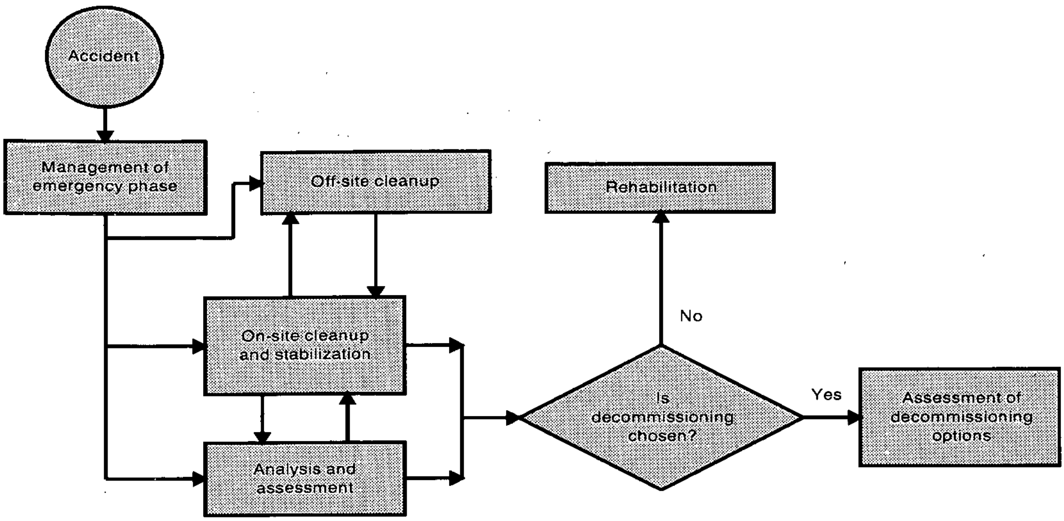

TECHNICAL REPORTS SERIES No. 346

Cleanup and Decommissioning of a Nuclear Reactor after a Severe Accident

INTERNATIONAL ATOMIC ENERGY AGENCY, VIENNA, 1992

CLEANUP AND DECOMMISSIONING OF A NUCLEAR REACTOR AFTER A SEVERE ACCIDENT

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FOREWORD

Although the development of commercial nuclear power plants has in general been associated with an excellent record of nuclear safety, the possibility of a severe accident resulting in major fuel and core damage cannot be excluded and such accidents have in fact already occurred. For over a decade, IAEA publications have provided technical guidance and recommendations for post-accident planning to be considered by appropriate authorities. Guidance and recommendations have recently been published on the management of damaged nuclear fuel, sealing of the reactor building and related safety and performance assessment aspects. The present technical report on the cleanup and decommissioning of reactors which have undergone a severe accident represents a further publication in the series.

A Technical Committee meeting on the present subject was held in Vienna from 24 to 28 June 1991 (IAEA Scientific Secretary, P.L. De of the Division of Nuclear Fuel Cycle and Waste Management). The meeting was attended by ten experts from nine Member States. The participants discussed and revised a preliminary report written by V.A. Kremnev (former USSR), R. Graf (Germany), J.H. Leng (United Kingdom), R.I. Smith (USA), A.A. Borovoi (former USSR), T.V. Efimova (former USSR) and the IAEA Scientific Secretary Z. Dlouhy of the Division of Nuclear Fuel Cycle and Waste Management. After the meeting, the report was revised by the IAEA Scientific Secretary, M. Laraia, of the Division of Nuclear Fuel Cycle and Waste Management, with the assistance of two outside consultants, J.H. Leng (United Kingdom) and C. Bergman (Sweden), and the final report was approved by the members of the Technical Committee. Acknowledgement is due to J.F. Zuber (Switzerland) for providing detailed information on the decommissioning of the Lucens reactor.

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

During the siting, design, construction and operation of nuclear facilities, the protection of operating personnel, the general public and the environment is a primary concern. Therefore, from the very beginning of the peaceful utilization of nuclear energy, nuclear power plants have been designed to deal with unanticipated occurrences and accidents. In order to minimize potential radioactive releases from the reactor core to the environment, containment systems have been used in most cases. During the last decade, some nuclear power plants have also been provided with emergency containment venting systems. Although the probability of severe accidents is low, their occurrence cannot be entirely excluded, as experience shows.

The Three Mile Island (TMI-2) accident in 1979 and the Chernobyl accident in 1986 accelerated studies in accident recovery. In addition to national efforts, many international organizations, such as the IAEA, the Nuclear Energy Agency of the OECD (OECD/NEA), the former Council for Mutual Economic Assistance (CMEA) and the Commission of the European Communities (CEC), have investigated or are investigating this area.

Since the early 1980s, IAEA documents have provided technical guidance and recommendations for emergency planning to be considered by appropriate authorities [1–4]. In particular, the IAEA has been active in planning for emergency preparedness against the possibility of situations where a severe accident may involve the need of on-site and off-site remedial actions and protective measures [5–8]. Guidance and recommendations have recently been published for the management of damaged nuclear fuel [9], sealing of the reactor building [10] and related safety and performance assessment aspects [11]. A catalogue of tools, methods and analytical techniques that would be of particular value during accidents, even those of much lesser magnitude than TMI-2 and Chernobyl, was recently published by the IAEA [12]. Operating experience and guidance on managing abnormal wastes, in particular those arising from accidents, can be found in Ref. [13]. The present technical report is a further step in the series of IAEA publications dedicated to this task. For the purpose of this report, a severe accident is one in which the reactor suffers major fuel and core damage. Such damage could occur without failure of the containment, as in the TMI-2 accident, and therefore with few off-site consequences, or with major containment failure as in the Chernobyl accident, where serious off-site problems arose.

Immediately after a severe accident has occurred, all efforts are directed towards bringing the accident under control and implementing those measures needed to protect the public and the workers, i.e. prevent recriticality, cool the reactor core, put out fires, and reduce or stop any major spread of radionuclides. These

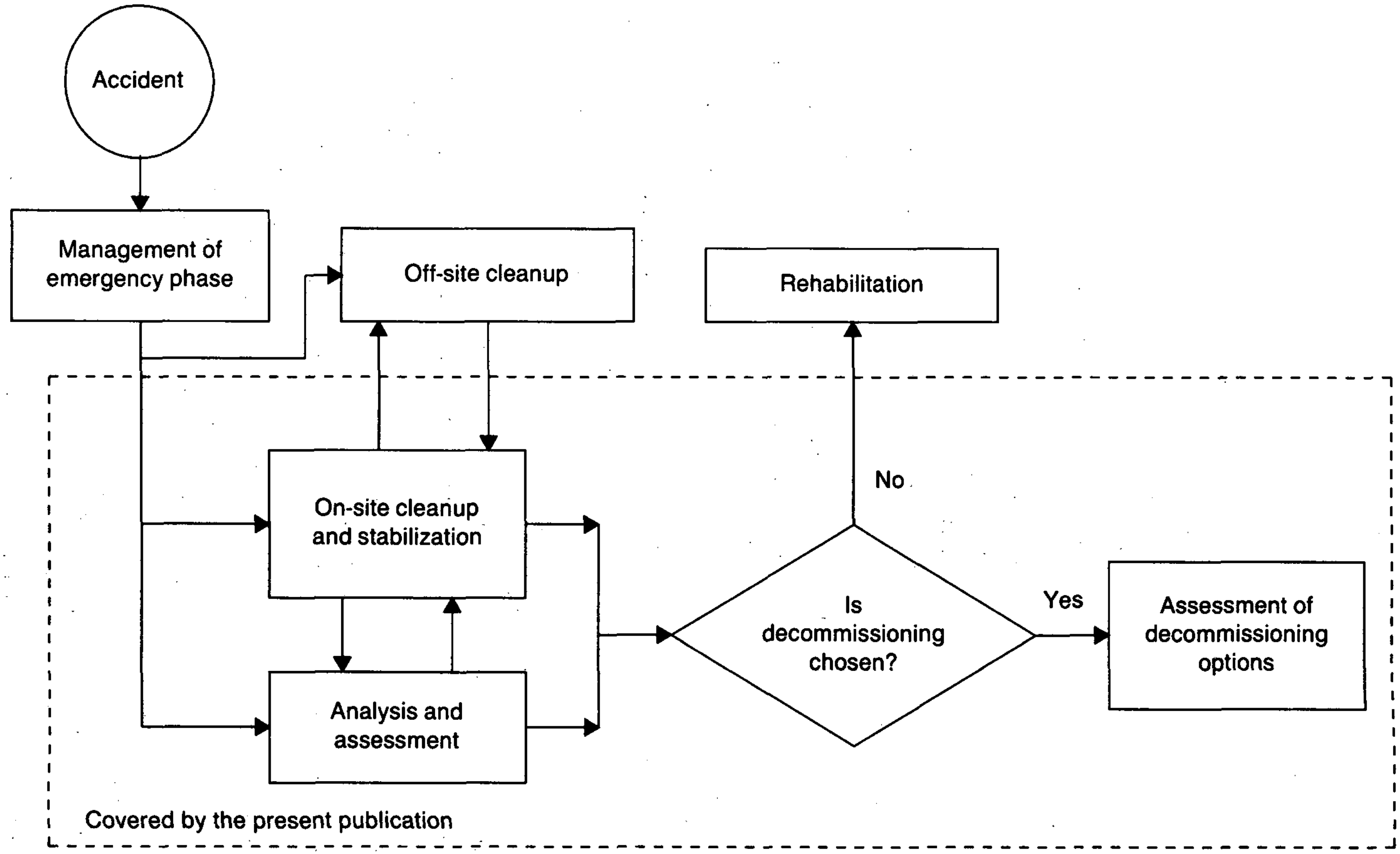

FIG. 1. Management sequence after a severe reactor accident.

emergency actions are all part of the management of the accidental situations and are covered in other IAEA reports [6, 9, 13]. Once the accident control is complete and the reactor facility is in a stable state, consideration of the possible options for managing the damaged facility and any off-site consequences should be initiated. After cleanup of the facility, the two main alternatives are:

(a) Rehabilitation and reuse of all or part of the facility;
(b) Decommissioning using immediate dismantling or dismantling after a period of safe storage.

The choice will depend largely on technical and economic factors such as the amount of damage to the core and containment, the spread of radioactive material, the availability of resources to complete the option and the need for the site. However, political factors and public attitudes can also have a significant impact and may result in decommissioning being chosen although rehabilitation might be the preferred choice from the technical and economic viewpoints.

1.2. OBJECTIVE

The objective of this report is to provide an overview of factors relevant to the identification of cleanup requirements and to the choice of a decommissioning option for a severely damaged nuclear power plant. A methodology is proposed to evaluate various options and to select an appropriate action in a particular accident situation.

The report should serve national authorities concerned with handling severe accidents, and the various organizations and operating personnel involved in collecting and evaluating the necessary information for the decision making process. It should also assist the regulatory authorities in identifying parameters relevant to the safety and environmental assessment of the selected option.

1.3. SCOPE

The report briefly describes the overall post-accident management sequences, including off-site cleanup, on-site cleanup and stabilization, and the possibility of rehabilitation and decommissioning. In Fig. 1, the interrelationships between the various activities are shown diagrammatically and the scope of the report in the context of these activities is indicated. However, the main part of the report presents a basic framework for identifying the cleanup requirements and determining the optimum decommissioning option for a severely damaged reactor.

Even if the reactor is to be decommissioned, there may be cases where parts of the facility can be reused after proper cleanup and rehabilitation. Once a decision is taken to rehabilitate a facility or part of a facility, all further actions related to the rehabilitation are outside the scope of this report.

It is recognized that, ideally, the taking of preventive actions during the construction of a nuclear facility might reduce the doses and costs for decommissioning should an accident occur. Generally, such actions are not considered justified for that sole purpose. However, they can be justified and implemented for other reasons, e.g. to facilitate operation, maintenance and normal decommissioning [14] and as a component of a programme to mitigate the consequences of severe accidents; as such they are, however, outside the scope of this report.

Although the information presented here is intended to cover severe accidents in nuclear power plants, it can also be used for other reactor types, such as research or prototype reactors. Although not specifically discussed, other facilities (reprocessing plants, fuel processing and fabrication facilities, etc.) are to a certain degree encompassed by the methodology of this report.

2. MANAGEMENT AFTER A SEVERE REACTOR ACCIDENT

2.1. MANAGEMENT STRATEGY

Immediately after the accident, the first activity is the actual management of the accident to a point where the immediate safety problems have been controlled. The main thrust is towards the cleanup and selection of the decommissioning option which is to be implemented. The content of each activity represented in Fig. 1 will be expanded in Sections 2.2–2.7.

2.2. MANAGEMENT OF EMERGENCY PHASE

Although outside the scope of this report, accident management includes the efforts aimed at bringing the situation under control. This comprises the steps necessary to ensure that:

(i) Recriticality cannot occur except as a result of significant changes to the system.
(ii) All fires and other severely exothermic chemical reactions have been terminated and controlled.
(iii) There is no continuing major release of radionuclides.
(iv) Heat removal is adequate to maintain safe temperatures for the remaining materials.

This phase also includes the implementation of protective measures deemed necessary by the emergency director [5].

2.3. OFF-SITE CLEANUP

Off-site cleanup is not within the scope of the report as it is covered in Refs [5–8]. To ensure that unnecessary off-site contamination is not created as a result of the on-site cleanup and stabilization, and, later on, of decommissioning activities, it is necessary to consider the off-site impacts during planning of the on-site work (Section 3.3).

2.4. ON-SITE CLEANUP AND STABILIZATION

Considerable guidance is given in Ref. [9] on the cleanup phase, but with specific emphasis on the handling of the fuel and associated waste. The present report is more general in that it considers cleanup of the plant and buildings and, in particular, recognizes that in severe accident situations it may be necessary to have a phased cleanup strategy interacting with analysis and assessment, each cleanup phase allowing access to further areas for detailed measurement and examination. Primarily, the post-accident on-site cleanup process is aimed at removing or fixing contamination, reducing radiation levels in working areas, removing fuel and gathering data on the extent of the damage caused by the accident. Because, in general, radiation levels will be higher than in a normally shut down plant, exposure control and dose reduction techniques will gain in importance and some or all of the methods given in Ref. [15] will need to be considered. The accounts of severe accidents given in the Annex demonstrate that the assessment and cleanup phases can last a considerable time. During the cleanup phase it is likely that waste types and volumes will differ from normal operational wastes and special provisions for treatment and disposal will be required, as discussed in Ref. [13].

As a consequence of the cleanup and assessment process, the information which will be necessary to decide whether rehabilitation is economically and technically possible will be accumulated.

2.5. THE DECISION — DECOMMISSION OR REHABILITATE

After the damaged reactor has been stabilized and cleanup has been initiated, the decision whether to decommission or rehabilitate can be considered. In the extreme case where damage is so severe that meaningful rehabilitation is impossible, this decision is simple; where damage is not so severe the decision to rehabilitate will be complex and should be based on a number of factors:

(i) Whether the costs of further decontamination, refurbishment of the plant and the potential additional measures which may be required will lead to a saving compared with the construction of a new plant, including an allowance for the decommissioning of both the old and new plants.

(ii) Whether the collective dose associated with the rehabilitation work, the operation of the rehabilitated plant and its final decommissioning will be unacceptable compared to the collective dose associated with the decommissioning of the damaged plant and the operation and decommissioning of a new plant.

