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Remote technology in spent fuel management

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FOREWORD

Spent fuel management has always been one of the important stages in the nuclear fuel cycle and it is still one of the most vital problems common to all countries with nuclear reactors. It begins with the discharge of spent fuel from a power or research reactor and ends with its ultimate disposition, either by direct disposal or by reprocessing of the spent fuel.

Continuous attention is being given by the IAEA to the collection, analysis and exchange of information on spent fuel management. Its role in this area is to provide a forum for exchanging information and for co-ordinating and encouraging closer co-operation among Member States in certain research and development activities that are of common interest. Spent fuel management is recognized as a high priority IAEA activity.

Within its spent fuel management programme, the IAEA has monitored the progress, the benefits and the implementation of remote technologies such as manipulation of remote tools, robotics, etc. Since the last IAEA meeting on remote technology related to the handling, storage and disposal of spent fuel, held in 1994, further progress has been recognized as the new spent fuel handling system designs include more and more applications of robot techniques. An Advisory Group Meeting on Remote Technology in Spent Fuel Management was held in September 1997 in order to bring together specialists working in this field and to collect information on new technical and economic developments.

The objective of the Advisory Group meeting was to review remote technologies in use for the complete range of spent fuel handling and spent fuel management covering wet and dry environments, to describe ongoing developments and to prepare a technical report.

The participation of the experts and their contributions made at the meeting are gratefully acknowledged. Special thanks go to A.W. Webster, Chairman of the meeting, and M.J. Crijns, who compiled and edited the report. The IAEA staff member responsible for the meeting was H.P. Dyck, Division of Nuclear Fuel Cycle and Waste Management.

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SUMMARY OF THE ADVISORY GROUP MEETING

1. INTRODUCTION

Remote technology is an integral part of spent fuel management (SFM). Extensive use of it is common to countries advanced in the use of intermediate and long term spent fuel storage or reprocessing to solve their backend problems. Reduced radiation exposure, improved safety, reliability and cost savings are all potential benefits of the application of remote technologies to the handling of spent nuclear fuel. Remote spent fuel processing from discharge from the reactor core to shearing operations, in the case of reprocessing, or long term storage, and final packaging, in the case of direct disposal, is already used to some extent as well as for fuel inspection and fuel repair in the reactor pool. More stringent operator dose limits and increasing pressure on the economics of the nuclear fuel cycle (with the goal to be faster, safer, more reliable, cheaper) may be addressed by recent advances in remote technology. These advances have concentrated on more efficient and cost effective systems that enable lower operator dose limits to be achieved in both existing and future facilities while reducing lifetime costs. Remote technologies are practised in order to support the back-end of the fuel cycle under normal operations and at off-normal events. They become more and more important with the handling of advanced (MOX) and high burnup fuel.

2. SYNTHESIS OF THE RESULTS

Remote technologies are increasingly used in nearly all steps of the nuclear fuel cycle. The overall objectives and benefits of remote technology are:

- man dosage reduction;
- faster processing and throughput;
- more reliable processes;
- further on improved quality assurance.

Making positive use of remote technologies needs, however, larger R&D activities for the development and commissioning of remotely operated processes. It was observed that each country present at the AGM focused on selected steps of the nuclear fuel cycle:

- fuel manufacturing: India;
- reactor services: Germany;
- cask handling and servicing: United States of America;
- spent fuel preparation for reprocessing: France and United Kingdom;
- spent fuel preparation for final storage: Sweden.

Other applications of remote technology were not reported during the AGM, e.g. MOX fuel manufacturing in France and UK, spent fuel packaging in Germany.

During the technical discussions, it was pointed out that automated processes for selected technological steps are preferred solutions in comparison to multipurpose robotics. It was also noted, that an adequate nuclear fuel design facilitates the later application of remote technology downstream in the nuclear fuel cycle.

The salient points in national progress are described below:

The front end facilities of the UP2-800 and the UP3 reprocessing plants at La Hague, France, are dedicated to spent fuel handling, i.e. fuel unloading, interim storage, dispatch and measurement. The operations, including maintenance, are performed remotely and automated. The facilities are operated from central control rooms. The use of automation at La Hague is aimed at reducing personnel exposure, increasing purposeful utilization of the equipment, ensuring safety, and improving fuel accountability.

Germany provides an example for remote handling of irradiated fuel assemblies in service activities and from final packaging of spent fuel at the end of its wet storage period (early encapsulation) Irradiated fuel assemblies need to be handled remotely At present computer controlled operation allows a more reliable process operation achieving shorter processing time and better quality due to in-process control

India's nuclear programme covers a wide scope, including fabrication and reprocessing of spent fuel from thermal reactors Sophisticated automation and remote robotics systems have been developed to aid the refabrication of second generation fuel and the dismantling of irradiated fuel from the fast Breeder Test Reactor at Kalpakkam, near Madras Maintenance operation, repair, decontamination and decommissioning activities have been kept in mind Remote systems developed are designed in such a way that they can be easily introduced into plants that are already in operation Fissile material accounting remote system maintenance and standardization of subsystems have been carefully looked at in the interest of long term and wide ranging applications Upgrading of remote systems installed and simulation of remote operations have been given importance in research and development activities

In the Russian Federation, remote technology is used in RBMK-1000 spent fuel management at the NPP site including spent fuel transfer and handling operations at at-reactor (AR) and away-from-reactor (AFR) reactor site (RS) facilities

The concept of final conditioning of spent LWR fuel in Sweden for disposal in hard rock makes use of remote handling in particular for the different steps in the encapsulation process within the copper canisters

The front end of the commercial reprocessing plant in the UK, including preparation for reprocessing uses remote handling Fuel design considerations for shearing/handling of advanced and higher burnup fuel are ongoing developments in these areas The technologies employed in the principal front end reprocessing plants at Sellafield are mature It is anticipated that advanced and high burnup fuel can be accommodated by the existing remote technologies The demands from other nuclear process areas such as decommissioning and plant cleanout are setting the pace of technology development from which the front end processes will benefit

The United States Department of Energy recently completed a topical safety analysis report outlining the design and operation of a centralized interim storage facility for commercial spent nuclear fuel Remote operation was required to maintain sufficiently low radiation doses, and robotic or telerobotic equipment was identified as a desirable solution The design and operational dose analysis results are leading to this identification and the general factors to be considered when specifying automation and robotics

Further, the USA uses a new computer simulation tool, the Radiological Environment Modelling System (REMS) to quantify radiation doses to humans working in radiological environments REMS produces more accurate dose estimates than most previous methods

3 CONSIDERATIONS IN PROCESS SELECTION

In the discussion it was clear that every application of remote technology is unique and in most cases it is not possible to copy an entire solution However, there are elements from existing solutions that might be used to compose new applications which must be tailored with regard to the special conditions

Once the task has been clearly defined and the main design requirements have been set, some important considerations to be taken into account when designing the process are described in the following paragraphs The listed considerations are dependent on each other and are not exclusive

Neither are they listed in any priority order Examples of how this process is applied can be found in the papers presented in this document

Remote versus hands-on operation

In nuclear technology the radioactivity of the items handled usually determines whether remote technology needs to be employed In accordance with the ALARA principle, remote technology is encouraged in order to reduce the collective radiation doses to the operating staff There may be factors such as low radiation level, low frequency, or ready access (by the operator) which can make hands-on operation the preferred option

Location of process

Where processes must be designed into existing facilities, many constraints are placed on the design Such constraints include available space, process interfaces, limitations on changing the existing process and operating philosophy If a new facility is being designed, it is important to minimize the volume of such a facility in order to minimize costs and environmental impact It is important, however, to also consider flexibility for future uses and to provide adequate space for maintenance

Throughput and capacity

Two of the most influential factors in process selection are throughput and capacity These govern the size and overall cost of the solution High throughput requirements can justify higher investment A process designed for higher throughput generally requires logistically optimized solutions with high reliability and provision of buffer stores

Quality

The quality to be achieved in the process is fundamental High quality is often achieved through a high degree of automation Another aspect is the way in which the desired quality is to be achieved The traditional technique of final testing requires a different approach than more modern techniques such as statistical process control

Lifetime

The lifetime of the process equipment to be considered is of major influence to the design Process equipment for long term use must fulfil different design criteria than those for short term use since design lifetime often dictates materials of construction, safety margins and choice of components The present experience is that facilities often operate longer than the original design lifetime

Lead time

Whatever the lead time is, the process equipment must fulfil all necessary design and safety criteria However, if the available time to prepare the process equipment is very short, less optimized solutions might have to be adopted If the lead time allows for more detailed design, better optimized equipment with better technical performance and economy can be provided

Licensing and safety

Licensing requirements are one of the major influencing design parameters The safety requirements differ from country to country, thus sometimes preventing direct transfer of equipment or processes already in operation in another country However, a positive licensing statement in one country may be used to support licensing elsewhere

Experience

Experience is a very valuable factor when designing a remote technology process It is usually gained through the process of designing, building and operating facilities Design guides published by the IAEA, ANS BNES, etc are also useful references Although new technology may seem beneficial proven technology is sometimes chosen to provide a higher degree of confidence Choice of solutions can be influenced by a regulator who may wish to see a working example of the process involved

Flexibility

When designing a process it is important to build in flexibility for current and anticipated future needs The degree of flexibility depends entirely upon the application and in certain cases for very specific tasks, however, it may not be necessary to take flexibility into account

Maintainability

Each process should be designed with specific consideration of maintainability Even processes which are essentially maintenance free should have provision made for possible failure If such provisions are made then the impact of failures on operational cost and plant outage are reduced A wide variety of remote technologies are available for specific maintenance operations

Simplicity

Simplicity of a process is often an advantage since it usually results in fewer failures of the process, lower capital cost and easier maintenance

Manual versus automatic operation

There are many factors that determine whether a manual or automated process is the optimum solution Manual operations are less desirable where high throughput and/or high radiation protection are required Automation provides enhanced productivity and can realize higher quality standards if required, but requires more complex machinery Flexible automation can rapidly respond to low volume, high radiation protection requirements The higher capital investment must be justified e g by less dose exposure and/or by higher process efficiency

Safeguards

It is of great advantage to incorporate safeguard issues at an early stage of design since this can avoid costly rework

Cost

Lifetime costs (including consideration of decommissioning and waste processing) should be the basis of cost comparisons rather than initial capital outlay alone A preferred solution may not always be possible to realize due to financial constraints on the project

4 TRENDS FORESEEN

This section reviews the trends foreseen in remote technologies for spent fuel handling, including automation, robotics, simulation, and expert systems It also discusses issues related to quality

Automation

The use of automation for most operations in the handling of spent fuel is widely spread throughout the nuclear industry world-wide Facilities, relying on automated operation of systems for spent fuel handling have been used for decades with good records of reliability, thus increasing the confidence in process automation These include

- at-reactor pools,

- the interim centralized storage facilities, e g the modular dry vault storage (MDVS) at Paks in Hungary, the Wylfa facility in the United Kingdom, and the CASCAD facility in France,
- the front end facilities of the reprocessing plants at Sellafield (United Kingdom) La Hague (France), and Mayak (Russian Federation)

