar, since the majority of the dose in both strategies is associated with the initial plant clean out and dismantling activities in structures external to the reactor containment. However, in the UK where an option considered was that the plant is buried beneath a mound [12], there is considerable potential for dose reduction.

Value of independence from off-site disposal facilities

An advantage of on-site disposal is that the volume of radioactive waste requiring off-site transport and disposal will be minimized. Where partial plant dismantling is undertaken, attempts should be made to incorporate all of these wastes within the primary containment envelope. On-site disposal may also provide the opportunity to dispose of conditioned operational and post operational cleanout wastes within the containment boundary, further reducing the waste volumes requiring off-site disposal. In some Member States, centralized radioactive waste disposal facilities may not be available or planned, thus making on-site disposal the only viable approach. In some other states, it may also be technically difficult or undesirable to move wastes to another location. When a new nuclear site is planned and licensing is applied for, it is normally now a requirement for preliminary decommissioning plans to be drawn up. However, this is not the case for some older installations for which on-site disposal could be particularly attractive.

Creation of an on-site disposal facility

One disadvantage with an on-site disposal strategy is the creation of a radioactive waste disposal facility on the original site. This action may raise environmental concerns because selection criteria for waste disposal sites are different from those used in siting reactors or other nuclear fuel cycle facilities. For many nuclear facilities the on-site disposal strategy requires essentially the same level of environmental assessment as a centralized disposal facility since unrestricted release may not be achieved for very long periods. For example, facilities such as commercial LWRs that contain significant quantities of very long lived radionuclides [13]. Accordingly, a similar level of site analysis and characterization of waste inventory as for a centralized near surface repository will be needed to approve the site for disposal. The confinement capabilities of any waste disposal site will depend on a combination of natural and man-made factors. While natural confinement factors at a given site might be inferior to those required for a near surface waste disposal facility, man-made engineered barriers applied in the on-site disposal case may still render the site suitable for long term confinement of radioactive waste. This suitability is usually assessed on a case-by-case basis. Compliance with release criteria would then have to be demonstrated through probabilistic or deterministic techniques as agreed upon with the regulatory body. In any case, surveillance and monitoring measures (similar to those in use at centralized waste disposal sites) are usually in place as long as institutional control of the site is required.

Another concern may be the very long commitment of the site to waste disposal which is a significant change in use from the original plan for the site. The lengthy delay period required to achieve unrestricted site release may constrain site use only to other licensed nuclear activities. For some sites this condition may be acceptable or even advantageous because it may capitalize on existing site monitoring capabilities and other existing infrastructures. There may also be a concern that the site area occupied by the engineered enclosure is constrained from reuse for conventional non-nuclear purposes in the foreseeable future.

Safety and environmental aspects

Evaluation of the safety and environmental consequences of using the on-site disposal strategy will generally be the most critical factor in gaining regulatory approval and public acceptance. The primary issue will be in providing assurance that release and dispersion of the radionuclide content in the disposal site is adequately retarded and meets the regulatory criteria throughout the decay period required to achieve unrestricted release. The decay period is likely to range between a few decades and thousands of years. The safety assessment will rely largely on the accuracy of the inventory data and characterization of the waste, and the associated corrosion and erosion rates for the barrier and waste materials. Persuasive arguments will have to be structured to show that engineered barriers have been adequately designed and constructed, and that corrosion and erosion rates and release pathways have been evaluated to demonstrate control of the disposal environment and releases. Barriers are particularly important for controlling the release of the shorter half-life radionuclides since these represent the principal types of contaminants present in many facilities. Also, it is believed that barriers can be designed to give assurance of durability for the 100 to 300 year periods involved. For the very long lived radionuclides, material corrosion and erosion characteristics and exposure pathways will be of greatest importance. The release of the contaminants will ultimately be determined by the corrosion and leaching rates of the activated materials because the period for decay to background may be too long for reliance on barrier performance. The safety assessment for such radionuclides are usually based on dispersion characteristics and are relevant for sites where the release rates and pathways can be shown to limit public and environmental exposures to within specified criteria. Two release scenarios can be envisaged: groundwater releases, and loss of containment or cover by external events. Both of these scenarios an be treated in a deterministic or probabilistic manner to show that the doses to the public are acceptable. Experience in disposal projects to date has shown that safety assessments can be developed which demonstrate minimal environmental impacts. The IAEA Safety Standard entitled Safety Assessment for Near Surface Disposal of Radioactive Waste, gives guidance in compiling a safety assessment [14].

Safety and environmental assessment usually considers the consequences of possible public intrusion in the long term. In the initial decay period of 100 to 300 years, planning for a near surface disposal project generally specifies that institutional controls must be in place to detect and guard against intrusion. A common regulatory position assumes that intrusion (with no knowledge of site history and conditions) will occur at the time that institutional control of the site is lost. A critical assessment of this position is contained in a recently published paper where the intrusion scenario is assessed in probabilistic and site-specific terms [15]. There is also further consideration of intrusion in a study of in situ disposal by mounding [12]. The materials used to construct barriers, and the disposed materials, may be arranged to provide some warning to intruders. Such warnings are unlikely to ensure absolute safety but, nevertheless, some consideration may be given to installing signs or plaques constructed of highly durable materials to provide permanent warning.

It should be noted that long term concerns could be somehow alleviated by removal of radioactive substances (e.g. decontamination or removal of highly activated reactor internals) prior to decommissioning.

4.2. POLITICAL ISSUES AND PUBLIC CONCERNS

In the previous section it was anticipated that safety assessments could be made and sustained for on-site disposal for some sites depending on local and technical issues and environmental factors. A safety assessment would also involve an environmental impact study that would be used in the process of gaining acceptance of the strategy by the public. The environmental impact study would also take account of numerous issues that would affect the local population directly or indirectly. On-site disposal is not without precedence because there have been instances [16–20], particularly in the 1960s and early 1970s, where it has been done successfully and without undue public concern. These instances have largely been restricted to small facilities (e.g. research reactors or demonstration plants) and principally where unrestricted release of the site has not been required or requested.

There was a trend in former years for regulators to discourage consideration of the on-site disposal strategy despite studies conducted by utilities and specialist nuclear organizations and consultants. For example, in the USA, the USNRC initially showed disfavour for the concept but recently has initiated a review and re-examination of the strategy, with the aim of either confirming previous concerns about its viability or moving forward with formal regulatory actions (see Annex A-11). These investigations may lead to future support for on-site disposal in the USA if a satisfactory case for viability can be made. These new investigations have been prompted by renewed interest by nuclear licensees, particularly commercial nuclear power reactor owners who are addressing the problems of decommissioning.

The major problem facing any initiatives to promote on-site disposal will be the creation of a disposal facility on a site where no waste disposal was envisaged. Any proposals for a nuclear waste disposal site give rise to significant public concern, which can also influence regulatory and government policy. It is therefore vital in making proposals for on-site disposal that the environmental impact is thoroughly explored and debated, and relevant comparisons are made with the other decommissioning strategies. In some cases, acceptance may be easier in that the site is sufficiently remote from population centres to be of lesser public concern, or that technical and safety difficulties and costs make total dismantling a less viable option.

In developing an approach to on-site disposal, the following steps should be addressed:

- a full and comprehensive environmental impact study;
- an inventory of wastes that will remain on-site;
- a preliminary safety assessment that addresses risks and hazards;
- discussions with the regulator and other appropriate government authorities to gain acceptance;
- a public awareness campaign to inform the public of the proposals and of any potential local economic benefits to be derived from the on-site disposal activities;
- public consultation to gain acceptance;
- formal planning and licensing applications leading to detailed designs;
- proposals for the institutional control period.

5. ON-SITE DISPOSAL OPTIONS

The primary difference between on-site disposal and other decommissioning strategies is that on-site disposal implies disposal of radioactive materials on-site and avoids transport to other locations or repositories. The strategy requires that the site becomes a near surface disposal site and would be subject to the regulatory and institutional controls that apply to such a repository. It may be necessary for some long lived wastes to be removed from site for treatment and disposal elsewhere. Both mobile and fixed radionuclide inventories must be defined.

The primary options available under the on-site disposal decommissioning strategy are illustrated in Fig. 1 and are listed below:

- In situ disposal where the reactor or nuclear facility is retained wholly or partly at its existing location; and
- On-site transfer and disposal, where the reactor or nuclear facility is moved to an engineered disposal facility at an adjacent location on the site.

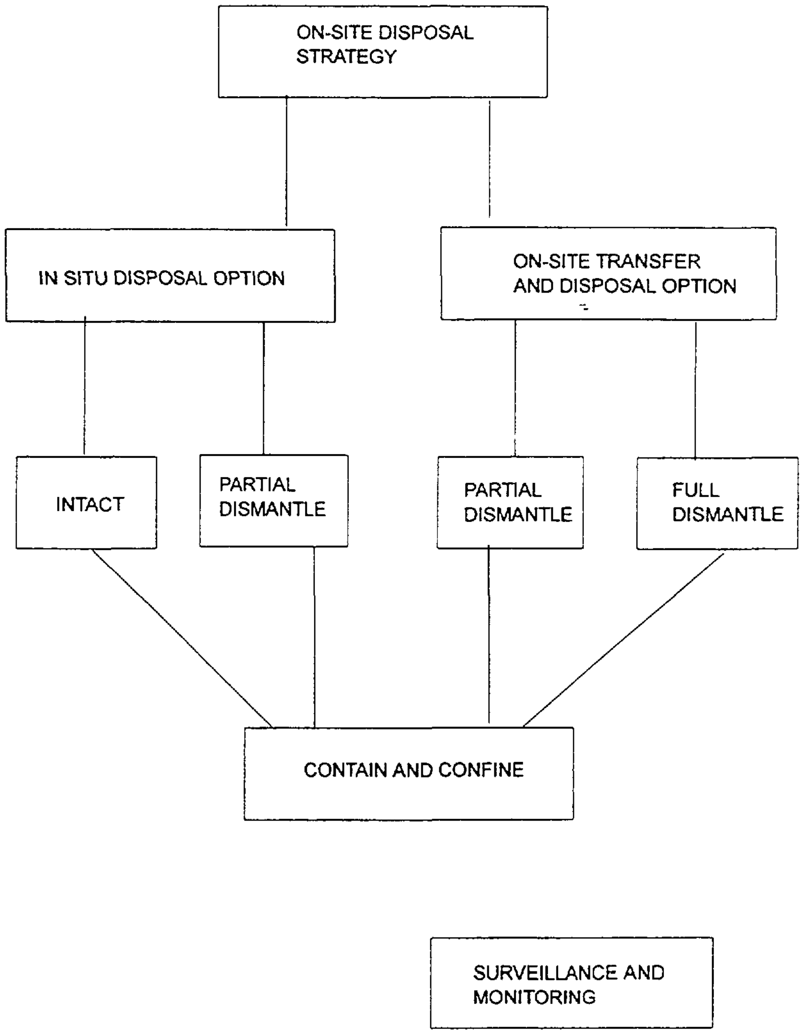

FIG. 1. Opinions available under the on-site disposal strategy.

These options may or may not require significant dismantling activities depending upon the size and complexity of the facility. The methods and techniques applied to incorporate either of these on-site disposal options are examined below. Where partial dismantling is indicated, this generally refers to the peripheral or auxiliary plant around the facility.

A very important distinction will arise when designing and licensing an on-site disposal facility. This distinction will mainly arise from the decay characteristics of the waste (short, medium or long lived). The realistic institutional control period for a site is usually considered to be between 100 and 300 years during which waste with short and medium decay periods will reduce to insignificant levels and unrestricted site access may be permitted. For long lived waste extending well beyond the institutional decay period, the safety requirements are considerably more rigourous and difficult and may preclude on-site disposal for certain sites.

On-site disposal is a form of near surface waste disposal and the IAEA has established criteria [5, 7, 14] for a shallow repository which may be used as guidance for determining whether a site is likely to be a suitable candidate for on-site disposal.

5.1. IN SITU DISPOSAL

For the in situ disposal option the major radioactive components (e.g. the vessel or primary containment structure) are disposed of at the existing location. Peripheral auxiliary plant would be removed as necessary and external peripheral radioactive materials would be placed within the primary containment. For fuel cycle facilities extensive decontamination may be necessary to remove mobile long lived contamination. The primary containment used during the operating life of the facility is likely to require enhancing as it was never intended as a long term engineered barrier for disposal. Enhancement could require the construction of an additional or secondary barrier to protect the primary containment from undue or more rapid degradation. The secondary barrier would also protect against external hazards (e.g. severe weather, flooding, aircraft impact, etc.). The secondary barrier could be constructed of durable materials such as concrete, copper or stainless steel. Alternatively or additionally, the facility could be protected further by covering with layers of natural materials such as sand, clay or rock to create an above surface mound. Erosion protection of the mound in the long term would be important and indigenous vegetation should be considered.

A few nuclear facilities have been constructed in below ground caverns. Similar considerations would need to be given to enhancing the primary containment but it is likely that the cavern would provide or be made to provide all the necessary features of a secondary engineered barrier. Filling of voids in the cavern is likely to be required to prevent subsidence in the long term. For in situ disposal the position of the mean water table level in the surrounding strata will be important. Some examples from the literature of studies examining in situ disposal are from the UK [12], from the USA [21, 22], and from Germany [23].

Encapsulation of part of the IRT reactor structure is proposed for Georgia [24]. Examples of facilities that have actually undergone in situ disposal are from Italy [25], Switzerland [26], and the USA [16–22, 27–29].

5.2. ON-SITE TRANSFER AND DISPOSAL

An alternative approach is the on-site transfer and disposal option where the configuration, complexity of operations or location of a nuclear facility are not conducive to in situ disposal . Under this option, the facility and associated contaminated and activated components are fully or partly dismantled and transferred to an on-site disposal location designed specifically for the facility being decommissioned. Since disposal is accomplished at a nearby (on-site) location, dismantling and segmentation activity can be minimized and major components can be moved intact. Because only controlled site transport is required, the movement of large and more cumbersome loads can be more easily accomplished. One major advantage of the on-site disposal and transfer option is that it allows for assessment, engineering, construction and regulatory approval of the disposal facility as a separate component of the project. That is, there is a distinction between the original operational environment and the disposal environment and associated barriers and controls. Examples from the literature of studies examining on-site transfer and disposal originate from Canada [30], Finland [31] and the USA [21, 32]. One-piece removal of the JRR-3 reactor in Japan is documented in [33].

5.3. CONSIDERATION OF FACTORS INFLUENCING IMPLEMENTATION

Since on-site disposal concepts rely on retarding the release of radionuclides for very long periods, probably beyond realistic institutional control capability, a number of factors must be considered prior to implementation; namely:

- Size, complexity and type of nuclear facility
- Inclusion of conditioned operational waste
- Location; geographic, topographical, demographic and local site conditions
- Integrity and durability of engineered barriers
- Geology and hydrogeology of the site
- Residual radioactive inventory and associated radioactivity decay profile
- Continued and future site use

These factors are discussed briefly in subsequent paragraphs.

Size, complexity and type of nuclear facility

For successful on-site disposal, it is important that the portion of the facility which contains the residual radioactivity has a configuration suitable for being enclosed within an engineered barrier. The on-site disposal strategy can be of significant benefit for very large components (e.g. a reactor vessel, building foundation structures, the monolithic concrete structures of a fuel reprocessing facility) where the sheer size and weight impose impediments to remote site disposal. Most reactors and many nuclear fuel cycle facilities are designed such that the highly radioactive areas are contained within shielded enclosures, and, in many cases, those enclosures can form part of the engineered barrier. Often, contaminated materials and equipment from auxiliary structures that support the reactor or fuel cycle facility can be dismantled and moved to within the engineered barrier for disposal. Other types of nuclear facilities (e.g. development laboratories, fuel handling facilities) may require dismantling and transfer of the radioactive materials to an engineered enclosure constructed on the site having the appropriate barrier characteristics.

It should be appreciated that on some sites different forms and levels of waste will or may have been disposed of in different locations. This will add to complexity of the site.

Inclusion of conditioned operational waste

There are usually significant volumes of operational wastes (e.g. ion exchange resins, filters, evaporator sludges, fuel processing waste, contaminated laboratory equipment, special operational wastes from non reactor facilities, etc.) generated during the operational and post-operational cleanout periods that will require disposal. These wastes could be included within the engineered enclosure for on-site disposal, provided that those wastes have been properly conditioned for disposal and meet the conditions of the safety assessment, and provided that there is sufficient space available.

Location; geographic, topographical, demographic and local site conditions

Nuclear facilities have been constructed at a very wide variety of locations. Some are in remote areas while others are close to or within large industrial or residential developments. Proximity to adequate cooling water sources is nearly always the case and these are usually provided by rivers, lakes, estuaries or the sea. If a long term decommissioning strategy is considered, the geographic location is likely to be important and many sites may be precluded. For coastal or estuarine sites erosion characteristics are important. Sites remote from population centres are more likely to be favoured, although it may be possible to consider the disposal of small research reactors within local industrial or scientific sites. The long term implications of disposal in developed areas are more critical.

For Member States that have only a few nuclear facilities located on a single site, it may be advantageous to consider on-site disposal at that site rather than develop a separate nuclear waste disposal site.

Integrity and durability of barriers

The primary purpose of an engineered barrier is to provide secure confinement of radionuclides and to retard their release into the environment. Since disposal is an irreversible process, it is usually assumed that the contained radionuclides will eventually be dispersed into the environment. The safety assessment for on-site disposal is only sustainable if adequate natural and engineered barriers are provided with sufficient durability to adequately constrain the rates of short and long term releases. The study of corrosion, leaching and dispersion characteristics of the contained radioactive material and associated release pathways from containment are part of the safety assessment. All reactors and fuel processing plants include confinement barriers appropriate for operation, but it is likely that these will need to be enhanced, probably by the addition of a secondary barrier. The type of materials and construction methods used for the secondary barrier are intended to produce the long term durability of the containment system. If there are concerns about the long term durability of the inner barrier materials when exposed to the environment, then consideration is usually given to covering the barrier with natural materials such as sand or rock, with an appropriate covering of vegetation that is self-renewing. The consequences of long term erosion of those natural materials and of the surrounding terrain on the total system containment capability is then considered. In addition, severe weather conditions, e.g. the onset of ice or flooding (permanent or temporary) is important for exposed barriers.

Geology and hydrogeology of the site

Site geology and hydrogeology are of significance in considering on-site disposal since these conditions will affect the long term capability of the containment system to retard release of residual radionuclides. Groundwater flow is of particular importance because it represents one of the primary pathways for radionuclide release. Safety assessments [14] identify the need to determine groundwater pathways and diffusion characteristics of the site geological strata (e.g. silt, clay, fractured and unfractured rock, etc.) for inclusion in the analyses of radionuclide dispersion from the disposal envelope. Site seismicity studies are important to confirm the long term integrity of the engineered barriers and the disposal environment.

Residual radioactive inventory and associated radioactivity decay profile

The radionuclide inventory and its characteristics are major factors in evaluating the long term safety impacts of on-site disposal on the environment and the public. The total amount of radioactive material and the time required to achieve background radiation conditions through radioactive decay are important in determining barrier characteristics. The radionuclides present, can be relatively short lived (up to 300 years for ^{90}Sr, ^{137}Cs, ^{60}Co, and other relatively short half-life radionuclides), or long lived fission products (^{99}Tc, ^{129}I), activation products (^{14}C, ^{36}Cl, ^{94}Nb, ^{59}Ni, etc.) and transuranics (Pu, U, Np, Am, etc.). The decay characteristics of the waste (long, medium or short lived) determine the design and safety assessment requirements for disposal.

Continued and future site use

It is important that any consideration of on-site disposal does not render the site totally unsuitable for future use. Reuse of the site for nuclear facilities is one option. Existing licenced sites are often an ideal or advantageous location for future nuclear activities. The continued use as a licensed nuclear site is attractive where feasible because surveillance and monitoring of the on-site disposal area can be supported by the ongoing activities and hence be more secure and economic. If non-nuclear uses are contemplated, then much more rigourous site clearance and controls will have to be instituted and some sites may be unsuitable for on-site disposal. Any new sites for nuclear facilities should consider all possible decommissioning strategies during the planning stages.

5.4. SUMMARY OF ADVANTAGES AND DISADVANTAGES

The advantages and disadvantages of the on-site disposal strategy relative to the other decommissioning strategies which involve removal and disposal at off-site locations (immediate dismantling, safe enclosure with deferred dismantling) are briefly summarized in Table I. These advantages and disadvantages are indicative and to some extent subjective because the strategy of on-site disposal is still developing. Many of the examples of implementation that have been given in this report were undertaken for expediency rather than a result of long term planning although this has not detracted from their success.

The suggested advantages and disadvantages given in Table I are therefore only intended to assist in weighing up different factors when selecting an optimum decommissioning strategy for a particular facility.

6. STUDIES AND EXPERIENCE WITH ON-SITE DISPOSAL

The experience base that exists from work carried out by a number of Member States on the consideration, study and use of on-site disposal as a decommissioning strategy falls into two main categories. First, there are numerous studies and proposals that examine use of this strategy. Second, a few actual projects have been implemented. A third category that has been considered is the manner in which the principles of on-site disposal can be applied to special cases, such as nuclear facilities shutdown as the result of an accident. The experience associated with each of these activities is summarized in subsections below, with additional detail presented in attached national annexes (1–11).

6.1. REPORTED STUDIES FOR ON-SITE DISPOSAL OPTIONS

Numerous studies have been undertaken, as part of a review, as part of on-going development of decommissioning strategies, or as examinations of specific special situations. These studies are summarized below.

TABLE I SUMMARY OF ADVANTAGES AND DISADVANTAGES OF ON-SITE DISPOSAL

ADVANTAGES		DISADVANTAGES	
Item	Comments	Item	Comments
Reduced cost	• Reduced dismantling • minimal off-site transport and disposal	Need for long term maintenance and surveillance (institutional period)	• may be multiple sites to monitor
Reduced worker dose	• Less dismantling • reduced waste handling	Difficulty of licensing and gaining public acceptance	• Change of site use from operations to disposal was never planned
Minimal off-site transport of waste	• Bulk of waste disposed of on-site	Proximity of site to population centres	• May only apply to some sites • public resistance
Reduced public interaction	• Fewer off-site activities	Very long term site commitment	• Change of site use
Continued use of existing site support facilities	• Less cost, use of existing personnel and infrastructure • less training	Increase of waste disposal sites within the country	• Public acceptance issue • Increases burden on future generations
Reuse of site and facilities for nuclear applications	• Uninterrupted nuclear licensing of the site	Deferred release of site for other uses	• May remain restricted for very long periods
For some nuclear programmes avoids need for central repository	• Country specific and may not always apply	Disposal may preclude other nuclear facilities	• Site size may be too small
Early on-site disposal may reduce monitoring costs	• Less surveillance than safe enclosure	May only be acceptable for certain nuclides	• long lived nuclides may be precluded
Possible early release of parts of site for non nuclear use	• Reduces boundary of licensed site	May be multiple disposal areas on the site	• more complexity for monitoring
Early disposal of nuclear facility eliminates future decommissioning activities	• Reduces burden on future generations	Multiple sites needing monitoring and surveillance	• increased overall cost
Severe site contamination may require on-site disposal	• accident or severely contaminated site	Additional complications in case site remediation is required in the future	• Extra barriers may render dismantling of the disposed structures more difficult
On-site transfer allows good design of new disposal facilities	• more robust safety, optimum design of barriers		

An environmental impact statement [21, 32] was prepared on alternatives for decommissioning of the original eight shut down plutonium reactors at the Hanford Site in the USA via on-site disposal. These analyses examined costs, radiation doses, and safety considerations for each of the alternatives, one of which was in situ disposal by mound burial. Other alternatives included immediate one piece removal, delayed dismantling and removal, and delayed one-piece removal with transport and disposal to another on-site location. Subsequent analyses [34] have focused on the preferred alternative (delayed one-piece removal).

An analysis for estimating decommissioning costs for the major fuel cycle facilities on the Hanford Site [22] examined several possible strategies, including dismantling and removal; dismantling, and transfer of radioactive materials to a regulated low level waste (LLW) disposal facility, with in situ disposal of non-radioactive materials; and in situ disposal of both radioactive and non-radioactive materials. No preferred alternative was reported.

An environmental assessment was prepared on alternatives for decommissioning of the Waste Calcining Facility at the Idaho National Engineering Laboratory in the USA [35]. The preferred alternative was in situ disposal by sealing all routes to preclude moisture ingress, grouting of all below-grade tanks and work areas, dismantling of above-grade structures and covering the encased process equipment and rubble with a concrete cap.

The use of shutdown reprocessing plant canyon buildings for permanent near-surface disposal of low level waste is being analyzed at the Hanford Site, with the canyons to be filled with packaged wastes and backfilled with grout [36]. Other low level waste may be placed around the outside of the buildings and the whole system covered with moisture-resistant covers and soil for final disposition.

Alternatives for final disposition of the Heavy Water Components Test Reactor at the Savannah River Site have been analysed [37]. While both dismantling and entombment were considered equally viable, dismantling was recommended as better meeting the goals of the long-range plans for the facility site.

Canadian studies have been concentrated on evaluation of on-site disposal for decommissioning CANDU reactors [30, 38, 39]. Three distinct on-site disposal options were examined as follows:

- emplacement underground directly beneath the reactor location;
- emplacement underground adjacent to the reactor building; and
- emplacement in a surface or underground vault within the nuclear station boundary.

The primary factors considered in these studies are cost, radiation dose, site feasibility, safety, environmental and regulatory aspects and public acceptance. The general conclusion is that the approach is technically viable for some sites. Future work must address the social and regulatory acceptability.

Similar studies are planned for two Canadian research reactors that are below ground installations extending into bedrock.

A study was conducted in Germany to evaluate the technical feasibility, costs and other impacts of on-site sinking of a reference reactor building [23]. The approach uses the caisson sinking technique with a closed working chamber beneath the building. Another technique, investigated in less detail is the sinking of the centre of the building, that comprises the biological shield, into a pre-constructed storage chamber. In its final state, the overall structure is a multibarrier system effective over long periods of time, confining the enclosed radioactive substances and forming a final repository. During the estimated five years of decommissioning work, the collective dose to the workforce will be between 0.75 and 0.95 person-Sv (a factor of 10 lower than immediate dismantling of the reactor building). Postulated accident scenarios result in negligible doses to the public. However, the intrusion scenario was not assessed. Concerning long term safety the study indicates that no releases to the environment are expected for thousands of years, even in the case of a long-lasting increase of groundwater corrosivity.

On-site disposal has been studied as part of its decommissioning plans for the Loviisa WWER reactors in Finland. The intention is to remove the pressure vessel, steam generators and pressurizer intact to the final repository, a rock cavern 100 m below the plant site surface and some 1 km from the power plant [31, 40]. The pressure vessel will be used as a waste package for the most active part of the plant (i.e. the reactor internals and the dummy fuel elements which have been used to prevent embrittlement of the pressure vessel material). The vessel will then be filled with concrete and closed with its original vessel head. The steam generators and the pressurizer will be stored in the same cavern as the reactor above the reactor prior to sealing. Detailed work plans have been produced.