For the economic side of the equation there will be a number of additional considerations which may be difficult to quantify and which can only be handled by some form of economic risk analysis. Examples are:

(a) The perceived value of not having to go through the full planning and approval process for a new plant on a new site, including the risk the application may fail or the process be excessively delayed.
(b) The risk that the refurbished plant may have difficulties in obtaining regulatory approval to recommence operation or could obtain such approval only after costly and time consuming modifications.

Finally, special local conditions such as public opinion and political considerations may weight the balance one way or another, making the detailed economics less important.

2.6. REHABILITATION

In the context of this report, rehabilitation refers to the cleanup and rebuilding of the reactor for reuse. However, on a nuclear site there are many installations which normally are not contaminated to any significant extent. If these are not affected by the accident, they can easily be put back into operation or be used for other non-radiological purposes. If simple decontamination to exemption levels is carried out, this can be considered as decommissioning to Stage 3 (see Section 2.7) and thus to be covered by this report.

2.7. ASSESSMENT OF DECOMMISSIONING OPTIONS

Because of the additional problems created by the accident, decommissioning of reactors which have suffered a severe accident will normally be subject to more technical uncertainty than is the case with a plant which has been shut down at the end of its economic or design life. As a result, it will be less straightforward to assemble the information necessary to make an optimum decision between decommissioning options. The main purpose of this report is to set out the steps required and the important parameters to be considered in arriving at a final decision.

In order to provide a well understood framework for the discussions, the IAEA definitions of a three stage decommissioning process are utilized. Under normal circumstances, fuel removal from the reactor is considered a prerequisite for decommissioning activities to proceed. The detailed definitions of the three stages are given in Ref. [16] and are summarized below:

Stage 1 (storage with surveillance). During this stage, the first contamination barrier is kept as it was during operation. But mechanical openings are permanently sealed. The containment building is kept closed and under institutional control. Surveillance, monitoring and inspections are carried out to ensure that the plant remains in good condition.

Stage 2 (restricted site release). In this stage, the first contamination barrier is reduced to minimum size by removing easily dismantled parts. Sealing of the barrier is reinforced by physical means and the biological shield is extended, if necessary, to completely surround the barrier. After decontamination, the containment building may be modified or removed if it is no longer required for radiological safety. Access to the building can be permitted. The non-radioactive buildings on the site can be used for other purposes.

Stage 3 (unrestricted site use). All materials, equipment and parts of the plant still containing significant levels of radioactivity are removed. The plant and site are released for unrestricted use. No further inspection or monitoring is required.

After a severe accident, the removal of all fuel may be impossible without at least partial dismantling to gain access and thus it may not be possible to fit the actually achievable states into the formal definitions of Stages 1 and 2; however, the concepts are useful as a baseline against which deviations can be discussed.

The final state of the site which can in practice be achieved, whether completion occurs within a few years or over a hundred years or more after the accident, will be very much dependent on the severity of the original accident. For Chernobyl, for example, it is difficult to imagine that it would be either cost-effective or dose-effective to attempt to return the site to unrestricted use (Stage 3), in the short or even medium term, while for TMI-2 Stage 3 would be an achievable target (see Annex). However, conversion of a damaged reactor to a waste repository is possible if a safety and environmental impact assessment shows compliance with relevant criteria and requirements.

It should be noted that there are national definitions of decommissioning alternatives which differ from those promulgated by the IAEA. Also, many Member States do not consider decommissioning completed until the site is available for unrestricted use.

3. FACTORS RELEVANT TO CLEANUP AND DECOMMISSIONING AFTER A SEVERE ACCIDENT

3.1. RADIOLOGICAL PROTECTION

The main objectives of all post-accident measures are to ensure that humans and the environment are adequately protected against exposure and contamination from radioactive material originating from the facility and to reduce residual contamination levels to a minimum. This protection should be achieved both during the different types of intervention such as sheltering and during cleanup and stabilization and eventual dismantling of the facility.

The International Commission on Radiological Protection (ICRP) has in its recent recommendations [17] expanded the system of radiological protection based on justification, optimization and dose limitation introduced first in Ref. [18] to more explicitly include cleanup and stabilization after an accident. This system was expanded in the IAEA Basic Safety Standards on Radiation Protection [19].

Facilities which have been exposed to a severe accident are most probably heavily contaminated. In addition, ancillary systems such as ventilation and waste management systems may not be fully operating. Although work in such environments requires extra radiological precautions to be taken, the possibility cannot be excluded that occupational exposure might be higher than during normal maintenance and decommissioning work. In all situations, however, the dose limits established by the national authorities must be met.

The optimization of radiological protection requires the evaluation of collective doses for all available options. Although in most cases this evaluation will be dominated by the occupational doses, the doses and risks to the general public as a result of the spread of radioactive material must not be neglected.

3.2. SAFETY CONSIDERATIONS PERTAINING TO THE FACILITY STATUS

Throughout each stage of the progression from the termination of the accident sequence, through plant cleanup and stabilization, to long term storage and eventual dismantling, there are general safety related issues that must be examined. These include:

(a) Determining the potential for recriticality as a result of long term changes in the geometry of the remaining fissile material or other long term physical, chemical or biological concentration mechanisms;
(b) Determining the potential for unacceptable heating to occur as a result of changes to heat removal paths reducing the capacity to remove the decay heat;

(c) Evaluating the potential for chemical reactions within the debris to cause either heat release or the generation of flammable or explosive gas mixtures or locally higher pressures;
(d) Controlling the spread of contamination, including leaching and other dispersive mechanisms;
(e) Maintaining the integrity of the containment as a structure and as a barrier to prevent further release (this issue should include an estimate of the probable operational life of such barriers without major remedial work);
(f) Utilizing auxiliary systems in cleanup and decommissioning work; and
(g) Evaluating the reliability of existing reactor logging and radiological monitoring equipment.

As the facility progresses through the stages of decommissioning, these safety related issues should be continually evaluated until the hazard has been eliminated.

The preliminary assessment to determine the hazards related to steps taken to stabilize the facility must of necessity be largely theoretical, based on what is known about the reactor materials and what can be seen by remote viewing techniques without interaction with the structure. One objective of the preliminary assessment is to establish what sampling and other monitoring procedures can be deployed without danger of creating new problems.

After the preliminary assessment of the hazards of intervention has been made, work can start on the main assessment of the status of the facility to permit cleanup to start. The difficulty in characterizing the state of the plant will depend on the spread of contamination through the building and the radiation field directly arising from the damaged reactor. Therefore, even at this early assessment stage, it may be necessary to decontaminate some areas as well as to erect temporary radiation shielding to get the required access. As emphasized earlier, it is essential to perform an adequate assessment of the potential consequences of the changes to the system which might result from efforts to characterize the facility. This in turn could in itself lead to further work requirements such as the introduction of a locally controlled and filtered ventilation system. Ultimately, the assessment work has to provide the data necessary to make the correct choice between the various decommissioning options.

The type of information which needs to be gathered and assessed includes:

(a) Inventory of the radioactive and fissile materials associated with the plant. This will involve calculations from a detailed irradiation history of the fuel charge, a detailed operational history of the reactor itself and as much information as can be collected on the exact composition, including trace elements, of all materials subject to neutron activation.
(b) Radiation contours around the reactor and in adjacent connected buildings, including the distinction between radiation coming directly from the damaged reactor and that coming from dispersed contamination.

(c) The quantity, isotopic content and time distribution of present and potential airborne contamination.
(d) The quantity and isotopic composition of the solid deposition and the degree of penetration into surfaces.
(e) The chemical and physical form of the radioactive materials, with particular emphasis on unusual phases or compounds which may have been generated either in the initial accident or as a result of efforts to terminate the accident.
(f) The mechanical stability of the remaining structures and, in particular, their residual resistance to seismic activity or any other disturbing agents such as severe gales and floods.
(g) Modifications to ventilation and cooling flow paths occasioned by structural changes as a result of the accident or remedial actions.
(h) Results of a detailed examination of the main reactor components, with estimates of the degree of damage.
(i) The status of auxiliary systems, including an inventory of working or repairable instrument sensors and a survey of the electrical integrity of all internal cable runs.

Stability of the system for some years is a prerequisite of all the final options. However, in this investigative phase it is important not to preempt the choices by, for instance, carrying out stabilizing work in such a way as to make dismantling much more difficult and costly. For example, stabilizing the contamination in certain areas using concrete would certainly reduce the potential for the spread of contamination but it would certainly affect the cost and difficulty of subsequent dismantling. Ease of physical access, whether by personnel or by remotely operated equipment, will markedly affect the costs of further work. Thus, an important part of the information gathering exercise is to give a physical description of all the potential work spaces. Where radiation makes human access difficult or impossible, photogrammetry[1] and remote video examination provide a useful method of constructing a three dimensional layout of all visible components.

3.3. ASSESSMENT OF SITE CHARACTERISTICS

In the assessment of the options for managing the damaged reactor, the radionuclide inventory, the facility and its containment, and the site should be viewed as an integral part of the overall system, where unfavourable characteristics in one component can be compensated for by favourable characteristics in another component.

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[1] Photogrammetry is a technique of remote surveying using precision multiview photography. Provided that a few fixed points are known within the field, computer application of geometrical principles will fix the positions and dimensions of all other items.

For example, where local geology does not exhibit high confinement, the damaged facility should be provided with efficient engineered barriers to prevent significant releases into the geosphere and the biosphere. Similarly, even with a high radionuclide inventory, immediate remedial actions can be delayed for limited periods of time in situations where the primary containment has not failed, while immediate actions have to be taken if this containment is breached.

Therefore, the selection of appropriate cleanup/decommissioning options should be made with detailed understanding of all three components mentioned above. The major site related characteristics important in the decision making process are summarized below.

3.3.1. Geology

No changes in local geology are expected even after a severe accident. However, if the prospect of creating a concrete structure around the damaged facility is foreseen, the actual geological situation should be reassessed from this point of view. For example, the behaviour of subsurface materials and their response to the stresses induced by a new construction will be of particular interest.

The stability assessment can be performed by means of standard methods for prediction of the physical behaviour of subsurface materials. The parameters necessary for use in numerical models include items such as:

— a description of on-site geological structure and subsurface stratigraphy, lithology, mineralogy and erosion characteristics;
— the physical and chemical properties of soil, such as their stress–strain relationships, static and dynamic strength properties and permeability parameters;
— seismic and tectonic stability characteristics.

The assessments of foundation stability should take into account bearing capacity, overturning and sliding. Methods of analysis, together with appropriate investigation programmes, are listed in specialized IAEA publications [20, 21].

3.3.2. Hydrology

The hydrosphere, including both the surface water and the groundwater, represents an important pathway by which radioactive materials can be dispersed from a damaged nuclear facility into the environment. Information on contamination of water bodies plays an important role in assessing the radiological impacts on the population after release of radioactive material into the environment.

On-site hydrology data, originally obtained from a pre-operational measurement programme (e.g. as part of the early site characterization) should be reassessed. Where necessary, reassessment using updated and additional data on hydraulically connected water bodies in the region must be done. The objectives of this task are:

— to assess the capability of waters to dilute and disperse radioactive materials released into the aquatic environment; and
— to use the data in the pathway analysis of the radionuclides released.

Particular attention should be given to sources of drinking water and water used for other purposes, such as irrigation or fishing.

Where the site has an existing borehole (sampling wells) monitoring system as part of its environmental programme, this should be used to support and validate any theoretical modelling. Where necessary, new sampling wells could be drilled to monitor the follow-up of the accident and cleanup/decommissioning activities.

The assessment procedures, together with information needed, are described in IAEA publications related to nuclear power plant siting [22–24] and waste repository siting [21].

For the purposes of the long term safety, and thus of the selection of the preferred decommissioning option, it is also necessary to include an analysis of the water transport pathways of radionuclides from the damaged facility.

3.3.3. Local topography

In general, even after a severe accident no major changes in topographic features are expected. Nevertheless, topography can play a role in planning and managing decommissioning activities, such as the provision of access for heavy transportation equipment, and creating storage/disposal sites for contaminated soils in natural depressions. In this respect, the major topographic features should be identified and evaluated.

3.3.4. Climatology

The climatological situation in the area is of importance for the assessment and thus for the choice of a detailed decommissioning option. Factors of importance include:

— Precipitation in the form of rain and snow which will influence erosion and surface water and groundwater flow;
— Wind (direction, speed, duration) which affects erosion and can cause spread of contamination;
— Temperature variation, especially freeze/thaw cycles which can have an impact on the design and lifetime of engineered structures.

3.3.5. Ecology

The interaction between all components in the ecological system which constitutes the local environment, and the way that this interaction is affected by the options chosen for the future of the damaged facility, are additional factors which have to be analysed and evaluated.

3.3.6. Local demography and socioeconomic aspects

The actual and projected population distribution surrounding the site that could be affected by the choice of rehabilitation or decommissioning option should be evaluated. Additionally, land and water use, socioeconomic aspects, including the locally available infrastructure (communication routes, electricity lines, health care facilities, etc.) need to be considered. Nutritional habits, including consumption of locally produced food, are important in the identification of foodchains and related radiological pathways.

3.3.7. Safety and environmental impact assessment

A safety and environmental impact assessment is a very helpful tool to evaluate the acceptability of a chosen option and can also be a component in the selection between different options. In many Member States there is a legal requirement that such an assessment be performed and reviewed by the authorities prior to approval of a decommissioning option.

After quantification of the parameters and definition of the methodology, the purpose of this assessment is to estimate the impacts on public and occupational safety of the post-accident cleanup and decommissioning of the given facility. Radiological and non-radiological impacts of both routine activities and possible industrial and transportation accidents during post-accident cleanup and decommissioning are evaluated.

The assessments are normally based on model calculations for which generic or preferably site specific parameters are required. These include:

- radionuclide source term, including the physical/chemical forms
- geometry and boundary conditions of the water bodies
- data of importance for water flow (hydraulic head, hydraulic conductivity, permeability, porosity, etc.)
- data of importance in the transport of radionuclides (solubility, distribution coefficient, etc.).

The content of the safety and environmental impact assessment may differ from country to country and also depend on the type of installation. In general, however, it will give a summary of all effects of the proposed activities on humans and the environment.

3.4. NATIONAL REGULATIONS

In general, regulations are in place that can be used to cover most aspects of the cleanup and decommissioning of a nuclear power reactor that has been involved in an accident.

It is unlikely that the time schedule for decommissioning a nuclear plant after a severe accident will depend on national legislation. The timetable, which most probably will be decided on a case by case basis, will be much more dependent on factors such as current status of the facility, site considerations, occupational and public exposure, and the availability of suitable equipment, experienced staff and waste disposal sites.

The cleanup and/or decommissioning of a reactor that has been involved in an accident are also subject to constraints imposed by statements, orders and amendments to the facility licence issued by the regulatory body subsequent to the accident. These constraints may relate to such activities as the controlled venting of the reactor building atmosphere, the use of special equipment for accident cleanup operations, the storage and/or disposal of radioactive wastes, and the release of processed accident water by evaporation to the atmosphere or by discharge to a river, lake or ocean. Statements, orders and amendments to the facility licence are of necessity specific to the particular reactor and accident and would be issued on a case by case basis.