Further automation of those systems may be foreseen as higher burnup and MOX spent fuel are arising, and as the regulatory limits for operator exposure become more and more stringent A good illustration of this tendency is the automation of the unbolting of the transportation cask lid that is currently manually performed at some facilities For example, the fuel receiving facility of the new reprocessing plant in Sellafield (THORP) is equipped with an automated system for unbolting the lid of transportation casks

With the increased reliability of automated systems, additional automation in the facilities for spent fuel handling may be considered for the improvements of activities such as quality assurance (QA) i e establishing and maintaining QA records, and material accounting Further applications of this technology may also be foreseen as the standardization and availability of automated systems develop

Future developments and improvements in the automation of processes will benefit from the continuous development of computers Eventually the work being conducted on sensors (improving sensitivity and resistance to radiation) and interfaces with programmable controllers (improving the transmission of information through the use of fibre optic) should increase the performance and the reliability of the systems

Robotics

Emerging trends in facility operation include the use of flexible automation such as robotics Flexibility, which includes multi-tasking and effective use of available space become strong arguments for using robotic and telerobotic devices Examples where such devices have an advantage include

- facilities where several cask and canister designs must be managed, such as will be the case at the CISF (Centralised Interim Storage Facility) in the United States of America,
- facilities where different designs of fuel assemblies are handled, dismantled, reconstituted or refabricated, such as the ISFSF (Interim Spent Fuel Storage Facility) construction project in the Republic of Korea,
- facilities with stringent restrictions on operational volumes, such as at BARC (BHABHA Atomic Research Centre) in India

However, we have to recognize factors which have been hindering the implementation of robotics in the nuclear industry Lack of relevant nuclear application experience and exposure to the technology appears to be the primary issue Greater acceptance will follow demonstrations of reliability and effectiveness of robotic and telerobotic systems

Simulation

Simulation is becoming more wide spread for validation and acceptance of process and facility design, and during the meeting, examples of simulation were shown from Sweden, the Russian Federation and the USA

Graphical simulation has been used to analyse costs for a central interim storage facility as reported in the 1994 IAEA-TECDOC-842 Here processes were simulated using models of actual equipment executing the required operations This resulted in high-confidence estimates of capital and operational costs, as well as throughput A similar type of simulation is reported in the 1997 US contribution, where human processes can be graphically modelled and verified, followed by the tracking of radiation exposure from all included sources This type of simulation results in a more precise tracking of exposure to all regulated points

With better dose and cost estimates, together with the visualization capabilities, simulation is expected to enhance communications with regulatory authorities, project sponsors and the public Further potential benefits include visual training for operators, and the ability to operate equipment directly from the simulation environment (thereby improving operational transparency)

Expert Systems

Expert systems are not currently used in spent fuel handling However, there are other applications in the nuclear industry (examples are presented in the papers from India and the Republic of Korea) Experience gained in these areas will give the confidence to apply expert systems to spent fuel handling

Quality Assurance Issues

As the design and implementation of new systems and equipment continues to evolve, the role of QA remains vital The participants of the AGM emphasized the following two aspects concerning the QA related to remote automated technology application in spent fuel management

- Process qualification for computer aided operations can provide a potential benefit for QA In-process control increases the reliability thus diminishing the requirements for end-product or status inspection, especially in areas difficult to access,
- Computer software qualification is a new challenging area of QA To assure acceptance of programmable systems by regulatory agencies, appropriate procedures for software development and testing must be further developed

5 CONCLUSIONS AND RECOMMENDATIONS

During the meeting the participants agreed upon the following conclusions and recommendations concerning the application of remote technology in spent fuel management

(1) When designing a process for remote technology in spent fuel management there are several important considerations Decisions relative to the selection of the desired level of technology should be made on a case by case basis, while considering all relevant aspects

(2) Automation is considered to be very well established Nevertheless, further development is expected for currently automated processes and for spent fuel handling operations not yet automated In addition, use of robotic equipment can be identified as a desirable spent fuel handling solution that is capable of providing flexibility with the possibility of high throughput Application of expert systems may provide further improvements in the future

(3) As the remote technology develops in general off-the-shelf products and systems will be adapted from other industries In addition improvements and spin-offs from spent fuel management applications may go back to other industries and other nuclear applications, such as

decontamination, decommissioning and waste management This will promote more attractive and wide-spread use of automated and robotic systems

(4) Design criteria of automated spent fuel handling processes should be taken into consideration during the design stage of any new fuel type

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AUTOMATION: A KEY TECHNOLOGY TO SAFE AND RELIABLE SPENT NUCLEAR FUEL HANDLING IN HIGH THROUGHPUT PLANTS

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Abstract

La Hague 30 year experience with nuclear spent fuel handling represents more than 48,000 assemblies handled in wet and dry environments. The front end facilities of the UP2-800 and UP3 reprocessing plants are dedicated to spent fuel handling, e.g. fuel unloading, interim storage, dispatch and measurement. The operations, including maintenance, are largely automated and are performed remotely from central control rooms. The use of automation at La Hague is aimed at reducing personnel exposure, increasing the purposeful utilization of equipment, increasing the reliability of operations and thus the safety of the facilities, and improving fuel accountability. The automation of the plants was designed to maintain a high achievable availability and flexibility of the facilities. Today, La Hague reprocessing plants have successfully reached their design capacity and handle fuel from utilities all over the world with a wide range of types and burnup. The future developments include a decision support system for operators.

1. INTRODUCTION

To date, with more than 45,000 fuel assemblies handled in the plants of the La Hague site, the experience with nuclear spent fuel handling represents approximately that of one thousand nuclear power plants during ten years of operation.

In the beginning of the UP2 reprocessing plant operation, as the throughput of arriving transportation casks was far below today's one, spent fuel handling operations in the HAO facility were performed locally with few remote operations. Today, about 1,500 tU of LWR spent fuel is annually transported from French and foreign nuclear power plants to the La Hague UP2-800 and UP3 reprocessing plants. The front end facilities of the plants are dedicated to spent fuel handling (unloading, checking, interim storage, dispatch and measurement). They are largely automated facilities, with a centralized control system.

At La Hague, the use automation to a great extent is aimed at increasing the purposeful utilization of equipment, increasing the reliability of operations and thus the safety of the facilities, reducing operator doses, and improving traceability. The major technical challenges faced by the designers were to maintain a high achievable availability and flexibility of the facilities.

This paper introduces the front end facilities of the UP2-800 and UP3 plants where operations for spent fuel handling are performed, describes the centralized control system and eventually concludes on future developments foreseen.

2. DESCRIPTION OF SPENT FUEL HANDLING AT LA HAGUE

The front end facilities of the UP2-800 and UP3 plants are dedicated to spent fuel handling (Fig. 1). Fuel unloading, checking and dispatching to the storage pools are performed in the NPH and T0 facilities. After storage, the fuel is transferred to the reprocessing units. Tunnels connect the NPH facility and the D pool to the R1 and T1 facilities where the fuel is cut and dissolved. Prior to cutting the burnup of each fuel assembly is measured and compared against the value announced by the nuclear power plant.

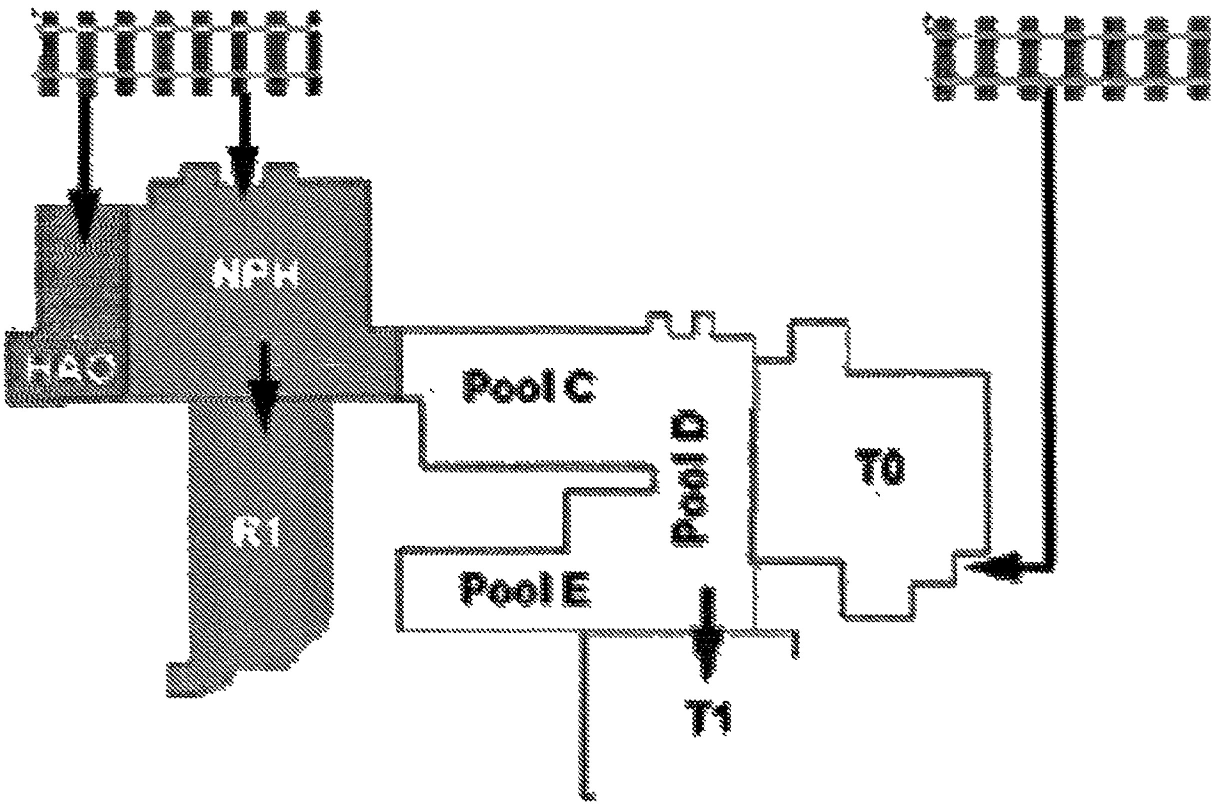

FIG 1 La Hague spent fuel handling facilities

2.1. NPH Facility: wet unloading

The NPH facility is part of the UP2 reprocessing plant in operation since 1970. It comprises a fuel unloading pool (Fig 2) and a fuel storage pool. It is connected to the other storage pools of the storage complex of the La Hague site which includes three other pools NPH also feeds the R1 facility.

The facility was commissioned in 1980. It has been originally designed with very few remote operations The cask preparation was mainly performed locally, with direct contact to the cask, and operators were conducting the fuel unloading operations while they were boarded on the cask unloading crane.

In 1989, the UP2 plant was upgraded. New facilities were built and others modified to bring the nominal capacity of the plant to 800 tU which was achieved in 1996. The programme for the new UP2 plant included modifications to the NPH facility, that were undertaken starting 1991. The objectives, among others, were to improve the reliability of cask handling equipment, operating conditions and to reduce the doses to the operators.

The automation of the fuel unloading operations (through the use of remote technologies) was chosen as one of the key upgrade to achieve these goals. A major challenge for the automation of NPH was to maintain the flexibility of the installation that is planned to receive any type of casks.