In two BWR reactors, Olkiluoto 1 and 2, operated by TVO in Finland decommissioning plans include segmentation of the reactor vessel and internals in a specially prepared work area within the reactor hall [41]. Resulting radioactive waste is emplaced in a final repository constructed 70–100m below ground in the crystalline rock of the power plant site. Accommodation of the decommissioning waste requires an expansion of the existing site repository. An additional option to remove the reactor pressure vessel intact using it as the waste package for the reactor internals has also been studied. The reactor vessel package is then transferred to the on-site repository. This approach is technically feasible and is the least expensive option [42].

Between 1989 and 1992, the UK nuclear electrical utility, Nuclear Electric, completely re-evaluated its decommissioning strategy, looking at nine different options and resulting in the selection of a proposed new strategy, Deferred Safestore. A multi-attribute decision analysis including three categories of factors, namely (i) environmental/safety, (ii) technical, and (iii) cost, showed that Deferred Safestore, is optimal. Dismantling takes place after 135 years of Safestore. The in situ option was studied in some detail [12] but did not score particularly well in the multivariate analysis because of the uncertainty in being able to secure the detailed safety assessment. There are long lived isotopes in the waste inventory although these are in activated material. Nevertheless, from the work conducted to date, it appears that an adequate safety assessment could be made for most Magnox stations, and possibly for AGRs. If the safety assessments can be made in detail and if the proposal was acceptable on wider grounds, the in situ disposal option becomes the most attractive choice particularly in terms of cost and dose to workers. Nuclear Electric therefore considered it highly desirable to retain this option as an alternative to eventual dismantling [12, 43, 44]. In the Nuclear Electric proposal, in situ decommissioning involves constructing a stable mound over the buildings and, in effect, burying them. All voids would be filled during construction of the mound, which would largely be formed by pumping sand from the local seabed or river [12, 44].

A somewhat innovative concept was recently illustrated, consisting of locating a nuclear power plant or parts thereof such that neither dismantling nor removal from the site are necessary (e.g. construction in an underground vault). The immediate implication of this concept is that a site would require licensing not only for construction and operation, but also for long term disposal [30, 45–47].

In the Russian Federation, on-site disposal is considered as an option for decommissioning of RBMK, WWER and fast breeder reactors at several sites. The final step of decommissioning is envisaged as either removal, safe enclosure or the in situ disposal option. Studies considering the in situ option propose protective barriers to inhibit radionuclide release into the environment for the entire radiation hazardous period. Since the geology of many reactor sites is not suitable for deep disposal, only near surface disposal is considered. Near surface disposal is limited to short lived radionuclides when the hazardous period of the inventory does not exceed the planned integrity of the engineered barriers (almost 500 years).

Although not strictly an in situ decommissioning study, the Georgian proposal [24] to entomb the IRT activated reactor components in the lower third of the reactor tank does incorporate several elements of the in situ disposal option. The initial step, by placing concrete to confine the existing radioactive material in the lower part of the tank, releases the upper tank portion for reuse. Future plans for dismantling and disposing of the total research reactor complex have not been developed. However, consideration could be given to disposal of all reactor components and waste within the monolithic reactor shield structure.

6.2. EXPERIENCE WITH ON-SITE DISPOSAL

Much of the on-site disposal experience has occurred in the USA and has extended from the 1960s to the present. No large commercial facility has been attempted. A project in Japan is reported but does not specifically imply disposal. Italy has accomplished on-site disposal of a small research reactor.

These experiences are discussed briefly below, with details given in the annexes.

Experience in the USA with the on-site disposal strategy began in the early 1970s with the BONUS [16], Piqua [17, 18], and Hallam [19, 20] power demonstration reactors. In all three cases, all special nuclear materials (i.e. fresh and irradiated fuel) were removed from the sites. The reactor vessel or block and associated contaminated activated equipment was contained within a sealed barrier enclosure, and external plant structures were removed. Because of the rather short operating life of these reactors (3 years or less), it was anticipated that the contained radioactivity would decay to unrestricted release levels within a reasonable institutional control period (120 years). Recently, portions of the sites of two decommissioned small prototype reactors at the Idaho National Engineering and Environmental Laboratory (INEEL), the Stationery Low Power Reactor I (SL-I) and the Boiling Water Reactor Experiment I (BORAX-I), which contained residual contaminated soil and debris and an old on-site burial ground, have had caps installed to prevent erosion and water intrusion. The caps are comprised of gravel, cobble and large rocks. Also at INEEL, the complex containing BORAX-II, III, IV, and V has recently been decommissioned by removing auxiliary structures, removing hazardous and contaminated material, backfilling the reactor pits which contain the reactor vessels with clean material, reinstalling the shield cover blocks, contouring the site and seeding with natural grasses [29].

A small test reactor (Air Force Nuclear Engineering Center Reactor) built to support studies on nuclear aircraft propulsion was entombed at the Wright-Patterson Air Force Base in Ohio in 1971 after about 3 years of operation. Non-structural radioactive components were removed, and radioactive cavities were filled with sand. Openings in the concrete biological shield were sealed by welding, and additional concrete shielding was placed to reduce radiation levels in accessible areas to <2 Sv per hour. The facility enclosure was upgraded in 1987 and remains in the entombed state today ([37], page 30).

Two other examples of on-site disposal at the Hanford Site are the Hot Semiworks Complex [27] and the 183-H Solar Evaporation Basins [28]. At the Hot Semiworks, the process facility and equipment, exhaust filter and stack rubble, and local LLW burial locations were covered with an engineered barrier consisting of a layer of ceramic warning blocks to deter any future intruders, coal ash, sand and gravel, a fabric cover, and revegetated top soil. At the 183-H basins, the structures were decontaminated with the contaminated materials removed from the site, and the structures demolished. An engineered barrier was placed over the back-filled basin locations to minimize water infiltration and erosion. Because a plume of chemical contamination remained beneath the back-filled basins, groundwater monitoring in that locale is expected to continue for up to 30 years.

Low-activity wastes from reprocessing waste storage tanks are being mixed with grout to form a matrix called Saltstone and placed into concrete vaults for disposal at the Savannah River Site. The same type of wastes are planned to be vitrified in steel containers and placed into concrete vaults for near-surface disposal at the Hanford Site [37].

Although not specifically an on-site disposal project, removal of the Japan Research Reactor No. 3 (JRR-3) to a storage facility adjacent to the reactor does demonstrate a major activity of the on-site transfer and disposal option. The removal of the 2250-ton reactor block (10 m 10 m 10 m) was performed as part of a programme to replace the JRR-3 core with an upgraded reactor core. The reactor block was raised about 3.7 metres, using a 12-cubic metre steel frame and a centre-hole jack system. The reactor block was then transported horizontally to a storage facility about 34 metres away using four 100-ton jacks mounted on steel rails. Finally, the reactor block was lowered 14 metres into the storage facility [33].

A small project was undertaken in 1987 in Italy on the decommissioning of the RB-1 research reactor at Montecuccolino [25]. Due to the low active inventory and dose (mainly from cobalt), it was proposed and agreed by the regulator that the reactor vessel could be left in the vessel well and capped over with concrete. Decay of the remaining active material is likely to reach clearance levels in a reasonably short time (less than ten years).

6.3. SPECIAL CASES

The containment of nuclear facilities where shutdown has been the result of an accident have some of the features of and incorporate the principles of on-site disposal to some extent. The post accident policy has generally been aimed primarily at interim control of radionuclide releases and at providing a secure storage period to allow radionuclide decay prior to final disposition. Accordingly, these projects are not identified as final disposal. Documented experience is outlined briefly below, with additional detail provided in the annexes.

The decommissioning strategy for the damaged Lucens reactor in Switzerland [26] may have been selected partly because of the underground construction of the facility. On completion of initial decontamination and dismantling work and following a period of natural decay, only 2.2 MBq of radioactivity remained in the reactor cavern in 1992. Due to the low residual activity and the presence of significant water flow at the site, a long term strategy of slow radioactivity release coupled with high river dilution was selected. The annual radiation doses to the critical off-site group were estimated to be negligible (2.6 Sv).

Another significant case is the well-known Chernobyl sarcophagus (or shelter). This term is applied to the whole complex of structures enclosing sources of radioactivity around Unit 4 of the Chernobyl NPP, which was destroyed in an accident in 1986. Immediately following the accident, 5000 t of boron compounds, dolomite, sand, clay and lead were dropped by helicopter onto the damaged reactor in an attempt to keep the fuel rubble subcritical and to control the discharge of radioactive material to the environment. More information on constructional features, monitoring and long term concerns of the sarcophagus are given in [48, 49]. As the sarcophagus was constructed under emergency conditions, it is believed that it is unstable and not capable of resisting earthquakes which are a risk in the Chernobyl area. During 1994–1995 a study was undertaken by the international consortium Alliance to assess the feasibility of constructing a new containment/shelter over the damaged Chernobyl reactor and its sarcophagus to effect on-site disposal. This study is described in the technical literature, [50]. However, the project, which was estimated to cost well over US $ one billion, was judged too costly and no practical implementation has followed the study [51, 52]. New concepts have been proposed and contracts are being negotiated for repair/stabilization of the shelter [53]. No decision has been made to declare Chernobyl a disposal site.

7. SUMMARY

The report has attempted to cover all relevant aspects of on-site disposal. It recognizes that on-site disposal is a very site specific strategy and there could be difficulties in licensing and gaining public acceptance. In summary the following topics were addressed:

A review was done of the on-site disposal strategy and attempts were made to define it more clearly.

Factors were examined which would influence the selection of on-site disposal as a strategy, particularly technical considerations such as cost and dose commitment, safety and environmental aspects, as well as political issues and public concerns. A table has been presented which lists the relative advantages and disadvantages of this strategy.

The options available within the strategy were elaborated and it was proposed that there are two main alternatives namely; in situ disposal (where the facility is disposed of within its original location) and on-site transfer and disposal (where varying amounts of dismantling are undertaken prior to relocation to an on-site disposal area).

In discussing the strategy, comparisons were made with the other two main decommissioning strategies which are early dismantling and deferred dismantling, the latter including safe enclosure, all followed by unrestricted release and reuse of the site.

It has been recognized that the inclusion of long lived radionuclides in the inventory for disposal will make licensing more difficult and a much more robust safety assessment will be needed for periods beyond the institutional control period.

A number of studies of on-site disposal were included and reviewed. These studies describe a variety of approaches to the subject and generally cover the range of options proposed in this report.

Some on-site disposal projects have been completed and the experience from those projects has been presented. Information was derived from the available literature.

A section on special cases especially dealing with decommissioning after accidents has been included. Not all of these cases have necessarily been declared disposal sites, but some may inevitably be candidates for on-site disposal.

8. CONCLUSIONS AND RECOMMENDATIONS

The following conclusions were drawn from the material collected and from discussions during the preparation of this report:

The on-site disposal strategy is a viable option that should be given due consideration.

This strategy depends on-site specific conditions and is not universally applicable.

There are valuable benefits that can be accrued from this strategy particularly in terms of cost, dose commitment and independence from off-site waste disposal facilities.

Examples and experience have shown that on-site disposal has been successfully accomplished.

Experience from shallow waste burial sites and mining/milling waste disposal should be taken account of in proposals for on-site disposal as a decommissioning strategy.

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ANNEXES

Eleven annexes have been included based on submissions offered by Member States or have been excerpted from published literature. Some of these are specific examples of where on-site disposal has been studied in some detail and also where implementation has been completed. Other annexes have been included where particular decommissioning activities e.g. on-site transfer of large components illustrate important aspects of on-site disposal. It should be noted that where particular decommissioning activities have been included to illustrate aspects of on-site disposal, this does not imply that the Member State has actually declared a policy of on-site disposal.

For Member States where detailed studies have been done e.g. UK, Germany and Canada, the option has been highlighted but any decision on implementation has been deferred. Even where implementation for certain sites has been successfully completed e.g. the USA, there is not a clear policy that this strategy will be adopted for other sites in that Member State. Finland has declared a policy of on-site disposal for their reactors.

Illustration of specific on-site disposal strategies does not necessarily imply compliance with IAEA Safety Standards.

ANNEX I

CANADA

On-site disposal of CANDU reactors has been evaluated in Canada for some time [I-1–I-3]. Several options specific to Ontario Hydro multi-unit reactor stations have been evaluated and reportedly provide an opportunity for significant reduction in decommissioning costs. Three on-site disposal options are described in [I-1] as follows:

- Underground beneath the reactor. In this option, the reactor unit and other large components are removed as integral assemblies and emplaced in an engineered vault underneath each reactor unit. This option requires a mined shaft and an emplacement vault about 53 m deep (Figs. I-1, I-2).
- Underground adjacent to reactor building. This option is similar to the previous one, except that the emplacement shaft and the vault are outside the reactor building (Fig. I-3). This option makes the removal of reactor units and excavation of shafts and vaults simpler. However, it requires a robust local transportation system.

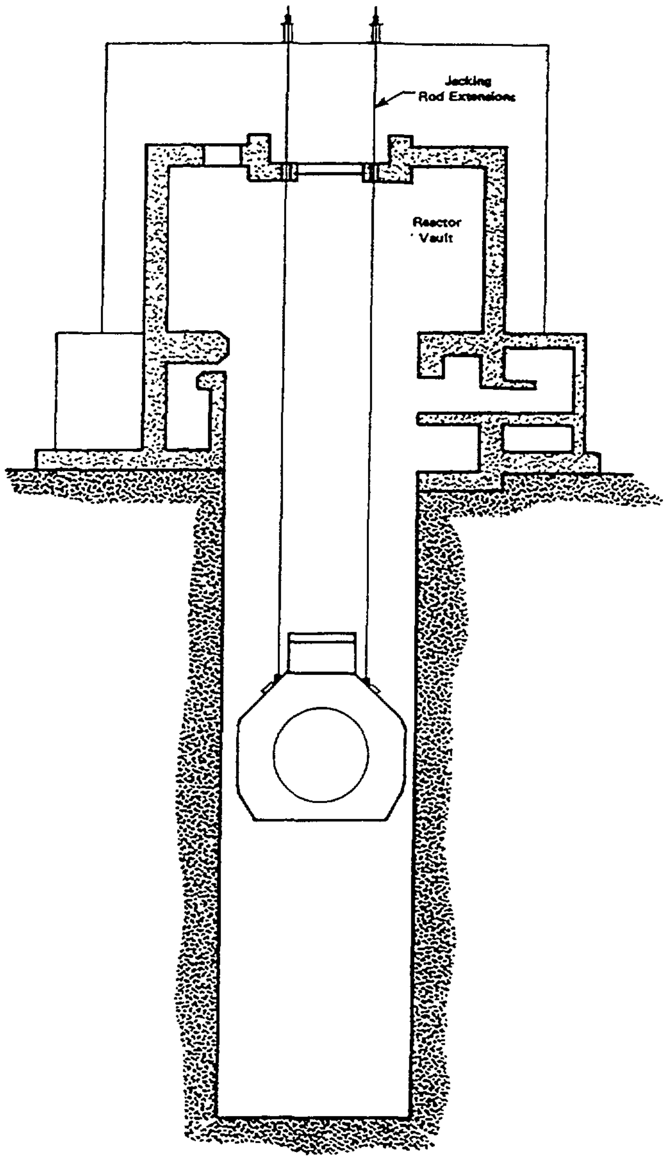

FIG. I-1. Bruce NGS A decommissioning one-piece removal and on-site burial (reactor is lowered into burial pit).

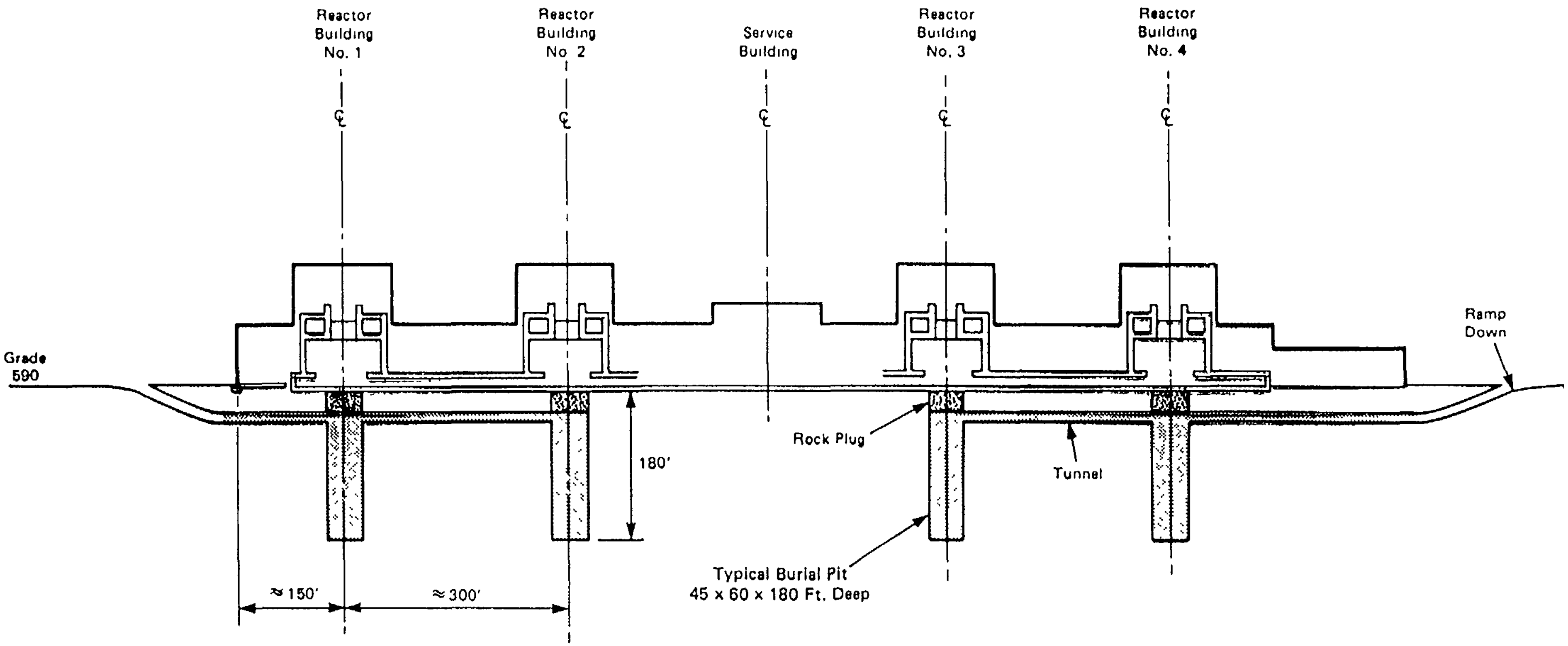

FIG. I-2. Bruce NGS A decommissioning one-piece removal and on-site burial (east-west longitudinal section through station looking north).

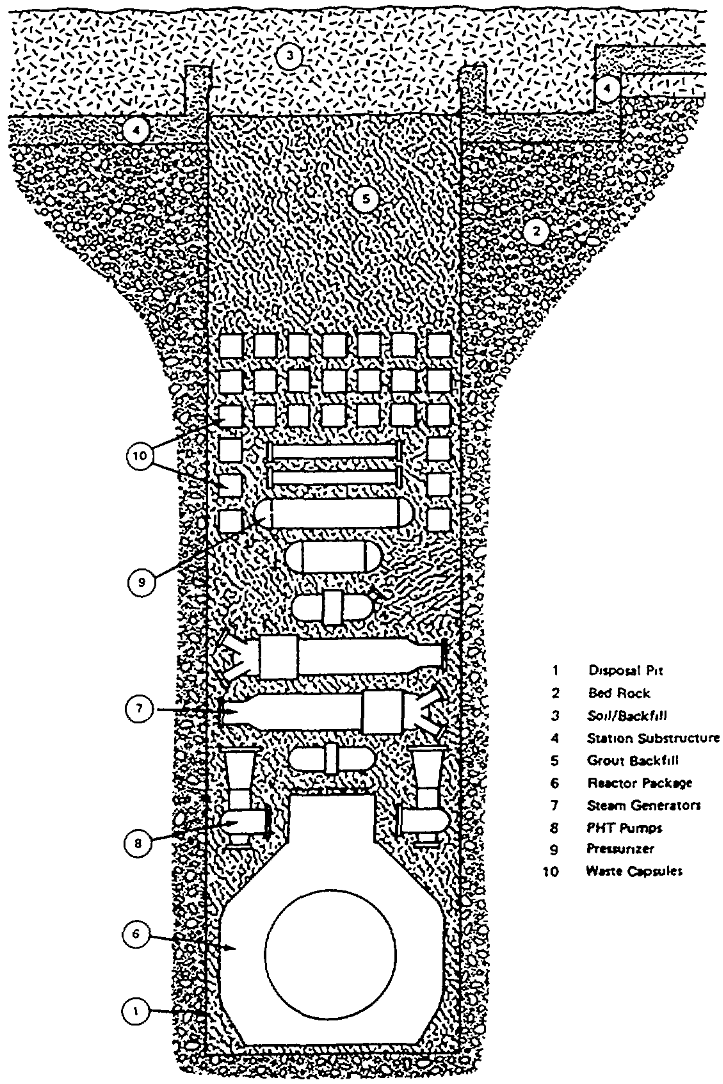

FIG. I-3. Bruce NGS A decommissioning one-piece removal and on-site burial (final arrangement of burial pit).

- Centralized in situ option. In this option, reactor assemblies and components from each unit with a multi-unit site are transported by a rail system to a common vault. The vault can be at ground level in embankments, near surface in trenches, or underground in a mined vault, all within the licensed site boundary. The common site is shared by all reactor units at that site. This option allows better restoration of station sites for reuse, since the reactor sites will be free from underground structures such a shaft and vaults containing buried components. It also has the advantage that underground activities for all units are centralized, providing engineering and cost advantages. There is also flexibility in choosing a location that better matches the geological and other environmental protection requirements. This option also requires a robust on-site transportation system to move material from each reactor to the central site.

The Ontario Hydro on-site disposal studies were undertaken primarily to present alternative decommissioning concepts for CANDU reactors for some sites (particularly the Bruce Nuclear Generating Site). Factors examined include cost, radiation dose commitment, site feasibility, safety, environmental and regulatory aspects and public acceptance considerations.

The studies conclude that on-site disposal offers a technically viable, low cost approach to decommissioning of CANDU reactors, particularly for remote sites. Encapsulation of the activated materials in existing primary containment with stabilization in a backfilled vault indicates adequate radioactive decay before any release to local waterways would occur.

Current work associated with Ontario Hydro reactors is focused on conventional decommissioning methods to demolish, process and remove wastes from the reactor site.

Similar studies are planned for two Canadian research reactors, the Nuclear Power Demonstration Reactor (NPD) and Whiteshell Reactor-1 (WR-1) (Figs. I-4, I-5). These reactors may benefit from the in situ option because both are below ground installations extending into bedrock. This condition allows naturally existing barriers (bedrock, clay soil structure, etc.) to be utilized as part of long term containment for the reactor and contaminated/activated components.

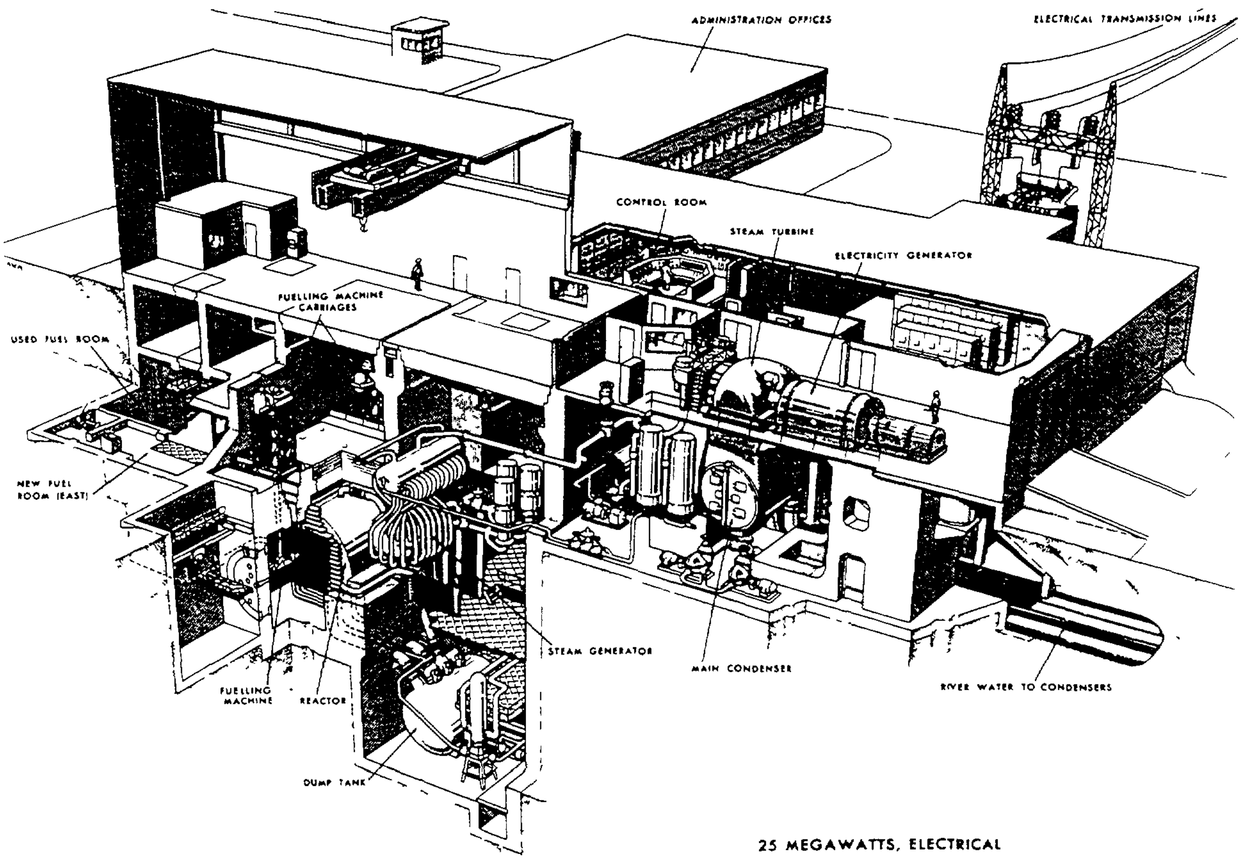

FIG. I-4. Nuclear power demonstration station Rolphton, Ontario.

Initially, these new studies are concentrated on the NPD reactor with current work in three areas as follows:

- a review of the characteristics of the radionuclide inventory;
- system materials corrosion/degradation effects; and
- identification and evaluation of possible barrier locations and materials.