An important area of concern in the post-accident cleanup and decommissioning of a nuclear reactor is the management of the large volumes of radioactive wastes (gases, liquids and solids) that result from the accident and from cleanup and decommissioning operations. In particular, certain post-accident cleanup and decommissioning wastes will have to be carefully evaluated with regard to characteristics such as specific activity, radionuclide content, total activity inventory and waste form. Ultimate disposition of these wastes will depend on the unique characteristics they possess and on the availability of suitable facilities for their handling, storage and disposal.

One regulatory issue which will have impact on the selection of a decommissioning option, and especially on the costs for disposal of the wastes, is the ability to exempt wastes and the site from some form of regulatory control. At present, there is international consensus on the basic principles for such exemptions [25] and the IAEA is in the process of establishing levels for the unconditional release of materials and also for the release of materials from nuclear facilities, including during decommissioning, for reuse and recycle [26]. However, exemption levels have not yet been incorporated into national legislations to any significant degree.

3.5. NEED FOR UTILIZATION OF THE SITE

An important consideration when selecting a decommissioning alternative is the present and future need for the site. Some alternatives may preclude future use of the site for nuclear power generation, or for any other purpose, for periods of up to one hundred years or maybe longer. Sites acceptable for nuclear power stations are often difficult to find and frequently even more difficult to get approved. A site already accepted and approved can be a valuable commodity, not only for nuclear power installations but for conventional electricity generation and other processes which can utilize the existing structures.

Selecting a decommissioning alternative (or rehabilitation) that will permit continued use of the site for nuclear power generation, even when that alternative may be more costly than other available choices, may be justified because of the long term need for the site. Thus, an economic analysis should be performed to compare the cost of decommissioning alternatives that would restore the availability of the site against the cost of developing and licensing another comparable site.

3.6. AVAILABILITY OF STORAGE/DISPOSAL LOCATIONS

Another factor to be considered in the selection of a cleanup or decommissioning strategy is whether there are locations available which could accept the radioactive waste for final disposal. In the absence of such facilities, interim storage of waste may be considered, either at a remote location or on the site. However, if no waste storage or disposal facilities are available or planned, there is an urgent need to initiate such planning. An on-site location should also be considered because of the obvious advantages of reduced waste transport.

If no off-site waste storage or disposal facilities are available, on-site storage under less stringent radiation protection and safety conditions than normally considered necessary may have to be adopted, since the alternative of not being able to do the work owing to lack of storage facilities would be worse. On-site disposal, however, should not be carried out until an assessment has demonstrated that the site meets the relevant national criteria for a repository.

3.7. AVAILABILITY OF CLEANUP AND DECOMMISSIONING TECHNOLOGY

The basic technology for cleanup and decommissioning is reasonably well known. However, during the planning stages, there may be problems caused, for example, by poor accessibility or by a specific operation considered for cleanup or decommissioning. In such cases it may be necessary to develop special tools or means for remote operation or handling.

The development work often has to be followed by testing of the new equipment, training of personnel, etc. These efforts can be costly and time consuming and thus it may be necessary to remain at Stage 1 or Stage 2 decommissioning for a considerable period of time, if safety or other factors allow these options [16].

3.8. FINANCIAL UNCERTAINTIES IN IMMEDIATE OR DEFERRED DISMANTLING

Funding for decommissioning is normally allocated (although the practical details vary from country to country) long before the decision for a planned decommissioning is taken [27]. However, in the case of premature decommissioning due to a severe accident, the expenses for cleanup and decommissioning will probably be much higher than for a planned shutdown and the funding allocated for normal decommissioning will not be adequate. Insurance mechanisms are available in some Member States to cover such extra expenses, although they are not always sufficient. The lack of immediate funding may result in deferred dismantling. In the case of a dismantling deferred for decades, however, events might occur which could endanger the later availability of funds. This situation has the potential for serious consequences unless the long term integrity of the damaged plant is ensured.

4. STRATEGY FOR CLEANUP

Following management of the emergency phase, post-accident activities include accident cleanup and decommissioning. Accident cleanup comprises those activities leading to defuelling of the reactor, partial or total cleanup of contamination and processing of wastes generated by the accident. If the facility is to be retired from service, decommissioning activities are considered to begin following the accident cleanup [28, 29]. However, a totally clear distinction between cleanup and decommissioning activities might be difficult to define and implement.

For each kind of activity and each option under discussion, data have to be collected to arrive at an appropriate decision and to explain methods and conclusions (especially in relation to dose estimates) to the supervising regulatory body. With a sound methodology and appropriate documentation and presentation of facts, public confidence can be assured.

4.1. GENERAL ASPECTS

Accident cleanup activities are necessary and would be approximately the same whether the reactor is ultimately refurbished or decommissioned, and, if decommissioned, would be independent of whichever decommissioning alternative is chosen. The rationale for this is discussed in detail in Section E.1 of Ref. [29]. Briefly stated, decontamination during the accident cleanup period (whether for eventual restart or decommissioning) cannot be too chemically corrosive or destructive, since this could compromise the integrity of systems that must remain intact during cleanup and decommissioning, especially if a delayed decommissioning alternative is chosen. In addition, major items of equipment such as the reactor vessel, reactor coolant pumps and steam generators may not be dismantled until after accident cleanup is completed since they form part of the primary system. However, the methods used to complete certain cleanup tasks may vary, depending on whether the decision is to restart or to decommission, and, in the latter case, on the decommissioning alternative chosen. The work required to complete specific cleanup tasks is, of course, determined by the severity of the accident.

Should there be a significant alpha contamination as a consequence of the accident, substantial problems will arise in addition to those normally associated with reactor decommissioning. Similar problems will, however, be encountered in the decommissioning of fuel reprocessing plants and experience from that field can be utilized. The main problems arise from the technical difficulties involved in monitoring the short range alpha emissions together with the low values of the acceptable levels. In bulk material such as cleanup and decommissioning waste these problems are multiplied so that there will be a substantial additional contribution to the costs of waste handling.

4.2. PREPARATION FOR POST-ACCIDENT CLEANUP ACTIVITIES

In Table I, typical working steps to prepare for the post-accident cleanup are listed and an indication given against each as to whether the activity will simply contribute to the total financial cost or will in addition add to the dose budget.

Initial entries into the contaminated and damaged areas are made for the purpose of obtaining data on the radiological and physical condition of the building. Very important for dose evaluation are measurements of contamination levels and radiation exposure values, particularly in areas where extensive work is to be carried out. Estimates of physical damage can help to define special tasks such as, for example, maintenance and repair of systems and equipment. Information about the operational status of the plant systems and services is needed for planning accident cleanup operations and for preparing documents for regulatory approval. In addition, documentation of all data and information gained during monitoring of the actual

TABLE I. PREPARATORY STEPS FOR POST-ACCIDENT CLEANUP

	Contributes to dose	Contributes to cost
Entry in contaminated area and data acquisition	×	×
Preparation of documentation for competent authorities		×
Design and fabrication of special equipment		×
Installation of special equipment	×	×
Development of detailed work plans and procedures		×
Selection and training of accident cleanup personnel		×
Filtration of airborne activity from the containment	×	×

state of the plant should be supplemented with precise documentation of the as-built state of the plant. Training of accident cleanup personnel in plant mock-ups helps to reduce exposure times significantly and makes it much easier to estimate the doses from actual tasks because approximate working times are known before the particular work starts.

Significant quantities of radioactive fission products are released to the containment building atmosphere as a result of a severe accident. The fission products include noble gases, iodine, and volatile and semivolatile radionuclides such as ^{137}Cs and ^{90}Sr. Most of the fission product noble gases have short half-lives and decay to insignificant levels prior to the start of building decontamination. The iodine isotopes also have short half-lives. The major contributors to radiation exposures inside the containment at times greater than one year after the accident are the relatively long lived caesium isotopes and ^{90}Sr, which plate out on building surfaces or are retained in the accident water. An exception is the noble gas ^{85}Kr, which has a 10.7 year half-life and which can also constitute a major radiological hazard to cleanup and decommissioning workers. The ^{85}Kr must be removed from the containment building atmosphere so that workers can begin the tasks necessary to clean the building, maintain instruments and equipment, and eventually remove the damaged fuel from the reactor core. (Removal of the ^{85}Kr from the reactor building atmosphere at TMI-2 was estimated to reduce the radiation dose rate for workers by a factor of about 4 [29].)

The release of slightly radioactive gases and liquids, to enable further entries into contaminated areas, is a principal contributor to doses to the public. Release of gases and liquids should be carried out in a carefully controlled manner and be properly monitored. It may be necessary to provide a new filtered ventilation system to control the situation.

The preparation of documentation that may be required to obtain regulatory approvals necessary to proceed with cleanup activities may be very time consuming and, therefore, is a critical factor in determining when actual cleanup operation can begin. Delays cannot be compensated for during the ongoing work and will directly influence the total cost.

For post-accident cleanup, it is cost effective to use those techniques and procedures that are in use to decontaminate a plant following normal shutdown. It is generally advantageous because estimates of the time requirements of known techniques in new situations may be more precise than estimates of time requirements of newly developed techniques, and time and costs for training of the staff are reduced. However, owing to the unpredictable circumstances of the accident, new decontamination techniques may have to be developed and tested in the field. Sufficient time for R&D work should then be made available. To avoid delays due to failures of equipment, reserve tools and spare parts should be kept available.

4.3. IMPLEMENTATION OF POST-ACCIDENT CLEANUP ACTIVITIES

The accident cleanup period is postulated to include the following tasks:

- initial radiation survey to determine the extent of contamination;
- processing of liquids contaminated during the accident (and by decontamination operations);
- initial decontamination of building and equipment surfaces and decontamination or removal of some equipment (typical initial decontamination steps for light water reactors are listed in Table II);
- removal of spent fuel (undamaged and damaged) from the reactor;
- conditioning, packaging and removal of wastes from the cleanup operations;
- periodical and final radiation surveys to determine the extent of residual contamination after decontamination, processing of liquids, defuelling and waste removal.

TABLE II. TYPICAL INITIAL DECONTAMINATION STEPS [29]

Use of the containment spray system for remote washing of contaminated surfaces
Removal and packaging of small items of contaminated equipment that are easily disposed of
Use of high pressure hose wash techniques for semiremote decontamination of building surfaces and equipment
Decontamination and refurbishing of essential support systems
Hands-on decontamination of selected areas
Local shielding of 'hot spots'

Some of these tasks require major facilities and equipment that have to be designed and fabricated during the phase of preparation for cleanup activities. It may become necessary to adapt the equipment as experience is gained in using it. Occupational doses due to decontamination work can be reduced by using an effective decontamination method and remotely operated equipment where practical.

Because of the uniqueness of the situation, planning and estimating cannot be precise. Thus consideration should be given to the effects on the overall timescales and costs of uncertainties in such parameters as the levels and extent of contamination, plant damage levels and problems with novel waste forms.

Work plans should include allowances for inefficiencies and maintain a certain degree of flexibility to changing data. This also implies flexibility in the cost schedule and the fundings since the need may arise for procurement of special tools and equipment.

If appropriate filter and waste treatment systems are adopted, no significant additional radiation exposure to the public is expected to occur. Selection of cleanup and decommissioning activities should ensure maintenance of the integrity of the various barriers between the radioactive inventory and the environment.

5. STRATEGY FOR DECOMMISSIONING

Following the completion of the accident cleanup activities, decommissioning activities may begin. As a result of the accident cleanup, the decommissioning activities are considered to be not greatly affected by the condition of the plant immediately following the accident. In addition, many of the uncertain conditions will have been removed during the accident cleanup; specifically the damaged fuel core will have been removed from the reactor, the large volumes of uncontained highly radioactive water will have been processed, the large areas of contaminated building surfaces will have been treated, and construction of necessary systems and structures will have been completed. Hence, decommissioning can be carried out in a more stable environment than the accident cleanup and the task may become similar to one of conventional decommissioning. Nevertheless, there would be certain impacts on the decommissioning from the accident and the accident cleanup activities, including increased levels and spread of contamination compared to normal decommissioning, the need to decommission systems and structures built and used during accident cleanup, and the potential need to store wastes generated by the accident (and during the accident cleanup period) on the site for an extended time period [30]. In addition, physical damage to the plant may compromise some systems and equipment needed for the performance of decommissioning tasks, thus necessitating repairs or substitutions and increasing the time and cost of post-accident decommissioning.

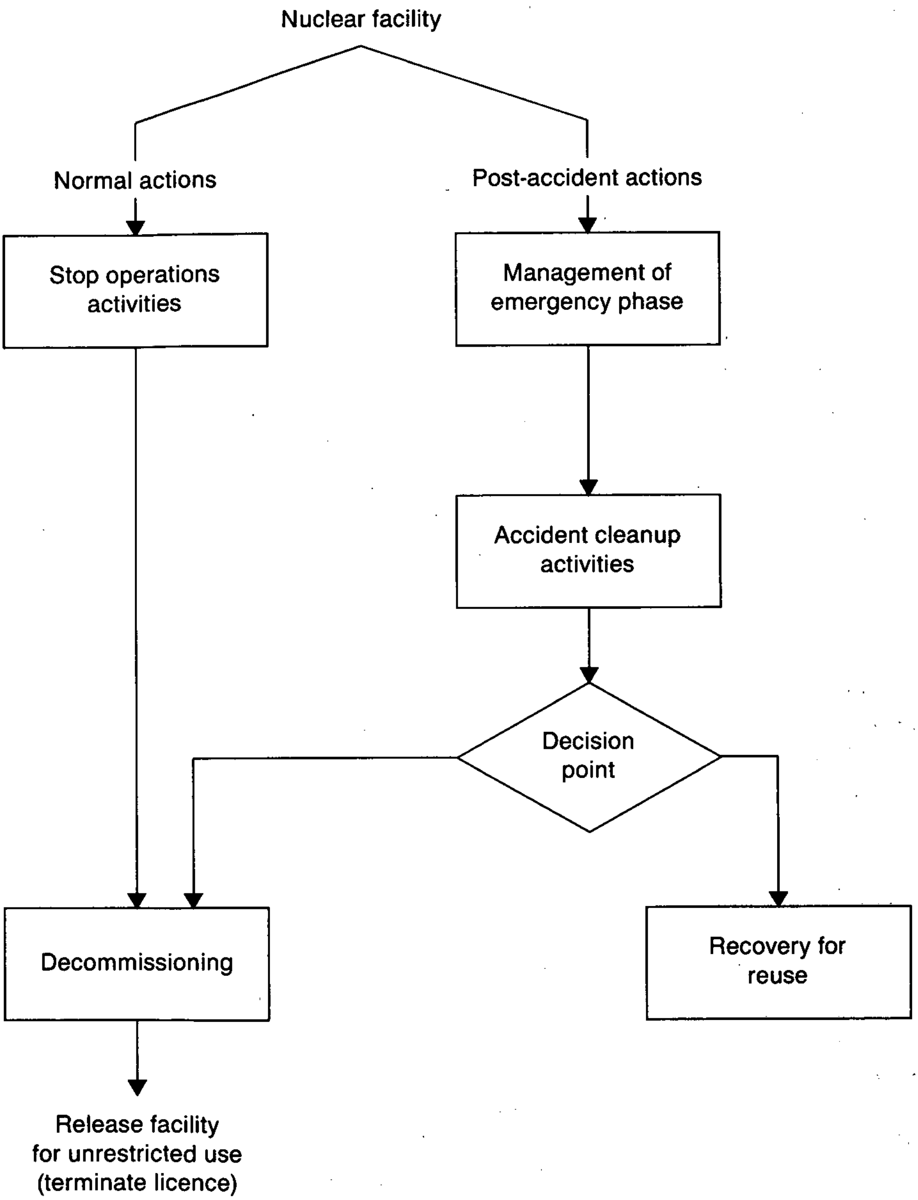

FIG. 2. Normal decommissioning and post-accident decommissioning: post-operation activities flow sheet.