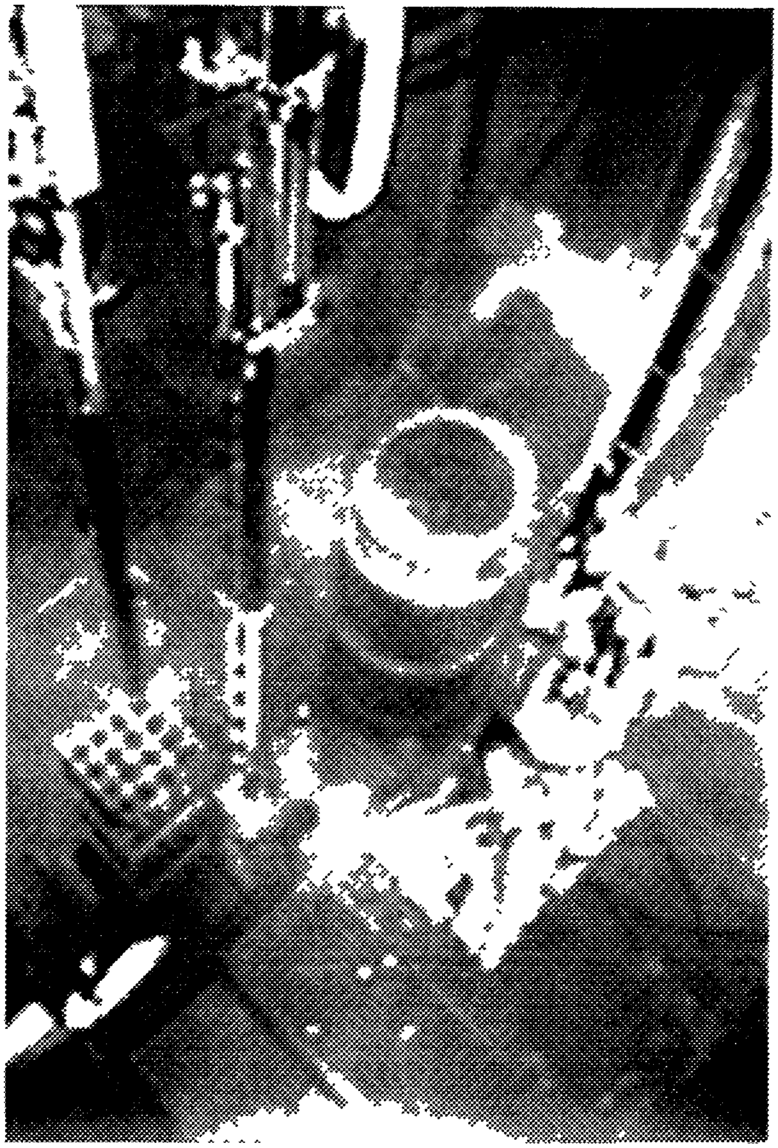

FIG 2 NPH Facility wet fuel unloading

Since 1994, the wet unloading operations at NPH are performed remotely from a control room Although the preparation of the casks (mainly cask connection to the process unit) is still performed locally, in a dedicated cell, semi-automatic sequences have been defined for the transfer and immersion of the cask, thus reducing the time required for the operations and consequently the operator exposure Once the cask is immersed in the pool, all fuel handling operations are performed from the control room

Due to the large variety of fuels received in the NPH facility, a wide range of automatic sequences have been developed to maintain the installation flexibility

2.2. T0 Facility: dry unloading

The T0 facility is part of the UP3 reprocessing plant. It operates since 1986 and comprises one cell for fuel dry unloading (Fig. 3). It is connected to the storage pool complex of the site through the D pool which also feeds the T1 facility.

The benefits of dry fuel handling lie in the diminution of operator exposure and the significant reduction of liquid and solid waste production.

The facility yearly unloads around 200 transportation casks, i.e. one cask a day. All operations are remotely performed and supervised from a central control room. Currently, four types of casks can be received in the installation: TN12, TN13, TN17 and LK100.

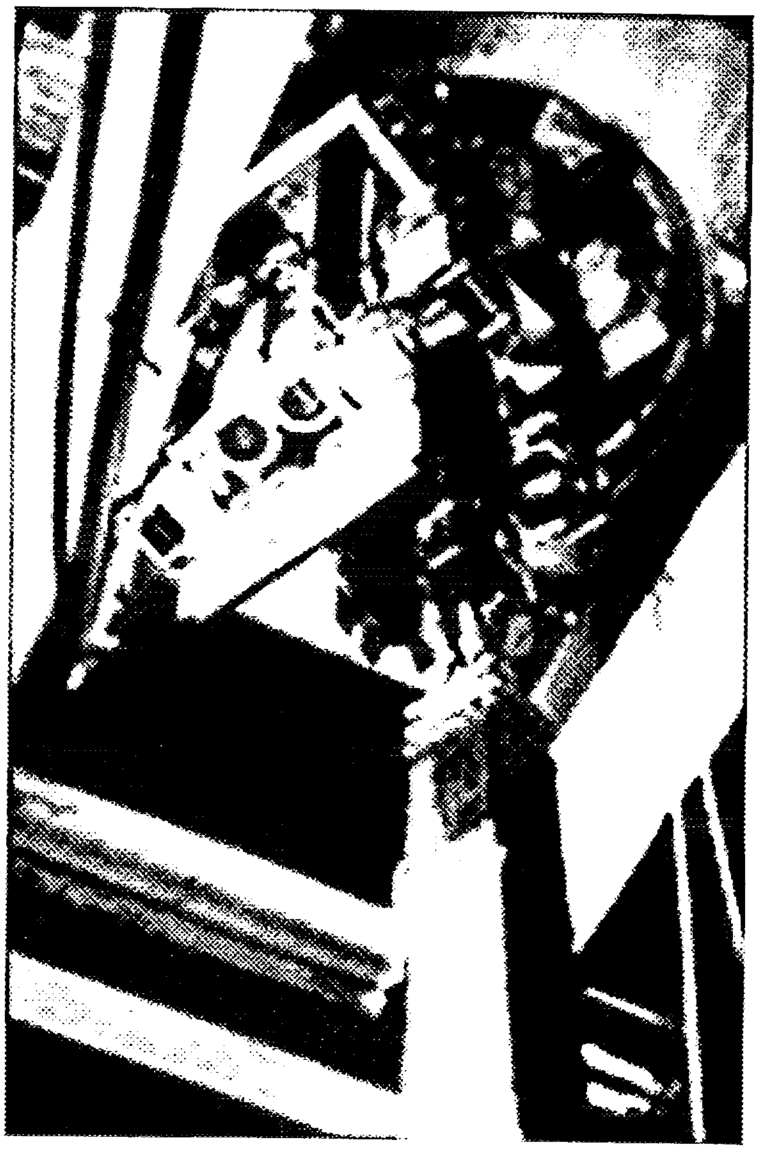

FIG. 3. T0 Facility: dry fuel unloading

2.3. Storage pools: spent fuel dispatch

In the spent fuel reception and storage complex of the La Hague site, fuel assemblies are handled in baskets: one basket for BWR fuel contains 12 assemblies, one basket for PWR fuel contains four assemblies.

The baskets are loaded in the NPH and T0 facilities and are moved to their storage position by the mean of an automated crane supervised from the control room (Fig. 4). For transfer to the R1 and T1 process facilities, the baskets are placed on a platform which moves along a ramp.

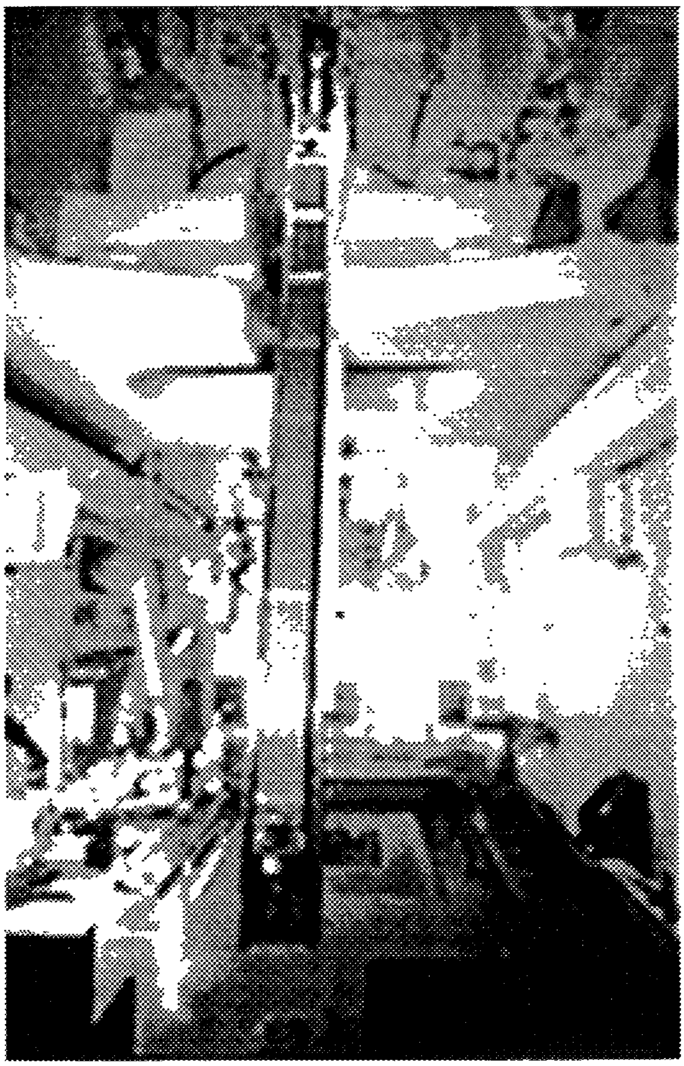

FIG. 4. Spent fuel handling crane

2.4. R1, T1: fuel assembly tipper bridge

In the R1 and T1 facilities, the fuel assemblies are retrieved from the baskets by the mean of a tipper bridge (Fig. 5) and introduced in a dry cell where the fuel burnup is measured.