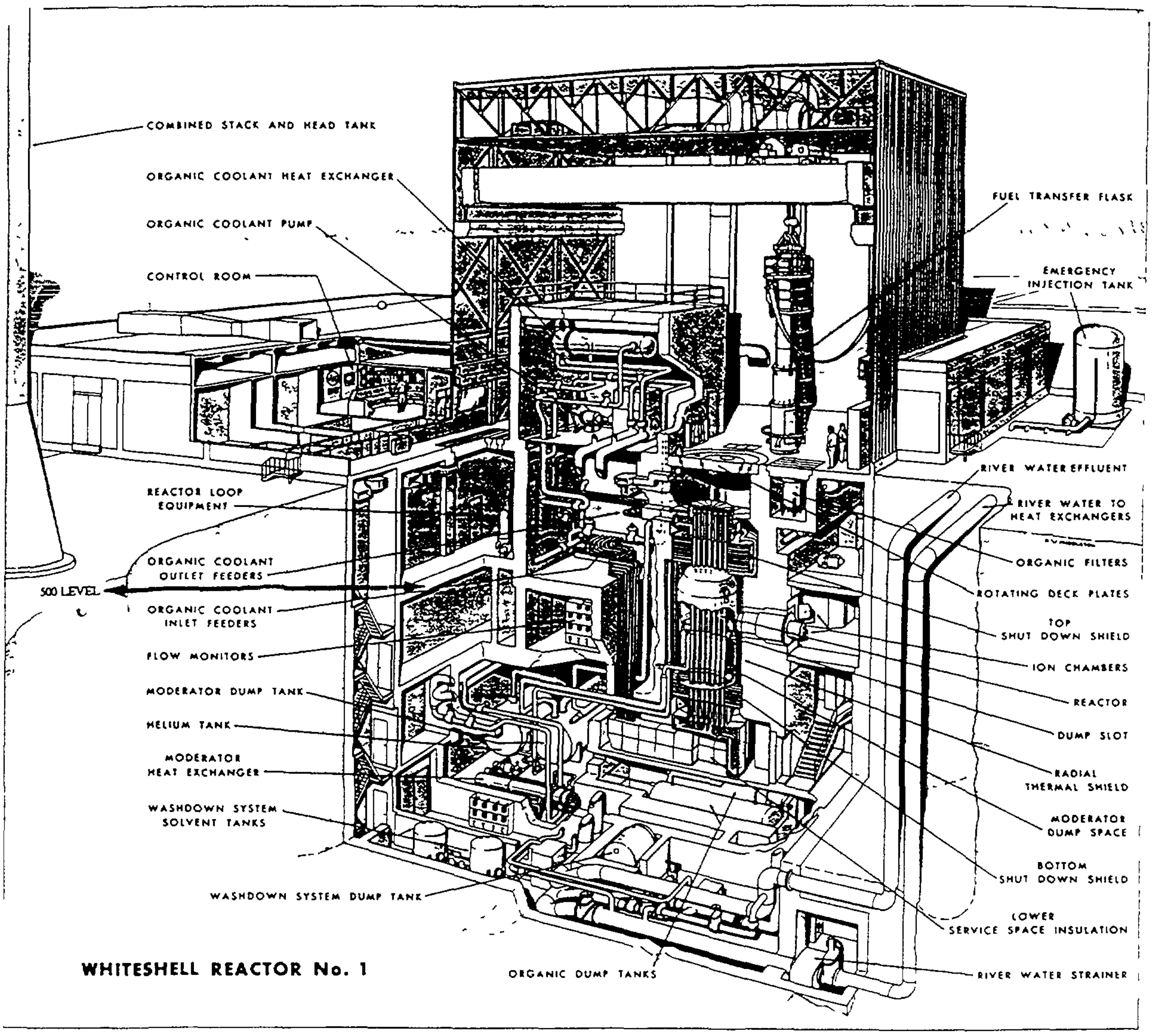

FIG. I-5. Whiteshell Reactor No. 1.

The evaluation will focus on the review and verification of existing radionuclide inventory data and will capitalize on scientific work already conducted on materials corrosion effects and on vault barrier materials as part of the Canadian nuclear fuel waste management program. The emphasis will be on how buffer materials in the disposal environment can reduce corrosion rates and hence diffusion rates and how barrier materials can be applied to inhibit release rates.

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Annex II

FINLAND

The structure of this paper is based mainly on the reference [II-1].

Nuclear power plants in Finland

The Finnish nuclear power programme consists of operation of four nuclear power plant units. Imatran Voima Oy (IVO) operates two PWR units, Loviisa 1 and 2 (2 445 MW(e)) at Hästholmen on the south coast of the Finland. Loviisa 1 started commercial operation on May 1977 and Loviisa 2 in January 1981. The other two units, Olkiluoto 1 and 2 (2 710 MW(e)) are BWRs and are operated by Teollisuuden Voima Oy (TVO) in Eurajoki on the west coast Finland. Olkiluoto 1 started operation in September 1978 and Olkiluoto 2 in February 1980.

Each plant has operated well — with high load factors and low personnel doses. Today both plants are also increasing power by 9–16%. Also the extension of plant life is now under discussion.

Environmental impact assesment studies for the new units has been started.

Low- and medium-level operating waste storage

Each nuclear power plant has an onsite final disposal facility for low- and medium-level wastes.

The repository at Olkiluoto was licensed and opened in 1992. It comprises two silos, one for low-level solid waste and the other for intermediate-level solidified waste. Both silos are situated in bedrock 70–100 metres below ground level. The silos are 24 meters in diameter and 34 meters deep. Waste is packed in 200 liter steel drums which are placed in concrete containers for disposal. The self-supporting containers are stacked in the silos without any additional supporting structures.

The repository for the operating waste in Loviisa was commissioned in 1997. The repository is located on the plant site in bedrock at depth of 110 meters below the ground level. Operating waste will be disposed of in two tunnels which have concrete floors and shotcreted walls. A hall for solidified waste has also been excavated.

Decommissioning of the nuclear power plants

The Finnish safety authorities have not set any specific regulations for the power plant decommissioning. However, the authorities require that the technical plans and cost estimates be updated at five year intervals. Hence, decommissioning and subsequent waste disposal are included as part of the overall waste management plans of power plants.

Like the final disposal of low- and medium-level operating wastes, the decommissioning wastes will be disposed of in the power plant site repositories. The principle is that the existing disposal repositories for operating wastes will be extended with new silos and tunnels in the future. The waste volumes expected from dismantling are of the same order of magnitude as the operating wastes.

The decommissioning plans can be based on immediate or deferred dismantling. A storage period between shutdown of the reactor and dismantling means that when dismantled the activity of structures and systems is lower than immediately after shutdown. The cooling period of a few decades would not prolong the timetable for the Finnish nuclear waste management plan, because the timetable for interim storage and final disposal of spent fuel covers the same period.

It has been shown that decommissioning will not cause any harm to the environment or to the public health, [II-2].

Technical plans for dismantling and waste disposal

The Loviisa units

The decommissioning plan for the Loviisa units is based on immediate dismantling after the end of the service life of the plant.

Radioactive dismantling wastes are classified into activated and contaminated wastes. The activated wastes include the reactor vessel and the reactor internals, control rod absorbers and extenders as well as thermal insulation plates and the biological shield of the reactor.

The contaminated dismantling wastes consist of process systems and structures from the reactor building, auxiliary building and waste buildings as well as the structures of the fuel storage pools and the radiolaboratory.

The wastes are placed in the on-site repository in the crystalline rock of the power plant site by expanding the existing repository for low- and intermediate-level operating wastes (see Fig. II-1).

The rock cavern is equipped with a bridge crane to facilitate the component handling in the cavern. The pressure vessel will be removed intact into the on-site repository along with the

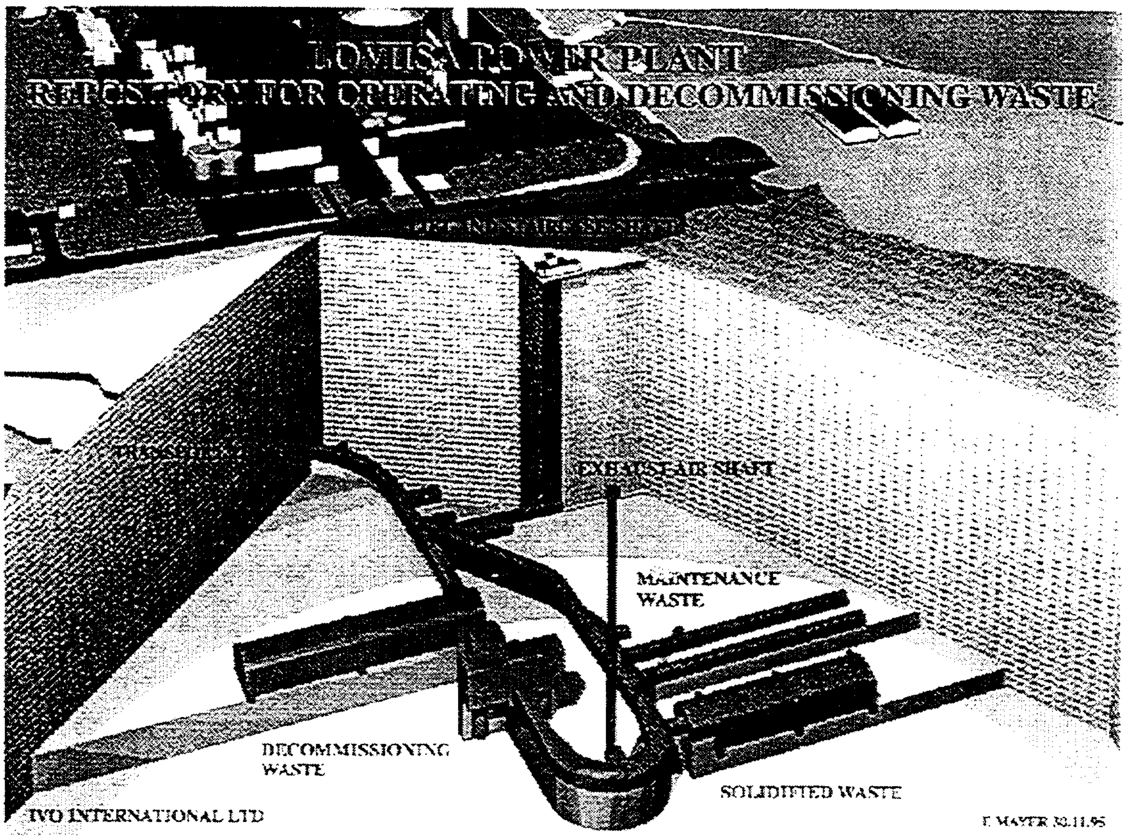

FIG. II-1. Final disposal facility of the Loviisa plant.

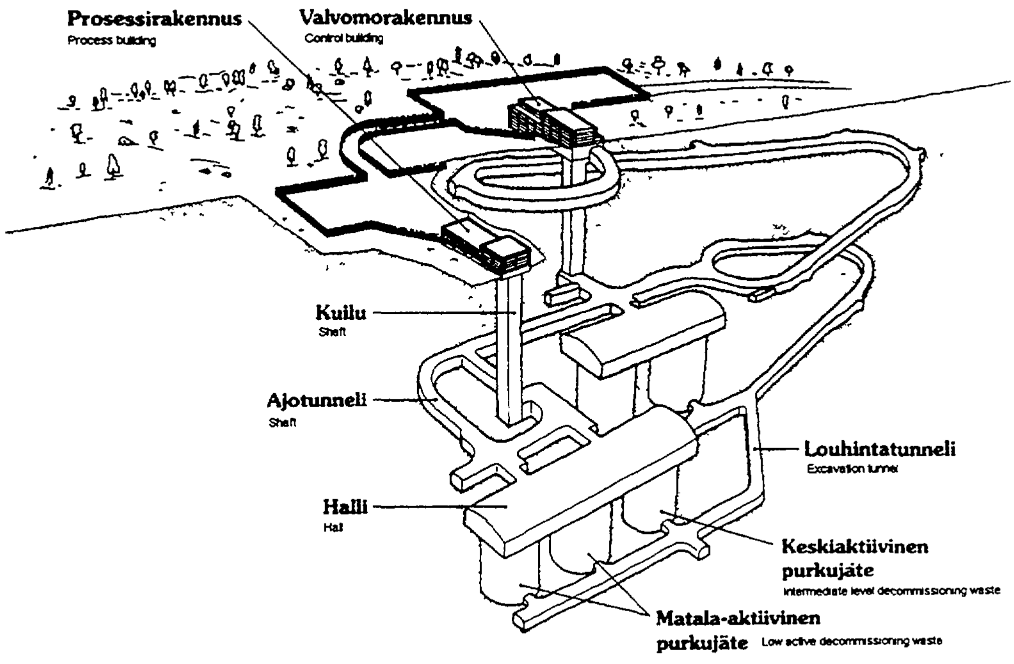

FIG. II-2. Final disposal facility of the Olkiluoto plant.

steam generators and pressurizers. The pressure vessel will be used as the waste package for the most active part of the plant, i.e. the reactor internals and the dummy fuel elements, which have been used to prevent embrittlement of the pressure vessel material.

Following placement and sealing of the reactor vessel in the bottom of the cavern the steam generators and the pressurizer will be stacked above the reactor before sealing the cavern (see Fig. II-3).

Reactor internals will be removed and transported using the steel shielding cylinder, originally used for power plant refuelling. Other wastes are packed in concrete or wooden containers.

The quantities of waste include 2600 tonnes of activated wastes and 5100 tonnes of contaminated wastes. The total volume of waste including packaging is approximately 12400 m^3. Total excavated volume for the decommissioning waste is about 45 000 m^3.

The Olkiluoto units

The decommissioning plan for the Olkiluoto units is based on deferred dismantling after safe storage of 30 years.

Radioactive wastes from dismantling and other wastes to be disposed of at the time of dismantling are classified into three main groups; activated waste, contaminated waste and very low-activity concrete.

The activated waste includes the reactor vessel, the reactor vessel internals, the inner layer of the biological shield and the thermal insulation of the reactor vessel.

According to the decommissioning plan, a chamber is constructed in the reactor hall for segmentation of the reactor pressure vessel and of the internals. The components are lifted in the chamber, where cutting and packaging is done by remote-controlled techiques. The biological shield is segmented in place. The activated waste is packaged into concrete containers or steel lined concrete containers dependent on radioactivity.

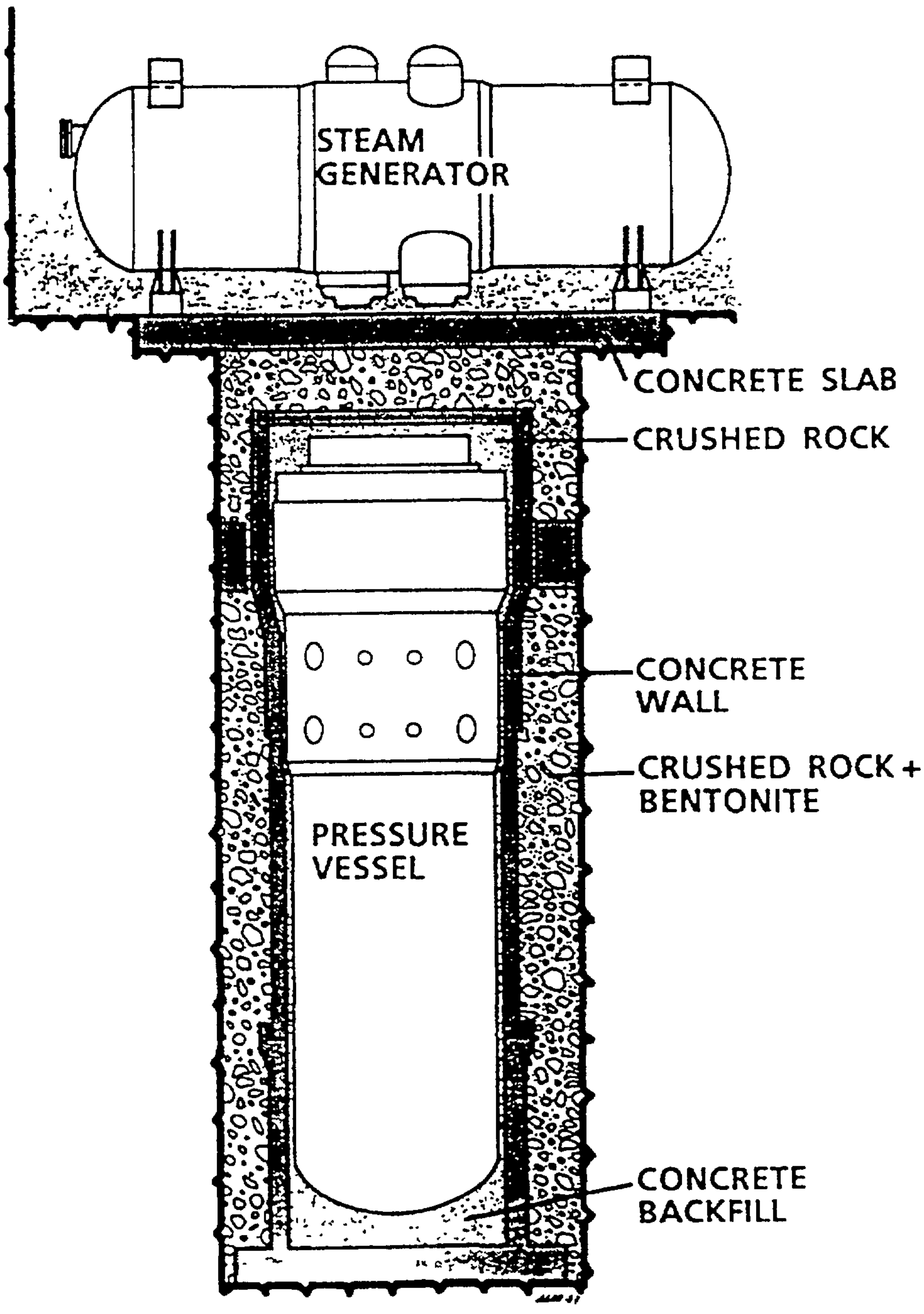

FIG. II-3. Closed reactor silo for Loviisa plant.

The contaminated dismantling waste consists of two categories; waste from dismantling of process systems and from fuel racks.

Contaminated systems are classified according to estimated activity levels. The actual activity levels of different systems and components will be measured prior to dismantling. Systems classified as radioactive are removed and packaged mainly in concrete containers. For pipe removal, a suitable cutting method is selected depending on the diameter and radioactivity of the pipe. Other items to be removed and packaged are valves, pumps and isolation materials. No volume reduction methods have been planned. The large components (e.g. tanks) are transferred in one or more pieces, without packaging directly to the repository. The fuel racks are also disposed of intact without cutting or packaging.

Very low-activity concrete, mainly from the dismantling of the outer layers of the biological shields, is packaged in wooden boxes.

The quantities of waste include 2500 tonnes of activated wastes, 5000 tonnes of contaminated wastes and 2600 tonnes of low-active concrete in total about 10 000 tonnes. The total volume of waste including packaging is approximately 23 $600m^3$. The total volume of excavated rock is about 96 $000m^3$.

In addition to the actual dismantling wastes, the activated core components replaced during the operational period (fuel channels, control rods, core instruments) will be disposed of in the same repository. Furthermore, the decommissioning wastes from the interim storage facility for spent fuel and from the fuel storage racks will be taken into account in the final repository plan.

The radioactive waste is emplaced in the on-site repository in the crystalline rock of the power plant site by expanding the existing repository for low- and intermediate-level operating wastes (see Fig. II-2).

Latest developments

As there are still decades before the dismantling of the power plants will be implemented, it is possible to develop and optimize the decommissioning plans as well as to utilize the experience obtained from projects in other coutries.

The following two studies were performed for comparison of alternatives: deferred dismantling of the Loviisa units (instead of dismantling soon after shutdown) and final disposal of the Olkiluoto reactor vessel in one piece (instead of segmentation of the vessels).

Deferred dismantling of Loviisa plant

The strategy for management of the Loviisa spent fuel was changed in 1995. Under the new strategy spent fuel will be stored at the power plant area longer than earlier planned. A prolonged schedule could offer possibilities for optimization of the decommissioning strategy, because administration of surveillance would not cause additional costs for overall waste management at Loviisa. However, a study [II-3] has shown, that it is not economically feasible to defer the dismantling.

Intact removal of Olkiluoto pressure vessel and its use as a waste package

The placement of the Olkiluoto reactor vessel intact into the repository could be a simpler and more economical method than segmentation. This approach would remove a need to construct separate segmentation chambers in the reactor halls and eliminate segmentation. In addition, disposal space could be conserved by placing activated internals inside the vessels before placing it in the repository.

A study [II-4] has shown that it is techically and economically feasible to remove the reactor pressure vessel intact and to use it as the waste package. It is also a good solution from a safety point of view [II-5]. The total weight of the packed reactor is about 1000 tons. The reactors may be positioned vertically in the independent shaft at the plant site, see Fig. II-4.

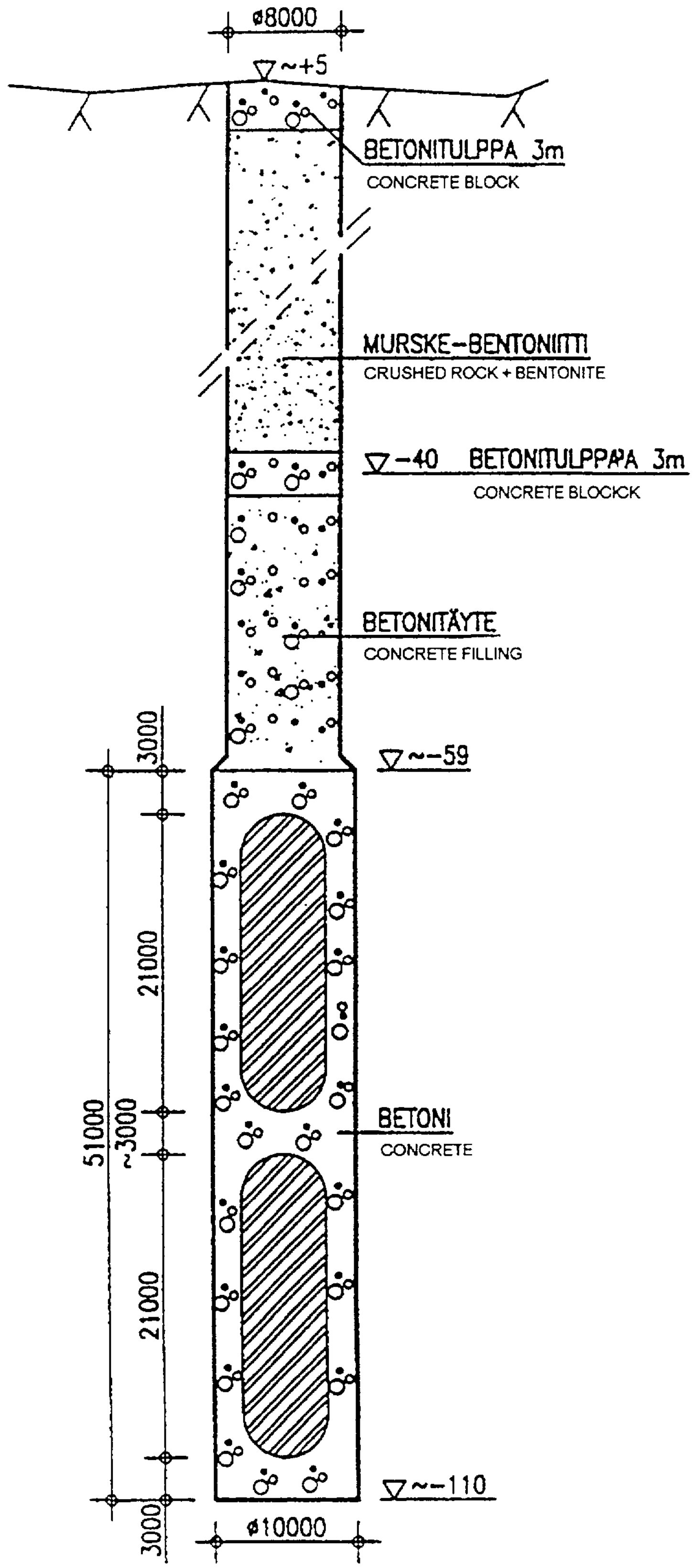

FIG. II-4. Olkiluoto reactors in the repository.

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Annex III

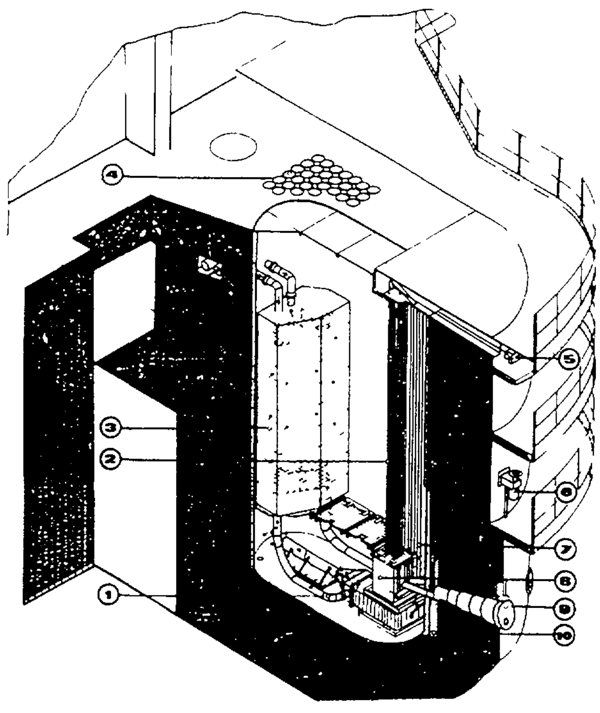
GEORGIA[1]

History and description

The research nuclear reactor IRT-2000 (later IRT-M) of the Institute of Physics, Georgian Academy of Sciences with thermal power 2 MW was put in regular operation in November of 1959. IRT type nuclear reactors, designed in the former Soviet Union, pertain to the group of light water pool-type reactors in which the usual (light) water is used as a moderator of neutrons and as a coolant. Light water is used also as a reflector and biological shielding.

The IRT type reactor was built not only in Georgia but in The Russian Federation (Moscow, Tomsk, Swerdlovsk), Belarus (Minsk) and Latvia (Riga) as well. In addition, the same type of reactors were constructed by The Russian Federation in Bulgaria (1961), China (1962), the Democratic People's Republic of Korea (1965) and Iraq (1967).