Figure 2 illustrates the differences between normal decommissioning and post-accident decommissioning. The technology for decommissioning of nuclear power plants after normal shutdown has been described in detail elsewhere [16, 31, 32].

5.1. GENERAL ASPECTS

There are two basic decommissioning options for a facility that has suffered an accident resulting in severe fuel damage and for which the decision has been made not to return it to service. These two options are:

- immediate dismantling (equivalent to Stage 3 decommissioning);
- long term storage followed by dismantling (equivalent to Stage 1/Stage 2 followed by Stage 3).

The selection is greatly influenced by the individual and collective doses and the costs which will be incurred as a result of a particular option. Dose estimates will need to have been reached by methods which are acceptable to the national regulatory body. Individual dose limits to be met will be set within the legal framework and in addition dose constraints may be given by the national authority to satisfy a requirement for exposures to be as low as reasonably achievable (ALARA).

The following sections provide a comparison of some of the radiation protection and safety related advantages and disadvantages of the two decommissioning options as well as guidance on the specific safety areas which need to be addressed. A comprehensive overview of factors relevant to the selection of a decommissioning alternative, after normal shutdown, is available in Refs [16, 33, 34].

5.2. IMMEDIATE DISMANTLING

If the extent and type of damage is such that it is practically impossible to envisage how the radioactive inventory can be adequately contained to allow long term safe storage to be an option, it may be safest to embark on a dismantling programme. Other reasons, however, such as the need for utilization of the site (see Section 3.5), could favour the selection of immediate dismantling. In any case, a preparation period of many months will be needed before cleanup starts. The cleanup itself will last at least several years. Dismantling can then start.

An advantage of immediate dismantling is the retention and utilization of plant expertise on the site during the actual dismantling. This expertise could lessen the potential for accidents and would avoid the dose associated with retraining of personnel. As waste is recovered and conditioned, the overall risk associated with the facility will steadily decrease.

The principal disadvantage of immediate dismantling is the high occupational exposure that could be incurred by the workforce, although this may be reduced by the extensive use of remote handling techniques. This exposure is a problem particularly with a facility that has experienced a major accident with the contamination spread throughout the facility (although the cleanup activities should have reduced this contamination considerably). There may also be a problem with the immediate availability of suitable waste disposal facilities to handle the various categories of waste resulting from the dismantling. Interim storage may be required. Repeated handling of the waste increases the probability of an accidental release and/or transportation related accidents, as well as increasing occupational exposures.

In addition to the general points noted above, certain specific considerations associated with the dismantling process are enumerated below. Similar considerations are also relevant to dismantling after long term storage but will be mitigated by the effects of radioactive decay.

Fuel: Normally, spent fuel is removed prior to decommissioning. However, after a severe accident and cleanup, there might be some fuel remaining in the facility. Prior to the commencement of dismantling, fuel which could not be removed because it exists as films, is fused to existing structures, or is inaccessible, must be evaluated to ensure that the dismantling process will not result in recriticality, interrupt the management of decay heat or cause additional relocation of the fuel.

Occupational doses: Immediate dismantling of a damaged facility will result in significant occupational exposure. There is the potential for both overexposures of individuals as well as a large collective dose to the workforce. Adequate radiological controls, including extensive use of remote manipulation techniques, must be employed to minimize occupational doses.

Containment and control of contamination: Dismantling will result in the movement of contamination. Most of the movement will be under controlled conditions. However, there is a significant potential for inadvertent releases owing to the destructive nature of dismantling and also to uncertainties in the location, physicochemical nature and magnitude of the post-accident contamination. Personnel control, and control of equipment, dust and debris are required. Doses to the public must be carefully monitored.

Transportation and disposal of radioactive material: Dismantling will result in large quantities of radioactive waste. This waste must be properly packaged and shipped to the appropriate disposal sites. If final waste disposal sites are unavailable then interim storage sites must be utilized. Safety considerations in the transportation of the waste should be addressed from both a radiological as well as an industrial accident perspective.

5.3. LONG TERM STORAGE FOLLOWED BY DISMANTLING

Deferring dismantling of the facility takes advantage of natural decay. Depending on the length of storage, a reduction in dose rates may be realized during actual dismantling. For most facilities, however, once the short lived fission products have decayed the occupational exposure will be reduced only by a factor of two for every 30 year period of storage. This is due to the fact that the radioisotope ^{137}Cs makes the largest contribution to the occupational exposure during deferred dismantling of a plant that has suffered an accident. A storage period of 100 years would result in a reduction in exposure by only a factor of 10. (This can be compared to the factor of almost 10^6 reduction in exposure from ^{60}Co, the controlling nuclide for decommissioning following normal shutdown.) Therefore, long term storage appears less attractive as an alternative for limiting occupational exposure from decommissioning following an accident than it is for limiting occupational exposure from decommissioning following normal shutdown, although other factors could make long term storage necessary [29].

Additionally, long term storage followed by dismantling may take advantage of technological developments during the period of storage that may result in additional dose savings. Technological developments in decontamination procedures and techniques, robotics and remote cutting could facilitate future dismantling of the facility. Also, a storage period would result in smaller quantities of high level and intermediate level radioactive waste being generated, since decay will result in waste moving down in category. The reduced inventory results in reduced transportation related impacts since a smaller inventory would have to be transported to the waste disposal site [9].

There are a number of safety related disadvantages to long term storage followed by dismantling. During the storage period there is gradual deterioration of the structures, systems and components designed to act as barriers between the contamination and the environment. Although this is a generic problem associated with delayed dismantling it is likely to raise special issues after an accident scenario where the status of the structure may be subject to uncertainty. The deterioration may be difficult to measure and the criteria for action to ensure continued safe storage may not be easily developed. Surveillance programmes would have to be developed and measures taken to ensure the safety of the personnel involved in these programmes. Finally, as the storage period continues, expertise in the layout, maintenance and operation of the reactor and knowledge about the accident lessens as personnel leave the facility so that at the time of dismantling there will be no one with personal experience of the facility. This expertise will have to be reacquired at the time of dismantling, with a possible corresponding penalty in occupational exposure.

To some extent the knowledge may be preserved by ensuring that all drawings are brought up to the current state, maintenance and operational procedures are carefully recorded and the whole documentation placed in a storage and retrieval system.

This is desirable in any event because even if dismantling starts soon after the accident stabilization and cleanup phases, it will continue over sufficient time for experienced staff to have left the project before completion.

In addition to the more general points so far discussed, there are a number of safety considerations which need to be addressed for a strategy involving long term storage prior to dismantling.

Fuel: (See applicable considerations in the corresponding paragraph in Section 5.2.)

Containment and control of contamination: Prior to storage of the facility, a programme must be developed to continue to maintain and improve the containment developed during accident stabilization and cleanup through the use of appropriate barriers such that the release of contamination to the environment is within national regulatory limits. Furthermore, within the facility a programme must be in place to control the spread of contamination. This control is necessary to permit access to the facility and therefore the continuance of the surveillance and monitoring programmes. This programme to contain and control contamination at the facility should evaluate the potential effects of both natural (rain, erosion, flooding, seismic activity), man-made (operator errors) and physical (corrosion, deterioration) phenomena.

Physical protection: During the storage period an adequate physical security plan must be developed to control access to the facility. Unauthorized access could result in injury to personnel as well as compromise barriers designed to protect the environment from contamination. Physical security should also extend to controlling the spread of contamination by animals or organisms that might gain access to the facility during storage.

Monitoring and surveillance: Continued safe storage of the facility can only be assured through an adequate monitoring and surveillance programme. The long term surveillance programme essentially provides the monitoring required to demonstrate that the measures taken under the heading of containment and control of contamination are meeting their objectives. The programme is thus divided into two parts: a radiological programme which monitors the final outcome in radiological terms and a technical programme which aims to anticipate and correct deterioration of the barriers before any radiological consequences occur.

The radiological programme will:

- monitor, record and evaluate any changes and movements of radioactive material inside the facility;
- monitor and assess the releases of radioactive material into the environment;
- monitor and assess the external and internal doses that the plant personnel and population around the facility may incur.

The technical programme will:

— monitor the physical condition of the engineered barriers;
— assess the performance of the geological barriers, if applicable;
— monitor any changes in the chemical and thermal behaviour of the material stored within the facility;
— feed any changes back into the original design assessments and instigate any corrective work required.

Maintenance of systems necessary to ensure safe storage: A number of plant systems might be required to be maintained operational during the storage period. On the basis of a fire hazards study the facility's fire detection system may be required to remain operational. Radiation monitoring systems might need to be operational, particularly during periods of surveillance.

Components and systems required to periodically drain and treat any water that has entered the facility would probably remain operational. Ventilation systems necessary to control particulates inside the facility and satisfy air quality requirements prior to manned entries may be required to be maintained.

An electric power supply for the above described systems will also be required. The reliability of this power must be considered. Other services such as communications systems and domestic water must be evaluated.

Although all the points raised under maintenance are equally applicable to the long term safe storage of an undamaged reactor, in the damaged situation the problems will be heightened by the residual contamination and radiation fields as well as other consequences of the accident on the integrity of services.

Maintenance of systems for future dismantling: Although it is extremely unlikely that it will be advantageous in terms of either dose or cost to maintain equipment for many decades purely because it will be useful for ultimate dismantling, an evaluation should be made in terms of the dose and cost of replacing with new equipment at the end of the storage period. In general, only equipment which can be effectively utilized as part of the maintenance for safe storage will easily justify its continued costs. However, some consideration should be given to retaining structures such as crane rails which have low maintenance requirements.

Although most of the preceding discussion focused on the storage period, safety considerations related to dismantling, as described in Section 5.2, must be evaluated near the end of the storage period. Many of the safety issues described in Section 5.2 will apply; however, it is likely that their significance would be reduced by the effects of radioactive decay. On the other hand, long term adverse mechanisms such as those resulting in increased adherence of contamination to the base material and hence reduced decontamination efficiency are also possible.

6. METHODOLOGY FOR SELECTION OF THE PREFERRED DECOMMISSIONING OPTION

Although there might be particular circumstances after an accident which favour one main decommissioning strategy, it is unlikely that either of the two alternatives, immediate and deferred decommissioning, will be precluded. For example, an accident resulting in widespread contamination and destruction of containment barriers would naturally call for immediate dismantling. However, as illustrated by the Chernobyl accident, that option is not technically and economically reasonable given the actual situation. On the other hand, a severe accident in which the containment remained intact would call for safe storage followed by deferred dismantling because of the intense radiation dose rates and high contamination inside. However, it might still be possible, with remote techniques and at extreme cost, to dismantle the facility, although the process may take many years.

Whatever constraints there are on different decommissioning strategies and options, an evaluation of radiation protection, safety, technical and economic parameters should be used to arrive at a solution. The final choice of the option will in many Member States be made at a political level, implying that political, public perception and other non-technical issues will have an impact on the final selection. Even if it is known that political considerations play a major role, it is necessary to present to the decision makers a solid technical/economic evaluation of the different options.

A flow diagram for selection of the preferred decommissioning option is presented in Fig. 3. A brief discussion of each element is given in the following paragraphs.

The diagram begins with a stabilized facility which is at least partly cleaned up and from which the accessible fuel has been removed. The first step is to perform an assessment of the facility based on the guidance provided in Section 3.

National legislation and its implementation as well as established policy may give directions as well as constraints for the choice of decommissioning strategy and must therefore be carefully considered at an early stage of the process (see Section 3.4).

Waste management and disposal issues may have a significant impact on the selection of the decommissioning strategy (Section 3.6). For example, if neither suitable waste storage nor waste disposal facilities are available, then long term storage followed by deferred dismantling may be the preferred option. Again, the importance of this issue warrants evaluation prior to a comparison of other factors.

Even if it is generally agreed that decommissioning of a facility after normal shutdown is feasible with existing technology [31], it may not be true for a severely damaged facility, in which case immediate dismantling is not possible. A technical feasibility study is therefore necessary since it may show that significant development work has to be done before dismantling is possible.

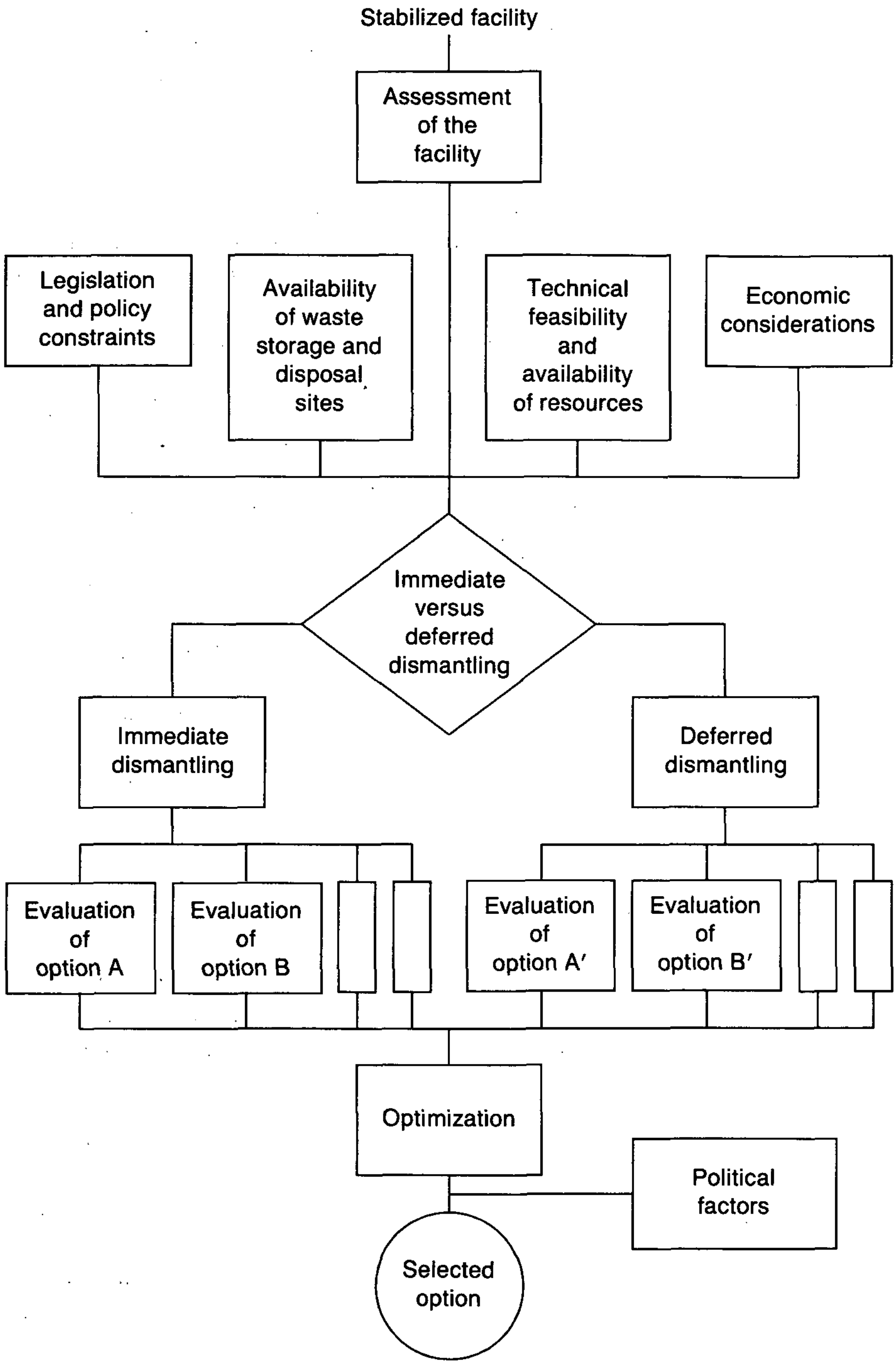

FIG. 3. The decision making process in the selection of a decommissioning option after a severe reactor accident.