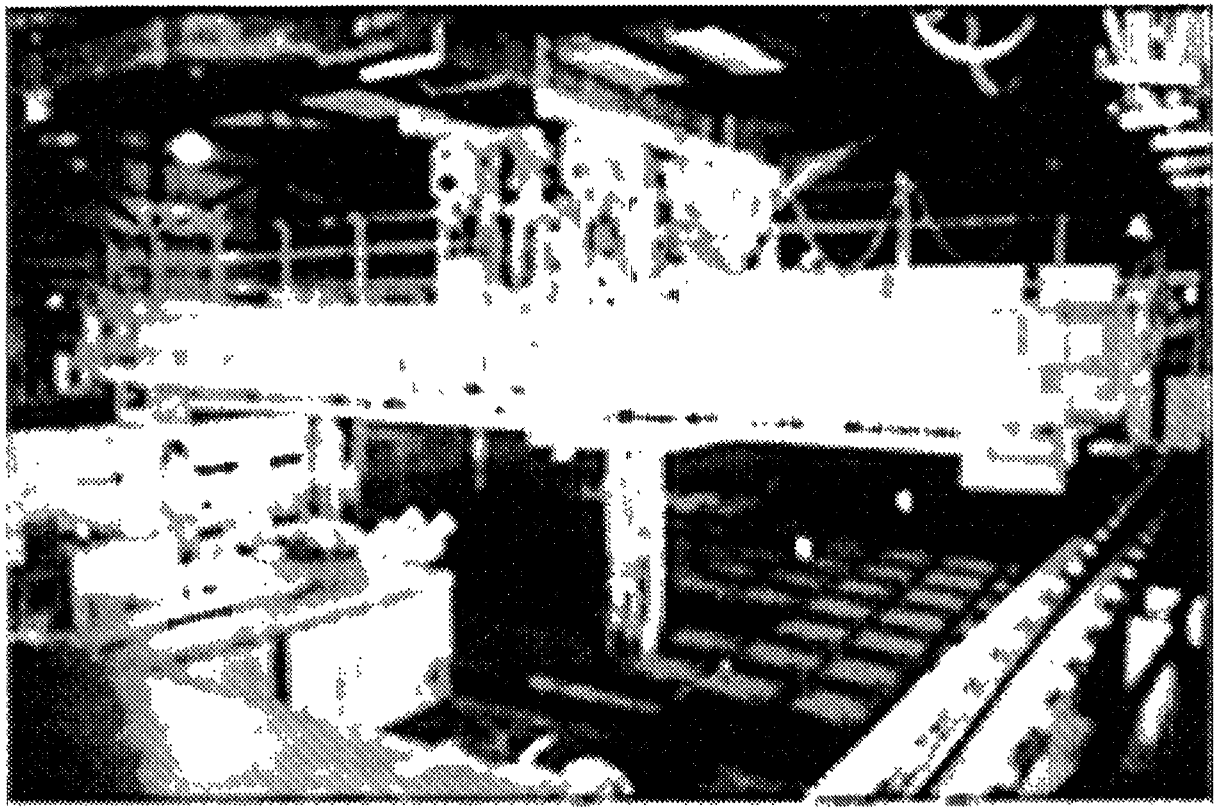

FIG. 5. T1 Facility: tipper bridge

3. LA HAGUE CENTRALISED CONTROL SYSTEM FOR SPENT FUEL HANDLING

Most systems described above are automated. Operations for spent fuel handling, including maintenance, are performed remotely and controlled from centralized control rooms according to the diagram Fig. 6.

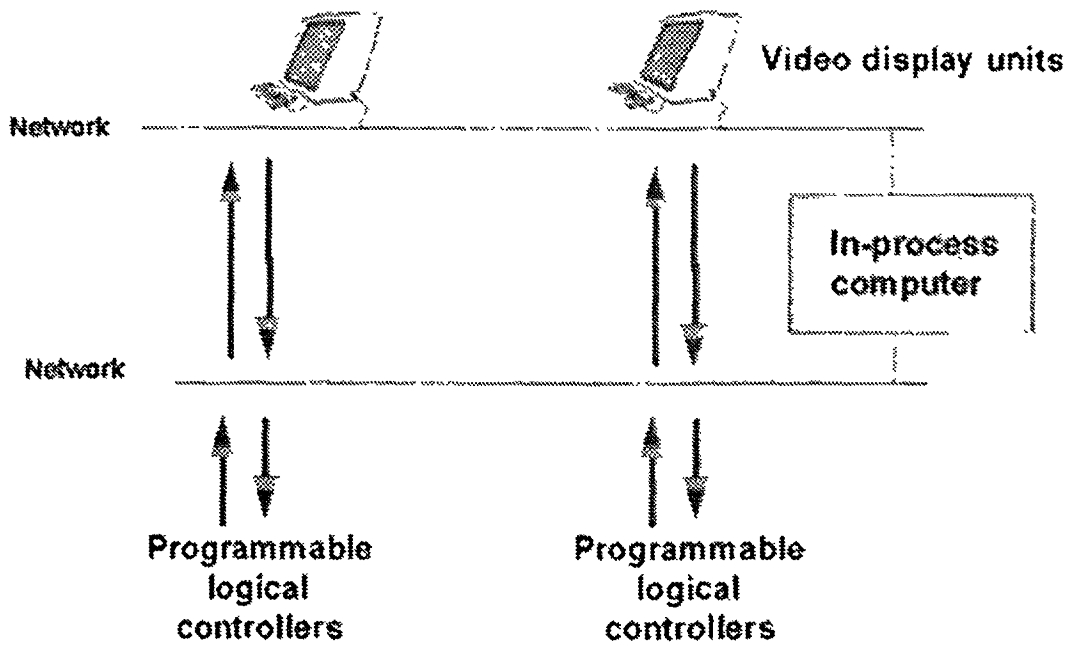

FIG. 6. Operating La Hague reprocessing plants

Data acquisition and control of the process are performed locally in each facility by the mean of process control systems (multi-loop control units and programmable logical controllers).

The main in-process management systems collect information from the process control systems and transfer them to the video display units in the control room. Thus, information available to the operator on the video display unit include information for process operation, facility surveillance, e.g. radiation control, criticality control, fire protection, and monitoring of personnel access to controlled areas.

The video display units display animated objects illustrating the process equipment and alarms. Man-machine interfaces (networks) allow the operator to control the process through the control boards (Fig.7).

There are two operating modes:

- the automatic control mode. The operator engage the process cycle from its control board and the sequences are actuated automatically;

- the remote manual control mode. The operator has to validate from its video display unit the end of each sequence (verifying parameters) for the next sequence to be actuated.

Whatever the mode used, the safety of the process is ensured through the implementation in the system of locked « permanent safety conditions » (CPS) which must be verified for the sequences to be resumed.

The surveillance and maintenance of the software is ensured by the software maintenance center (CML) which provides for software control and management of software updates and upgrades It also includes support programs for modifications and tests

In case of the unavailability of the normal operating system, an additional control mode is provided called « the safety mode » which uses a dedicated hard wired control panel

FIG 7 UP3 Control Room

4 FUTURE DEVELOPMENTS AND CONCLUSION

Thanks to a conception allowing a high achievable availability and flexibility of the facilities, the La Hague reprocessing plants have successfully reached their design capacity and handle fuel from utilities all over the world with a wide range of types and burnup

The plants are operated and maintained using automation and a centralized control of operation The automation of operations at COGEMA La Hague represents

- 150 video display units,
- 700 programmable logic controllers,
- 1200 multi-loop control units,
- 35 in-process computers

The future developments of the operating system include the development of a decision support system for the operators

XA9948977

EXAMPLES OF REMOTE HANDLING OF IRRADIATED FUEL ASSEMBLIES IN GERMANY

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Abstract:

Examples for the remote handling of irradiated fuel in Germany are presented in the following areas:

- fuel assembling pool service activities;
- early encapsulation of spent fuel in the pool of a nuclear power plant (NPP) at the end of the wet storage period.

All development in remote fuel assembly handling envisages minimization of the radioactive dose applied to the operating staff. In the service area a further key objective for applying advanced methods is to perform the work faster and at a higher quality standard. The early encapsulation is a new technology to provide the final packaging of spent fuel already in the pool of a NPP to ensure reliable handling for all further back end processes.

1. INTRODUCTION

Spent fuel assemblies (SFAs) need to be handled remotely in order to protect the operating personnel from the radioactive radiation originating from the fission products contained in the fuel. In the beginning of nuclear technology, remote handling occurred mostly manually from behind a shielding placed between the irradiated fuel and the operators. Shielding is provided either by water of the spent fuel (SF) pool or by the structure of the hot cells in which the irradiated fuel is handled. Handling of irradiated fuel has been improved considerably by applying computer controlled automation of the processes. This results not only in a more reliable performance of the handling processes, but also in much shorter process time and in a considerable reduction of the radioactive doses applied to the operating staff. In the paper examples of advanced remote fuel handling from the following two areas are presented:

- fuel services in the pool of a NPP;
- preparation of SF for long-term interim storage in the pool at reactor site.

2. EXAMPLES OF REMOTE HANDLING IN FUEL ASSEMBLY SERVICES

2.1. The MULTI-INSPECTION system

The MULTI-INSPECTION system has been developed by Siemens to minimize time and expense of the visual inspections of fuel and control assemblies required during the course of a refueling outage. In the past, these components were held in the refueling machine for the duration of the inspection and inspected in a temporally installed system. With the new system, the components are placed in receptacles located in the SF pool so that the refueling machine is only required for a short time to position components.

While the inspections are being conducted, the refueling machine can be used to insert and remove SFAs, to perform a mast sipping test or to shuffle control assemblies and flow restrictors. This cuts the time required for refueling activities by as much as three days, as was confirmed during two refueling outages at the Gösgen PWR plant in Switzerland, in which this kind of inspection system was first installed (Figures 1 and 2).