The pool of IRT-2000 reactor is a tank made of 6mm thick aluminium alloy sheets surrounded by 1.8 m thick biological shielding of reinforced concrete. The height of the pool is 7.8 m, the length 4.5 m, the width 1.9 m and the internal volume about 60 m^3. The pool is filled with distilled water up to the height of 7.2 m and is covered with a lid of organic glass. See Fig.III-1 for an illustration of the reactor layout.



1-Cooling pipe. 2-Control channels. 3-Hold-up tank. 4-Dry assemblies for storing radioactive samples. 5-Servomotors of control rods. 6-Slide valve. 7-Vertical experimental channel. 8-Reactor core. 9-Horizontal experimental channel. 10-Heat screen.

FIG. III-1. View of reactor pool after the second reconstruction.

[1] This is not intended by Georgia to be a final disposal operation but illustrates many of the procedures that would be needed for disposal.

In 1967–1968 the nuclear reactor IRT-2000 was subjected to the first large-scale reconstruction in order to increase reactor power (about two times) and widening the possibilities of experiments within the core.

The reconstructed reactor IRT-M (M-modernized) operated mainly at a power of 4 MW and operated reliably till 1973. In 1973 the reactor was subjected to the second large-scale reconstruction. The main aim of this reconstruction was the substitution of the aluminium tank of reactor with a stainless steel tank because of corrosion of the old aluminium tank. In addition to the reconstruction, there was modernization of the cooling system as well as the regulation and control systems allowing an increase of reactor power up to 8 MW.

In March 1990 the Academy of Sciences of Georgia, taking into account the limited residual work resource of the reactor and expected large investments necessary for evaluation of seismic stability of buildings and the reactor, and the adverse reaction of the public after the Chernobyl tragedy, decided to withdraw finally the reactor of the Institute of Physics from operation and to decommission it.

During 26 years of operation the nuclear reactor of the Institute of Physics was in operation for 70 000 hours. More than 6000 MW days heat energy was produced which corresponds to consumption of nuclear fuel Uranium-235 amounting to 7.5 kg.

Decommissioning

The Institute of Physics carried out the following steps after the decision to shut down:

- The reactor core was unloaded and all spent fuel assembles in the reactor dispatched from Georgia to the reprocessing facilities.
- A full inventory of radioactive waste kept in dry assembles within the reactor biological shielding and in other special places was drawn up.
- All the low radioactive wastes from the reactor were moved to a special storage repository but excluding the large scale units. The storage facility has not been functioning since 1993.
- Measures have been taken for ensuring prolonged storage of high radioactive and large-scale wastes accumulated in the reactor during many years of operation (experimental channels and devices, various technological elements of the reactor, etc.). Transferring of these wastes to the special repository was not possible because of difficulties in preparation for transportation, absence of special large-scale protective devices for safe transportation and encapsulation and consequently the waste was retained in the reactor and its dry storage channels.
- The fully unloaded reactor tank and its inner equipment is currently kept under its radiation protecting water layer, the parameters of which limits are maintained under constant control.

The measures mentioned above secure the full safety of the reactor. They also secure the radiation safety excluding extreme and hypothetical emergencies resulting, for example, from destructive earthquakes, direct hit of an aircraft to the reactor tank, etc.

Table III-1 contains data of all kinds of radioactive waste existing in the reactor with indications of their total activity, the most important radionuclides and volumes.

TABLE III-1. INVENTORY OF MATERIALS AFTER SHUTDOWN (MAY 1998)

Radioactive materials:	Activity (Bq)	Most important radionuclides	Volume (m^3)
Activated materials:			
(1)Heat shield and concrete of biological shielding	2.0 10^{11}	^{55}Fe, ^{60}Co, ^{152}Eu, ^{154}Eu	18
(2)Core construction & components	3.0 10^{12}	^{55}Fe, ^{60}Co	0.6
(3)Experimental cryogenic channels and their installations	2.0 10^{12}	^{55}Fe, ^{60}Co	5.5
(4)Activity generator of the loop	8.2 10^{10}	^{152}Eu, ^{154}Eu	0.7
Contaminated materials			
(1)Pipelines and equipment of the first contour	6.5 10^{8}	^{137}Cs, ^{60}Co	15
(2)Pipelines and equipment of the low temperature experimental systems	2.3 10^{9}	^{60}Co	2.7

One of the strategies for decommissioning of research reactors which is accepted in some countries is immediate dismantling after short-term storage. For dismantling, the requirements are large repositories for waste, special cutting tools and equipment and facilities for conditioning and transporting highly radioactive waste. Many countries do not have these facilities or it would be uneconomic to invest in them. One of such countries is Georgia.

For other countries the more acceptable variant is long term storage strategy which foresees conversion of the reactor into the intermediate state and its full dismantling only after long delay periods. In such cases, owing to reduction of activity, the total expenditure will be lower despite the additional expenses necessary for control, supervision and protection of reactor during the delay periods. Besides, in this case, the necessity of designing special techniques no longer arises in the immediate future and the danger of radioactive contamination of environment is minimized.

Despite the above advantages of the strategy for long term storage, it is still not acceptable for conditions in Georgia because:

It requires considerable financial expenses, necessity for services, control and for providing reliable physical protection during the whole period of long term storage of nuclear facility.

It cannot provide the guaranteed radiation safety in case of a postulated emergency.

A shut down reactor without any provision for more research (e.g. low power reactor) excludes the possibility to attract young specialists for appropriate skill and training in the nuclear field.

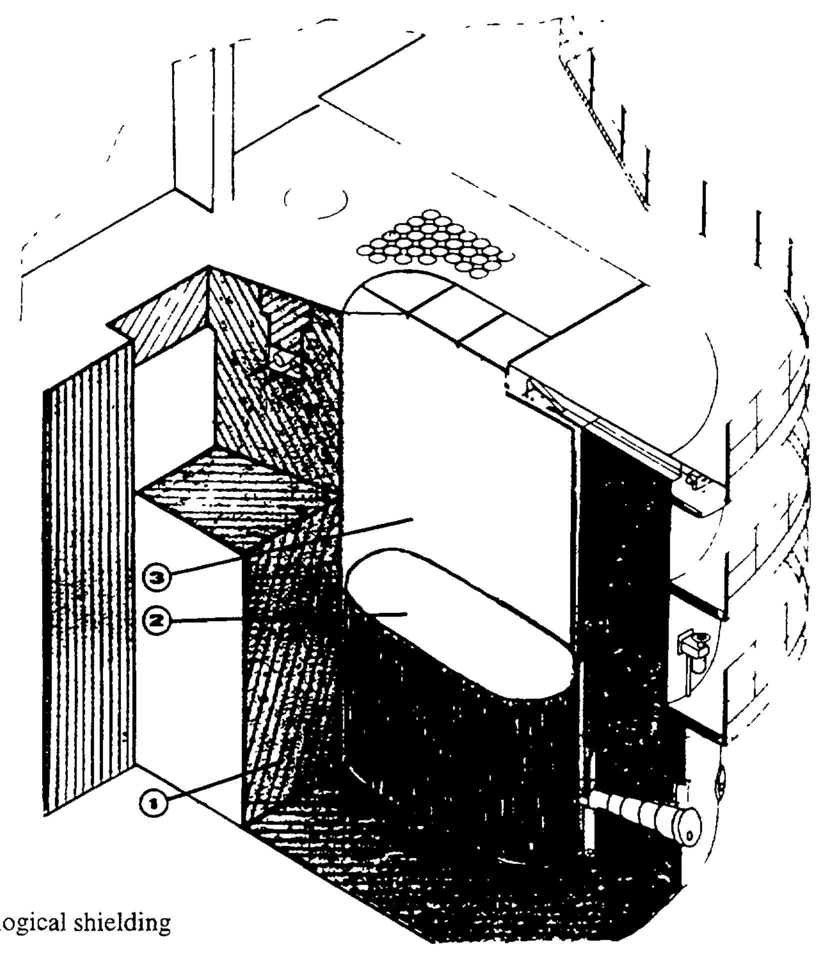
Naturally, the most advantageous and reasonable strategy is conversion of the reactor into such a passive state which does not demand special control and supervision, guarantees its safety even in extreme situations for long periods.

The preferred option which fully satisfies these requirements is immobilizing the most radioactive lower part of the reactor tank (approximately 1/3 of its total volume) and inner cavities of horizontal experimental channels with concrete thereby encapsulating the radioactive waste.

As a result of the preferred option the real possibility of the useful re-application of the rest part of the tank arises. For instance, a critical assembly or low-power reactor can be designed in it operating with quite intensive sources of neutrons. This will have very important benefits for Georgia such as education and training of specialists in the field of reactor physics and atomic energy.

It should be mentioned that the option to encapsulate the lower section of the reactor requires a relatively small amount of concrete (20 m^3) to be added to the existing huge reactor block which has a volume more than 300 m^3. This will not increase difficulties of full dismantling, especially after long delay periods (more than 50 years) because due to radioactive decay, the residual activity will eventually only be from low concentrations of long-living radionuclides, such as ^{41}Ca, ^{59}Ni, ^{94}Nb, ^{152}Eu and ^{154}Eu.

For reduction of harmful radiation to personnel in the process of concreting the lower part of reactor tank, it is decided to carry out the concreting under the protective layer of water. Figure III-2 shows the additional concrete added to the lower third of the reactor block and the space available above for other uses.



1. Reactor biological shielding
2. 20m^3 concrete matrix
3. 40m^3 space for other uses

FIG. III-2. View of reactor pool after concreting its lower part.

The decommissioning option is quite favourable and acceptable because it is

- Radiation safe and ecologically clean during construction,
- Radiation and seismically stable;
- Comparatively not labour-consuming and easy to fulfil;
- Does not result in large financial or material expenditure;
- Provides the opportunity to install a low power experimental nuclear facility

The technology of carrying out the above mentioned process has been developed together with the Institute of Building Mechanics and Seismic Stability of the Georgian Academy of Sciences. The recommendations regarding the use of various sorts of concrete used usually for the building of nuclear structures have been prepared

The proposed decommissioning strategy for the reactor of the Institute of Physics was discussed in detail with the Institute of Reactor Technologies and Materials of Russian Scientific Center "Kurchatov Institute"(1995), at the department of Building of Nuclear Installations of the Moscow State University of Building (1997) as well as in the framework of a NATO grant for two German nuclear research Centers (Rossendorf, 1995; Karlsruhe, 1996) and confirmed by the Presidium of the Academy of Sciences of Georgia

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Annex IV
GERMANY

The approach to decommissioning nuclear facilities followed by the utilities and operators in Germany is the conventional method of safe enclosure and dismantling in the combination suitable for the specific plant situation. Also, there is the official resolution on disposal of radioactive wastes which has exclusively been directed towards deep geologic repositories in salt dome or hard rock host environments. Additionally, there are legal problems associated with who owns the ground under a site and for how long; the operator of the plant often possesses only a 99 year lease. And certainly, the matters of long term safety and public acceptance remain unresolved.

This situation may change according to recently and more frequently voiced ideas of installing near surface repositories for low level wastes (LLW) in the southern part of Germany, in order to facilitate the local acceptance for the deep repositories located in northern Germany. Near surface repositories or shallow land burial grounds need long term safety considerations, environmental impact assessment, and acceptance similar to those required for structures to be disposed of on-site.

The advantages and disadvantages of on-site disposal are not discussed here because they are sufficiently and correctly dealt with in the main part of the report. The following is based on a study which was prepared for the Federal Minister for Research and Technology. A short English version of the study was issued in January 1988 [IV-1]. It does not provide any official view; technical feasibility is the only scale on which it can be monitored. The sinking of a portion — i.e. the area comprising the interior of the biological shield — was also investigated. As it does not contribute further information, it is not discussed further. For details see [IV-1].

Sinking

A possible alternative or complement to the conventional decommissioning techniques is the on-site sinking either of a reactor building in total or of its central part sufficiently deep into the ground. The investigation presumes that the decommissioning work begins — after a preceding planning period — about five years after final reactor shut down and that the fuel elements, the operational wastes, and the secondary loop water have been removed at that time. All other contaminated and activated components of the core area and of the biological shield as well as the primary loop and pool water remain in the building. Five years are required for the decommissioning consisting of the steps

- preparation of site, building and components (2 years)
- sinking of the building (1 year)
- sealing of the sunken structure (2 years).

The reference technique (total sinking) uses the caisson sinking technique with a closed, pressurised working chamber beneath the building. In its final state the sunken structure is a multi-barrier system effective over long periods of time, confining the enclosed radioactive substances, and forming a final repository. During the five years of decommissioning work the maximum collective staff dose exposure is between 0.75 and 0.95 person-Sievert. Releases of radioactive substances are to be expected from the cementation of pool and primary loop water only. The maximum individual dose due to these releases amounts to about 0.16 Sv/a over a time period of about 1.5 years during the preparation step. A consideration of possible adverse events does not result in any accidents causing radioactive releases of concern. It also is shown that external events do not interfere with the sinking process in any way that could terminate the progress of the work.

Concerning long term safety the findings are, that under continuation of the present site and groundwater properties the barrier efficiency remains intact. Postulated accident scenarios - i.e. the removal of the outer barriers up to the steel containment - result in maximum annual individual effective dose equivalents (including 50 years dose commitment) of less than 1.5 Sv. The maximum annual collective dose achieved by consumption of all contaminated release water for drinking and irrigation water will not exceed 70 Sv/a (7 mrem/a).

The total cost of decommissioning a reactor building by sinking is 149.6 million DM (based on 1987 value). The breakdown is: planning and quality assurance 6%, preparation (step 1) 61%, sinking (step 2) 17%, sealing (step 3) 16%. Decommissioning of a reactor building using presently envisaged "classical" techniques requires 256 million DM (based on 1985 value) [IV-2].

Reference case

Reactor The reference reactor selected is a 1200 MW(e) PWR. Data presented in Table IV-I can be taken as representative of a number of commercial reactors in Germany.

Site Fifteen NPP sites were examined. In most cases the reactors are located on considerably deep gravel type soil formations near rivers and are constructed on flat foundations. Exceptions are coastal sites with buildings on pile foundations. Sites, at which the base rock approaches the surface — i.e. within 60 m of surface level — are less frequent. Under all those aspects it can be concluded that sites at rivers and on deep gravel formations are representative with respect to the sinking technique and the long term behaviour conditions. Correspondingly applicable and representative data on geology and hydrology were provided by the complete set collected for the Karlsruhe Research Centre site.

Technique In general, structures are sunk by the caisson technique. This procedure is characterized by the fact that the caisson is sunk from the level at which it is positioned while excavation is under way. A distinction is made between open caissons and caissons with a closed working chamber. For the task "total sinking", i.e. taking the entire structure deeper, only a caisson with closed a working chamber will be appropriate. This method is combined with all the advantages, such as full understanding of the soil and groundwater conditions and control for an accurate and efficient sinking of the structure. Sinking an entire reactor building and applying the working chamber method was selected as the reference technique.

Radionuclide inventories The respective data about the distribution of radionuclides, corresponding radiation levels and heat power in the various areas are summarised in Table IV-II. Figure IV-1 gives an overview of the temperatures expected at different areas in the building as a function of time. It shows the importance of the effect of decay heat generation in the insulated body submersed in groundwater (about 5°C). From these findings it can be concluded that the temperatures, the temperature gradients and their changes over time can be dealt with, particularly as the more realistic time schedule of the investigation puts the time of insulation 3 to 4 years later.

The insulating process itself, i.e. the cementation of all empty space in the building, will also contribute a heat source which must be taken into account.

TABLE IV-I. DATA OF THE REFERENCE REACTOR

Type of reactor	1200 - 1300 MW(e), PWR	Base plate reinforcement	no details [1)]
Diameter of building	~60m	Spent fuel	removed [2)]
Total height of building	~62m	Operational wastes	removed [2)]
Total weight of building	~1700 kt	Total activity	3.3 E17 Bq [3)4)]
Total operation time	~40 years	Decay heat power	~13 kW [3)]
Mass distribution	no detail [1)]		

(1) Adaptation to the conditions of a real plant is possible by means of a hydraulic system used.

(2) Approximately 5 years after final shut down of operation.

(3) Approximately 5 years after final shut down, spent fuel and wastes removed

(4) Most recent value: <2 E 16 Bq - according to [2]

TABLE IV-II. ACTIVITY DISTRIBUTION IN THE REACTOR (BQ); HEAT POWER IN THE CENTRE AREA (W)

Component/area	Time after shut down (years)					
	5	10	50	100	500	1000
Pressure vessel body	5.5E14	2.3E14	7.2E12	4.6E12	4.1E11	1.4E11
Pressure vessel plating	3.2E13	1.6E13	4.7E12	3.3E12	2.8E11	8.6E10
Core installations/satellite parts	1.4E16	7.2E15	1.7E15	1.2E15	1.0E14	3.4E13
Core liner	3.1E17	1.2E17	3.9E16	2.8E16	2.8E15	1.2E15
Pressure vessel insulation / biological shield liner	3.1E13	1.5E13	1.9E12	1.3E12	1.3E11	4.8E10
Biological shield plus reinforcement	2.4E12	1.1E12	6.4E10	2.7E10	1.2E10	7.8E09
Activation total (Bq)	3.3E17	1.3E17	4.0E16	2.9E16	2.9E15	1.3E15
Heat power total (W)	1.3E4	6.5E3	6.5E2	5.2E2	2.3E2	2.1E2
Primary loop contamination	1.3E14	4.8E13	1.4E12	4.6E11	8.0E10	8.0E10
Pool water contamination	8.9E10	5.3E10	7.2E9	2.2E 9	2.2E 5	2.2E 0
Primary coolant contamination	4.3E12	3.3E12	3.7E11	3.1E10	1.4E 6	3.4E 5

Technique applied

The term sinking comprises all activities (Table IV-III) until the structure has reached its final position below surface and has been insulated from the ambient environment. All phases are accompanied by monitoring, quality control, radiation protection and plant surveillance.

Phase I

Preparation of equipment Preparation shall mean the work by which the nuclear power plant is made fit for sinking. To this end, connections between the reactor building such as process lines, and connections to service buildings, must be severed and sealed. The preparation phase will last approximately 2 years and comprise:

- the severance of all connections between the reactor building and the peripheral facilities, including removal of the outer cladding,
- the preparation of the area around the building for erection of the diaphragm wall and provision of the shafts,

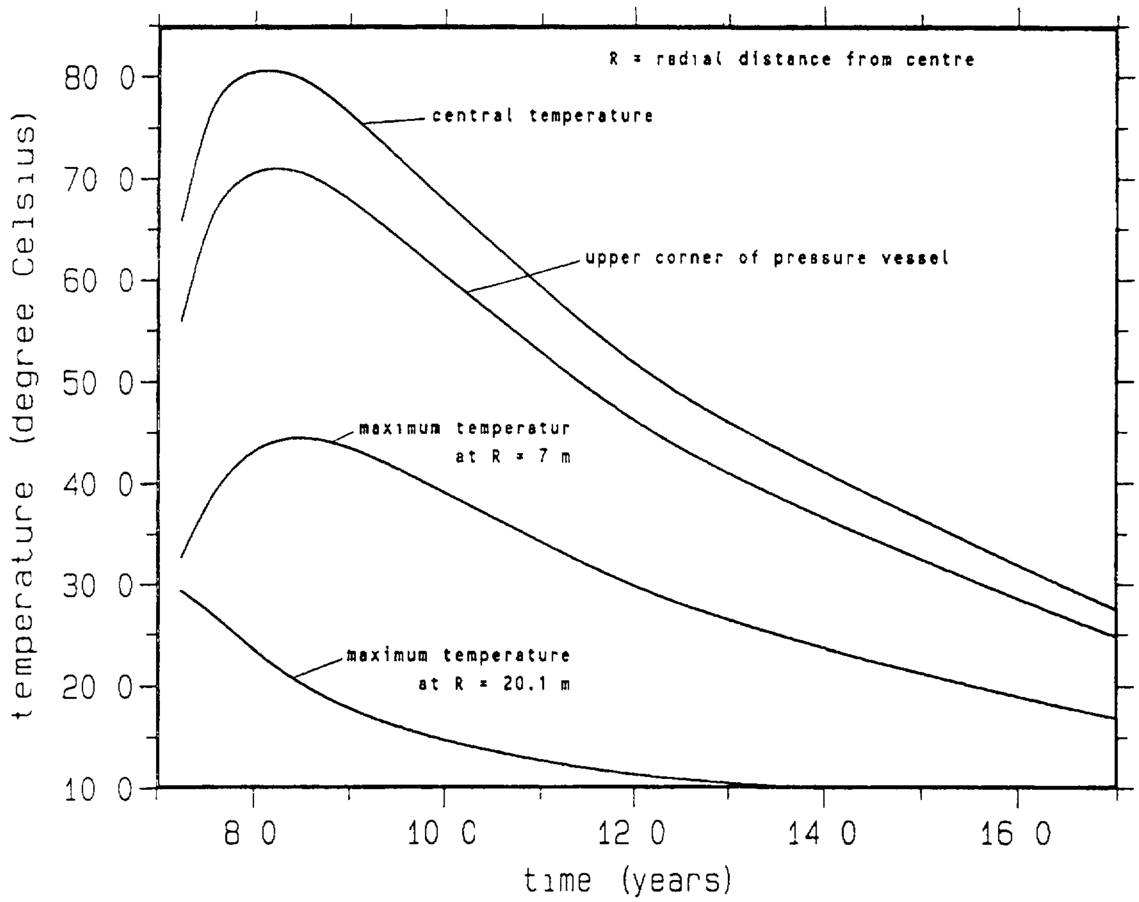

FIG IV-1 Temperatures ın the ınsulated structure

TABLE IV-III PHASES OF SINKING

Phase 0 Plannıng and documentatıon		
Phase 1 Preparatıon	Phase 2 Sınkıng	Phase 3, Insulatıng
Desıgn work Preparatıon of plant equıpment, cuttıng connectıons, ventıng system etc Strengthenıng of buıldıng for sınkıng, possıbly by means of ınterıor reınforcement Installatıon of draınage system (dıaphragm wall ınjected bottom) Constructıon of caısson (cutters and workıng chamber) Protectıve measures (seal, new outer cylınder) Processıng of contamınated water (prımary cırcuıt, pool)	Control of loads Implementatıon of sınkıng operatıons, draınage, pressurızatıon Control of statıcs Excavatıon and conveyance of soıl	Beneath bottom Fıllıng ın of workıng chamber and press cavıtıes wıth sealıng materıal Fıllıng ın of ınterıor cavıtıes Fıllıng ın of resıdual prımary cırcuıt volume ın heat exchanger, RPV annular cavıty and buıldıng Top coverıng (sealıng, concrete cover, demolıton work on cupola, backfill)

- the ınstallatıon of auxılıary ventılatıon and power supply systems after the plant systems have been severed,
- all work to be carrıed out ınsıde the reactor buıldıng ınvolvıng severance, sealıng and concretıng, to the extent that such are necessary prıor to sınkıng, ın partıcular wıth respect to the prımary coolant and pool water,
- and radıatıon protectıon for thıs work

Drainage installations The structure ıs sunk ınto soıl strata contaınıng groundwater The workıng chamber ıs pressurızed The pressure ıs determıned by the effectıve water pressure at the level of the bottom of the workıng chamber Thıs prevents water from enterıng the working chamber. The working chamber with the cutters is constructed beneath the base of the structure. The working chamber consists of an outer cutter ring approx. 60 m in diameter, and a bulkhead approx. 4.0 m thick. The bulkhead is also supported by an inner cutter ring approx. 25 m in diameter. Hydraulic presses between the base of the structure and the ceiling plate, and between the ceiling plate and inner cutter control the sinking operation. The caisson is constructed in part using mining techniques, and in part using underpinning techniques, protected by the diaphragm wall.

Phase II

Sinking activities The sinking phase comprises the period from commencement of earth excavation in the working chamber up to the planned depth of sinking being reached. It lasts about one year. The total weight of a 1200 MW(e) reactor building is approx. 170 kt, the weight of the caisson approx. 60 kt. Hence the total weight to be sunk is approx. 230 kt. Acting counter to the weight being sunk are the buoyancy exerted on submersion of the body into the groundwater, and the frictional forces exerted as a result of the pressure of the soil on the outer walls. The difference of the forces increasing and decreasing the load results in the cutter loads which have to be imparted into the soil. In the initial phase, cutter loads to be absorbed by the soil are high. In later stages, these loads are reduced by buoyancy and friction increasing the effective cutter load.

Sinking scheme (Figure IV-2) The principle behind the planned sinking concept is to enable the sinking operation to be carried out under control, continuously, and smoothly with respect to the structure. Hence loads which might be imparted into the structure, arising from variations in the cutter forces being imparted via the structure base, are to be kept small. For this reason, hydraulic presses are installed between the working chamber ceiling plate and the inner cutters. As a result it is possible to increase and decrease precisely the forces acting through the inner cutters, or to shift loads onto the outer cutter. Supported by the excavation of earth, sinking is initiated alternately at the inner and at the outer cutter, and in this manner the structural body is stacked down in stages.

Phase III

Insulating activities The insulating work is carried out after having reached the required depth, the activated and contaminated residual inventory being permanently and securely shielded from the environment, and the multiple barrier system erected. First of all, the working chamber and cutter substructure areas are backfilled. A bottom bitumen seal is applied around the top of the hydraulic equipment, after which the hydraulic press cavity is filled. The cavities inside the building itself are filled, in sequence, with special concrete.

Several areas of high radiation level (RPV shielded zone and steam generators - primary side) are filled in taking special safety precautions. Sealing materials are also used in the interior cavity, either to prevent the development of cracks or fractures, or to bring about a spontaneous sealing of unavoidable cracks. Above the working level a steel covering equivalent to the steel shell is installed, and the bitumen seal and concrete cover put into place. Finally, the lateral and shaft areas are filled in. The result is a closed, compact, impervious and shielded block (Fig. IV-3). Over a period of one year after insulating, the enclosed radionuclides heat up the entire structure to temperatures which are just below 70°C in the centre of the RPV (Fig. IV-1). This heating up and subsequent gradual cooling down take place in such a way that the stress build-up and stress reduction processes remain controllable, and are unable adversely to affect the overall barrier structure.