No decommissioning work can be done without financial, human and material resources. The problems of funding may give rise to serious constraints on the selection of decommissioning strategy and may lead to deferred dismantling being the only possible alternative.

The next step is to analyse the different detailed options. For immediate dismantling, variations in the use of remote handling technology as well as in aspects such as operation sequence, waste treatment, conditioning, transport and disposal can be considered.

For long term storage followed by deferred dismantling, the variations include parameters such as the level of cleanup in the facility prior to long term storage, the monitoring and surveillance programme, and the length of the storage period before dismantling.

For each option identified, the planning must be performed in sufficient detail to permit estimation of the radiation doses to the workers and to the public that would result from each option. Similarly, the costs and other resource requirements for each option as well as other non-radiological impacts must be estimated.

If more than one option is possible, an optimization of radiation protection in which the radiation dose and its resulting detriment are balanced against the costs (financial and other) should be carried out. Various standard methods are available for doing this. The most frequently used are the cost–benefit and the multi-attribute approaches [35–38].

Since the formal decision is often made on a political level, the impact of sociopolitical considerations, public attitudes as well as other non-technical factors may be significant. It is outside the scope of this technical report to analyse these factors but it should be pointed out that the decision makers must be properly advised about the health, environmental and economic effects of the options considered.

7. CONCLUSIONS AND RECOMMENDATIONS

A number of important conclusions can be derived:

— Methodological and technical approaches utilized for the cleanup of a damaged facility and the selection of an appropriate decommissioning strategy should draw on the practical experience available from previous decommissioning activities, in particular post-accident situations.

— Efficient monitoring of the damaged facility and its surroundings should be initiated at the earliest time after the accident and be continued as long as there is a potential risk of release of radionuclides to ensure adequate protection of humans and the environment.

— Once the situation at a nuclear facility experiencing a severe accident has been stabilized and the post-accident cleanup is completed, any further steps taken should comply with basic guidance on the decommissioning of nuclear facilities.
— The choices available for decommissioning after an accident are severely limited if there is no established waste disposal or storage site.
— Administrative mechanisms to ensure the availability of appropriate funds in the event of an accident should be identified by each Member State so as to prevent a situation in which a severely damaged facility has to be kept in an unsafe condition because of a lack of funds.

A number of areas may be recommended for future study. These include:

— investigation of chemical and physical interactions between damaged fuel and barrier materials
— investigation of mechanisms for migration of radionuclides through barrier materials
— development of criteria for the selection of appropriate barrier materials and analysis of appropriate combinations of different natural or man-made barrier materials
— development of easy monitoring and analysis of radionuclides of interest
— development of methods for repairing damaged containments
— development of remotely operated technologies to allow examination of the physical and chemical state of damaged fuel, to assess whether recriticality is possible, to retrieve samples for analysis and to assess temperature and radiation fields
— development of remote manipulation and robotics techniques to remove and encapsulate damaged fuel and other high activity components.

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Annex

REVIEW OF FOUR ACCIDENTS

Each serious accident is likely to be unique so that the state of the plant after the management of the emergency phase will vary not only because of differences in the accident sequence and final outcome but also because of differences in the original plant design. However, it is useful to summarize for four historical examples the initial accident, the control measures and the final state after cleanup or decommissioning activities.

In spite of the uniqueness of each accident, there are some common factors such as:

- damage to the fuel, fuel support structure and other reactor internals, making conventional defuelling impossible;
- spread of radioactive contamination to areas within the building which would normally have been clean or have contained only trace levels of radioactivity.

On the other hand, the scale of the problem can range from a situation in which the buildings and containment are physically undamaged to one in which major demolition was caused. Table A–I gives a summary of post-accident cleanup and decommissioning experience. The four accidents described in this annex span the above range of severity and are presented in chronological order:

- Windscale Pile 1
- Lucens Experimental Reactor
- Three Mile Island Unit 2
- Chernobyl Unit 4.

A–1. WINDSCALE PILE 1

The Windscale Pile 1 reactor, which was primarily constructed to produce plutonium, was of a simple graphite moderated uranium fuelled design. The fuel cladding material was aluminium and the cooling used forced air convection. Cooling flow was once through with eight 2300 hp (1.7 MW) blowers drawing air through filters from the atmosphere, whence it was forced through the horizontal fuel channels, collected and discharged through a 125 m chimney at the top of which it again passed through filters. There were roughly 180 t of fuel and the full thermal power output was approximately 180 MW. Construction was completed in 1950.

Since this was a graphite reactor operating at a low temperature, it was necessary to release the Wigner energy stored in the graphite lattice in a controlled manner

TABLE A-I. SUMMARY OF NUCLEAR REACTOR POST-ACCIDENT CLEANUP AND DECOMMISSIONING EXPERIENCE

Facility name and location	Reactor type	Power level (MW(th))	Year of accident	Status following accident cleanup
NRX, Canada	Research, pool	10	1952	Returned to service
Windscale Pile 1, UK	GCR	180	1957	Pre-decommissioning phase
NRU, Canada	Research, heavy water	200	1958	Returned to service
SL-1 Reactor, USA	Military, BWR	3	1961	Decommissioned
PRTR, USA	Research, heavy water	—	1965	Returned to service
Enrico Fermi, USA	Fast breeder	—	1966	Returned to service
Lucens, Switzerland	Experimental, heavy water	30	1969	Decommissioned
Three Mile Island, USA	Commercial, PWR	2800	1979	Still in accident cleanup. 'Monitored storage' envisaged by the end of 1993
Chernobyl, Ukraine	Commercial, RBMK	3200	1986	Confinement structure in place. No decision made as to eventual plant status

and on a regular basis to avoid unplanned releases occurring during reactor operation. Several such planned releases had been carried out successfully prior to October 1957 and a further routine release was then commenced.

During the 1957 release one region of the reactor overheated and caught fire. Unsuccessful attempts were made to blanket the fire with CO_2 gas but finally the fire was extinguished by a combination of flow starvation and the dousing of the graphite with thousands of cubic metres of water.

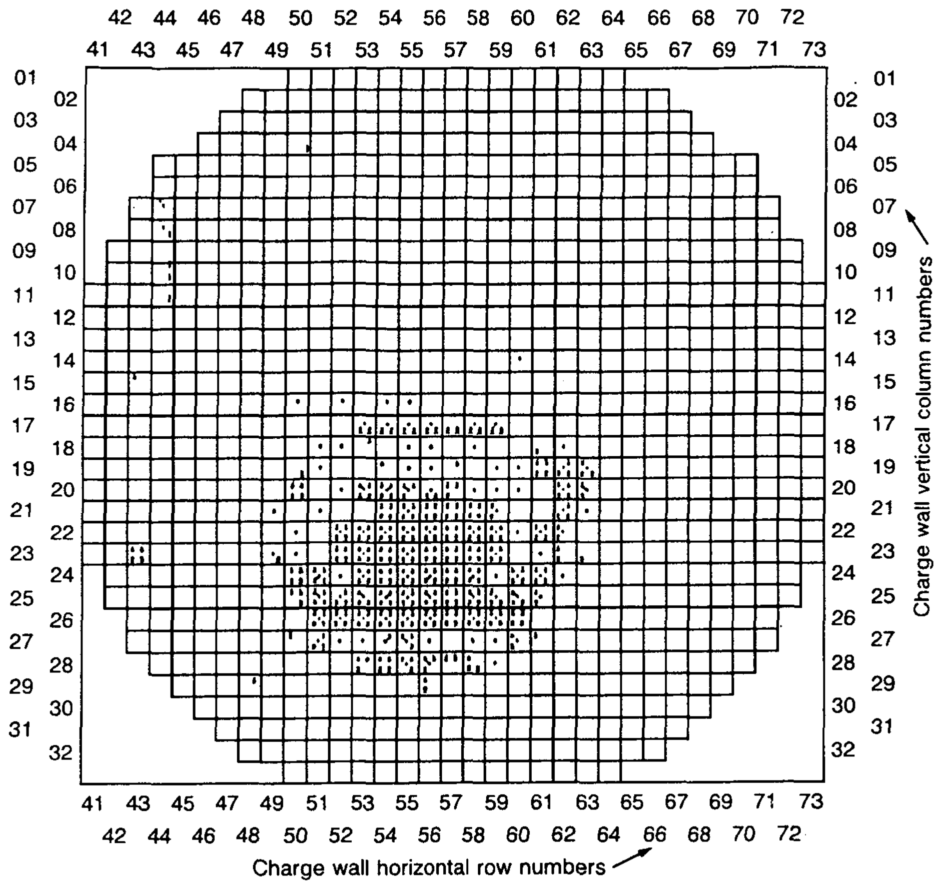

FIG. A-1. Pile No. 1; fire damaged zone viewed from charge face.

In the immediate aftermath of the accident all the fuel which could be discharged by the standard technique was removed, leaving possibly 10% of the fuel charge in the damaged core (Fig. A–1). All control and shutoff rods were run into full insertion and physically secured and the drive mechanisms removed. Because the reactor was on the same site as a large chemical reprocessing plant, there were no special problems associated with the treatment and disposal of the water used to quench the fire, which because of the design of the fuel handling route had largely flowed into the fuel pond.

At the end of the immediate post-accident phase the situation was as follows:

(a) The reactor was thermally stable, and no risk of criticality existed.

(b) The entire outlet side including the chimney was severely contaminated.

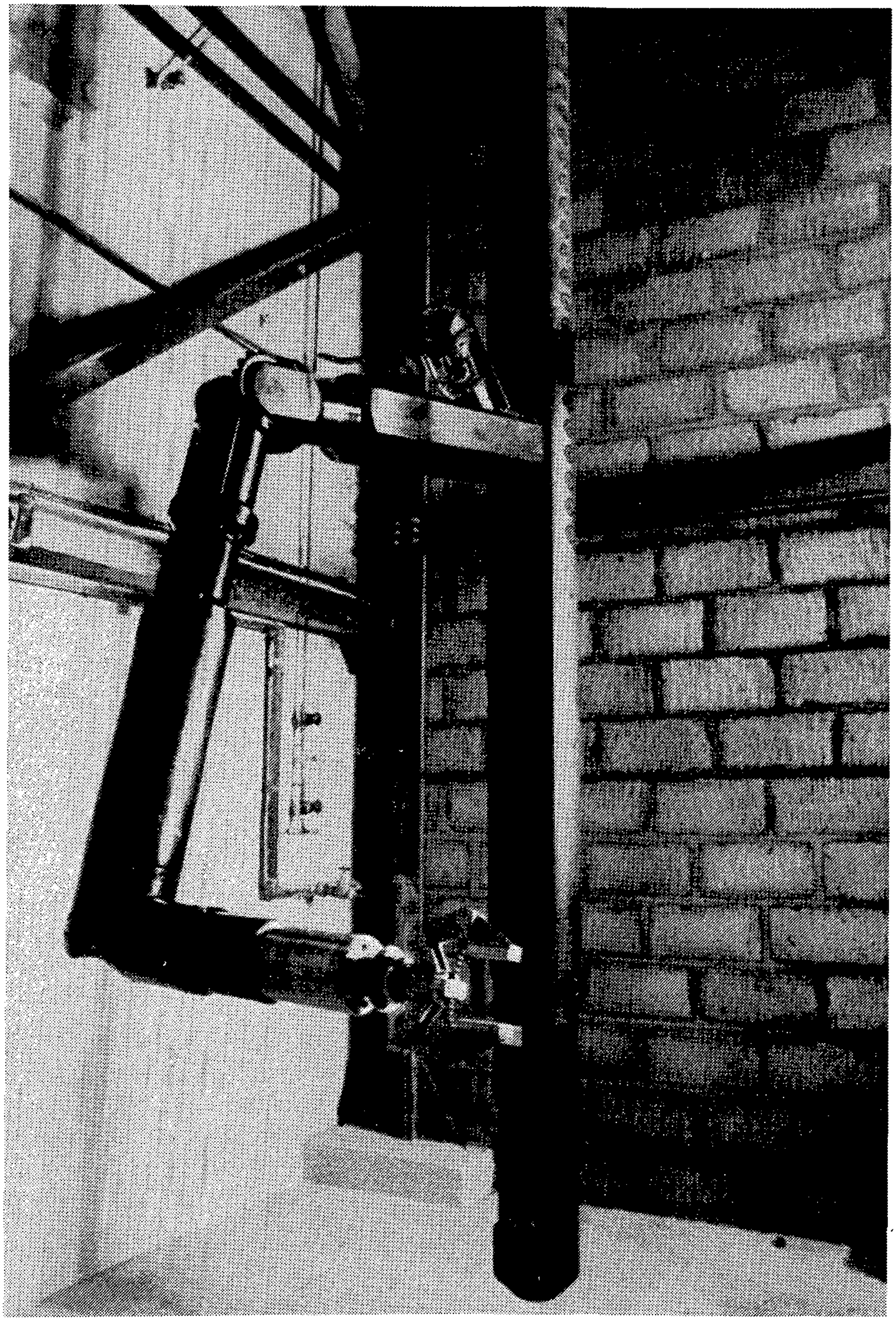

FIG. A-2. *Mast-type remote manipulator.*

(c) The damaged region was not visible and conditions within it were largely unknown.
(d) There was a significant quantity of highly active debris and fuel in the air and water ducts and at various other points inside the biological shield.

However, there was no significant physical damage outside the core region itself and little contamination within the normal working areas. The inlet air system was cut and blocked, leaving only a small residual flow, adequate to maintain thermal stability.

The reactor was placed under a surveillance regime for several years, with periodic reviews of the physical condition of the reactor and the technical situation. By 1982 it was decided that remote technology had advanced sufficiently to embark on an initial pre-decommissioning phase. The programme comprised four main activities:

(1) The provision of a fully engineered ventilation system and complete isolation of the reactor from the chimney;
(2) The removal of all the residual fuel and debris external to the core, mostly from the water and air ducts;
(3) The cleaning, draining and sealing of the water duct, including isolation from the pond;
(4) The removal of the upper chimney section, which was constructed in engineering brick with steel support work, to leave the concrete lower section.

Provision of the ventilation system is almost complete although the final installation has revealed hitherto unsuspected problems caused by the fact that the detailed as-built state differs from the original drawings. These differences were revealed only by the civil work done to prepare the air dam sites.

Two remote manipulators have been acquired, one of the mast type with a long reach (Fig. A–2) to remove debris from the discharge face and one mounted on a tracked vehicle capable of operating in both air and under water (Fig. A–3) to clean up the air and water ducts. Both are based on existing systems developed for other purposes in difficult environments.