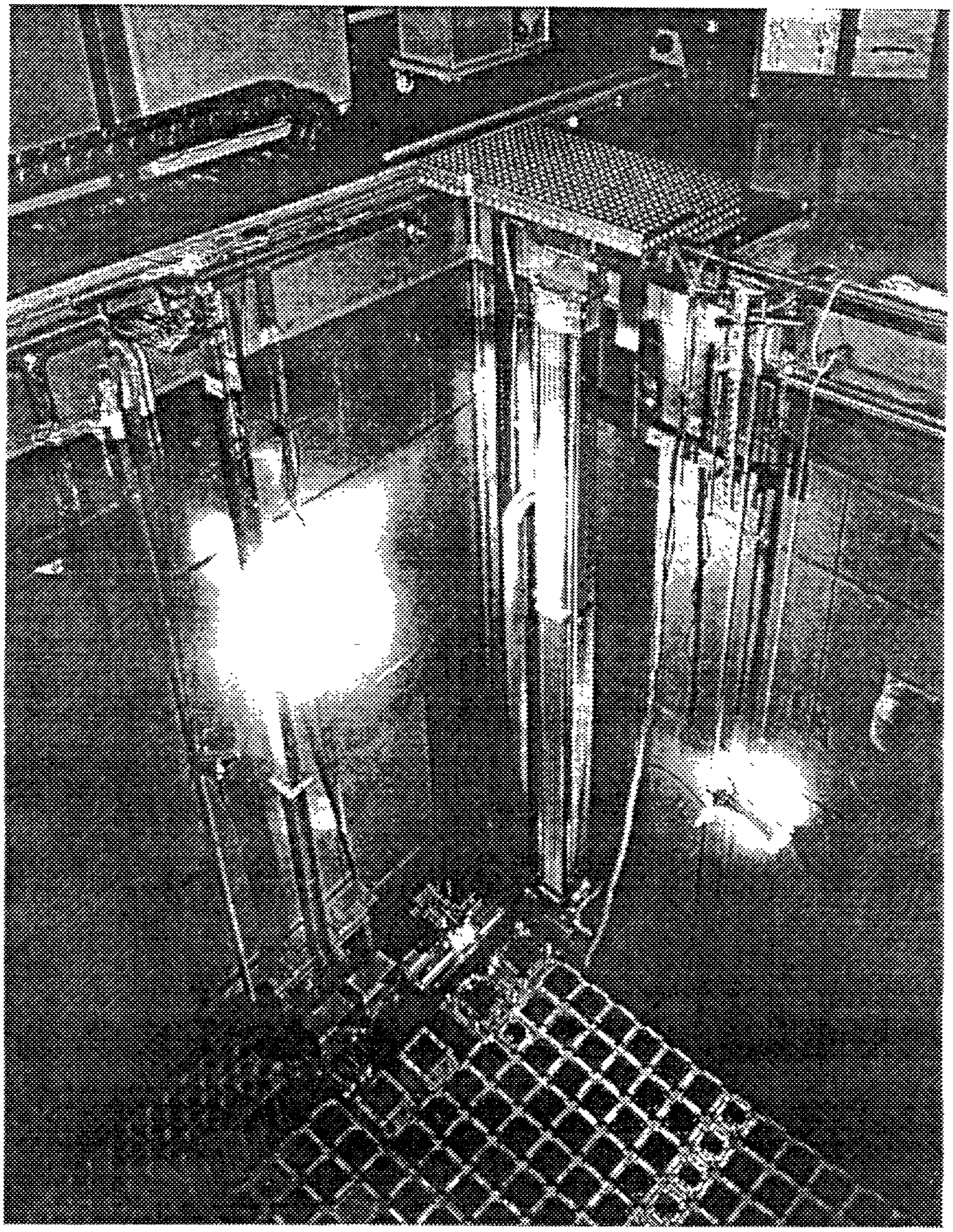

FIG. 1. MULTI-INSPECTION at Gösgen

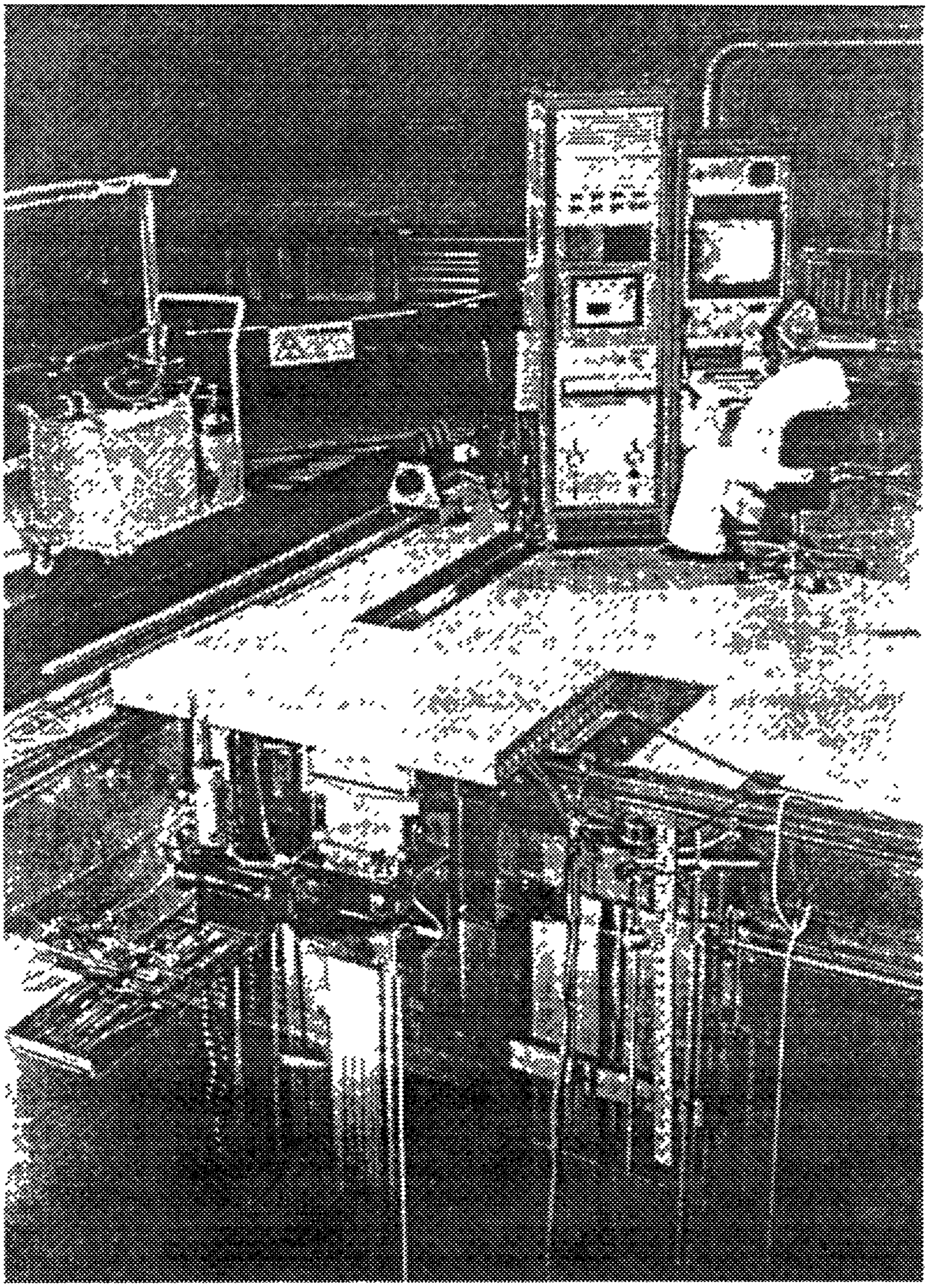

FIG 2 Visual inspection systems

The MULTI-INSPECTION system installed at the Gösgen plant consists of an inspection manipulator, a control cabinet and two fuel assembly holding boxes. Each box is freely accessible from one side, and is open for visual inspection. The SFAs or control assemblies placed into the boxes can be individually rotated to allow inspection from all sides. The compact, portable control cabinet is used to control the overall system from the edge of the fuel pool. The inspection manipulator is fitted with one or two underwater video cameras which are mounted on a co-ordinate-controlled traveling table which in turn is mounted on a mast-guided carriage. This arrangement allows the cameras to traverse the entire length of the fuel assembly, including the top and bottom end pieces. The carriage can also be raised to the top edge of the fuel pool, where the cameras can be removed and replaced above the surface of the water.

The inspection manipulator can be used for a wide variety of applications by simply attaching the appropriate equipment modules. For example, the system permits optical measurement of bowing and dimensional changes. In addition to the timesaving semi-automatic measurement of the oxide layer thickness on peripheral fuel rods, the same measurements can be made on all interior fuel rods at any height using the INOXIS system.

The MULTI-INSPECTION system is available in several different versions, so it can be tailored to meet the requirements of individual power plants. The fundamental principle of operation, however remains the same. At the Grafenrheinfeld PWR plant in Germany, for example, the new fuel elevator has been appropriately modified. A different system design has been implemented at the Philippsburg 2 PWR plant in Germany, in which the dual inspection boxes are hung in backpack fashion on a fuel storage rack.

2.2. The AUTOMATIC CO-ORDINATE-CONTROLLED CARRIAGE

Siemens uses an AUTOMATIC CO-ORDINATE-CONTROLLED CARRIAGE (Fig. 3) to perform fuel reconstitution when testing of the integrity of cladding tubes is also required. This new device is a further development of the carriage already used for transferring fuel rods to new fuel cages. It is integrated into the existing reconstitution device, and makes fully automated testing of the integrity of the cladding tubes of all the fuel rods in a fuel assembly possible. The fuel rods are grabbed as before by the fuel rod exchange device and examined using the eddy-current method as they are withdrawn. Friction force is also measured during this process. Fuel rods which are within permissible tolerances are reinserted, and defective ones replaced.

Using the new co-ordinate-controlled carriage (Fig. 3), the fuel rod inspection for one fuel assembly takes only about ten hours, i.e., three service personnel shifts per fuel assembly less than needed with earlier inspection methods.

3. EARLY ENCAPSULATION

The encapsulation technology provides the possibility to encapsulate both PWR and BWR SFAs at the end of the wet storage period. The encapsulation process takes place in the SFA storage pools at reactor site. It is based on well approved service technologies in all individual process steps. Among other consideration there is the advantage that the final packing of the SF occurs in using reliably performing processes conducted by those engineers which are familiar with the design, fabrication and operation performance of the fuel (Fig. 4).

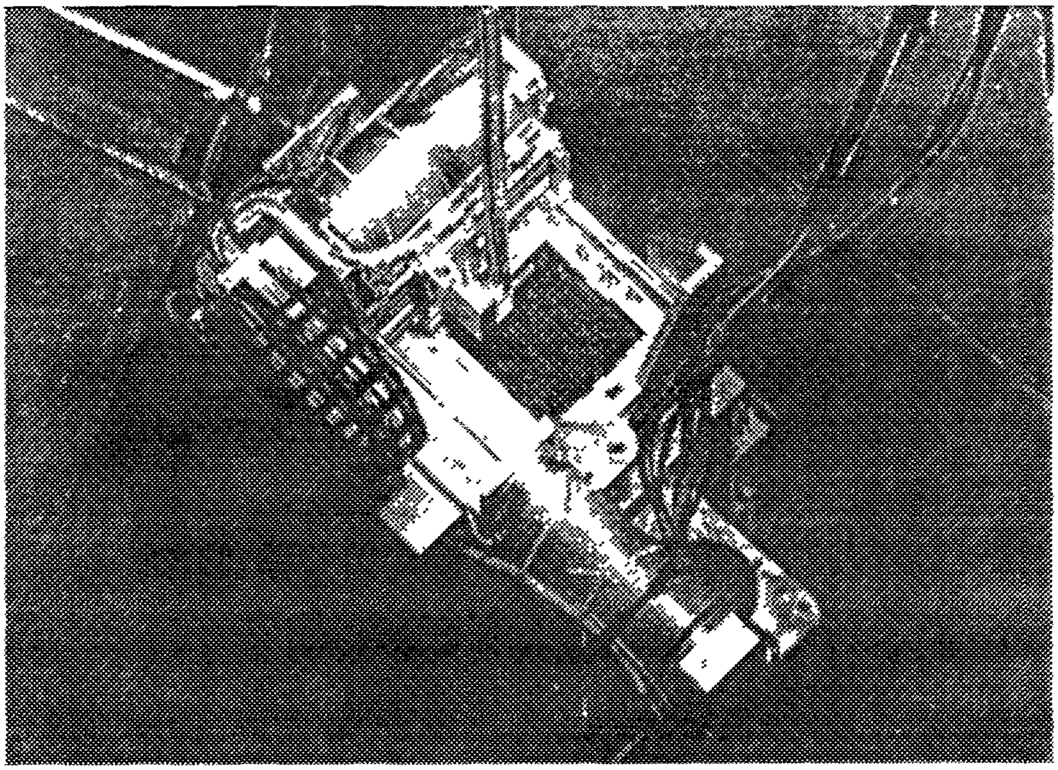

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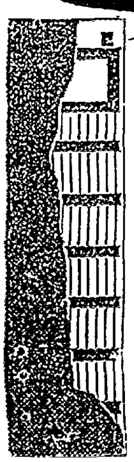
FIG 3 The Siemens co-ordinate-controlled carriage (shown in operation on top of a SFA)

no longer extended ARS pool storage, extremely long interim storage AFR and complex interfacing between front end & back end of the fuel cycle even for FA with increased burn up



Each spent FA is loaded into a capsule. The capsule interior is dried and made inert, and the capsule is seal welded. The encapsulation is performed just prior to dual porpose cask loading as a sevice

- *the capsule assumes the barrier function instead of the cladding*
  - → *the FA EOL condition plays no longer a major part if the cask loading is performed*
- *the capsule is designed to take all loads from handling processes and in storage*
  - → *the capsule minimizes the number of interfaces between front end and back end and allows for seperate optimization*
- *the encapsulation provides an early conditioning of the spent FA for the repository*
  - → *adequately designed SS capsules provides a better long term storage performance than a high burn up FA in its EOL condition*
- *individually encapsulated spent FA provides the smallest decay heat per package*
  - → *this is the only chance to limit the the interim storage for U-MOX-FA even with higher burn up to < 100a for all kind of geological rock formations discusset for the final repository*

FIG 4 Early encapsulation creates benefits throughout the entire backend of the nuclear fuel cycle

3.1. The equipment

The encapsulation process occurs remotely and will have the capacity of several SFAs per day. The early encapsulation makes use in all subsequent steps of the back end of the fuel cycle from the available and approved technologies like transportation and storage casks and final disposal techniques. Only the cages in the SFA-casks needs some slight modifications. Fig. 5 exhibits the back end of the fuel cycle when early encapsulation is selected for the final packaging of the SF already in the SF pool at reactor site.