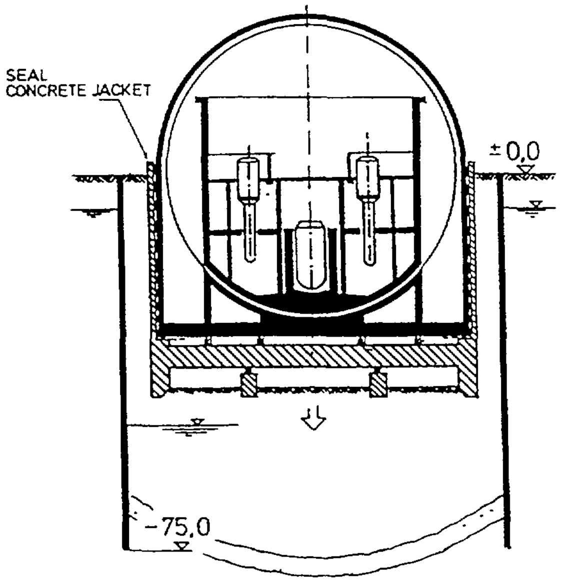

FIG. IV-2 Sinking phase — intermediary state

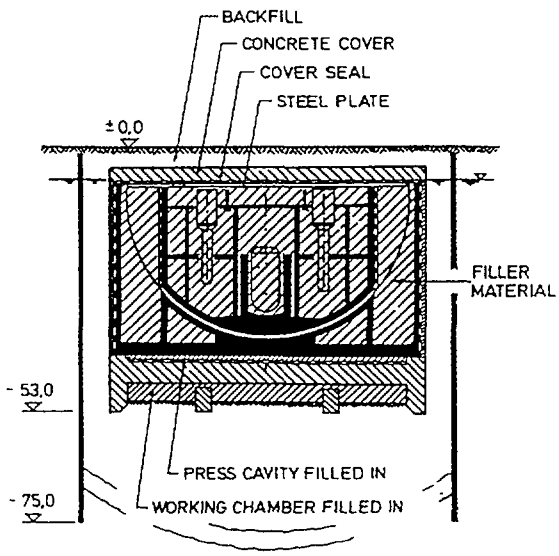

FIG. IV-3. Final phase — insulation complete.

Safety achieved

General note The safety aspects concern the workers' safety (i.e. radiation and conventional protection), the environmental safety during normal operation sinking procedures and under accident conditions, and serious interfering external events possibly affecting or complicating the process of sinking.

Concerning conventional safety, the common regulations in use are to be supplemented with those governing mining activities as well as work in pressurized atmospheres. The latter is compulsory too, since technical and administrative measures for this type of work have to be provided in any case - even if atmospheric pressure is normally applied — because an incidental groundwater inrush cannot be completely excluded.

Radiation exposure of staff The radiation exposure of staff working during preparation, sinking, and finally insulating the reactor building is assessed as the product of the local dose rate levels at the various working areas and the residential or working times at the respective areas. The dose rate at a working place depends on the dose rate of the inventory and its reduction by shielding and decontamination measures, of which only the first type is used here. It is to be noted that the scheduling of component filling measures — i.e. before or after sinking — makes use of the radiation level decrease with time, since this level is dominated by ^{60}Co with a 5.4 year half-life. The local dose rates in some areas are far too high for human activities. Shielding measures reducing the level by two to three orders of magnitude are to be used. Remote handling techniques are to be applied in the case of filling the pressure vessel insulation and its annular space.

Collective dose of staff The subsequent estimates, partly based on data derived from [IV-3, IV-4], are to be considered as upper boundaries. The working times used in the estimates are the total person hours according to detailed time and staff deployment plans. The residential times of the health physics personnel is included in the person hours of phase 1 and 3. Concerning phase 2 (sinking) lasting about 1 year, an amount of 10 000 hours is assumed for survey, inspection, and maintenance personnel spending 2000 person hours inside and the rest outside the building. The findings are shown in Table IV-IV. Thus, the collective dose of staff might range between 0.75 and 0.95 person Sievert (75 to 95 person rem) or below, as the estimate is rather pessimistic.

Individual staff exposure The work activities are to be planned in such a way that the dose any individual member of staff might be exposed to is as low as possible and does not exceed the regulatory limits. The individual working time of about 1700 hours per year excludes violations of those limits, provided that the average local dose rates do not exceed 29 Sv/h or 70 Sv/h respectively. According to Table IV-IV, this will be ensured for most cases if the appropriate shielding measures are prepared and used. For those areas where the dose rate levels exceed the values mentioned, the individual annual residential times are considerably less than 1700 hours.

Normal procedure impact to the environment The work performed inside the reactor building does not generate any release of radioactive material at all, because demolition and cutting of contaminated or activated parts is completely avoided with this technique. The only source of release is the cementation of the water contained in the fuel storage pool and the primary cooling loop. The period of release is one year during the normal working hours. The resulting releases as gases (vapour and ^{85}Kr), as aerosols mainly retained in the venting system filters, and tritium with the vapour and the aerosols.The release route is the venting system via the stack. The resulting calculated maximum individual doses covering all exposure routes (whole body dose inhalation 26 nSv/a and ingestion 130 nSv/a over the period of about 1.5 years when the water is solidified in cement) are by far smaller than those calculated from the actual releases of the operating NPP.

Accident considerations According to common use, the accidents are arranged in "internal events" and "external events". Only those events are considered in some detail which could result in radionuclide releases and subsequently increased radiological effect in the environment or could seriously affect the progress of the work.

Internal events The consideration comprises the phases 1 and 3, i.e. the periods between 5 and 7 years and between 8 and 10 years after reactor shut down, respectively, since activities are going on inside the building only during these periods. The characteristics and the

TABLE IV-IV. TOTAL COLLECTIVE STAFF EXPOSURES DURING DECOMMISSIONING BY SINKING

Type of work	Phase/ area [1]	Time (person -hours)	LDR [2] (µSv/h)	Dose (10^{3} person-Sv)	LDR (µSv/h)	Dose (10^{3} person-Sv)	LDR (µSv/h)	Dose (10^{3} person-Sv)
Preparation at the outer wall		3290		6 1		1 4		14 3
pipe cutting	1/i	780	3	2 3	0 7	0 5	7	5 5
pipe sealing	1/i	360	3	1 1	0 7	0 3	7	2 5
penetration sealing	1/i	900	3	2 7	0 7	0 6	7	6 3
adaptation of venting and electrical supply	1/o	560	0	0	0	0	0	0
others	1/o	690	0	0	0	0	0	0
Cementation of Water		16800		301		251		167
filling steam generator (secondary side)	1/i	2200	23	51	20	44	25	55
filling pressure vessel	1/i	3100	20	60	30	93	10	31
filling pressurizer	1/i	200	21	4	15	3	10	2
filling main coolant pipes	1/i	370	43	16	20	6	10	4
treatment of leftover water	1/i	3660	23	83	14	51	10	37
general activities	1/i	3840	23	87	14	54	10	38
general activities	1/o	3430	0	0	0	0	0	0
construction work for sinking		327775						
site and caisson construction	1/o	175940	0	0	0	0	0	0
sinking	2/o	141	0 835	0	0	0	0	0
surveillance during sinking	2/i	2000	20	40	20	40	20	40
surveillance during sinking	2/o	8000	0	0	0	0	0	0
Filling of inner rooms		98 065		525		469		656
filling steam generator (primary side)	3/i	400	25	10	14	6	17	7
filling ring space	3/i	5210	15	80	20	156	17	133
activities below pool floor	3/i	9955	20	199	20	199	17	169
activities at pool floor	3/i	7905	15	119	10	79	7	55
activities outside the safety cylinder	3/i	58400	2	117	0 5	29	5	292
activities outside the building	3/o	16195	0	0	0	0	0	0
Insulation Tasks		64410		27		7		67
filling of excavations	3/o	29640	0	0	0	0	0	0
construction of upper insulation	3/I	6840	2	14	0 5	3 4	5	34
construction of concrete cover	3/I	6650	2	13 3	0 5	3 3	5	33
removal and filling work	3/o	21280	0	0	0	0	0	0
Total summation		**510370**		**899**		**771**		**944**

insignificance of the radionuclide inventories provide evidence — even without discussion — that most events cannot occur at all or would result in virtually negligible radionuclide release if they occurred.

External events The access shafts to the subsurface working area are protected even against unusual rainfall. The effects of an explosion-caused shockwave do not endanger the progress of the work due to the enormous mass of the building and the short duration of the effect. Damages of the foundation soil caused by mining activities are site specific and can be excluded at nuclear power plant sites.

Sabotage can damage the structure of the caisson or of the building - according to time and location of the action - to such an extent that the progress of sinking is endangered. However, the normal security of nuclear sites is assumed as sufficient.

Earthquake effects, particularly during the sinking process, were extensively and carefully investigated in a separate exercise. The finding is, that the event would be controllable and would neither require to cancel the sinking nor result in radioactive effluents.

Tidal waves at coastal sites or floods at river bank sites could flood the caisson or — in later stages of the sinking process — the building. This is prevented by specific measures according to site conditions. An airplane crash, together with a major fuel fire, and the resulting mechanical shock, could damage the sinking hydraulics. Repair and reestablishment are easily possible.

It can be summarized from the discussion that events which might cause an exposure in the environment or endanger the progress of the process are not likely to occur.

Long term behaviour

System of engineered barriers

As shown in Fig. IV-3, the sunk structure forms a barrier system consisting of:

- an outer containment of reinforced concrete (80 cm),
- a bitumen safety layer (12 cm),
- the reinforced concrete cylinder and base plate of the original building and a new cover (80 cm minimum),
- the steel safety containment shell (3 cm), completed with a welded equivalent steel sheet substituting the cut off section at the upper part,
- the concrete structures inside the steel shell - i.e. the originally existing concrete structures and the concrete fillings in the originally empty rooms and spaces as well; in this respect, the inner concrete cylinder is particularly important, as its walls, its bottom plate above the building base plate, and its newly constructed cover plate jointlessly enclose the parts of the reactor,
- the fixation of the radionuclides encapsulated either in a matrix of concrete or by the filling material inside and around the components or by their origin as constituents of the metallic and concrete materials inside the biological shield.

The materials chosen as well as the method of their application ensure extreme durability and longevity of the barrier system which control an intrusion of groundwater as well as a release of radioactive inventories. The steel structures embedded in the concrete are protected by the groundwater itself which prevents carbonation of the concrete and, thus, depassivation of the steel environment Only if the pH value of the groundwater (normally at about 12 5) fell below 9, carbonation of the concrete and subsequent depassivation of the steel could occur

But even if the groundwater table fell and the concrete structure was exposed to oxygen, the time required for the complete carbonation of the outer wall of 80 cm concrete would add up to at least 250 000 years More aggressive water like sea water could achieve this result in only 2500 years However, such carriers could not be used as drinking water or for irrigation purposes They block themselves as exposure routes. Relatively fast degradation of the outer barriers can rather be expected from micro-organisms and plant life

Because of that situation, an accident scenario was postulated which affects the efficiency of barriers in a way that eventually causes a release of radioactive substances into the groundwater

- at any point in time, for instance 2000 years after sinking (500 as well as 100 000 year scenarios were also investigated), the barrier effect of outer concrete wall, bitumen layer, building concrete , refill etc down to the steel safety shell is annihilated and the safety shell is penetrated by pitting corrosion,
- the percolation of groundwater into the containment region of the sunk reactor begins and takes about 230 years to reach the closest radioactive inventories effecting mobilization,
- the leaching processes gradually transfer accessible radionuclides into the water at rates which are governed by the respective diffusion or corrosion processes and limited by the available oxygen, dissolved in groundwater and generated by radiolysis,, after another 230 years, the safety shell is saturated with groundwater (exception RPV),
- the release of contaminated water into the surrounding groundwater begins after another 40 years when the percolating fluid reaches the outflow point,
- the release rate slowly falls due to decay for about 40 000 years until the inventory of the RPV becomes accessible to the carrier medium, then the real maximum occurs (see Fig IV-4),
- the transport of dissolved radionuclides with the groundwater flow through the soil to the point, where water is taken from a well for further use,
- incidentally, this well is at the exact place where the contaminated water leaves the sunk structure and where the radionuclide concentration of the groundwater has its maximum (see Fig IV-5),
- all the contaminated water is used for irrigation and drinking water for humans and animals as well, thus opening all exposure routes via food chains,
- no credit is taken for decreasing concentrations, retention in soil etc - the scenario is the most conservative one imaginable

Whether this process begins 500, 2000 or 100 000 years after sinking, affects the further results on exposure only marginally The reason is that the very long lived nuclides dominate the contribution to dose The release rate versus time functions were determined for the relevant radionuclides according to the above described reference scenario. Those nuclides of importance (see Table IV-V) not shown in Fig IV-4 have release rates of less than 10 Bq/year and are not further considered, therefore The parameters determining the radionuclide concentration immediately outside the sunk reactor are the concentration inside, the release area, the soil porosity, and the groundwater distance velocity The results obtained with this approach are shown in Fig IV-5 for the reference scenario They show the source concentrations of the relevant radionuclides versus time. Migration of radionuclides with the groundwater, dispersion effects etc. were investigated in detail and dimensions. These are not stressed here because their contribution further lowers the exposure of individuals.

Radiation exposure assessment

The radiation exposure caused by the use of groundwater contaminated in the case of an accidental release (scenario) is assessed in a way as close as possible analogously to the procedure used for the determination of repository safety in the Projekt Sicherheitsstudien Entsorgung (PSE, Project Safety Studies of the Back End of the Fuel Cycle). Thus, a direct comparison of repository release scenario is possible. The exposure is assessed in terms of the maximum individual effective dose equivalent taking into account a 50 year dose commitment due to incorporated radionuclides ([IV-5], volume 4). The pathways of exposure considered

TABLE IV-V. SELECTION OF THE RELEVANT RADIONUCLIDES

Nuclide	Half-life [years]	Activity at sinking [Bq]	Activity after2000 years [Bq]	Rank
^{10}Be	1.60E+06	2.08E+04	2.08E+04	
^{14}C	5.73E+03	6.16E+09	4.84E+09	7
^{36}Cl	3.00E+05	8.81E*08	8.77E−08	
^{39}Ar	2.69E+02	2.14E+10	1.24E+08	
^{41}Ca	1.03E+05	5.05E+09	4.98E+09	6
^{59}Ni	7.50E+04	1.06E+15	1.04E+15	1
^{63}Ni	1.00E+02	5.25E+16	5.01E−10	4
^{92}Nb	3.60E+07	2.74E+04	2.74E+04	
^{93}Mo	3.50E+03	2.68E+10	2.20E+10	5
^{94}Nb	2.40E*04	1.53E+14	1.49E+14	2
^{99}Tc	2.10E+05	8.17E+10	8.17E+10	3
^{129}I	1.57E+07	3.44E+05	3.44E+05	

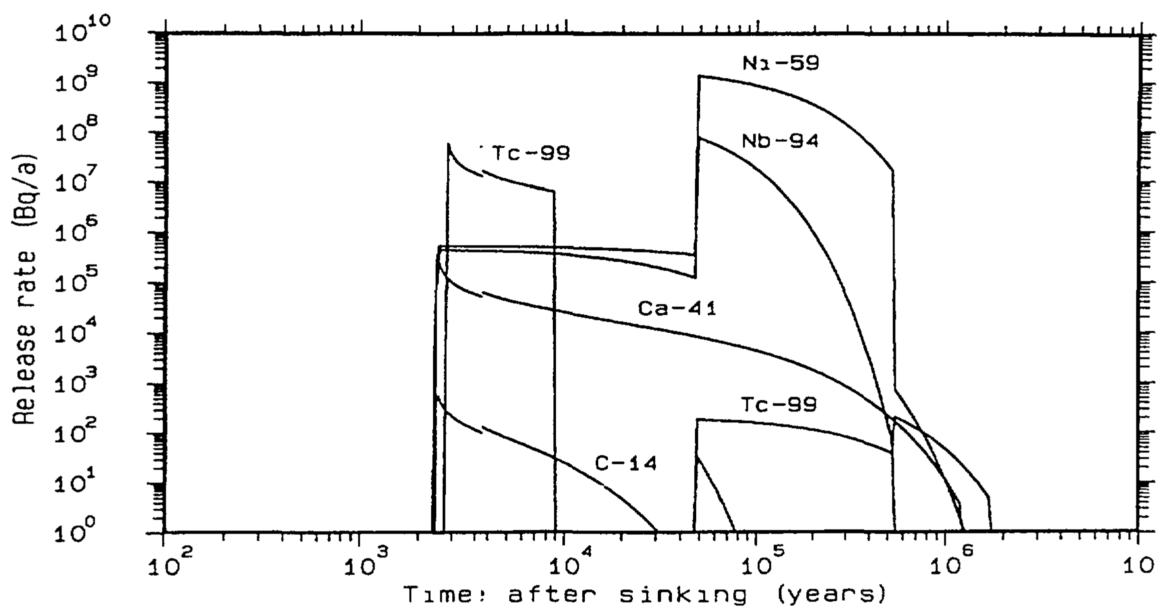

FIG. IV-4. Release rates according to the postulated scenario.

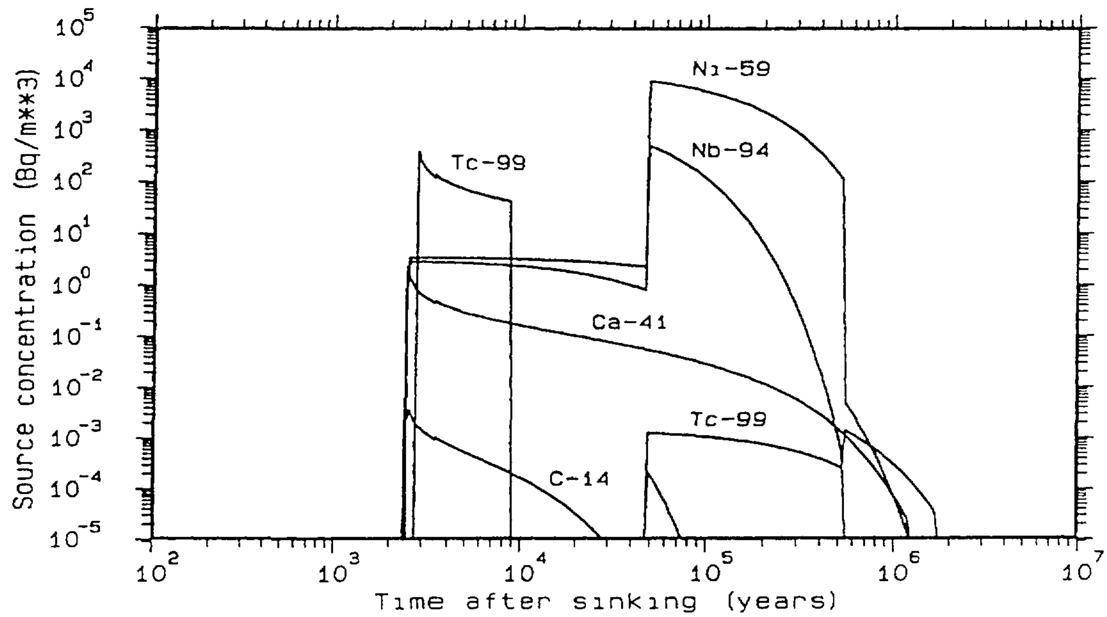

FIG. IV-5. Corresponding source term concentrations.

are intake of drinking water and food. The drinking water is assumed to be taken from the groundwater directly without any processing. The food is taken either from agricultural areas directly or via food chains from pastures, both irrigated with groundwater. The maximum doses are derived from the maximum concentrations immediately at the source (0m distance). The dose results are summarized in Table IV-VI.

Furthermore, it has to be noted that the dose maxima are displaced in time in such a way that the summation of the different contributions is not necessarily the maximum exposure. Thus, a summation of doses according to the time behaviour of the concentrations is shown in Table IV-VII, demonstrating that the conservative maximum doses would be less than 2 Sv/year (0.2 mrem/year) and that they are almost independent of the start in time of the scenario.

A further result was that under the assumption that all contamination released per year is consumed via the exposure pathways, the resulting maximum of the annual collective dose (dose equivalent with 50 year dose commitment) would be less than 70 Sv (7 mrem/year).

Cost and time

The time schedule and cost estimate have been drawn up with reference to the 1300 MW(e) reference power plant, with the following boundary conditions. The reference power plant is a hypothetical power plant. Its dimensions, volume and load-bearing structure correspond to those of 1300 MW(e) pressurized-water plants. Detailed, concrete technical specifications on elements in the interior zone, equipment etc were not available. Hence estimates had to be made for these zones. Their effect on the results of the time schedule and cost estimate is however of subordinate importance. They have been taken into account in the statements of quantities by making additions.

With respect to the substratum, the sinking activities have been based on broken ground extending to the required depth of sinking. The groundwater level is approx. 6.9m below the surface of the site.

Prices are based on 1987 levels. The study has investigated total sinking.

TABLE IV-VI. INDIVIDUAL EFFECTIVE DOSE EQUIVALENT FOR 1 YEAR INTAKE AND 50 YEARS DOSE COMMITMENT IN NSV/YEAR (1 NSV = 1.E-9 SV)

Scenario	500 year case			Reference case (2000 years)			100 000 year case		
Distance (m)	0	2000	3000	0	2000	3000	0	2000	3000
1st maximum (y)	*950–1100*			*2200–2500*			*100 000*		
^{59}Ni	2 2E-1	3 2E-2	1 6E-2	2 2E-1	7 0E-2	5 4E-2	2 7E+2	8 1E+1	7 0E+1
^{94}Nb	6 6	9 8E-1	4 9E-1	6 6	2 1	1 6	1 6E+2	5 4E+1	4 9E+1
^{99}Tc	1 9E+2	5 3E+1	2 5E+1	1 9E+2	5 7E+1	3 8E+1	2 3E+1	5 7	5 0
2nd maximum (y)	45 000			45 000					
^{59}Ni	5 4E+2	1 1E+2	5 4E+1	5 4E+2	2 2E+2	1.6E+2			
^{94}Nb	8.2E+2	1 6E+2	8.2E+1	8.2E+2	3 3E+2	2.5E+2			
^{99}Tc	5 7E-4	1 1E-4	5.7E-5	5 7E-4	1 5E-4	1.3E-4			

TABLE IV-VII. FOR DIFFERENT SCENARIOS AND TIME PERIODS ACCUMULATED DOSES ACCORDING TO THE VARIOUS NUCLIDES CONTRIBUTING; MAXIMUM RESULTS FOR CONTAMINATIONS AT 0M MIGRATION DISTANCE; IMMEDIATELY ADJACENT TO THE SUNK REACTOR.

Scenario	Time period (years)			Dose (nSv/y)
reference case	2500	until	9000	200–20
(2000 a)	9000	until	45 000	5
	45 000	until	550 000	1400–5
500 y case	950	until	7500	200–20
(500 a)	7500	until	43 000	5
	43 000	until	550 000	1400–5
remote case	100 000	until	550 000	450–5
100 000 a)				

Construction time, cost and staff required The construction time is crucial to the cost estimate. The total construction time obtained on the basis of familiar and fully developed construction techniques is five years. The initial planning period required is two years. The assignment of personnel is shown in Table IV-VIII, schedule, costs and rates are indicated in Table IV-IX. A total of 2765 person-months or 553 000 working hours was estimated. The monthly budget over the 5 years of construction work varies from almost 4 million DM per month in the initial phase of preparation to an average of 1 million per month for the insulation work in phase 3.

Comparison The cost of direct disposal of a comparable reactor building [IV-2] is 256 Million DM (price levels Sept. 30, 1985), where the conventional shutdown and decontamination techniques are used, work is commenced 2 years after the plant has been taken out of operation and the duration of the works was estimated as being approximately 10 years. The cost advantage of keeping more material for disposal on-site will be increased further if, as is to be anticipated, the limits for material re-use are made more restrictive, and the costs for decontamination, waste conditioning, transport and disposal increase. There is also an advantage in terms of time as the implementation is faster.

TABLE IV-VIII. SEQUENCE OF OPERATIONS AND WORK INVOLVED (INCLUDING RADIATION PROTECTION)

Time (a)	Phase	Activities	Work (person-hours)
0		shut down of reactor	
0–5	0	planning, engineering	
5–7	1	preliminary work	196 000
		Interior:	
		Separation	1200
		Insulation	2000
		Filling	17 000
		Exterior:	
		Diaphragm wall	70 000
		Shafts	42 400
		Cutters	63 400
7–8	2	Sinking	152 000
		Interior: Monitoring	10 000
		Exterior: Sinking	142 000
8–10	3	Insulating	153 000
		Interior	
		Filling	75 000
		Insulation	13 500
		Exterior	
		Insulation	13 500
		Filling	51 000
5–10	1,2,3	Management	

TABLE IV-IX. DETAILS ON COSTS AND TIME

	Time (years)	Cost (million DM)	Rate (million DM/month)
Planning & QA		8.58	
Preparation	2	91.32	3.80
Sinking	1	25.59	2.13
Insulation	2	24.11	1.00
Total	5	149.60	

REFERENCES TO ANNEX IV

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Annex V

ITALY

The RB-1 reactor (10 W thermal power) consisted of an external multiplication zone fed with enriched uranium, moderated and reflected by graphite, and a measurement zone formed by a hollow where the lattices under examination were placed. The reactor as a whole — a cylinder of 286 cm diameter and 300 cm height — rested on a steel base plate; all were located in a steel vessel, with inner dimensions of 340 cm diameter and 495 cm height.

Direct irradiation measurements of the vessel and base plate showed low activation. The contact exposure showed a peak value of 0.3 Sv/h. The activity measurements on samples from the vessel, lid, base plate and reactor upper structures registered the presence of ^{60}Co. The specific activity of ^{60}Co ranged from a peak (corresponding to the reactor centre line) of 3.5 Bq/g for the vessel and detector supports to less than 1.0 Bq/g in the remaining parts. The maximum specific activity of ^{55}Fe derived from a conservative estimate was calculated to be 8.0 Bq/g.

A threshold of 1.0 Bq/g for the unrestricted release of reactor components had been established. In principle, this could have led to a decision to dismantle and remove the reactor vessel. However, the very low activation level did not justify such an expense.

A decision was made to dispose of the reactor vessel in situ, based on the following considerations:

In about 10 years the ^{60}Co and ^{55}Fe would fall below the limits for unconditional release.

Vessel flooding would not result in discharges affecting the environment; in fact, the total oxide release was estimated at 0.45 g, which is insignificant.

The vessel could conveniently be filled with debris from demolition of the components above floor level. Concrete would then be poured in to fill the voids, forming a level floor. This is conventional work, involving no radioactive material.

The RB-1 premises are within the Montecuccolino site, where other nuclear reactors and laboratories are operating. Therefore, the probability is low that any future building work would transfer radioactive material to the environment.

The RB-1 decommissioning project was completed in 1989 [V-1, V-2].

REFERENCES TO ANNEX V

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Annex VI
JAPAN

Background

Construction of a final repository for low level radioactive waste arising from research and medical activities is under consideration in Japan. The low level radioactive waste arising from the operation of nuclear power plants is disposed of into the Rokkasho low level radioactive waste disposal facility. All radioactive wastes are stored in waste storage facilities until construction of the final repository in the Japan Atomic Energy Research Institute is complete. Since the storage space has been limited in recent years, efforts have been made to reduce waste arising in JAERI. One-piece removal is considered as one of the options for minimizing radioactive waste arising from decommissioning nuclear facilities. In decommissioning the Japan Research Reactor No. 3 (JRR-3) and the nuclear ship Mutsu, the reactor blocks were removed from the original locations in one piece for long term storage on-site. This is not considered to be disposal as the items will eventually be sent to the final repository. The decommissioning of the Japan Research Reactor No. 2 (JRR-2) is in the planning stage; one piece removal will be applied to dismantling of JRR-2 in the same manner as JRR-3.