At the same time, studies were started to determine the long term decommissioning strategy. The first preliminary study showed that the uncertainties about the physical state of the fire damaged region were so large that it was impossible to make a reliable comparison between the various options. Essentially there were steps in each of the options which could not, with existing knowledge, be shown to be practicable.

The main problem with the Windscale reactor is that the fire damaged region is inside the physically massive graphite core. External examination gives little indication that there has ever been an accident and none at all about the internal state. Access to the front and rear faces of the graphite is difficult but possible; access to

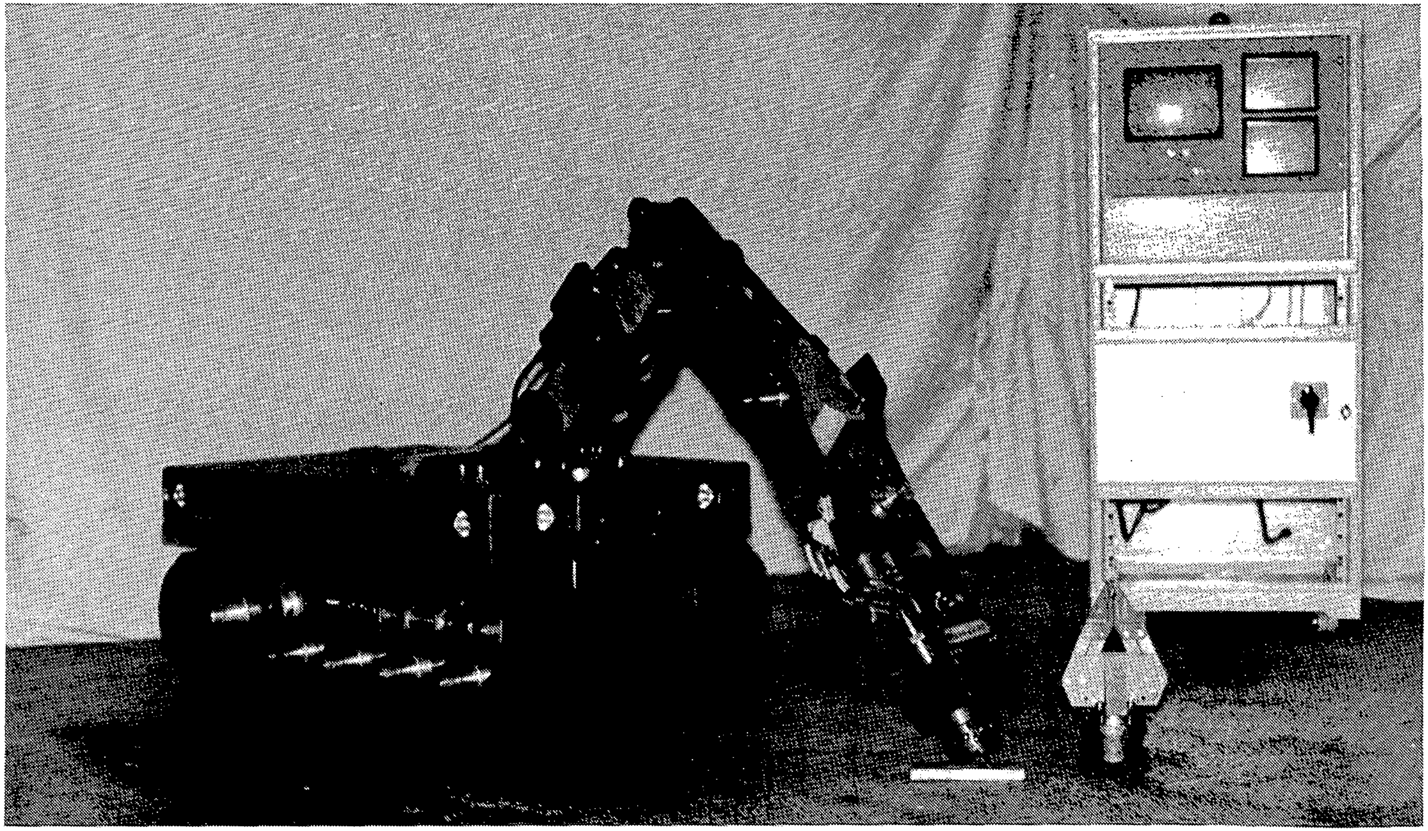

FIG. A-3. Remotely operated recovery vehicle.

the sides, top and bottom is virtually impossible. Thus not only is it difficult to deploy equipment to carry out standard NDT techniques but also many such techniques are inapplicable because they have inadequate penetration range to produce worthwhile results. On the other hand it is very difficult to produce safety arguments to support intrusive examination without some understanding of the state of the fire damaged region.

Work on the pile will inevitably be time consuming and costly, with a high development content and an associated degree of uncertainty. In order to proceed, it has been proposed that multiattribute utility analysis be used to systematize the decision process. This technique has already been used in several published generic studies of decommissioning strategy. Its strength is that it enables complex decisions to be handled in a structured way and also allows the identification of critical uncertainties so that the development work can be correctly focused onto the key issues.

Initially, a wide range of options and variations between options are being studied, ranging from immediate dismantling to long term safe storage followed eventually by hands-on dismantling. As information is generated and fed into the model it is anticipated that it will be possible to eliminate a number of unpromising options, leaving the final detailed optimization to be performed on the remaining two or three most promising alternatives [A–1, A–2].

A–2. LUCENS EXPERIMENTAL REACTOR

The experimental nuclear power station at Lucens was equipped with a CO_2 cooled, D_2O moderated experimental reactor built in an underground cavern in a hill near the town of Lucens, Switzerland. Each of the 73 reactor fuel elements consisted of slightly enriched metallic uranium rods clad with a Mg–Zr alloy and situated in seven channels bored in a graphite block vertical that was centred in a Zircaloy pressure tube, with pressurized CO_2 for cooling the fuel rods. The vertical pressure tubes were located in an aluminium calandria that contained the D_2O moderator.

On 21 January 1969 an accident occurred in which one of the peripheral fuel elements overheated and its pressure tube ruptured. The accident is described in detail in Ref. [A–3]. The decommissioning of the reactor following the accident is described in Ref. [A–4].

The immediate cause of the accident was water that entered the CO_2 coolant system through a defective shaft gasket, causing corrosion of the cladding of several fuel elements. Corrosion products settled at the bottom of coolant channels, partially blocking the coolant flow. Interruption of the coolant flow resulted in the melting and subsequent ignition of one of the fuel elements and in the rupture of the pressure tube separating the CO_2 coolant from the heavy water moderator. The reactor coolant expanded into the moderator tank, increasing the pressure in the tank and causing the fracture of its rupture disks. A mixture of contaminated D_2O steam and CO_2 then entered the upper reactor vault, carrying with it fission products and vaporized fuel fragments. A large portion of the liquid heavy water moderator dropped onto the floor of the lower reactor vault. Apart from CO_2 and noble gases, very small amounts of solid fission products were released into the reactor cavern.

Some airborne radioactivity leaked through a cable penetration from the reactor cavern to the adjoining plant cavern and, after the main ventilation of the plant was stopped, also to the control building. After analyses of the atmosphere in the reactor cavern, the cavern was vented through the stack via iodine filters.

The accident destroyed one fuel element and seriously damaged the moderator tank of the reactor so that a decision was made not to restart the facility. The period from January 1969 to March 1970 was used for initial decontamination of the reactor cavern, for recovery of as much of the heavy water as possible, for drying the cavern atmosphere and removing tritium vapours, and for remote inspection of the region within the biological shield where most of the damage was concentrated. This was followed by disassembly of the upper reactor vault components, cutting and removal of the pipes that carried the CO_2 coolant to and from the reactor, repeated decontamination of the reactor cavern, and fixing of the remaining contamination by painting. The reactor was defuelled (except for the damaged fuel element) by an ad hoc procedure. The pressure and calandria tubes were then cut and removed. The calandria was sectioned and packaged for disposal. After the upper portion of the calandria vessel had been removed, the damaged fuel element and its pressure tube were recovered.

Final steps in the dismantling included the disassembly of the highly contaminated systems and the decontamination and disassembly of the station for the treatment of radioactive material. Decontamination was completed during the last half of 1972 and the first half of 1973.

Dismantling a small plant after a short operational life interrupted by a serious incident leading to severe damage and contamination cannot be identical to decommissioning a nuclear power plant after 30 or 40 years of full power operation. Nevertheless, even if the life of the plant had been short, erroneous use of Co alloys for the shock absorbers of the control rods and for the mouthpiece of the fuel elements resulted in high radiation levels (between 0.5 and 1 Gy/h). The decommissioning of Lucens suffered in addition from a lack of space (underground construction being costly), from a lack of sufficient floor loading capacity and from the inadequacy of the lifting equipment.

It was decided in 1988 to return the site to non-nuclear uses. The plan for final decommissioning calls for the following measures [A–5]:

(1) Filling up with concrete two of the three caverns (reactor and fuel pond) still containing up to 40 GBq (1Ci) of disseminated radioactive substances, fixed by paints since 1973, in order to immobilize this radioactivity and to ensure mechanical stability within the bedrock (sandstone).

(2) Filling up with concrete the original basement of the plant cavern (lower part of the turbine hall and some auxiliary rooms), leaving the upper part free for other activities.
(3) Construction of a supplementary drainage system around the caverns in addition to the existing one, which is still in operation and will remain so. The new system is designed to function passively and without the need for maintenance for the next 30 years and is aimed at attracting groundwater from the bedrock, collecting it by means of a piping system and conducting it to a collecting pond permitting control and monitoring.
(4) Construction of a new passive pipeline for conducting this drainage water under gravity from the control pond directly to the nearby river, thereby preventing this water from flowing back into the water table.
(5) Removal of the heavy containers in which contaminated and/or activated parts of the reactor and of its equipment have been sealed and their transfer to a future repository for conditioning and/or final storage.

The detailed draft project covering items (1)–(4) above was forwarded in 1988 to the safety authorities for review and assessment. In their safety assessment report, the latter agreed to the project, but imposed the obligation that during one year the functioning of the passive drainage system (item (3)) and of the pipeline to the river (item (4)) be observed, in order to show that they would function for at least 30 years without any maintenance.

The construction permit for items (1)–(4) was granted in December 1990. Construction work began in June 1991 and will probably be completed during 1992. The 'one-year observation phase' imposed in the construction permit will probably begin in 1993 after completion of the finishing work (grouting injections).

Release of the whole site of the former Lucens plant will then be pronounced in two steps. The first step will cover the release of the underground part of the site, including the drainage system and the pipeline to the river, as well as the major part of the site area; a decision on this first step will be based on positive results from the observation phase and is expected in early 1994. At that time, the nuclear part of the Lucens site still maintained under regulatory surveillance will be reduced to the small part of the area on which the heavy radioactive waste containers are temporarily stored (item (5) of the above list). The second step (release of the entire site) will be taken after the effective removal of these containers to a repository. For the time being (1992), the date of this removal is unknown [A-5].

A-3. THREE MILE ISLAND UNIT 2

Three Mile Island Unit 2 (TMI-2) is a pressurized water reactor. The reactor has three independent cooling loops. Heat generated by the fission process within the reactor core is removed by means of the primary coolant to two steam generators

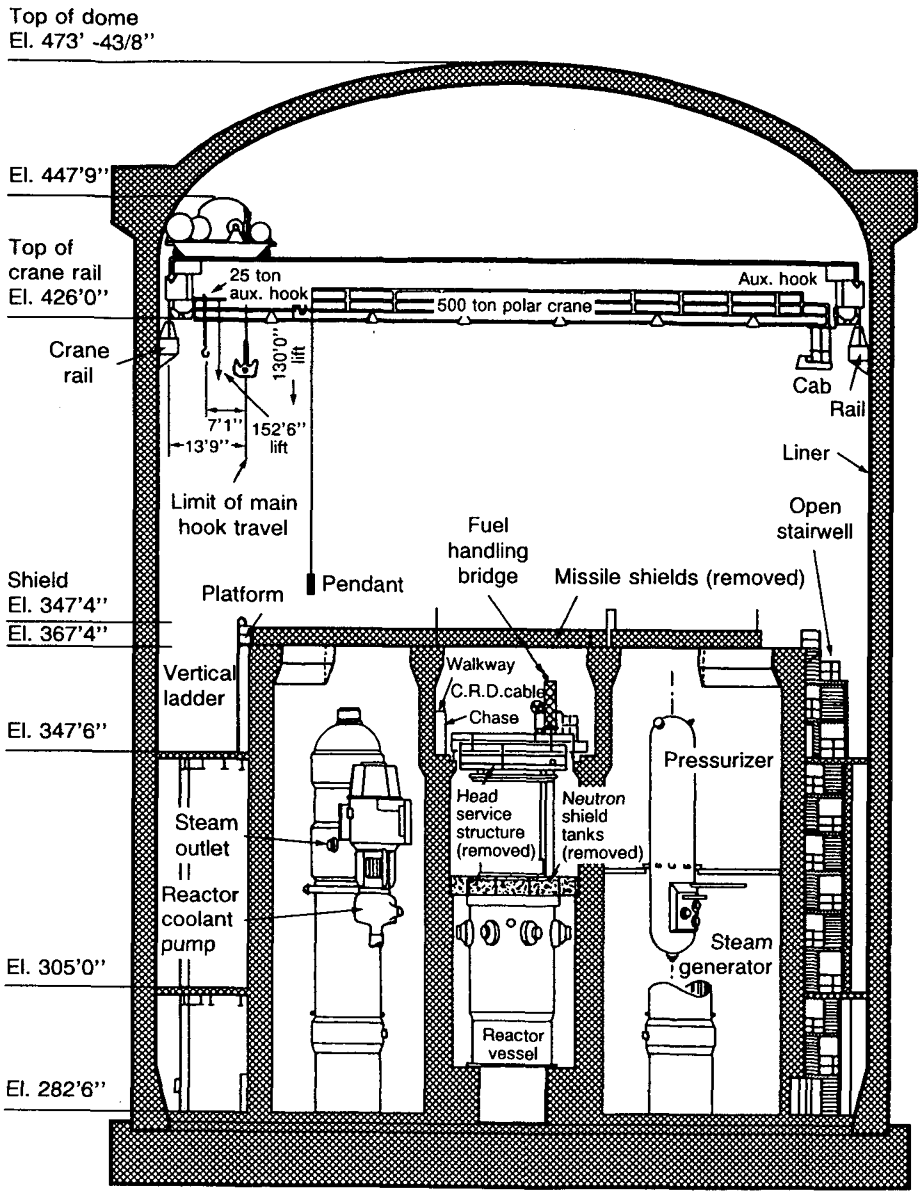

FIG. A-4. TMI-2 reactor building cross-section [A-6].

where steam is produced to operate a turbine. Between the issuance of its operating licence (8 February 1978) and the date of the accident, TMI-2 had operated for about 95 effective full-power days. TMI-2 was operating at 97% power when the accident occurred.

On 28 March 1979, a malfunction occurred to components that maintain the flow of coolant water to the steam generators in the secondary loop. This resulted in a loss of ability to remove heat from the primary loop. This caused the coolant water temperature and pressure to increase rapidly. This, in turn, caused a relief valve on the pressurizer to open. Steam and water were discharged to the reactor coolant drain tank which is located in the basement and is equipped with a pressure limiting rupture disk (Fig. A–4).