The transportable encapsulation device (Fig. 6) consists of :

- the encapsulation station with 2 individual process modules for the encapsulation working in parallel;
- one joint service module which contains all necessary tooling and the welding device;
- 2 intermediate SFA storage positions for receiving the SFA from the storage racks and after completion of the encapsulation process for passing the encapsulated SFA for either further storage in the pool or for being immediately loaded into a waiting transport and storage cask;
- a first transportation cover to lower the empty capsule in a dry manner down into the encapsulation module;
- a second transportation cover with an integrated SFA drying device to transfer the SFA from the receiving position to the encapsulation module and to dry the SFA;
- a third transportation cover with an integrated He-leak-detector system to remove the encapsulated SFA from the encapsulation module, to provide necessary leak testing and to forward the encapsulated SFA to the second intermediate storage position.

3.2. The process

Fig. 7 describes the complete encapsulation process. For handling of heavy components the overhead crane in the containment provides the necessary support, the transportation of the SFA is carried out whether by the fuel manipulator crane or by the auxiliary hoist available at the fuel manipulator crane. The encapsulation process can be described by the following characteristic process steps:

1. On the floor aside of the SF storage pool the empty capsule is raised into a vertical position and filled with Argon (Fig. 8);
2. The Argon-filled capsule is transported with its cover lid in a dry manner down to the encapsulation module, connected to the locking device and lowered down to the encapsulation module (Fig. 9). The cover lid of the capsule is now removed into a waiting position;
3. Parallel to the process steps 1 through 3 a SFA is taken with the SFA handling machine from the storage racks and brought in the first intermediate storage position (Fig. 10);
4. The SFA is subsequently removed from the first intermediate storage position by the second transportation cover, docked on top of the encapsulation module which contains the empty capsule and is drained;
5. After drying the SFA to the specified value by heating it with electrical heaters and by subsequent vacuum the SFA is ready to be inserted into the encapsulation module;
6. The SFA is lowered down into the capsule. The cover lid of the capsule is taken back from its waiting position and positioned on top of the capsule now containing the SFA;
7. The capsule is now closed by an automated TIG-(tungsten inert gas) welding process. During the welding process a valve - contained in the cover lid of the capsule - is kept open and allows an pressure equalization between the capsule and the encapsulation module thus avoiding a material blow out in the very last moment when the weld tightens the capsule closure;

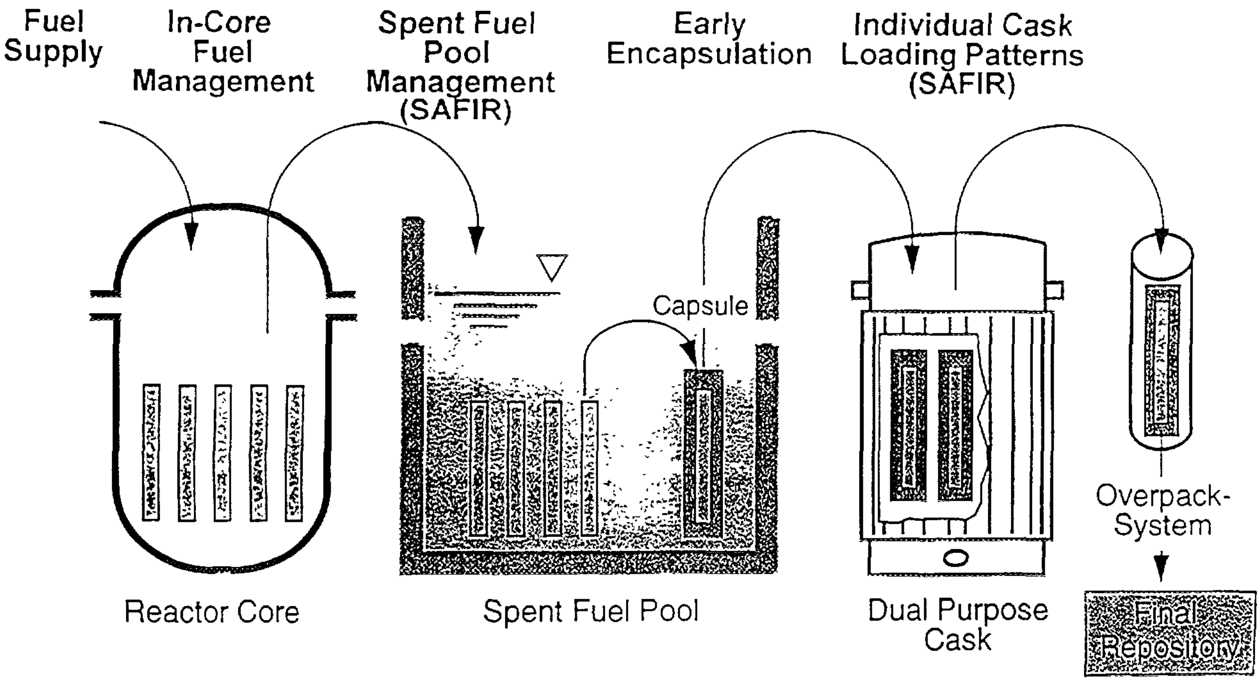
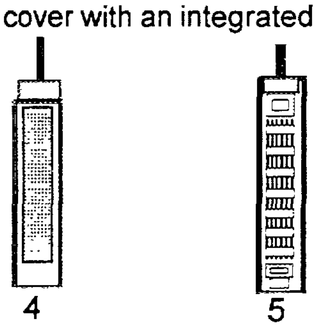


FIG. 5. Optimization of the entire process

The transportable encapsulation device consists of:

- the encapsulation station with 2 individual process modules for working in parralel **(1)**
- one joint service module which contains all automated tooling **(2)**
- 2 intermediate FA storage positions for receiving the FA from the storage rack and after completion of the encapsulation process for passing the encapsulated FA for storage or for loading to transport and storage casks **(3)**
- a first transfer cover to lower the empty capsulein a dry manner down to the encapsulation station **(4)**
- a second transfer cover with an integrated FA drying device **(5)**
- a third transfer cover with an integrated He-leak-test device **(6)**



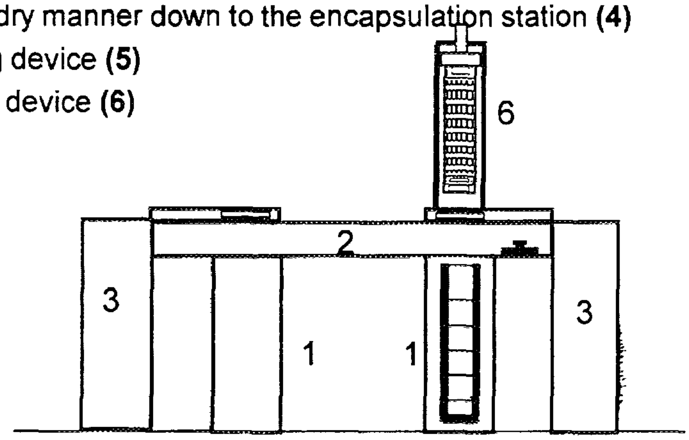


FIG. 6. Early encapsulation - the equipment

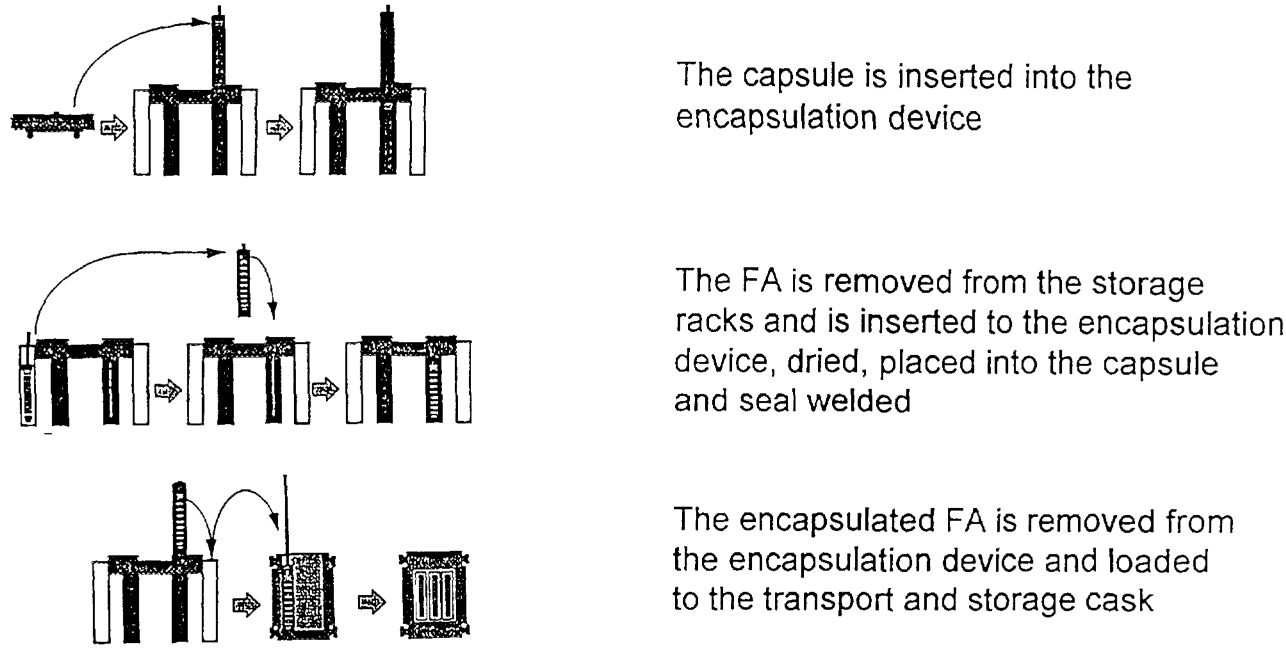


FIG. 7. Early encapsulation of a SFA occurs in a movable service equipment within the pool of a reactor