Decommissioning of research reactors

(1) Japan Research Reactor No. 3

The Japan Research Reactor No. 3 (JRR-3) was a heavy water moderated and light water cooled swimming pool-type research reactor with metal uranium fuels. It attained criticality at September 1962 and continued operating until 1984. The maximum neutron flux was 2×10^{13} n/cm^2 s with 10 mW in thermal output. During operation, it contributed to neutron beam experiments and isotope production. To achieve a requirement for higher flux capability, JAERI decided the JRR-3 should be reconstructed to further enhance the experimental capabilities in 1984. The decision was made to remove the reactor block in one-piece to the adjacent storage location.

In the dismantling activities cores were drilled to separate the reactor block from the building. The reactor block was transported out of the reactor building, through a temporary opening. The reactor block was transported 34 m to the temporary shelter over the storage facility. The steel frame carrying the reactor block was fixed by stops, and the reactor block was lowered to the final position for long term storage. The top opening was then closed with reinforced concrete.

The new JRR-3M reactor was constructed in the original location in the reactor building. The first criticality was achieved in March 1990 and its utilization was started in November, 1990.

(2) Japan Research Reactor No. 2

The Japan Research Reactor No. 2 (JRR-2) is a heavy water moderated, light water cooled CP5-type research reactor. It uses highly enriched uranium fuels to obtain a neutron flux of 1.8 10^{14} n/cm^2 s with 10 MW thermal output. It attained criticality in October, 1960 and continued in operation until 1996 for neutron scattering experiments, irradiation tests of nuclear fuels and materials, radioisotope production, boron neutron capture therapy, etc. It was finally shut down due to degradation of components after 36 years of operation. The JRR-2 decommissioning project started in August, 1997. The project was divided into 4 major phases; shutdown activities, reactor safe storage and removal of cooling systems and reactor body. It will be completed by 2007. The one-piece removal method will be applied to dismantling of JRR-2 in same manner as JRR-3. The building will be reused for hot laboratory experiments.

(3) Nuclear ship Mutsu

The nuclear ship Mutsu (NS Mutsu) was the first Japanese prototype nuclear power research vessel. It contained a pressurized light-water moderated reactor with 36 MW thermal output. The NS Mutsu operated between 1991 and 1993 with an integrated power generation of 81 090 MWh. After the experimental voyages, it was decided to decommission the NS Mutsu in February 1993. The one-piece removal method for the reactor was selected. A big barge was used to lift the NS Mutsu from the water. The reactor room was then removed by cutting the body of the NS Mutsu. It was transported to a storage facility constructed at the home port. A museum, which exhibits the history and some parts of the NS Mutsu vessel and the containment facility of the reactor room, was opened in July, 1996. The ship itself has been re-used as a non-nuclear ocean research vessel.

One-piece removal of JRR-3

The cooling system was first drained, disconnected, and sealed. The system was then flushed with light water and the piping was cut away from the reactor body, using suitable tools. The pipes were completely filled with resin and sealed with steel alloy plates on the outside wall of the reactor.

Core-boring to separate the reactor block from the building structures started after the preparatory work. For horizontal concrete cutting, three core-boring machines were placed symmetrically and operated with precision control of the cutting depth. Horizontal core-boring was completed in 11 days resulting in 210 metres of cores. The boring machine cooling water was re-circulated to reduce the volume of radioactive liquid waste.

The reactor block was separated from the building structures by continuous core-boring of the 3m thick floors. Vertical cutting was performed by 7 core-boring machines. It took nearly a month to complete 259 core borings totalling approximately 820 meters in length.

The reactor block weighing nearly 2250 tons was raised 3.7 meters in approximately 15 hours. The reactor block was jacked-up by controlling the jack stroke equalizers and sub-cylinders devised to distribute load to each jack uniformly and by monitoring the axial force at 12 points of the lifting rods. The suspended reactor block with the steel frame (approximately 2500 tons in total weight) was transported horizontally 33.6m to a location above the storage facility in 7 days at a rate of 5 meters per day. During the horizontal transportation, the propelling force and the strokes of lateral jacks were controlled and the straightness of the steel frame was monitored by laser theodolite. Figure 1 illustrates the structure of the one piece removal system. A propelling force of about 80 tons was required to move the reactor block and the steel frame. The coefficient of rolling friction of the rollers was 0.128 cm. The reactor block was lowered 13.5 m into the storage facility in 3 days controlled in a similar manner to the jacking-up process. Figures VI-2, VI-3 and 4 show, respectively, the lifting frame of the reactor block, transportation in the horizontal direction, and lowering of the reactor block into the final location for long term storage.

The one-piece removal, including preparatory work, was completed in about 12 months requiring 14 300 person-days of effort. The maximum individual radiation exposure was measured to be 4 mSv. In addition, dismantling of cooling systems required 5300 person-days of effort over a 7 month period. The maximum individual radiation exposure was measured as

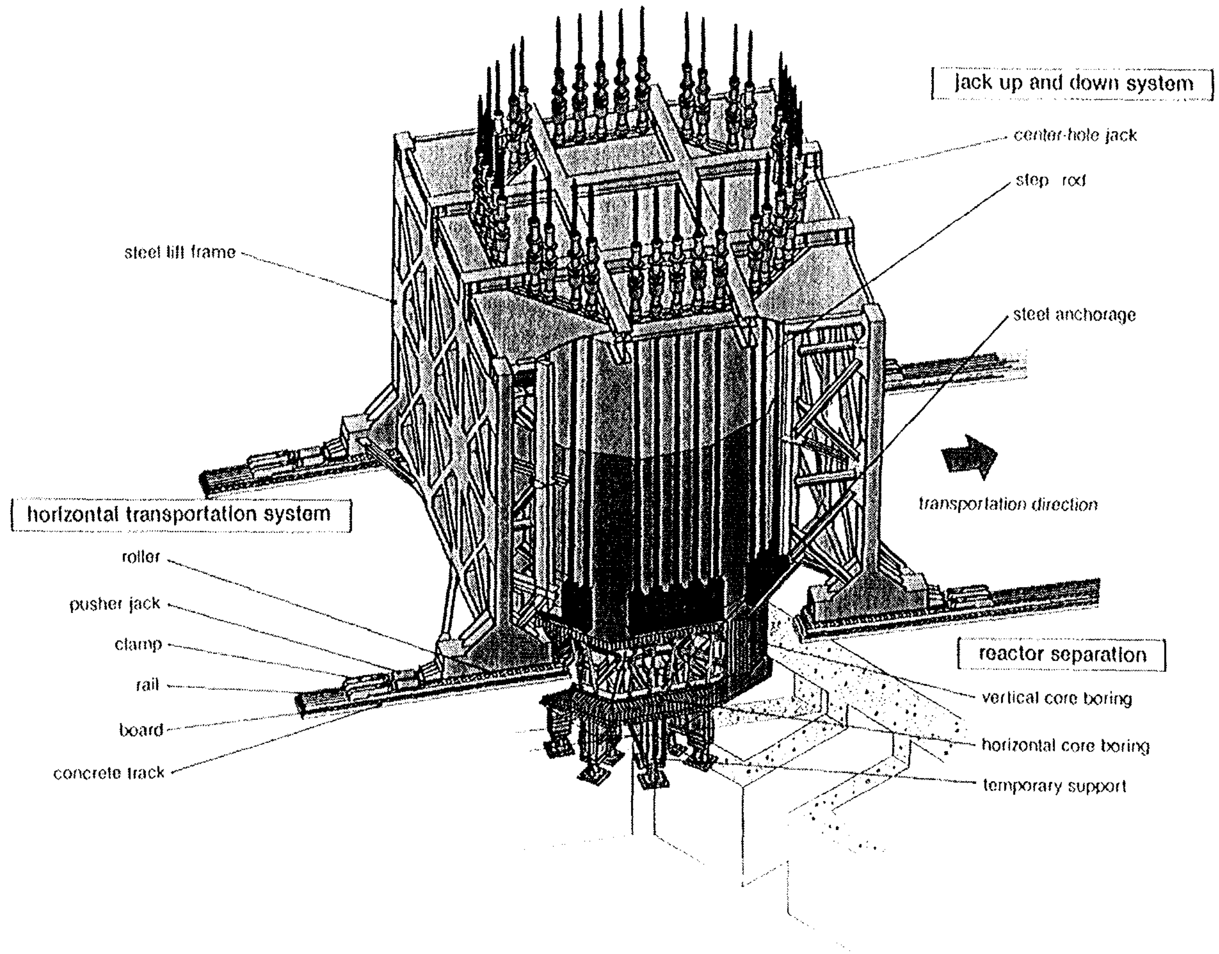

FIG. VI-1. Structure of the one piece removal system.

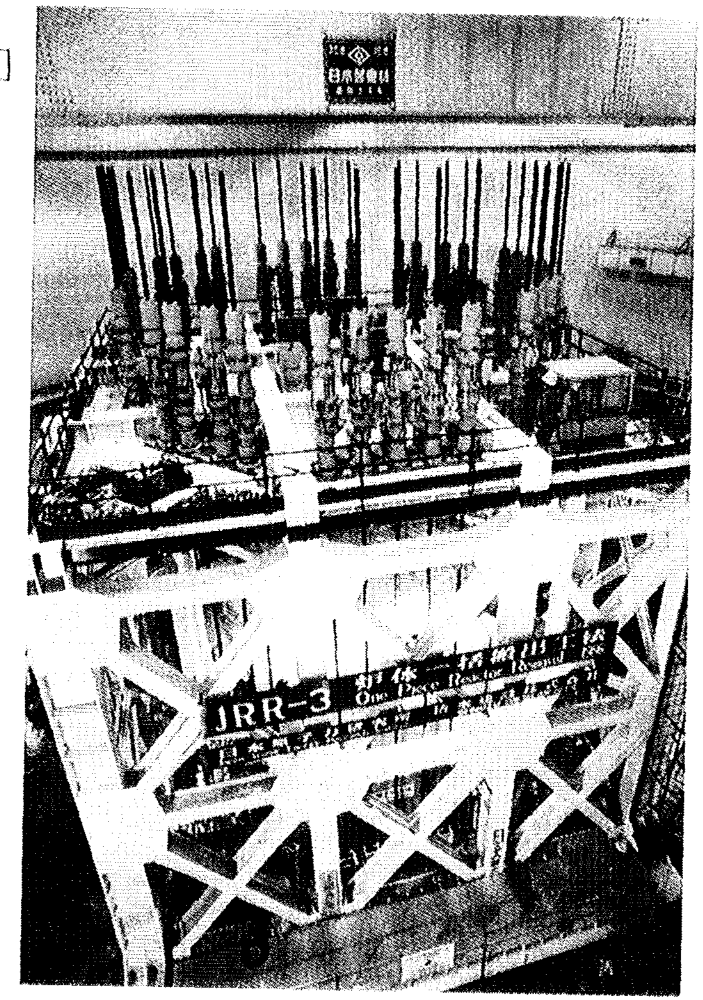

FIG. VI-2. Lifting frame of the reactor block.

FIG. VI-3. Transportation of the reactor block in horizontal direction through reactor containment.

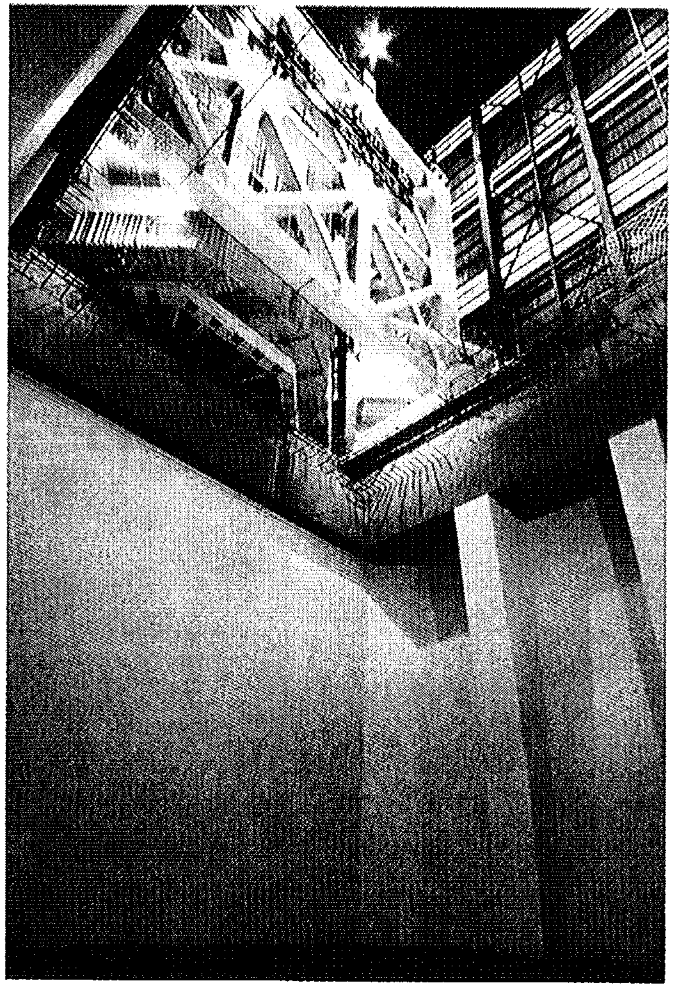

FIG. VI-4. Lowering the reactor block into final storage facility.

13 mSv. The waste arising from the dismantling activities consisted of 2200 tons for the reactor block, 540 tons for contaminated materials and 110 tons of secondary waste.

It was confirmed that the one-piece removal method was effective in minimizing environmental contamination, radiation exposure of workers and waste produced.

Storage and disposal strategies

The above decommissioning operations, including the one piece removal of reactor blocks, do not necessarily involve disposal of waste because no disposal site is currently available. The reactor blocks have been placed in long term storage and it was demonstrated that on-site transfer of large heavy items can be undertaken without technical difficulties.

Decisions about disposal, on-site or off-site, have not been taken yet.

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Annex VII

RUSSIAN FEDERATION[1]

The problem of non reactor and power reactor decommissioning in the Russian Federation has been under study since the end of the 1980s. The conceptual designs for decommissioning of the Leningrad, Novovoronezh, Beloyarsk and Armenian NPP, as well as the uranium-graphite reactors at PO "Mayak", Mining Chemical and Siberian Chemical plants have been developed.

The scope of work performed made it possible to formulate a tentative assessment of radiation hazards during decommissioning, the main provisions of which are shown below.

The uranium-graphite reactors and the power plants with RBMK and BN reactors operating in the Russian Federation are intended to be decommissioned in several steps [VII-1–VII-3] as follows:

(1) Transfer of the facility into a nuclear-safe state

(2) Conduction of a detailed engineering examination

(3) Preparing of the reactor plant for preservation

(4) Preservation of the reactor plant and its cooling systems

(5) Dismantling and disposal.

At the first step the nuclear fuel is discharged from the reactor core, cooling pools and the reactor plant buildings and is dispatched to the chemical reprocessing plant.

At step two the physical and radiological condition of the systems, equipment, reactor plant structures as well as stores, repositories and the site area are examined.

The preparation of reactor plant for preservation comprises decontamination of equipment and rooms, dismantling of some equipment and service lines which are not involved in the preservation process and are only of minor radioactive contamination. It also comprises processing and conditioning of accumulated operational liquid and solid wastes, and any decommissioning wastes and placing these wastes into metal and concrete containers, which are placed in stores or unused rooms at the reactor plant.

At the preservation step, activities are carried out to protect and preserve systems, equipment and unit structures for the period of time determined by the reactor plant decommissioning programme.

The final step of reactor plant decommissioning is dismantling, long term storage or in situ disposal.

The final step involves complete dismantling of the reactor, auxiliary systems and equipment, full or partial dismantling of the common plants rooms and structures, removal of all operational and decommissioning wastes, bringing the site conditions to a state which satisfies the normative documents and rules.

Long term reactor plant storage assumes that the preservation conditions provide safety and care to ensure the possibility of dismantling and final disposal (in situ) when this is required.

[1] This text is an edited version of the original text submitted by VNIPIET.

The in situ reactor disposal assumes the creation of protection barriers preventing radionuclide release into the environment from the reactor, reactor systems, equipment and rooms during the whole radiation hazardous period of wastes.

The main principles of this concept are to be reviewed for applicability to the uranium-graphite reactors.

Due to the depletion of operating resources at the reactors at the Siberian and Mining - Chemical plants, alternatives for decommissioning this plant were studied.

From the results of detailed engineering examination of one of the uranium-graphite reactors, the reactor and its auxiliary system contamination levels were assessed. The following elements were found in the graphite samples taken from different places: ^{3}H, ^{14}C,^{36}Cl, ^{41}Ca, ^{55}Fe, ^{60}Co, ^{59}Ni, ^{63}Ni, ^{94}Nb, ^{93}Nb, ^{54}Mn, ^{90}Sr, ^{134}Cs, ^{137}Cs, ^{144}Ce, ^{152}Eu, ^{154}Eu.

The graphite core total activity is about 29 000 Ci.

The reactor metal structure activity is due to ^{55}Fe, ^{60}Co, ^{59}Ni, ^{63}Ni radionuclides with a total radioactivity, after 30 years cooling of about 1200 - 13 000 Ci.

The average dose equivalent rate of the primary circuit piping is 0.8 - 14 mR/s. Some areas have dose rates up to 100 mR/s.

High radiation levels of the reactor structural elements, the absence of remote control means and mechanisms for the reactor plant equipment dismantling and lack of necessary waste treatment facilities showed that the dismantling alternative would need significant material, financial resources and incur significant doses.

Long lived radionuclides contained in the structural materials, the absence of adequate examination and safety assessment results would not allow a decision on in situ disposal of reactor plants. Therefore, preservation and long term storage for at least 100 years was recommended for the reactors.

Radioactive wastes formed during plant preparation for preservation and the wastes collected in the storage facilities which do not meet the up-to-date safety requirements are to be treated and placed in shielded reinforced concrete containers. The shielded containers are located in unused rooms of the reactor plant.

To protect the environment from radioactive contamination, barriers are erected around the reactor, which protect the reactor both from natural and man-made events (earthquakes, hurricanes, shock waves, collapse of structures and aircraft crash) and from any possible terrorist group actions.

As the primary barrier, the structural material of the reactor core strengthened by introduction of natural materials, such as sand, bentonite or their mixture is to be used.

A second barrier consists of metal structures surrounding the reactor and forming the reactor. The metal structure tightness is achieved by cutting off the reactor channels passing through the metal structures and sealing off all holes.

For a third barrier, concrete fencing structures around the reactor well are used. To build this protective barrier, all apertures and holes are filled with concrete. A tight enclosure not imposing any load on the reactor itself, is built above the reactor well and supported on the central hall floor. Metal beams are installed over the enclosure with reinforced concrete plates for impact protection against dropping items placed on them.

The reactor building, itself serves as the fourth barrier.

The accepted technology of long term reactor plant preservation does not rule out a future decommissioning strategy of dismantling and final disposal on the site.

The decommissioning strategy for a radioactive facility by in situ or on-site disposal mainly assumes the use of near surface disposal techniques due to the lack of favourable hydrogeological conditions for subsurface disposal in many cases.

The Russian concept of radioactive waste management allows near-surface disposal alongside deep geological disposal.

Safety is governed by the normative document "Near-surface radioactive waste disposal. safety requirements" within the framework of the "Federal special purpose program on radioactive waste and spent nuclear material management for 1996–2005". From this it is only permitted to send the short lived radioactive wastes for near-surface disposal with a potential hazard period not exceeding the effective life of engineered barriers. This effective life is limited to 500 years.

The required level of man and environmental protection for near-surface disposal is to be provided by the collective protection features of engineered barriers. The geological medium (natural barrier) serves as a redundant barrier for protection in the event of a catastrophic accident causing destruction of all engineered barriers.

The engineered barriers in the case of near-surface disposal must limit the release and retard the dispersal of radionuclides by localizing the range of their possible migration within the barriers.

The protection given by the radiological shielding will maintain the level of radiation impact upon the public within the prescribed limits.

The protection measures against direct radiation is a specified quota (upper limit) of the annual risk limit and, for normal conditions, is the effective exposure dose rate equivalent limit.

In agreement with Radiation Safety Norms [VII-4], the upper mortality risk limit will be within 5.10^{-5} per annum (risk limit for public) and 1.10^{-7} per annum (the risk at which the radiological hazard can be exempted from the radiation safety norms and from control).

The upper risk limit for the near-surface disposal is still to be assessed. The near-surface disposal safety is to be provided irrespective of the long term control and maintenance need. This requirement is met by the 1.10^{-7} per annum upper risk level.

As the protection function in the case of near-surface disposal is performed by the engineered barriers, this disposal technique may be achieved on all sites, but only for a limited group of radionuclides in the waste.

It is believed that decommissioning of a radioactive facility using the near-surface strategy for disposal in situ requires dismantling and removal of structural materials containing long lived radionclides (^{59}Ni, ^{63}Ni, ^{93}Nb) which are to be disposed of in deep geological formations.

REFERENCES TO ANNEX VII

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Annex VIII

SWITZERLAND

In January 1969 an accident occurred to the 10 MW experimental gas cooled reactor at Lucens. This reactor was housed in underground caverns. A fuel channel became blocked, overheated and the resulting overpressure ruptured the pressure tube and irreparably damaged the calandria tank. The internals were extensively dismantled over the next few years and decontaminated. Decommissioning challenges were given by lack of space, lack of floor loading capacity and inadequate lifting equipment. The decommissioning that was done was described at an International Symposium in Seattle [VIII-1].

In 1988, after some years of safe enclosure, it was decided to decommission and dispose of the remaining facility in situ and return part of the site to non nuclear uses. This was reported at a conference on decommissioning at Avignon in France [VIII-2]. Two of the three caverns (reactor and fuel pond caverns) were filled up with concrete to immobilize the waste within the bedrock. The lower part of the turbine hall and auxiliary rooms were also filled with concrete leaving the upper part free for alternative uses. An extensive drainage system was installed to monitor groundwater which was attracted inwards to a collecting and monitoring pond. This system has a minimum 30 year life. A discharge pipeline was also installed to divert effluent to a nearby river. The activity in the effluent was found to be extremely small and it was permissible to discharge it for dilution in the environment. Activated waste from the partial reactor dismantling work has been stored in heavy steel containers and it was reported that they are to be sent to a separate site at Wuerenlingen.

All the above construction and sealing work and the establishment of an effective monitoring system was initiated in 1994 followed by a monitoring phase which has not reported any significant release of activity.

It was reported in 1997 that the non nuclear areas in the caverns have been converted into a cultural archive for the Canton of Vaud. It houses a library, museum, a restoration workshop and secure storage for cultural objects [VIII-3].

The above description was recorded in more detail in Ref. [VIII-4].

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Annex IX

UKRAINE

The Chernobyl Nuclear Power Plant Unit 4 was a vertical pressure tube graphite moderated, boiling water cooled reactor (RBMK). There were four RBMK units operating at the Chernobyl site when the accident occurred. The accident took place on 26 April 1986 during a test of the turbine generator system prior to shutdown of the unit for planned maintenance. The test was apparently conducted under unproven conditions.

The initiation of the test under these unique, unsafe conditions caused a significant insertion of positive reactivity, resulting in a prompt critical condition. The resulting rapid power rise melted some of the fuel, ruptured the fuel cladding and injected fragmented and molten fuel into the coolant channel. The interaction of the coolant with the hot fuel fragments produced steam very rapidly. Sufficient force was generated to destroy much of the reactor, lift the top plate off the reactor and eject core material, breaching the roof of the reactor building.

Between 27 April and 10 May, 5000 t of boron compounds, dolomite, sand, clay and lead were dropped by helicopter onto the damaged reactor in an attempt to keep the fuel rubble subcritical and to control the unchecked discharge of radioactive material to the environment.

In the period following the accident, specialists began considering how they might isolate the reactor building itself, which continued to cause high levels of radiation. A number of approaches were considered to contain the destroyed unit to prevent further emissions.

Finally, on the basis of radiation measurements and the determination of the status of the fuel in the core, as well as an analysis of the remaining structure of the reactor building, engineers designed a structural covering with a span of 55 m that used the remaining walls and the top of the building as supports. The confinement system was designed to provide shielding and to reduce the danger of the spread of radioactive materials from the damaged facility. Other specifications laid down for the design of the 'sarcophagus' or shelter were: minimum construction time through the use of simple, reliable and proven methods; removal of residual heat and radiolytic hydrogen; minimum dose to building workers; and provision for performing monitoring and diagnostic work on the state of the active mass. Outer protective walls were built along the perimeter; inner concrete partition walls were built in the turbine hall between Units 3 and 4; and a protective steel roof over the turbine hall completed the structure. The outer structure of the shelter was therefore to be shaped by a number of buttressing elements rising in echeloned tiers, the dimensions and forms of which were determined in part by the features of the structure they enclose as well as the contaminated debris that could not be moved [IX-1].

Design work and construction on the encasement of Unit 4 proceeded quickly, allowing Unit 4 to be enclosed inside its concrete and steel shell by mid-November 1986. Since the structure was constructed in great haste, there is no assurance that the damaged building, including several hundred rooms and halls, will remain stable. Some degradation of the Unit 4 confinement system, possibly producing changes to the nuclear fuel geometry, could result in: fuel criticality and related radioactive releases; radioactive dust releases; reduced fuel cooling and, again, increased radioactive releases. To check and diagnose the condition of the structure, the temperature is measured in the space under the cover over the central hall and on the upper surface of the cover over the reactor vault, as well as in the components of the lower base plate and the surface of the covering over the pressure suppression pool. In order to refine data on the location and intensity of heat sources, the heat flux is measured continuously at the accessible points of the areas under the reactor and on the upper surface of the destroyed core. Gamma radiation is monitored in all maintenance areas of the plant, at most of the other accessible locations in the Unit 4 building and also in the space under the covering and on the upper surface of the destroyed core. The concentrations of hydrogen, carbon monoxide and water in the air are also monitored continuously.

In order to detect any chain reaction in the damaged fuel, neutron sensors have been installed. To prevent any possibility of a fission chain reaction in the reactor vault, a liquid neutron absorber was introduced. Vibroacoustic sensors were also installed to monitor the mechanical stability of the fuel mass and the structural elements of the shelter by recording any acceleration, velocity and vibration caused by shifts of major components. A set of computers monitor these sensors. Over 100 boreholes were drilled into the reactor pit where the reactor core was located before the accident and into the premises under the reactor. These boreholes have permitted remote observation of previously inaccessible rooms, including estimation of structural damage and determination of the location of the fuel fragments, finely dispersed fuel and fuel-containing mixtures. Also, samples of materials were obtained during drilling, enabling a more complete characterization of the fuel and its interactions with the materials surrounding the reactor. More details on the state of the Chernobyl sarcophagus and its monitoring systems are given in [IX-2].