A key factor in the accident was that the relief valve failed to close when the pressure returned to normal. Water continued to be discharged through the open relief valve into the drain tank. As the water level fell, the fuel became exposed, resulting in intense heat which caused severe fuel damage.

So much water and steam were discharged through the relief valve that the storage capacity of the drain tank was quickly exceeded, causing the rupture disk to burst and discharge huge amounts of radioactive coolant to the reactor building sump and basement. Radioactive coolant water in the reactor building sump was automatically pumped into the auxiliary sump tank in the auxiliary building. Since this tank was already about half full, much of the water spilled into the auxiliary building, which was not designed to contain radioactive material.

After the fuel damage occurred, radioactive materials were transported through the primary coolant system via the letdown line to the makeup and purification system in the auxiliary building. Because this liquid is a stream of primary coolant directly from the reactor, it contained significant amounts of radioactivity. As a result of liquid leaks in the makeup and purification system, large amounts of radioactive material were released into the auxiliary building. No longer held under pressure, krypton, xenon and other volatile radionuclides evolved from the water into the auxiliary building atmosphere.

Exposed surfaces in the reactor building and auxiliary and fuel handling building were contaminated with material in the reactor coolant and with radionuclides that became airborne as steam escaping from the reactor coolant system condensed during and shortly after the accident. After the accident, the water in the basement was heated by residual heat from the reactor vessel, evaporated, condensed on the walls, and drained down onto the floors and back into the basement. This period of evaporation and condensation contributed to the permeation of radionuclides into porous surfaces, such as concrete, and to the incorporation of radionuclides into corrosion layers as iron surfaces rusted [A–6].

The accident at TMI-2 generated significant quantities of radioactive waste. Approximately 2500 m^3 of water with an activity of 1 TBq/m^3 were released into the auxiliary building, fuel handling building, service building and diesel generator

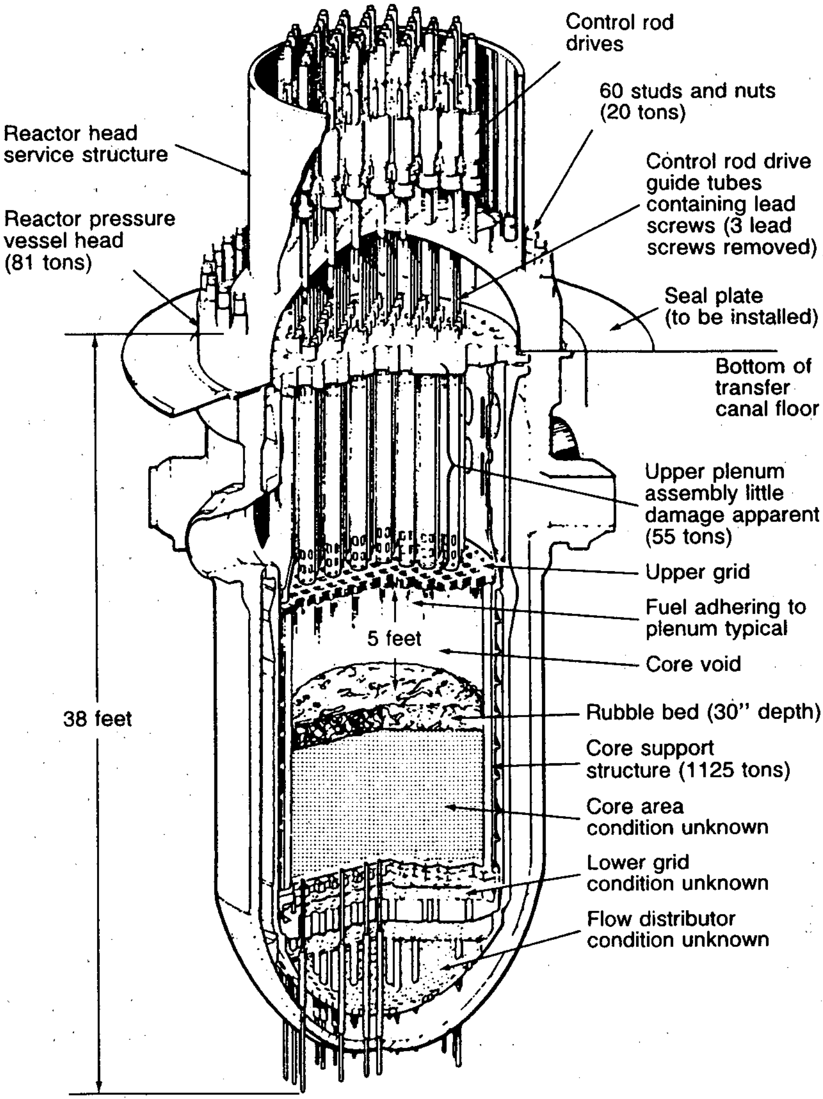

FIG. A-5. Cutaway view of TMI-2 vessel showing status in October 1984 [A-6].

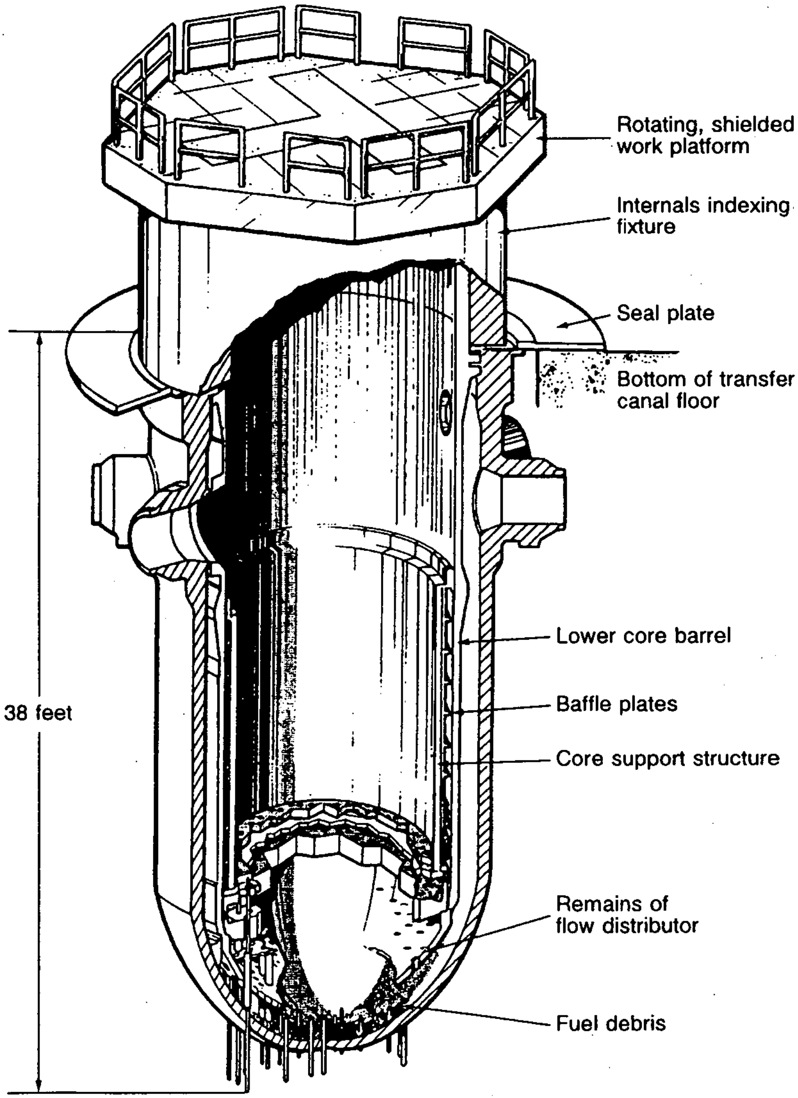

FIG. A-6. Cutaway view of TMI-2 vessel following completion of defuelling efforts [A-6].

building. Another 2000 m^3 of water with an average of 7 TBq/m^3 was contained in the reactor building. In addition, a large amount (approximately 60%) of the core inventory of noble gases and volatile nuclides was released to the reactor building. On inspection, the reactor core was found to be largely broken down to rubble. Of the original 130 000 kg of core material (fuel plus structure), only 20–30% was found to remain as partially intact fuel assemblies [A–7].

The situation was stabilized — with the reactor vessel full of water, the lower portions of the reactor containment building filled with water, and the remaining portions of the containment building interior highly contaminated. Residual noble fission gases were released to the environment at a carefully controlled rate some time after the accident [A–7, A–8]. The containment building remained sealed for an extended time period, with the first re-entry by personnel occurring 475 days after the accident.

The water in the auxiliary buildings was treated by conventional organic resin technology in a system known as EPICOR II. A proprietary mix of conventional anion and cation bead resins was used in the processing campaign. The EPICOR vessels used were 1.5 m^3 liners containing roughly 0.9 m^3 of ion exchange resin. Of the 75 liners deployed, 50 were highly loaded, containing about 74 TBq each of mostly ^{137}Cs and ^{90}Sr. The United States Department of Energy (USDOE) accepted these wastes for disposal at its Idaho Falls Facility [A–7].

Reactor building and reactor cooling system liquids were treated with the submerged demineralizer system (SDS). The SDS was installed in the reactor building fuel pool and utilized zeolite material to function as an ionic sieve to remove caesium and strontium activity from the waste stream.

Each SDS liner was 0.3 m^3 in volume and contained 0.23 m^3 of ion exchange resin. The liners were each capable of containing 2200 TBq of ^{137}Cs (total nuclide activity 3700 TBq). Approximately twelve liners were used during the campaign. Waste shipments were made using high integrity containers in shielded shipping casks. The USDOE also accepted these wastes for disposal at Idaho Falls [A–9].

Cleanup of the upper portions of the containment building and defuelling of the reactor has been performed over the ten-year period since the accident. The high radiation levels initially present in the containment building work areas necessitated the use of a variety of remote and semiremote decontamination techniques. The most effective method for surface decontamination was found to be a simple water flushing and scarifying of concrete surfaces [A–6].

Defuelling the reactor presented many unusual problems because of the severe damage to the fuel, and the presence of a massive bed of fused rubble and resolidified once-molten fuel (Fig. A–5). In addition, severe problems were encountered with maintaining water clarity owing to the growth of microorganisms and the continued resuspension of fine particles in the water. Since all defuelling was conducted using remotely operated tools and video monitors, the inability to see through the water within the reactor vessel greatly hampered operations.

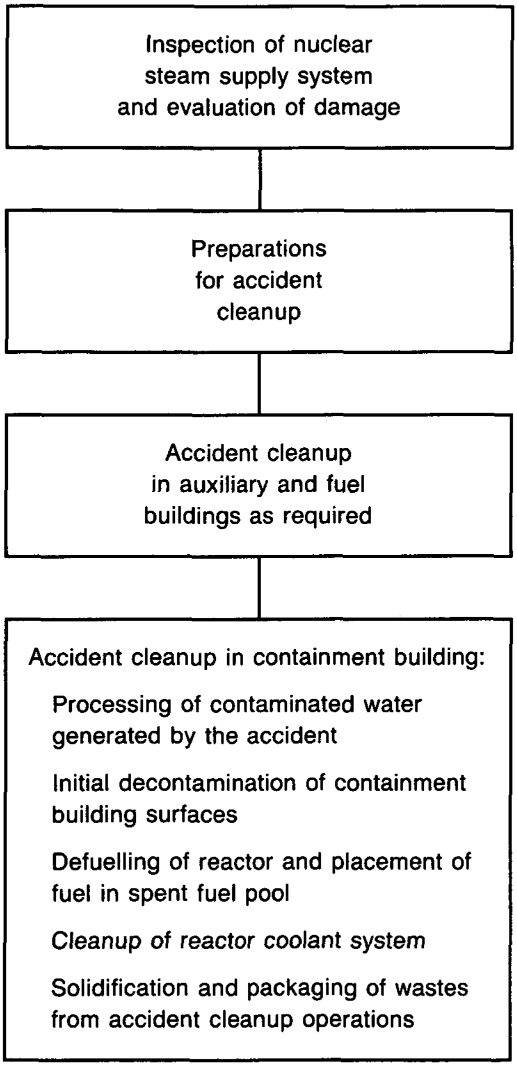

FIG. A-7. Sequence of accident cleanup tasks at TMI-2.

Using various drilling and grapple tools, the core material was loaded into disposable fuel canisters that contain poison material and catalytic recombiners. Each canister was designed to accommodate one fuel assembly (or a similar quantity of rubble). Canister storage prior to shipping was maintained in the spent fuel pool. A shipping cask was designed to accommodate seven canisters for transportation to a USDOE site. Defuelling of the reactor vessel was completed in early 1990 (Fig. A–6). More than 99% of the original core was removed and shipped off the site. Estimates of the remaining quantities of fuel (in kg) are as follows [A–10]:

Auxiliary and fuel handling building	11
Reactor building outside the reactor coolant system	72
Reactor vessel	609
Reactor coolant system (ex-vessel)	90
	Total 782

Although substantial progress has been made in the decontamination of the reactor building and the auxiliary and fuel handling buildings, considerable work remains. It is estimated that there is still approximately 1600 TBq of ^{137}Cs and 90 TBq of ^{90}Sr in the reactor building [A–6]. Figure A–7 graphically describes the sequence of accident cleanup tasks at TMI-2 [A–8].

Current plans call for placing the facility into long term monitored storage until such time as Three Mile Island Unit 1 is ready for decommissioning (around the year 2014), at which time both units will be decommissioned [A–6, A–11].

A–4. CHERNOBYL UNIT 4

The Chernobyl Nuclear Power Plant Unit 4 was a vertical pressure tube graphite moderated, boiling water cooled reactor (RBMK). The chief design features of RBMK reactors, such as that at Chernobyl, are the following:

(1) Vertical channels containing the fuel and coolant, enabling local refuelling while the reactor is in operation;
(2) Fuel in the form of bundles of cylindrical fuel elements made of uranium dioxide in zirconium tube-type cladding;
(3) A graphite moderator between the channels;
(4) A boiling light water coolant in the multiple forced circulation circuit, with direct steam feed to the turbine.