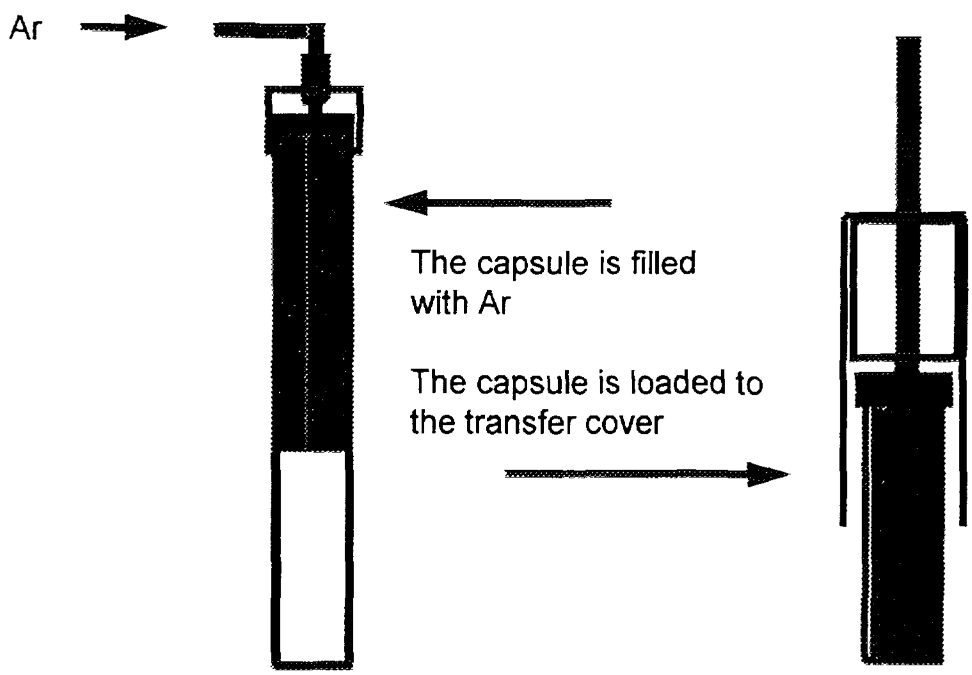


FIG. 8. Early encapsulation - the process

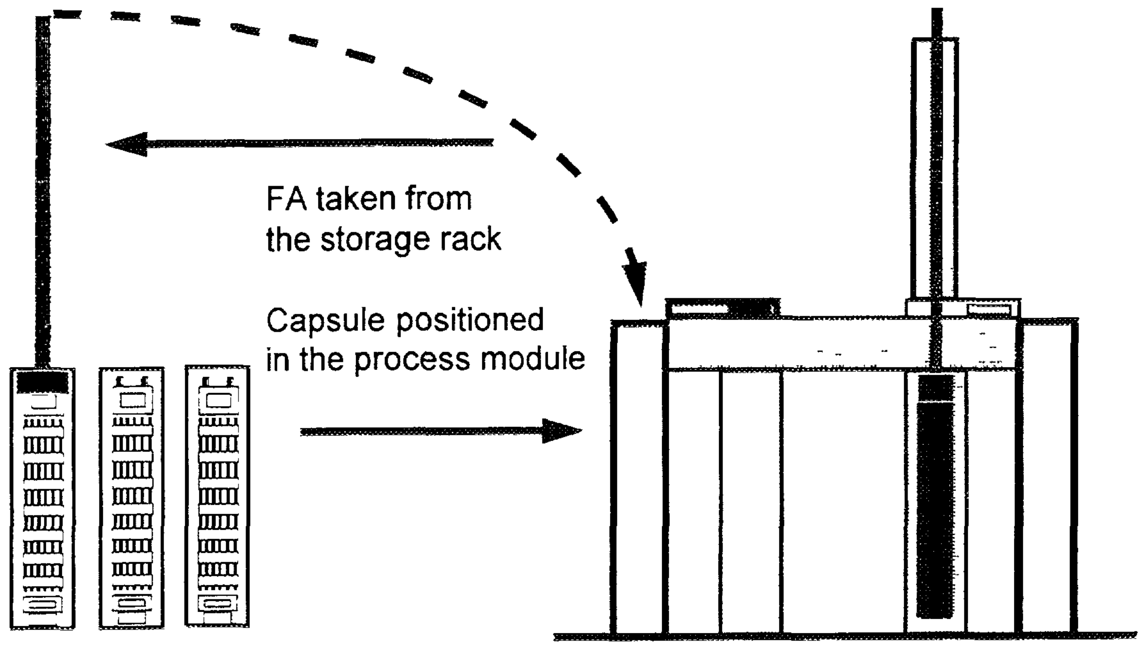


FIG 9 Early encapsulation - the process (continued)

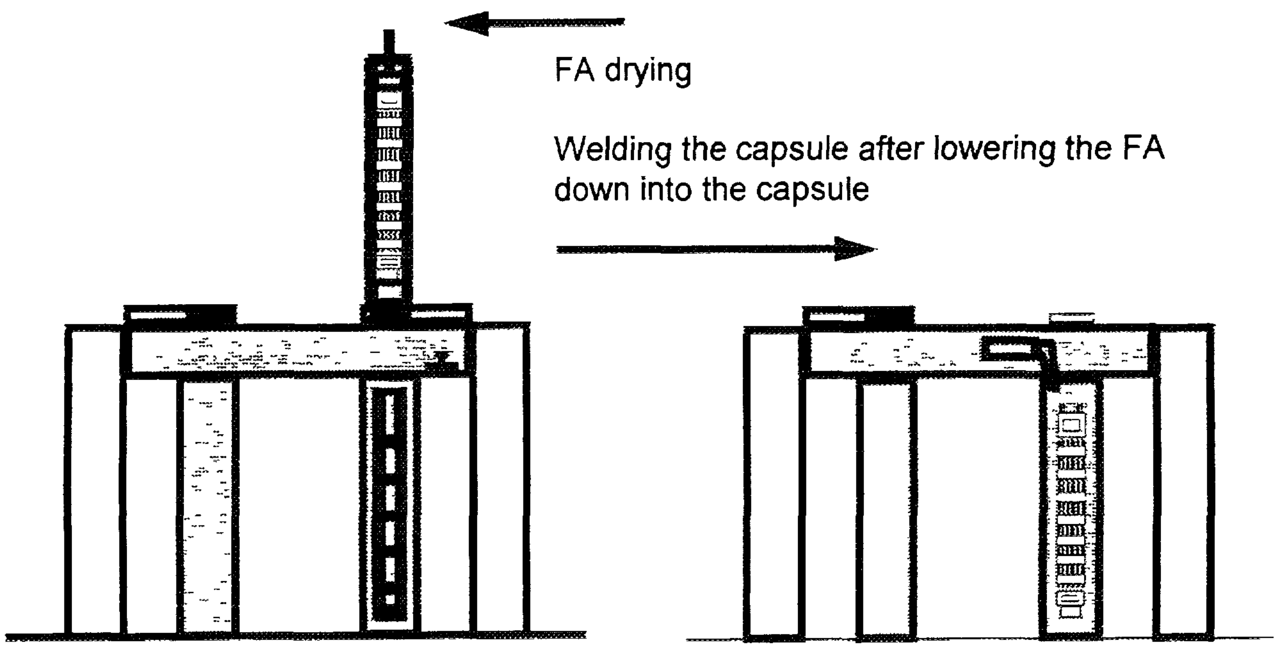


FIG 10. Early encapsulation - the process (continued)

8 After completion of the main welding the still open valve in the capsule cover lid is connected to a gas supply system providing the right capsule internal gas pressure by feeding Argon and Helium. The Helium is later on needed to perform the He-leak test. Finally this gas valve is closed mechanically and welded (Fig. 11),

9 The encapsulated SFA is now tied up into the third transportation cover with the integrated He-leak testing system. Having passed the leak tight test successfully the transportation cover is disconnected from the encapsulation module and brought to the second intermediate storage position from where it might be transferred to the waiting storage position or to the storage and transportation cask.

The encapsulation process will performed remotely and automated The „hands-off" operation mode in normal operation allows to minimize the doses applied the operating staff The process can be operated in the forward direction to encapsulate the SFA and in the backward direction to decapsulate the SFA. Decapsulation is possible from each intermediate process step and from the final process step to ensure that the SFA can be removed from the encapsulation station independently what has happened.

Each electrically driven machinery is designed redundantly or can be remotely replaced by a new device As a final measure each electrically driven motion can also be operated manually using long handling tools.

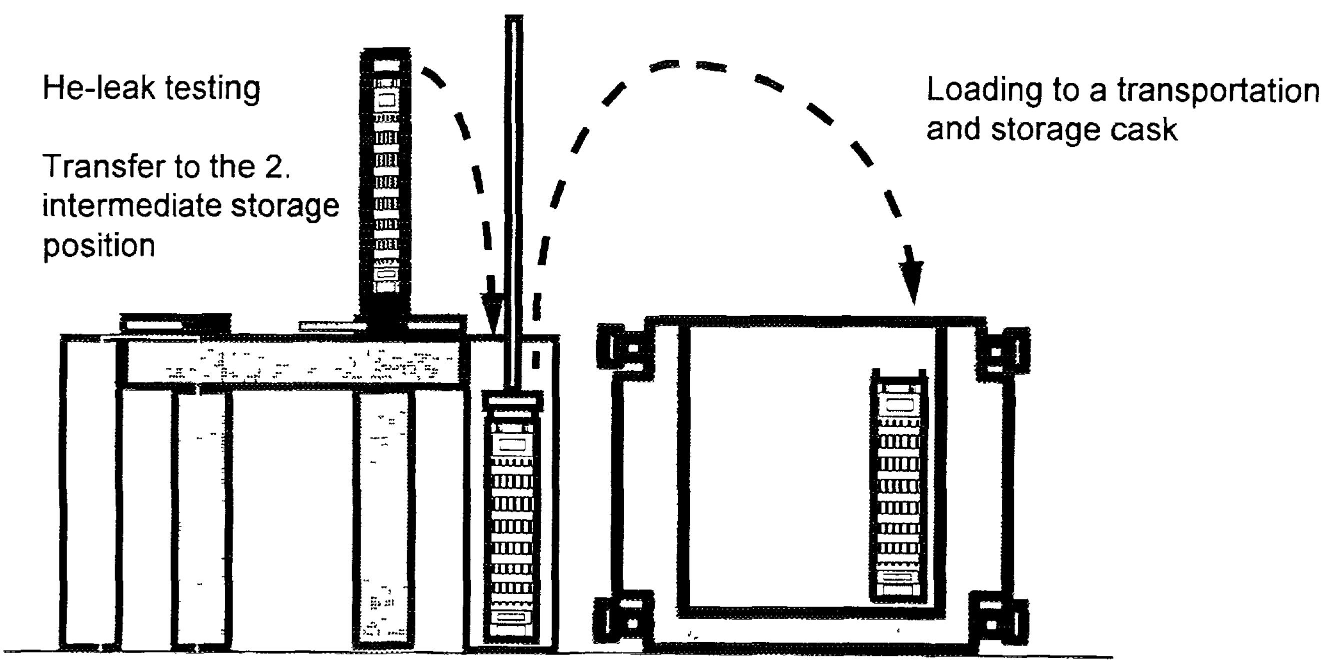


FIG 11 Early encapsulation - the process (continued)

3.3. Selected considerations of technology

Exemplary for all other process steps the following steps will be discussed in greater detail to illustrate how the available know how is integrated in the SFA encapsulation process:

- From the two possible welding processes - laser welding and TIG welding - the TIG-welding was selected because of its larger scope of experience under remote service conditions. TIG-welding can be performed with and without material supply. Experience shows that the material supply during remote welding might be troublesome therefor it was decided to weld without any additional material supply during welding. Argon will be used as the welding atmosphere. Normally some hydrogen is added to Argon for better focusing the welding arc. Since we see safety related questions together with the hydrogen addition we will perform the welding in pure Argon. Gas cooled welding heads instead of water cooled were selected to avoid problems with water in case of malfunction. Since the operation temperature of such welders is somewhat higher we had to envisage an increased wear of the W-cathode;
- With respect towards long interim storage periods the SFA drying process during the encapsulation is of great interest. When water rinsing is completed there are still water films adhering to all surfaces of the SFA. On the fuel rod surfaces those water films will be easily evaporated by the support of the decay heat. The removal of residual water in the lower ends of the guide does not occur so nicely since the heat of evaporation must be taken only from the heat capacity of the SFA structure. Without any additional heat supply from external sources this water may freeze during vacuum drying. Therefor our concept foresees that in first step heat is supplied to the SFA structure from a heating system integrated into the drying device. Vacuum drying starts if the cold spot in the structure is heated to > 90°C. A second heating/drying cycle assures that relay all water is reliably removed. This drying procedure was developed experimentally first. In a second step this procedure was modeled theoretically. Finally the design of the drying equipment was performed by using this experimentally verified design code.