The sarcophagus is one of many, although the largest and most dangerous radioactive waste storage facility resulting from the Chernobyl accident. It constitutes a nuclear and radiation hazard. The area is one of geological instability with the estimated frequency of an earthquake of force 6 once in 100 years. There is also some concern about the possible effects of bad weather. Both earthquakes and storms could cause a large release of radioactive dust and thus recontaminate the environment. The threat of this is enhanced in view of uncertainties as to sarcophagus physical state. Some of the structures, in particular, the upper plate of the reactor biological shield, with a weight of 2000 t, and some of the internal supports for the sarcophagus are in unstable, poor or unknown condition. It is evident that should these structures fail, radioactive dust would be released.

The situation inside the building deteriorates with time due to changes in temperature and moisture content, radiation effects, etc. resulting in structural changes, corrosion of the materials and redistribution of fissile and other radionuclides. Some evidence of these changes is following. Fuel and graphite dust originating from the disintegration of the fuel elements and the graphite moderator blocks is accumulating. Hydrogen is measured in growing concentration. The neutron flux has some fluctuations and generally has risen since the accident in 1986 which is attributed to water getting into the sarcophagus, naturally as well as due to the suppression of radioactive dust [IX-3].

Complete dismantling of the remaining structures ('green field' option) appears unfeasible at this time. Moreover, the green field option requires preliminary hermetic sealing of the shelter for the period needed for the development of dismantling technology.

This dangerous situation requires an urgent conversion of the sarcophagus into ecologically safe system. In view of the complexity of the task for this purpose in 1992 the Ukrainian government organized an international competition. The competition demonstrated that in proposed projects and technical solutions, approaches to the management of radioactive waste confined to the sarcophagus, as a final objective, were a measure for projects adequacy and maturity. Technically, and in the view of reaching the final objective, a phased approach to the conversion of the sarcophagus into ecologically safe system received a major support during the international competition and later feasibility studies by Alliance, a consortium comprising a group of finalists of the competition.

The phase approach proposed by Alliance [IX-4] includes stabilizing of the original sarcophagus and building a new structure (Shelter-2), followed by processing of radioactive waste, final dismantling of all structures and disposal. A period of 10 years would be necessary to build a new shelter which would both provide a containment for protecting the environment from uncontrolled releases of radioactive materials and function as a dismantling complex for the safe retrieval of radioactive waste during the next 20–25 years. The new structure would have cost well over $ 1 billion. An alternative proposal - which did not collect much acceptance - was to stabilize the fuel in situ by filling the sarcophagus with concrete and create a permanent giant monolith. The extremely high cost of Shelter-2 and other technical factors (like the continuing stay of fuel-containing material (FCM) in the shelter made it difficult for the Ukrainian government and potential donors to pursue this approach. Instead, plans were developed to improve the existing structure to ensure that it continues to contain the highly radioactive remnants of the RBMK reactor.

In April 1997, the Ukrainian and teams from major industrialized countries approved a Shelter Implementation Plan (SIP) developed by a commission of experts from the European Commission's Tacis programme, the US DOE and the Ukraine government.

The plan takes into account Ukraine's view that nuclear materials must be extracted from the sarcophagus. It features structural stabilization and partial disassembly to minimize the risk of collapse, and contains measures on shielding and access for safe management of FCM buried in the debris. The system of diagnostics is to be improved, including nuclear materials control. Creation of a new shelter to provide better protection and security is also possible. The plan is broken down into 22 individual tasks within five major areas:

(1) Reducing the probability of shelter collapse, including a design for stabilization, in particular: stabilization of the roof; stabilization of five weak sections of the sarcophagus; structural investigation and monitoring; seismic characterization; and geotechnical investigation.

(2) Reducing the consequences of accidental collapse: emergency dust suppression system; increase of emergency preparedness; and dust management.

(3) Increasing nuclear safety: contained water management, FCM characterization, criticality and nuclear safety tasks.

(4) Increasing worker and environmental safety: development and installation of a radiological protection programme; introduction of an integrated monitoring system; creation of an integrated database (configuration management); measures for shielding and access; and increasing industrial safety and fire protection.

(5) Long-term strategy and study of conversion of the shelter to an environmentally safe site: confirmation of a preliminary FCM removal strategy; development of technologies to extract FCM; development of a safe confinement strategy.

The overall cost of the project (to be completed by 2005) is estimated about $750 million. More technical and organizational details on the SIP approach are given in [IX-5–IX-7].

An important outcome of the SIP study was the identification of a number of priority tasks which are termed early biddable projects (EBP). These are essentially of an investigatory and preparatory nature and will serve to guide the early stages of the project. They are grouped into four technical areas [IX-6]:

(a) **Civil engineering:** Structural stabilization design; Integration and mobilization; Structural investigation and monitoring; geotechnical investigation; safe confinement strategy.

(b) **Operations and monitoring**: Seismic characterization and monitoring; radiological protection programme; industrial safety, fire protection; infrastructure and access control; integrated monitoring system; integrated database/configuration management.

(c) **Emergency systems**: Emergency preparedness; dust management; emergency dust suppression system; criticality control and nuclear safety; contained water management.

(d) **Fuel containing material (FCM):** FCM initial characterization; FCM removal and waste management strategy; FCM removal technology development.

Three to six companies were selected for each bidding of the four packages. Recently, the Ukrainian government and the European Bank for Reconstruction and Development agreed on projects that will allow work to begin in improving the sarcophagus over Chernobyl-4. In addition to technical factors of extreme complexity, handling the financial aspects of Chernobyl-4 stabilization remains a major issue.

A comprehensive overview and progress report of Chernobyl-4 decommissioning activities is given in recent papers at two international conferences [IX-8, IX-9]. In particular the nuclear and civil liability has been dealt with in [IX-9] and it is noted that the Ukraine has acceded to the Vienna Convention on Civil Nuclear Liability. Funding for the shelter implementation plan is being obtained and administered by the European Bank for Reconstruction and Development (EBRD). It has been reported [IX-10] that US$103 million has been made available for 1999. The first contract for remedial construction work was placed recently with the collective construction enterprise, Ukrenerbud [IX-11].

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Annex X

UNITED KINGDOM

Background

In 1990 studies were started on a range of options in the UK for the eventual decommissioning of nearly 40 large commercial gas cooled reactors that would need to be dealt with in the next 2 to 3 decades. Berkeley Power Station (2 reactors) was already shut down and some of the other older stations were due to follow in the next few years (three are currently shut down). Detailed studies had commenced in 1986 on two stations (Berkeley and Bradwell) to establish the technical problems and costs of all decommissioning activities right through to stage 3 dismantling and site clearance. These extensive studies were completed in 1990 and confirmed that the costs of dismantling the complex gas cooled reactors were high especially if early stage 3 dismantling and site clearance was considered. The studies of alternative options covered the commonly considered stage 1, 2 & 3 scenarios, safe enclosure scenarios (called 'Safestore' in the UK) and the in situ disposal scenario. Altogether 9 options were considered including variations and these were ranked in order of preference using multivariate attribute decision analysis. In all cases the fuel was removed from the reactor. Cost and environmental factors including radiological safety were dominant selection parameters. In situ disposal ranked high in terms of low cost and was considered acceptable for some reactor types in terms of environmental impact. It was not put forward as UK decommissioning policy however as there were concerns about public acceptance. The preferred UK option involved stage 3 dismantling deferred for more than 100 years and there was therefore no need to make an early decision and the in situ disposal approach was left in abeyance as a future option.

Study of on-site disposal for Commercial Power Reactors [X-1]

Selection of reactor types for study

In the UK all nuclear power plants are coastal except for Trawsfynydd power station which is located in Wales some 50 km from the sea. There are also two groups of reactor types; those with steel pressure vessels and contained within a concrete bioshield and those where the concrete bioshield is also the pressure vessel and containment. There is also an important variation in terms of radioactive inventory since some reactors contain large quantities of stainless steel which is important in terms of long term exposure risk because of the durable nature of stainless steel. In general Magnox reactors are largely constructed with low carbon steel while the advanced gas cooled reactors (AGRs) used stainless steel.

Four NPPs were chosen for engineering and feasibility studies viz. Berkeley (old Magnox, shut down, complex layout, coastal), Trawsfynydd (Magnox, inland site), Oldbury (Magnox, concrete pressure vessel, compact layout, coastal) and Heysham II (AGR, large construction, very near coastal shoreline). These four stations were considered to be representative of all the UK stations.

Berkeley was studied in the most detail because it was already undergoing stage 1 defuelling and is the most complex Magnox station. An additional more detailed study of the Heysham AGR site was done because of technical problems and because Heysham is a major site containing two AGR stations Heysham I and Heysham II.

The radiological studies however considered nearly all the UK commercial power station sites in terms of their hydrology and strata to enable preliminary pathway analysis to be done to ensure that there were no insurmountable problems in terms of making the radiological safety assessment.

Study methodology

Three specialist consultants were engaged to undertake the detailed studies (1 for engineering and 2 for radiological studies). They were selected for their particular expertise in heavy civil engineering work and for expertise in radiological impact and safety studies. It was believed that sea-dredged sand from shallow coastal waters would be a suitable material for covering the site and a specialist sea-dredging contractor was consulted. The studies were conducted over a period of about 9 months.

Gas cooled reactors are very large structures and it was considered that it would not be viable to move the reactor vessels or sink them lower into the ground. It was also believed that it was not desirable to undertake any major dismantling of the main radioactive components.

Engineering feasibility and design

Three of the four sites studied were coastal, and it was confirmed that it was feasible and economic to use sea dredged sand. This would be brought ashore in barges to a conveyor system which would deposit the material in stock piles adjacent to the building ready for placement. For the inland station spoil from a large slate quarry could be used and transported economically. The superstructure of the stations are not contaminated and it was shown to be economic to remove these as they were only light steel structures and so reduce the mound by some 20–30 metres. The stability of the mounds was addressed by limiting the slope to 27°, creating berms (level terraces) on the slope at about 10m intervals and by establishing rock filled toe drains around the mound to remove excess water and prevent slumping. In some cases walls had to be built on the shore line to limit sea erosion. The mound would be covered in top soil and planted with indigenous grass. The design life of the mound was required to be 10 000 years. Settlement of the mound was addressed and three important mechanisms could be foreseen. These are settlement during construction, settlement of the underlying state and collapse of internal voids if these remained. It was decided to fill all internal cavities within the reactor structures with grout. This operation although necessary accounted for a large proportion of the costs and the dose commitment to workers. None of these factors however were high compared to the total dismantling option. It was not considered necessary to cover the whole mound with impervious clay and water was allowed to drain or seep in naturally to be taken away by the toe drains. Concrete capping was included over the reactor vessels themselves to provide additional protection against water ingress into the vessels. The inherent structure integrity of the reactor bioshields was seen to be important.

Figures X-1–X-3 show the concept, feasibility considerations, drainage considerations and the material handling and placement. Estimates have been made of the materials required. As much as 1 million m^3 would be needed for the mound and about 1/3 million m^3 for infilling (grout).

Radiological assessment

Eleven sites were investigated in terms of their geology and hydrogeology using what data was available. From this a hypothetical site was devised with the basic assumption that whatever happens rainwater would infiltrate into the mound and radionuclides would move towards the human environment. The worst conditions for exposure of individuals were assumed where water returns with the least dilution. Doses to populations would be

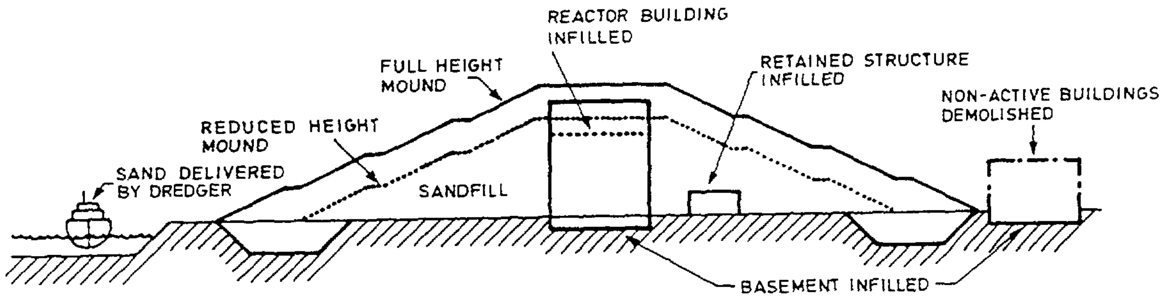

FIG. 1. In-situ decommissioning concept.

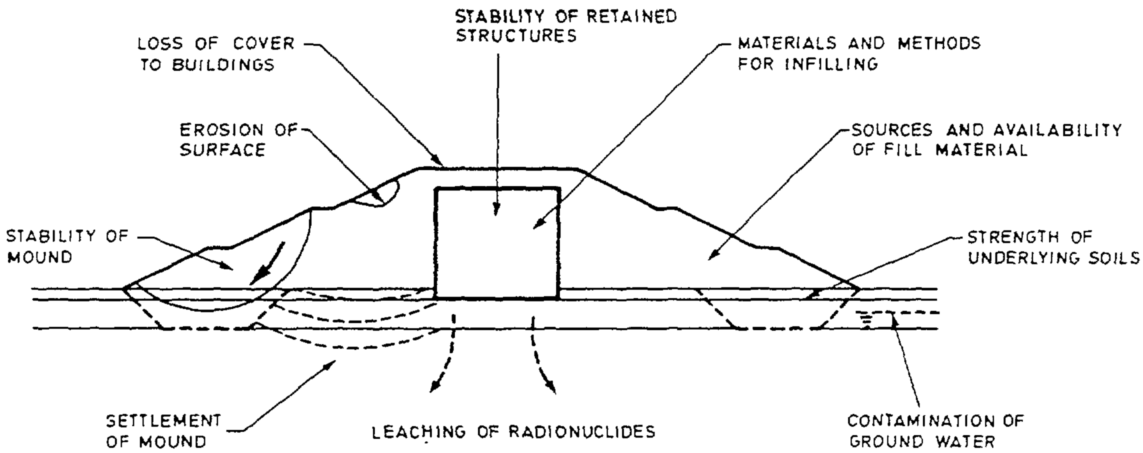

FIG. 2. Feasibility considerations.

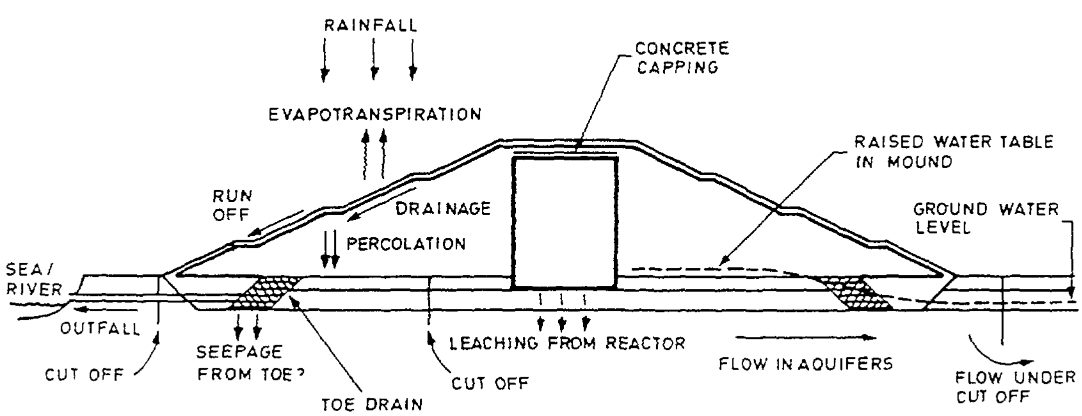

FIG. 3. Drainage considerations.

maximized if hypothetical site conditions were chosen which had high dispersion and dilution The site was assumed to be on an estuary but close to the sea with local streams which drained to the estuary Various exposure scenarios were chosen including the undisturbed site (reference condition), sea level rise, glacial erosion, earthquake, sea level fall, gas generation within the mound, human intrusion and subsequent site re-development, e g a housing estate on the mound in years to come (beyond an institutional control period of 300 years) Dose and risk calculations were done for each of the above scenarios and referred to a risk of fatal cancer of 10^{-6} per annum per individual Since this was for a hypothetical site no dose or risk values are published but comparisons showed that the risks are similar to those from post institutional management of low and intermediate level waste near surface disposal sites

Groundwater pathway studies showed that doses to critical groups such as farmers, fishermen, houseboat dwellers etc were acceptable The peak dose would occur within about 100 years assuming dispersion of radionuclides by groundwater as predicted by established pathway analysis

For various release scenarios studied the waterborne dose peaks between 50 and 100 years The doses to critical groups varies between 270 and 880 Sv and never exceeds 1 mSv Even this is believed to be pessimistic and is about the same order as natural background in the UK

^{14}C release from the graphite core of UK gas cooled reactors gives the main airborne collective dose to the world population The collective dose to the world population from this amounts to 2 9 10^3 Person Sv after 10^3 years The total local and regional exposure amounts to 72 person Sv truncated at 10^4 years and is dominated by consumption of drinking water

Human intrusion was a particular and special risk which could not be dismissed It was concluded that human intrusion risk would be little different from shallow waste disposal repositories beyond the period of institutional control Human intrusion risks were regarded to be higher than those assessed by groundwater pathways The presence of corrosion resistant materials was a significant factor in the preliminary assessment of risks from different reactor types

Conclusions

It was concluded that from the engineering and radiological aspects some sites would be more acceptable than others It was decided to defer any decision on in situ decommissioning at present and leave it as an option for future generations since no reactor dismantling is planned before about 135 years from shutdown

BNFL decommissioning strategies

Long term strategy has also been established for non-reactor nuclear sites owned by BNFL This is as follows

- The Springfields (fuel fabrication plants) decommissioning programme assumes that all significant contamination will be removed through the decommissioning and demolition process and that the majority of the waste arising will be disposed of as authorized waste by controlled burial at Clifton Marsh, other waste classified as LLW will be disposed of at Drigg A further review of environmental risk to determine the overall 'end of life' condition of the site is ongoing and further discussion with the Regulatory Authorities will be held

- Capenhurst (enrichment plants) strategy is for the following scope of work to be undertaken: removal of all buildings and foundation slabs (including roads and miscellaneous underground services) with the resulting holes backfilled using clean subsoil and topsoil, i.e. reinstatement to a condition of unrestricted use within the nuclear licensed site boundary.
- The Sellafield (reprocessing plants) end point is based on achieving an acceptable level of risk. Considerable effort within BNFL is currently addressing all aspects of Sellafield's long term strategy.
- A work programme is currently being formulated for the Calder Hall and Chapelcross Magnox reactor sites, the recent integration with Magnox Electric will widen the expertise available to deal with this topic. Safestorage is the most likely option.

UKAEA decommissioning

The United Kingdom Atomic Energy Authority (UKAEA) have not directly conducted on-site disposal of a facility, but have obtained special authorization to dispose of activated shielding and other components from the NIMROD facility (Rutherford Laboratory) in the Meashill Disposal Trench at Harwell.

The UKAEA did consider in situ disposal as one of the options for the Dounreay facility but, due to the particular circumstances on the site, they decided to defer any final decision for long term disposal. The current policy is to continue with long term care and maintenance after stage 1 decontamination work.

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Annex XI

UNITED STATES OF AMERICA

On-site disposal of nuclear facilities has been accomplished for a few reactors and for some non-reactor nuclear facilities in the USA. The principal reactor cases are described in the first section of this annex. Some examples of on-site disposal of non-reactor nuclear facilities are described in the second section. The current regulatory climate in the USA for on-site disposal of licensed nuclear facilities is discussed in the third section.

On-site disposal of shutdown reactors in the USA

In the more than 50 cases of reactor decommissioning in the USA since 1954, only-five installations have utilized the on-site disposal strategy. The rest were either immediately (within a few years following final shutdown) dismantled, or were placed into safe storage for an extended time period and eventually dismantled. Most of the reactors that were dismantled were small research reactors, with only a few power reactors being dismantled. Three of the five reactors, which were disposed of on-site using the in situ option, were relatively small demonstration plants built by the US Atomic Energy Commission in the early days of nuclear power development, namely:

The Hallam nuclear power facility,

The Piqua nuclear power facility, and

The Boiling Nuclear Superheater Power Station (BONUS).

The fourth and fifth reactors to be disposed of using the in situ disposal strategy were a small 10 $MW_{thermal}$ water-cooled facility designed to support studies on nuclear aircraft propulsion (AFNECR), and a complex containing early reactor experiments built during the development of boiling water reactors (BORAX-I through -V).

More recently, the US Department of Energy (DOE) has selected the on-site disposal strategy (on-site transfer and disposal option) for the shutdown plutonium production reactors at the Hanford Site. In addition it has been reported recently [XI-18] that US Regulatory and Department of Transportation approval has been given to dispose of the Trojan reactor vessel at the low level waste (LLW) disposal site near Richland, Washington. Each of these cases is described briefly in subsequent subsections.

The unconditional release levels specified at the time of the disposal of the three demonstration power reactors were:

(a) the external radiation hazards are "safe" if the surface dose rate from each component is less than 0.2 mrem/h .

(b) the internal radiation hazards are "safe" if:

- specific activity and solubility of the residual radioactive materials are such that the applicable non-occupational maximum permissible concentrations cannot be exceeded;
- the total activity, times the fraction deposited upon ingestion or inhalation, is less than a non-occupational maximum permissible body burden; and
- replacement of the total amount of the element in the standard man by the radioactive isotope of that element taken from the reactor did not exceed the

allowable non-occupational body burden. This criteria is not applicable for elements such as europium for which the intestine or lung is the critical organ.

The **Hallam nuclear power facility** was a demonstration plant, located in Hallam, Nebraska. The station was initially operational in 1963, and was finally closed in 1966, with on-site disposal (in situ option) completed in 1969. The reactor was graphite-moderated, cooled using liquid sodium, and was designed to produce 256 $MW_{thermal}$. Problems with leakage of sodium into the graphite moderator and the anticipated costs for repair led to the early closure of the plant. All irradiated nuclear fuel and all bulk sodium was removed from the plant, with the residual sodium rendered inert. All residual radioactive sodium was transported to a federal nuclear installation for storage and eventual disposal. Heat exchangers and other system components were dismantled and removed. Remaining radioactive components and materials were sealed in the underground vaults of the plant. All penetrations were seal-welded, the reactor was sealed beneath two plates of 0.5 in. steel which were welded in place, and the entire entombment was covered with plastic film, tar and earth. An estimated 300 000 Ci (~1.1 10^4 TBq) of residual radioactivity were contained within the enclosure at the time of on-site disposal. The State of Nebraska periodically inspects the site for structural integrity and radioactivity containment, but no monitoring systems were installed. The details of the on-site disposal activities are reported in reference [XI-1].

The **Piqua nuclear power facility** was a demonstration plant, located in Piqua, Ohio. The station was initially operational in 1963 and was finally closed in 1966, with on-site disposal (in situ option) completed in 1969. The reactor was organically cooled and moderated, and was designed to produce 45 $MW_{thermal}$. Problems with the organic cooling system led to the early plant closure. The irradiated nuclear fuel, selected reactor core components and other radioactive materials were removed to a federal nuclear installation. The organic coolant and moderator was disposed of by burning. The reactor vessel, thermal shield, grid plates, and support barrels remained in place. The vessel penetrations were seal-welded, the vessel was filled with sand, and the enclosure penetrations were plugged. The enclosure was sealed with a waterproof barrier and a concrete cover. Contaminated piping and equipment was either decontaminated or removed from the reactor building, which was converted into a warehouse. An estimated 260 000 Ci (9.6 10^3 Tbq) were sealed within the enclosure at the time of on-site disposal. Analyses predicted that the contained radioactivity would decay to unrestricted release levels after about 120 years. No monitoring systems were installed. Details of the on-site disposal activities are reported in Refs [XI-2, XI-3].

The **Boiling Nuclear Superheater Power Station (BONUS)** was a demonstration plant, located in Rincon, Puerto Rico. The station was initially operational in 1964 and was finally closed in 1967, with on-site disposal (in situ option) completed in 1970. The reactor was cooled and moderated using boiling light water, and was designed to produce 50 $MW_{thermal}$. Difficulties with the superheat system led to the early plant closure. The irradiated nuclear fuel, selected radioactive materials, and unirradiated nuclear fuel were removed to a federal nuclear installation. The penetrations through the lower portion of the reactor building were plugged and sealed, including a concrete slab which sealed off the upper surface of the engineered barrier enclosure (i.e. the reactor building). Figure XI-1 shows the reactor vessel being loaded with selected items for entombment. Figure XI-2 shows the final concrete slab being placed to form the upper engineered barrier. An estimated 50 000 Ci (1.9 10^3 TBq) of radioactivity were sealed within the engineered barrier structure at the time of closure, comprised of about 71% ^{55}Fe, 29% ^{60}Co, and <1% ^{63}Ni. The allowable dose rate at 1 cm from the enclosure surface was required to be <0.2 mR/h on the average, with hot spots not exceeding 1 mR/h. A hazards analysis for the engineered barrier structure, assuming a severe earthquake followed by a tsunami, concluded that such an accident would not result in unacceptable radiation doses. Details of the on-site disposal activities are reported in [XI-4].