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. In the process of

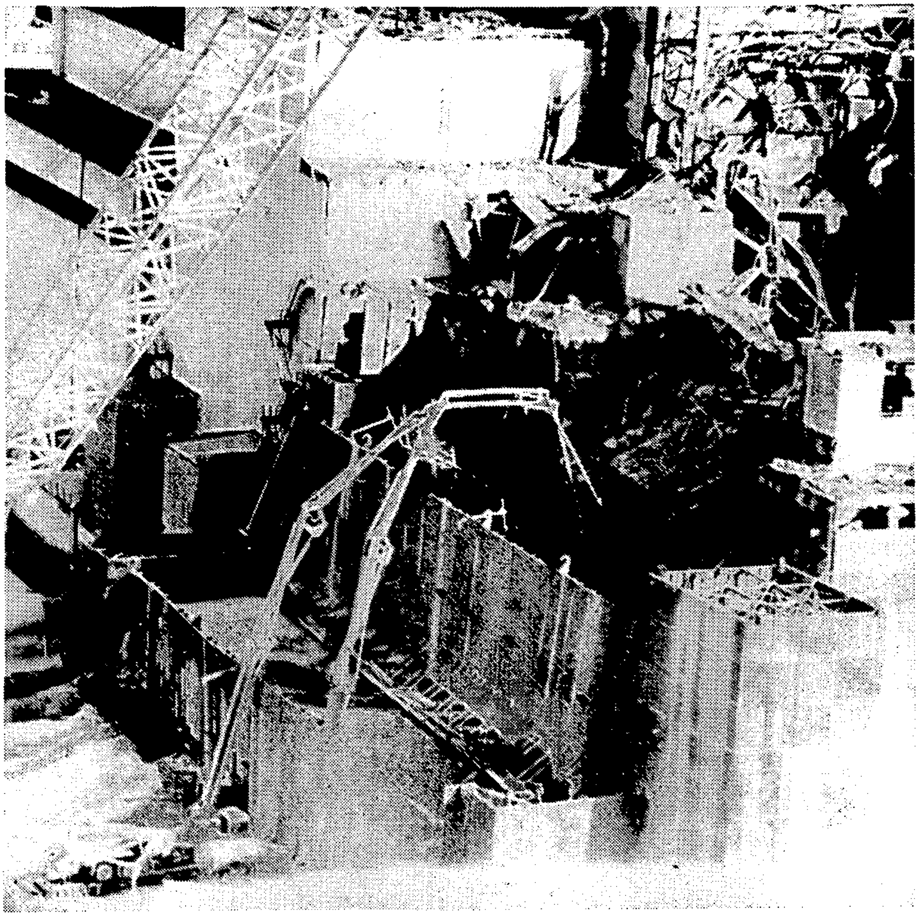

FIG. A-8. Building the second terrace on the north side of Chernobyl Unit 4.

establishing the test conditions for the reactor, the operators brought the plant to an unstable operating condition. However, they chose to run the test from this unstable condition. To prevent the reactor from automatically shutting down, they purposely bypassed several safety systems.

The initiation of the test under these unique 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.

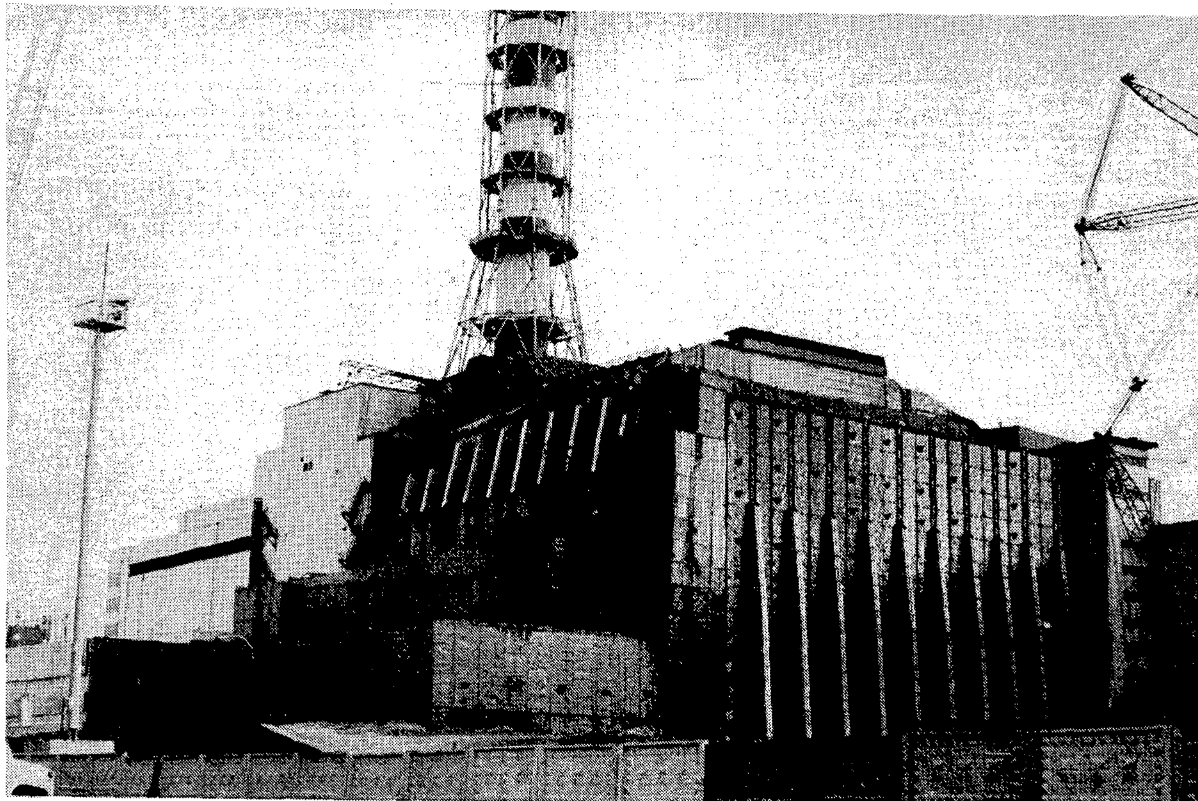

FIG. A-9. Chernobyl Unit 4 after completion of the post-accident confinement system.

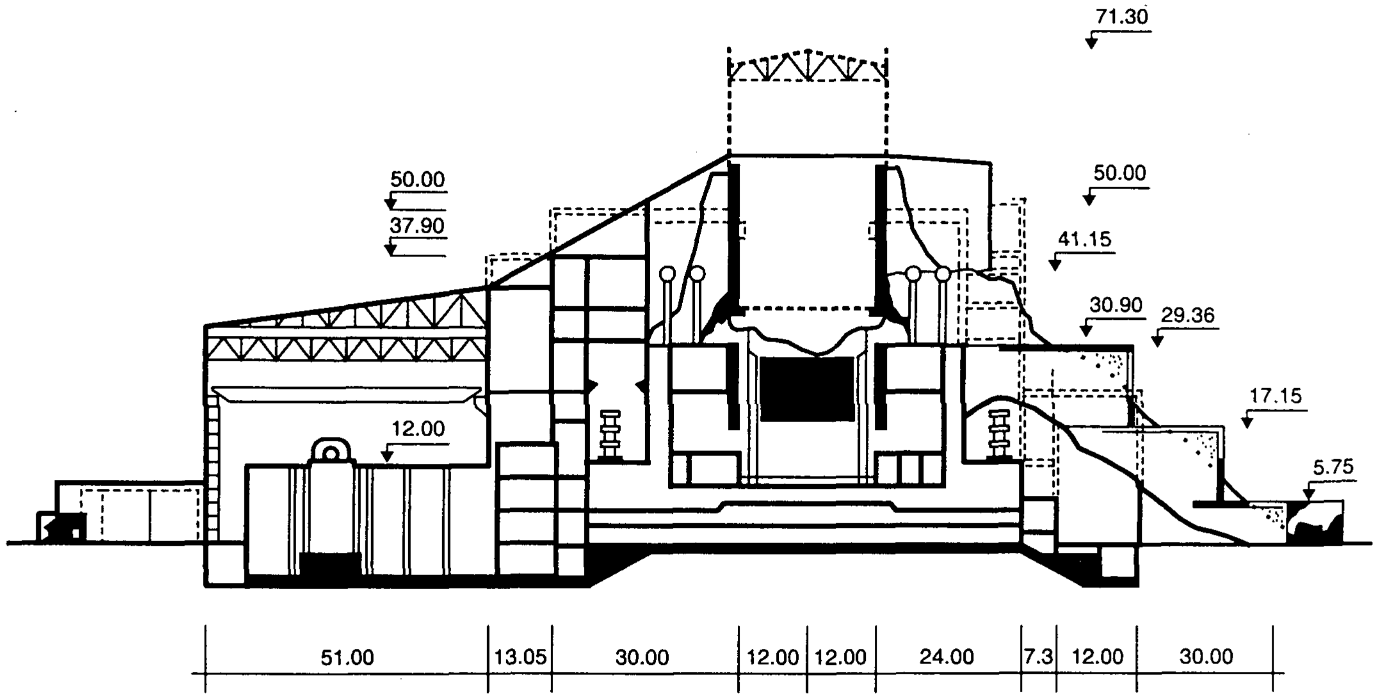

FIG. A-10. Chernobyl Unit 4 entombment plan.

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 [A–12].

By 6 May 1986, the discharge of radioactive material from the facility had decreased by several orders of magnitude. Nitrogen gas was pumped into a space under the reactor building to cool the fuel remaining in the reactor.

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 (Figs A–8 to A–10) 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 [A–13]. The shelter was designed to resist even severe environmental conditions for 20–30 years. Outer protective walls were built along the perimeter; inner concrete partition walls were built in the turbine hall between Units 3 and 4; a metal partition wall was installed in the turbine hall between Units 2 and 3; 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. The surface layer of soil in the area adjacent to Unit 4 was removed to local disposal sites. This area was then covered with concrete and asphalt and the surface levelled for self-propelled cranes and other machinery [A–14].

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 (Figs A–8 to A–10). While the confinement system has been designed to resist even severe environmental conditions, there is still 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

TABLE A–II. MASSES AND ACTIVITY OF THE MAIN LONG LIVED RADIONUCLIDES LEFT INSIDE THE CHERNOBYL SHELTER

Nuclide	Mass (kg)	Activity (TBq) (round figures)
Sr-90	42	2×10^5
Cs-137	54	2×10^5
Ce-144	32	4×10^6
Pu-238	1.4	1×10^3
Pu-239	400	1×10^3
Pu-240	170	1.5×10^3
Pu-241	48	2×10^5
Am-241	1	1.5×10^2
Cm-242	0.25	3×10^4
Cm-244	0.057	1.5×10^2

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 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, and the ventilation exhaust is monitored for the presence of shorter lived iodine isotopes. 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

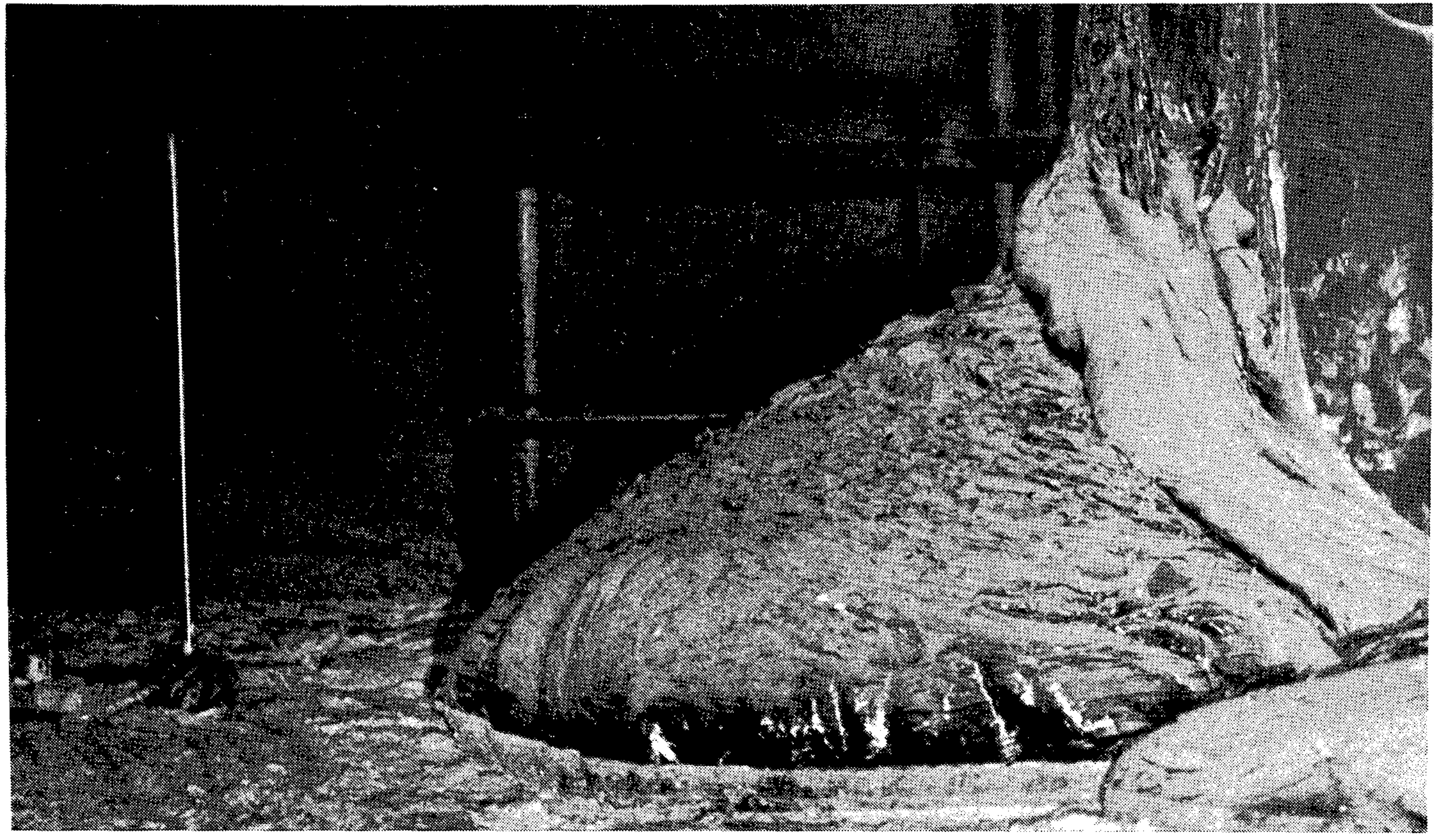

FIG. A-11. Photograph of solidified melt of the fuel-containing 'lava' found in one of the column reactor rooms at a distance of a few tens of metres from the Chernobyl reactor core.

fuel and fuel-containing mixtures. Neutron and gamma detectors and heat control devices could be installed through the boreholes. 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.

Table A–II shows the masses and activities of the main long lived radionuclides inside the shelter. Measurements confirmed that about 97% of the fuel inventory remained in the reactor vault. The location and physicochemical characteristics of the fuel are extensively described in Ref. [A–15] (Fig. A–11).

Some disturbing observations were made in 1990 during examinations of fuel-containing materials:

— The embrittlement of 'lava-like' masses, their gradual destruction and the formation of fuel dust (the peculiar lava-like materials were produced by unusual physicochemical fuel interactions with reactor and building materials during and after the active phases of the accident).
— The gradual reduction in size of fuel particles and their transformation into a more dangerous, finely dispersed fraction.
— The formation of soluble uranium compounds at the boundaries of fuel-containing materials.
— The gradual leaching of radionuclides by water from the fuel-containing masses. (When it rains, a considerable amount of water penetrates into the shelter through slots in the roof and walls.)

For the above reasons, great attention was devoted in 1990 to the development of safer storage of the nuclear fuel.

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.

The option which is colloquially called 'Arch' envisages a new envelope being built outside the existing shelter in the next few years. The main specification would be to build a structure which is strong and hermetically sealed so that any collapse of the destroyed unit structures inside does not affect the overall structural stability or the radiation conditions. This design would ensure the safe condition of the structures inside.

Of three solutions studied by the Scientific Council, a Government advisory body, the one called 'intermediate concreting of the reactor section' was selected to address the radioactive dust hazard. The option calls for filling all the reactor section compartments of the plant with a concrete mixture, up to a height of around 41 m. This would allow dismantling of the shelter's metal structures above this height. Such mass concreting would increase the stability of the structure as well as capture and confine all the particulates. However, no firm decisions have yet been taken on how to proceed with the long term sealing or decommissioning of the reactor.

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