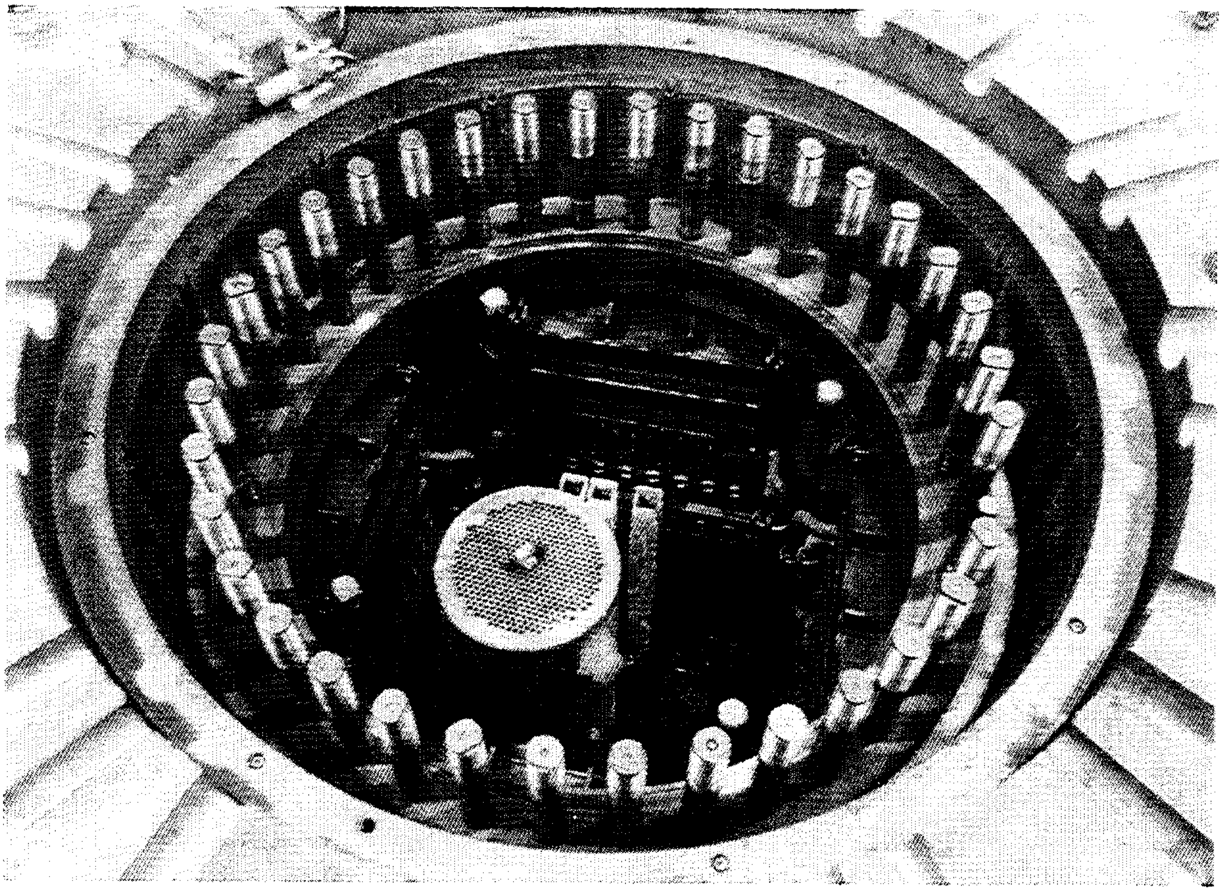

FIG. XI-1. The BONUS reactor vessel being loaded with radioactive items to be entombed.

Air Force Nuclear Engineering Center Reactor (AFNECR): The AFNECR was a small water-cooled, 10 $MW_{thermal}$ test facility, located at Wright-Patterson Air Force Base in Dayton, Ohio. The facility began operation in 1967 and was closed in 1970. The reactor and its associated experimental facilities were located within a domed steel containment structure that extended from about 50 ft below grade to about 110 ft above grade and was about 82 ft in diameter. The reactor was entombed within the enclosure during 1970 and 1971 by removing non-structural radioactive components, filling radioactive cavities (reactor tank and test cells) with sand, sealing the outside of the concrete biological shield, and placing additional concrete shielding as necessary to reduce the radiation levels in accessible areas to less than 0.2 millirem per hour. The facility remained in that condition until 1987, when a study was conducted to review further options for completing the decommissioning of the facility. As a result of this study, the entombed structure was upgraded and entombment was continued. All radioactive materials external to the structure were removed, including underground tanks (Ref. [XI-5], p. 30).

Boiling Water Reactor Experiment (BORAX-I, II, III, IV, and V): The BORAX reactor complex began with a small, open-top BWR (BORAX-I) that operated at 1.4 $MW_{thermal}$, and was deliberately destroyed in a safety experiment in 1954. After the experiment, the site was

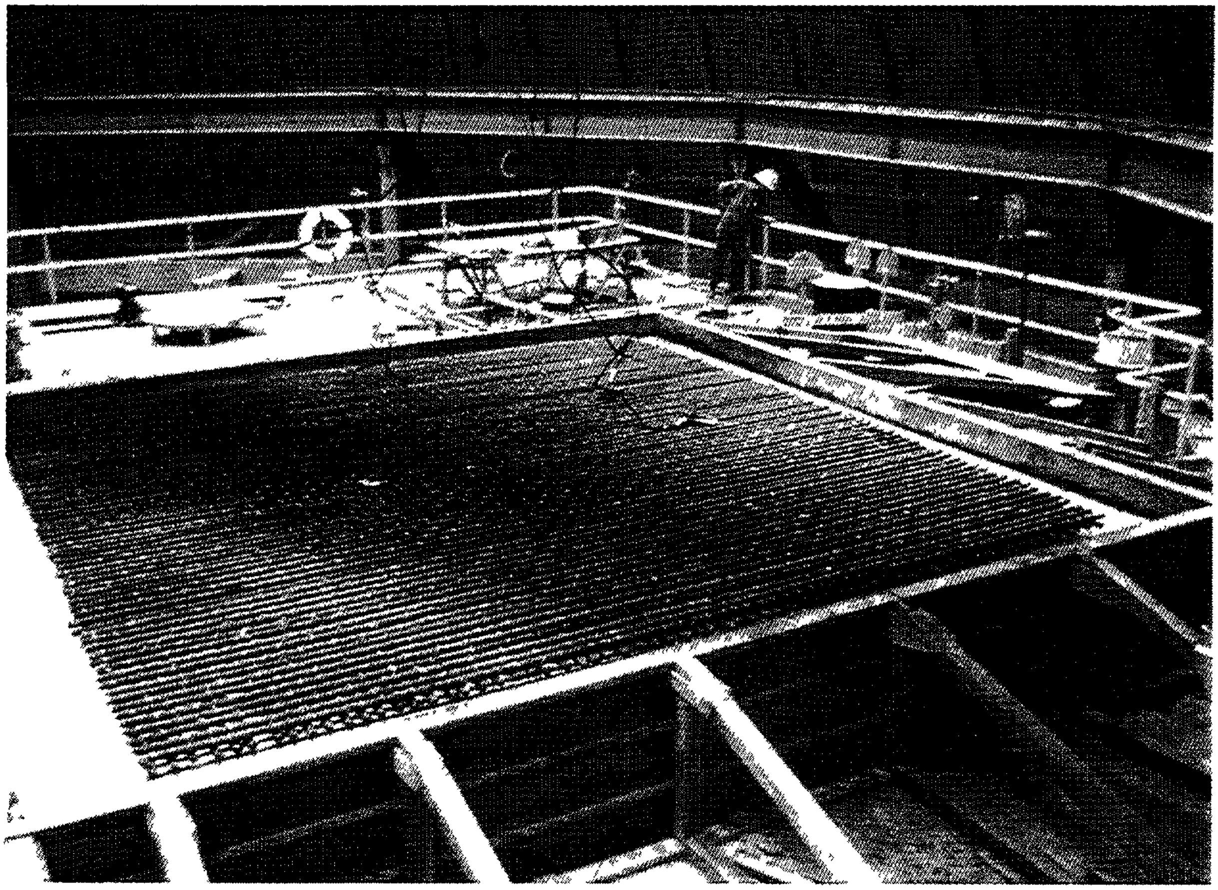

FIG. XI-2. The upper engineered barrier being placed over the BONUS reactor.

cleaned up and the reactor vessel was buried in place. A biobarrier cap was placed over the site during Fiscal Year 1997. BORAX-II (6 $MW_{thermal}$), BORAX-III (15 $MW_{thermal}$), and BORAX-IV (20 $MW_{thermal}$) reactors utilized a common set of auxiliary facilities and the same reactor vessel but with different fuel designs and configurations. Following shutdown of BORAX-IV in 1958, the reactor pit containing the reactor vessel and related components was backfilled with sand and capped with concrete. BORAX-V was installed in a new reactor vessel located in an expansion of the reactor building, but also utilized the common set of auxiliary facilities. BORAX-V was placed in standby in 1964, and the facility remained in safe storage until it was radiologically characterized in 1979. Removal of auxiliary structures and components began in 1985 and continued for about 10 years. Final decommissioning of the reactor building began in 1996 with removal of asbestos (which was bagged and placed in the reactor pit) and lead shielding which was recycled or disposed of, and extensive sampling of soils and concrete around the site. Contaminated materials were disposed of as low-level or mixed wastes and clean concrete rubble was sent to a sanitary landfill. A 2-inch thick steel shielding lid was placed on the top of the vessel and the reactor pits were backfilled with clean material and the shield cover blocks reinstalled. The site was regraded to natural contours and planted with native grasses, with warning signs installed on the site perimeter fence. The work was completed in 1997, at total cost (1994–1997) of just over $1 million [XI-6]

Hanford plutonium production reactors: The on-site disposal strategy has been selected for the eight graphite-moderated, water-cooled reactors that were constructed at the Hanford Site between 1943 and 1955 and were operated between 1944 and 1971. After shutdown, each reactor was placed in safe storage and monitored under a maintenance and surveillance programme.

The DOE has analyzed several decommissioning strategies for the reactors, including no action, immediate one-piece removal (on-site transfer and disposal), safe storage followed by deferred one-piece removal (on-site transfer and disposal), safe storage followed by deferred dismantling (on-site transfer and disposal), and in situ disposal. Because the environmental impacts of these strategies did not offer a strong basis for selection among the strategies, the DOE selected safe storage followed by deferred one-piece removal as the preferred decommissioning option for the Hanford reactors [XI-7], based on its review of environmental impacts, total project costs, and the results of the public hearing process. The preferred option is one type of on-site disposal as it consists of the decontamination and demolition of the peripheral support structures for the reactors and transport of each reactor block intact from its present location to another area (200-West) of the Hanford site for disposal in a low level waste (LLW) disposal facility, leaving the original site suitable for release. At present the 105-C reactor, the first of the eight reactors to be decommissioned, has been placed in long term (75 years) safe storage by decontaminating and demolishing the surrounding support facility structures, reducing the immediately adjacent building walls to the level of the top of the reactor shield and placing a new stainless steel roof over the remaining structure, with the building access points being secured. The footprint of the facility has been reduced from about 5000 m^2 to about 1500 m^2. Monitoring of groundwater beneath the reactor will continue and the interior of the enclosure will be inspected about every five years. Similar actions will be taken for the remaining seven reactors, with the possible exception of B-Reactor which has been designated a national historical engineering monument.

The in situ disposal option examined for these reactors assumed that the peripheral support structures were decontaminated and demolished, the reactor block and its shields were sealed within an engineered barrier enclosure, and the enclosure was covered with a protective earthen mound. Surfaces within the facility that are potentially contaminated would be coated with a fixative to ensure retention of contamination during subsequent activities. The major voids beneath and around the reactor block would be filled with grout or gravel as a further sealant and to prevent subsidence of the final overburden. Piping and other channels of access into the reactor building would be backfilled with grout or similar material to ensure isolation of the reactor from the surrounding environment. Finally, the reactor block enclosure and the spent fuel storage basin, together with the contained radioactive material and added gravel and grout, would be covered to a depth of at least 5 m with a mound containing earth and gravel. Rip-rap (layers of increasing diameter rocks) on the sides of the mounds would ensure long term structural stability and provide protection against erosion in case of a flood.

Trojan research reactor: Portland General Electric Co. (PGE) has obtained regulatory and transportation approval to send the reactor vessel with its internals intact by barge on the Columbia river for disposal [XI-8]. The Washington Department of Health has agreed that the waste package meets state and federal requirements for waste classification and disposal. The proposed disposal site is the US Ecology Inc. Low Level Radwaste (LLW) facility near Richland, Washington and shipping is proposed in the third quarter of 1999. Some of the internals are greater than Class C waste but PGE is using USNRC guidance on concentration averaging of the whole package. Disposal has been allowed in one piece because of the robust nature of the vessel, the internal filling with concrete and the fact that most activated components are stainless steel. In addition, one piece shipment is expected to result in lower doses to operators. The cost saving is significant being $15 million less than for cutting the vessel into at least 40 pieces.

On-site disposal of non-reactor nuclear facilities in the USA

The nuclear facilities that have been disposed of using the on-site disposal strategy in the USA are owned by the DOE and are located on major DOE sites (e.g. Hanford, Idaho National Engineering and Environmental Laboratory [INEEL], Savannah River). Several examples of these types of disposal are described in subsequent subsections, together with descriptions of several studies related to on-site disposal.

Hanford Hot Semiworks Complex The Hot Semiworks Complex on the Hanford Site operated as a pilot plant from 1949 to 1967 to develop two different methods for fuel reprocessing (Redox, Purex) and a method for separating strontium from high-level liquid waste. The complex was maintained in safe storage from 1967 until 1983 when the decision was made to decontaminate and decommission the plant. After evaluation of alternatives, the approved method of decommissioning was a combination of partial dismantling and on-site disposal, based upon the stability of the end product, cost of the project, and projected impacts. The below-grade process facility was sealed, the exhaust filter systems and ducts were grouted, and several on-site low-level waste disposal areas and the ventilation stack rubble were all covered with an engineered earthen barrier, consisting of a layer of earthen material at least 4.6 m in thickness placed over the surface of the buried waste or any contaminated surfaces. The liquid waste storage tanks were grouted, but, before they were covered over, it was discovered later that some radioactive liquids remained beneath the grout in at least one tank. Final action to reach closure for the tanks remains to be accomplished. A review of the availability of acceptable nearby native material for use in the construction of the barrier disclosed that the bottom ash from a nearby coal-fired plant steam plant would function at least as well as soil for construction of the main body of the barrier. At the grade level below the barrier, a layer of ceramic discs was emplaced to warn any inadvertent intruder into the barrier at some point in time after loss of institutional control, which is assumed to be 100 years from the present. A filter bed composed of sand and gravel was placed on top of the bottom ash to preclude downward wicking of moisture from the overlying topsoil to the fly ash. A geotextile (fiberglass drainage fabric) was placed over the filter bed, and a thick layer of topsoil covered the fabric layer. The topsoil was revegetated with a shallow-rooted bunch grass to enhance evapotranspiration. More details are given in [XI-9].

183-H solar evaporation basins: In situ disposal was the selected decommissioning option for the 183-H solar evaporation basins. Decommissioning activities for this facility include removal of solid waste, decontamination (shotblasting) of basin concrete walls, solidification and removal of liquid waste drums and sampling, and rubblizing and removal of the decontaminated basin concrete walls, floors, and footings for backfilling adjacent clearwells. Some of the soil beneath the basins was excavated and disposed of in the Environmental Restoration Disposal Facility. About 6600 cubic yards of material was disposed of as LLW. Some chemical contaminants remain in the soil at depths ranging from 18 to 40 ft. beneath the basin floors. The remaining soil depressions were backfilled with clean material. The last task was the installation of a top soil cover to enhance the moisture storage and lateral draining while minimizing water infiltration, erosion, differential settling and sedimentation, and long term maintenance. A multi-layered engineered barrier was not required. Because of the residual chemical contamination in the soil, a pump and treat (ion exchange) process is being applied to the underlying groundwater system for an extended time, and groundwater monitoring will be required for a minimum of 30 years from the date of closure [XI-10].

Disposal of **immobilized low-activity waste from reprocessing waste storage tanks:** Current plans at Hanford [XI-11] for disposition of the large volume of low-activity wastes arising from the cleanout and decommissioning of the many reprocessing waste storage tanks on the site are focussed on immobilizing these wastes by vitrification into steel boxes. The steel boxes are to be placed into below-ground concrete vaults located in the 200 East Area, in the vicinity of the waste storage tanks in that area. When filled, the vaults are to be closed, covered with soil, and capped with moisture barriers. Retrievability has to be maintained for 50 years following vault closure. An estimated 200 000 m^3 of vitrified waste is expected to be dispositioned in this manner. To be eligible for this mode of disposal, the radioactivity of the waste must not exceed Class C levels, as defined in the Code of Federal Regulations, Title 10, Part 61 (10 CFR 61).

A related approach is being carried out at the Savannah River Site. The Saltstone Facility, which began operations in 1990, is an integral part of the site's high-level radioactive waste treatment and disposal capability. Salt cake and concentrated salt solution make up approximately 93 per cent of the 34 million gallons of material in the site's radioactive waste storage tanks.

Pretreatment of the tank waste separates soluble salts from insoluble sludge to generate a salt solution, which is further treated to remove all but 0.1 per cent of the radioactivity. The decontaminated low-level, radioactive salt solution is then sent to the Saltstone facility for stabilization and disposal. At the facility, the decontaminated salt solution is mixed with cement, fly ash, and blast furnace slag to form a hydraulic waste form, which is then pumped into large concrete vaults. Hydration reactions occur as the waste form cures, which results in a solid material having very low leaching properties. Tests show that any waste leached from the concrete will remain within drinking water standards established by the Environmental Protection Agency. In addition, wells near the edge of the disposal site will be monitored to ensure that the groundwater meets these standards.

Disposition of large reprocessing canyon buildings: An approach is under study at Hanford [XI-12] for using the deactivated large reprocessing canyon buildings as receptacles for low-level wastes. While several variations on this approach are being considered, the approach currently most favored would place a layer of low-level wastes into the canyon and backfill the layer with grout. Then, another layer of waste would be placed and grouted, continuing until the canyon volume was filled. Additional layers of lower activity wastes would be placed around the outside of the canyon building and the total disposal system would be covered with soil until the building and the surrounding wastes are covered with a sloping mound. The mound (and included canyon building) would be capped and covered with moisture barriers, creating what are essentially above-grade LLW disposal sites. The canyon buildings are all located in the 200 East and 200 West areas at Hanford, in the vicinity of the reprocessing waste storage tanks and major LLW burial facilities, and would be within the region of the site that will remain under federal control, probably in perpetuity.

Waste calcining facility: A project is under preparation at INEEL to partially dismantle and stabilize the waste calcining facility (WCF) at the Idaho Chemical Processing Plant. WCF was used from 1963 until 1982 to evaporate and oxide liquid high-level radioactive waste in a high-temperature fluidized bed. About 2300 m^3 of calcined waste is stored at the Calcine Solid Storage Facility. WCF is a heavily reinforced concrete structure with one ground-level and two below-ground levels. DOE wants to close and dismantle WCF to reduce the risk of radiation exposure and hazardous releases and to eliminate the need for long term surveillance and maintenance. Foreseen actions include:

- Filling the below-grade vessels and operating corridors with grout to prevent future subsidence and maintain the integrity of the closure cap;

- Disconnecting and/or blocking all lines in or out of the facility to prevent moisture from entering the WCF; and
- Dismantling the superstructure and covering the encased process equipment and rubble with a concrete cap to minimize future water infiltration [XI-13].

Decommissioning cost study: A study was conducted to estimate the costs for decontamination and decommissioning of the major facilities on the Hanford Site [XI-14]. Three options were evaluated in this analysis:

(a) complete decontamination and dismantling of the facility, with the facility structures demolished to 5 ft. (1.5 m) below grade, and off-site disposal of all wastes;

(b) complete decontamination and dismantling of the facility, with the facility structures demolished to 5 ft. (1.5 m) below grade, and with disposal of LLW at a central on-site disposal facility and disposal of clean rubble in the remaining below-grade portions of the facility; and

(c) decontamination and dismantling of the facility to 5 ft. (1.5 m) below grade, with the LLW placed in the remaining below-grade portions of the facility. The facility structure is demolished to 5 ft below grade, with any remaining below-grade volume filled with clean rubble.

In all options, the residual below-grade structures are filled and covered over with clean top soil, which is planted with native vegetation. Any hazardous materials are transported to an on-site hazardous waste disposal facility, and any transuranic wastes are transported to an off-site TRU disposal facility. Excess clean structural rubble is transported to an on-site clean landfill. The study results showed significant cost advantages for the third option (in situ disposal).

HWCTR decommissioning analysis: The Heavy Water Components Test Reactor was built on the Savannah River Site to test components for use in the heavy water cooled and moderated plutonium production reactors at that site and was operated from March 1962 until December 1964 when it was permanently shutdown because of problems with heavy water leaks and fuel failures. The facility was placed in standby for a year and was then retired in place. The facility was placed into a condition approximating safe storage in 1975 and has continued in that state ever since. A study of decommissioning options was carried out in 1976, which concluded that dismantlement, protective confinement (safe storage), or entombment would be equally acceptable, but no actions were taken at that time. In 1994, four auxiliary buildings on the site were demolished and disposed of as clean waste. In 1995, detailed radiological contamination surveys of the facility were conducted, and asbestos thermal insulation was removed from piping and components. Additional radioactive contamination surveys and visual inspections of the facility were conducted in 1996 which supported the preparation of an analysis of removal alternatives for the HWCTR [XI-5]. This study examined dismantlement, partial dismantlement and interim safe storage, conversion to beneficial reuse, and entombment, and concluded that while both dismantlement and entombment were equally viable, dismantlement better suited the the long-range plans for the site.

Regulatory climate for on-site disposal of power reactors in the USA

The three on-site disposal (entombment) projects discussed above occurred in the early days of nuclear power development in the USA, when both the development and the regulation of nuclear reactors was carried out by a single agency, the US Atomic Energy Commission, and prior to the creation of the Nuclear Regulatory Commission (NRC) with its structured body of rules and regulations for dealing with reactor construction, operation, and decommissioning. Since those early days, a number of concerns have arisen regarding the complexity and long term safety of on-site disposal of radioactive materials at the reactor locations. After an effort of many years, the Decommissioning Rule [XI-15] was issued in 1988, which established the basic regulatory framework for decommissioning of all licensed nuclear facilities. This rule was supported by the generic environmental impact statement on decommissioning [XI-16], wherein the potential impacts of decommissioning the various types of licensed nuclear facilities were examined, based on a series of detailed technical studies on decommissioning individual types of nuclear facilities. The 1988 Decommissioning Rule was amended and revised in 1996 [XI-17] to clarify ambiguities in the rule and to modify procedures to reduce regulatory burden, provide greater flexibility, and allow for greater public participation in the decommissioning process. No significant changes were made to the rule requirements for the on-site disposal strategy.

In the 1988 rulemaking, the NRC allowed a period of up to 60 years for a reactor licensee to complete the decommissioning of his facility. This period was predicated upon the assumption of a safe storage period of up to 50 years and a deferred dismantlement period of up to 10 years or less. No specific connection was made between the 60-year period for completing decommissioning and the on-site disposal strategy. The possibility to select on-site disposal was left open, however, by allowing consideration of a decommissioning strategy that provided for completion of decommissioning beyond 60 years after final shutdown for power reactors if necessary to protect public health and safety. Factors set forth in the rule which could be considered in evaluating such an extended decommissioning period included: the unavailability of disposal capacity at low-level waste (LLW) disposal sites; the presence of other nuclear facilities on the site; and other unspecified site-specific considerations, thereby providing some regulatory flexibility.

Following on-site disposal, the facility would remain under the nuclear license, with surveillance and maintenance by the licensee, until it could be determined that the contained radioactivities had decayed to unrestricted release levels. For power reactors that have operated for the duration of their operating license (40 years maximum), the unrestricted release criteria would certainly necessitate the removal and off-site disposal of the highly activated reactor vessel internals, perhaps the removal of some of the activated biological shield materials, and perhaps the removal of the entire reactor pressure vessel itself, all prior to sealing the engineered barrier enclosure for on-site disposal.

In addition to the removal of the materials described above, an accurate inventory would be required of the quantities of the various radioactive species remaining within the enclosure at the time of sealing, to assure that the activity levels of the residual radioactive materials within the enclosure will have decayed to unrestricted release levels by the end of the licensed entombment period. Measurements necessary to determine this inventory would require a considerable investment of time, funds, and worker radiation dose. Also, the actual duration of the licensed entombment period could depend upon the quantities and species of radioactivity present in the inventory at the time of closure. The possibility of entombment periods exceeding 100 years raised concerns regarding the structural integrity of the engineered barriers, and the viability of maintaining institutional controls over enclosure sites for extended time periods.

An additional concern arose with the potential for creating a number of single-purpose LLW disposal sites in locations that may not be well-suited environmentally for long term LLW disposal. As a result of these various concerns, power reactor licensees were discouraged from seriously considering the on-site disposal strategy for decommissioning.

Historically, on-site disposal (entombment) has not been considered a viable decommissioning strategy by the NRC, for the reasons outlined above. However, entombment of power reactors is now being reconsidered by the NRC because of interest expressed by some power reactor licensees. Entombment is being re-evaluated relative to todays regulatory environment, for the purpose of determining whether those previous concerns that discouraged entombment are still valid. Some of the issues being considered include:

- re-evaluation of previous assumptions regarding long term (>>60 years) integrity of the enclosure structures;
- the possibility of other entombment scenarios such as long term storage followed by entombment for extended time periods;
- re-evaluation of previous assumptions regarding the stability and longevity of institutional controls for entombed facilities for extended time periods;
- re-evaluation of the possible impact of the entombment strategy on the volume of LLW arising from decommissioning, and on disposal space availability at LLW disposal sites;
- consideration of the impacts of one-piece removal of the reactor pressure vessel and its internals on the cost and dose comparisons of entombment with immediate dismantlement and safe storage followed by deferred dismantlement; and
- consideration of the recent rule amendments for License Termination of Licensed Nuclear Facilities [XI-18] for both unrestricted use and conditional release scenarios.

A new set of regulations governing the use of the on-site disposal (entombment) strategy for power reactors may arise from the on-going review of this decommissioning strategy.

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GLOSSARY

A selection of definitions taken from IAEA Radioactive Waste Management Glossary, 1993 except where noted.

barrier. A physical obstruction that prevents or delays the movement (e.g. migration) of radionuclides or other material between components in a system, e.g. a waste repository. In general, a barrier can be an engineered barrier which is constructed or a natural barrier which is inherent to the environment of the repository.

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.

decommissioning. Actions taken at the end of the 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 national legal and regulatory requirements, 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 some nuclear facilities used for mining and milling of radioactive materials or the disposal of radioactive waste.

decommissioning option. One of various decommissioning strategies which may be considered when decommissioning is being planned. A variety of factors, such as further use of the site and the availability of technologies and waste management facilities, will influence which decommissioning strategy is ultimately chosen.

dismantling. The disassembly and removal of any structure, system or component during decommissioning. Dismantling may be performed immediately after the 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.

institutional control. Control of a waste site (e.g. disposal site, decommissioning site, etc.) by an authority or institution designated under the laws of a country or state. This control may be active (monitoring, surveillance, remedial work) or passive (land use control) and may be a factor in the design of a nuclear facility (e.g. near surface disposal facility).

on-site disposal[1]**.** Decommissioning activities which encompass final disposal of the nuclear facilities or portions thereof within the nuclear site boundary.

repository. A nuclear facility (e.g. geological repository) where waste is emplaced for disposal. Future retrieval of waste from the repository is not intended. (See also disposal)

repository, geological. A nuclear facility for waste disposal located underground (usually more than several hundred metres below the surface) in a stable geological formation to

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[1] This definition is specific to this publication.

provide long term isolation of radionuclides from the biosphere Usually such a repository would be used for long lived and/or high level wastes

repository, near surface. A nuclear facility for waste disposal located at or within a few tens of metres from the Earth's surface Such a repository is suitable for the disposal of short lived low and intermediate level wastes

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 hazards

site. The area containing, or that is under investigation for its suitability to construct, a nuclear facility (e g a repository) It is defined by a boundary and is under 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, low and intermediate level. Radioactive wastes in which the concentration of or quantity of 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

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