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OVERVIEW OF REMOTE TECHNOLOGIES APPLIED TO RESEARCH REACTOR FUEL

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Abstract

This paper gives a brief overview of the remote technologies applied to research reactor fuels. Due to many reasons, the remote technology utilization to research reactor fuel is not so widespread as it is for power reactor fuels, however, the advantages of the application of such techniques are obvious.

1. INTRODUCTION

Since the first nuclear research reactor went critical in 1942, over 550 research reactors have been constructed world-wide and, of these, approximately 300 are currently in operation. Approximately 80% of the reactors currently in operation around the world are over 20 years old and 56 % are over 30 years old [1].

These research reactors were designed and constructed according to the industrial standard and practices in the country of origin at the time of construction.

The design, operation and utilization philosophy of nuclear research reactors is fundamentally different from that of nuclear power reactors. In addition, utilization may lead also to more frequent system modifications than for the power station systems, sometimes including even the reactor itself.

The same difference exists between the spent fuel manipulation systems and the equipment for power reactors and research reactors. This is due to different reasons, some of them are follows:

- Only some research reactors burn significant quantities of fuel each year;
- The frequency of the spent fuel manipulations is much less than it is for power reactors;
- The spent fuel assemblies used in research reactors, in general, are considerably smaller in size than the power reactors fuel assemblies;
- Application of the remote technology was not so developed and widespread at the time of construction of the majority of the research reactors, as it is today.

2. SPENT FUEL MANIPULATIONS IN RESEARCH REACTORS

Most of the research reactors began their operation in the early 60's and 70's. Spent fuel production is a much slower process than it is for the power reactors. So the built-in storage capacities were enough for many years of operation of the research reactors in the first periods. From some research reactors, after shorter or longer periods of storage, the spent fuel was transported for reprocessing.

In the last 10-15 years some modifications were carried out related to the fuel cycle, that resulted in increased research reactor spent fuel production. Two of the major contributing factors are:

- research reactor power upgrades for higher power capacities;
- enrichment reduction to LEU (less than 20 % of U-235 content).

Until 1988, the USA as the main supplier of HEU for research reactors took back the spent fuel for reprocessing. Russia, as an other important supplier of HEU for research reactors has never taken back the supplied fuel from the countries of utilization.

At the beginning of the 1990's, an increasing number of fuel accumulated at the site of research reactors, approaching the capacity limit of their storage facilities.

As the reprocessing capacities were and are being limited for research reactor fuel due to high cost and the originally designed storage capacities reached their limit, additional storage areas were necessary. It was provided by re-racking or modifying the existing compartments of the reactor facility to make it possible to store the spent fuel there.

In some countries new storage facilities were constructed usually on the site, so the spent fuel transportation has been carried out within the site boundary.

However, since last year the USA re-opened the spent fuel take back policy with modified conditions it can be seen from the above description, that the demand on the spent fuel manipulations related to the spent fuel management considerably increased in present time.

From the beginning of the 90's more and more power reactor operating country were forced to review or make some correction on their spent fuel back-end strategy. The most preferable solution became the so called intermediate dry storage provided by separate storage building as MVDS and CASCAD, or storage in different type of casks (CASTOR, TN etc.). In countries with limited nuclear industries (having just few power reactors and other kind of nuclear facilities) the spent fuel intermediate storage provided by a central facility could be a good solution. This central storage facility can accommodate different kinds of fuel, if during the design stage of the facility careful considerations are taken. That kind of mutual utilization of storage facilities will claim for additional spent fuel handling manipulations at the research reactor facilities.

Spent fuel transfer (handling and transport) from research reactors to the central storage facilities will be necessary. In some cases, the shipping cask may be so large and heavy, that it is impossible or undesirable to load the spent fuel inside the research reactor facility. A transfer flask may then be used to transfer one or more fuel assemblies at a time, from the research reactor facility to the large cask outside the facility.

There is a wide range of spent fuel types for power reactors, but for research reactors this range is even wider. In some research reactor facilities different types of fuel were and are being used during the lifetime of the facility. Fuel types used in research reactors change more frequently than in power rectors. Research reactor fuel can be divided onto three groups; plate type assemblies (MTR in many countries), multi-tube assemblies (BR-2 in Belgium, DIDO in UK, IRT-3M or VVR-M in Russia) single- or multi-rod type assemblies (Slowpoke fuel in Canada, TRIGA in many countries, SM-2 in Russia). Although the majority of the research reactor spent fuel are of MTR or TRIGA types, a significant percentage of experimental and exotic fuels exist at research reactors around the world. The research reactor fuel assemblies are usually 60-90 cm long, but there are exceptions (Slowpoke fuel pin is just 30 cm long, while another multi-rod type Canadian fuel, NRU is 275 cm long). In the Russian designed research reactors, a large variety of fuel assembly geometry have been used. The active part of these fuel assemblies vary in length from 35-200 cm. All this means that, the spent fuel manipulation systems must be compatible with different kinds of fuel assemblies [2].

Until now most of the research reactor spent fuel are being stored in wet conditions. Taking into consideration the fact, that 80% of these reactors operates more than 20 years, it means that an increasing number of spent fuel are in water for 20 or 30 years. In the future to avoid any possible degradation of spent fuel assemblies in considerable amount at the same time, spent fuel canning or transition from wet storage to dry storage will be necessary. This will claim for additional fuel handling manipulations, not applied now in research reactors.

Considering the above described rising demand on research reactor spent fuel manipulation, it is obvious, that application of remote technology could provide a great benefit, especially in radiation protection (occupational dose reduction) and operation reliability.

3. REMOTE TECHNOLOGY APPLICATION IN RESEARCH REACTOR SPENT FUEL HANDLING

Research reactors constructed in the 60's and 70's were designed and operated with a very limited number of remote operations. Because of different reasons, until now, there is no real progress in use of remote technologies at research reactor facilities. Some of the main reasons are as follows.

- The necessary investment and utilization cost of the remote technology equipment is too high for the research reactor operators;
- In most cases, operators have ample time for spent fuel manipulation carried out one by one;
- There are not too many generally applicable remote equipment, especially designed for research reactor spent fuel manipulation.

In spite of these facts there are some areas, where remote technology is applied in research reactor facilities related to spent fuel management.

3.1. Spent fuel handling

Usually it is provided by long tools with manual operation in the reactor pool, or in the at-reactor pool. Application of even a simple spent fuel handling machine is very rare, however there are examples e.g. at Cirus Research Reactor, in India [3]. Remote hand tool application at power reactors is unique, used only for relatively low volume of operations. For research reactors manipulation this is the most conventional tool.

Spent fuel transfer between the reactor and the at-reactor pool is mainly provided by remote handling tools. Some reactor compartments have a direct connection between the reactor and the storage pool, where the spent fuel can be transferred. It is provided by a buggy system arranged in a channel or through a chute used for example, in some Russian designed research reactors where the fuel transfer cask is placed over the reactor pool.

3.2. Spent fuel transportation

This is mainly provided by spent fuel transportation casks or flasks. The range is very wide from the very simple ones to the highly sophisticated types, designed for off-site transportation as well. While many casks have been approved internationally for use in the shipment of spent fuel, only a few are suitable for the shipment of spent fuel from research reactors (CASTOR M2, or Russian TK-19).

Transfer of the spent fuel assemblies from the storage position in the pool into the transfer cask is provided by a lifting equipment installed on the cask or separate handling equipment is used to carry out the cask loading process. Casks used for off-site transportation usually do not include spent fuel lifting systems. So, for loading these casks an intermediate cask or other kind of remote fuel handling equipment, with necessary shielding arrangement, is used.

3.3. Spent fuel examination

There are some spent fuel inspection systems that are used for research reactor spent fuel examination. That kind of equipment can be divided into two groups; systems providing direct visualization of spent fuel assemblies and systems utilizing different level of tomography.

Equipment providing direct visualization is manufactured nowadays with fiber optic equipped with a CCTV system. Earlier versions of that equipment used in research reactor facilities were only a periscope applied for spent fuel identification, defect detection or to check correct spent fuel position, during manipulations.

In some research reactor facilities, Computer Aided Tomography are in use for fuel examination. Neutron beam produced by the reactor is used for this purpose. A multi-axis robot has been applied to carry up the fuel to the neutron beam [4].

3.4. Spent fuel cutting

In some cases at the research reactor facilities, it may be necessary to trim away those portions of the fuel assembly which do not contain fuel (e.g. grid plate fittings, end boxes) in order for the assembly to fit into the shipping cask basket, for example at Demokritos Research Reactor in Greece [3] or at DHRUVA Research Reactor in India [4]. For this purpose, underwater cut off saw is used in these facilities. Depending on the fuel geometry and whether both end of the fuel assembly must be cut, different equipment arrangement exists (saw with single blade or double blade, different spent fuel assembly fixing frames, etc.)

3.5. Spent fuel canning

Spent fuel assemblies beginning to leak during the operation in the reactor or during storage should be canned to avoid the continuous radioactive release into the surrounding environment. For this purpose tubes which can be closed by welding or bolting may be a good solution. As the spent fuel assembly leak in research reactor is not a frequent event until now, and the long term storage of research reactor fuel assemblies in wet condition presents good results, the reactor facilities do not have equipment for canning the spent fuel assemblies in large numbers. Taking into consideration the amount of spent fuel assembly quantities stored in wet condition for longer and longer time, the demand on equipment for spent fuel canning will raise. Techniques to can the research reactor spent fuel assemblies already exist, however further progress will be necessary applying more and more remote equipment to meet the requirement.

4. FUTURE DEVELOPMENT

It can be seen from the previous chapters, that the frequency of research reactor spent fuel manipulation, including spent fuel handling, transfer, examination, etc., will increase. The necessary manipulations will include more and more complex operations not designed or intended to do at the reactor facility earlier. Remote operations can be a key factor in performing these spent fuel management manipulations.

The main area where real progress will be necessary in the near future is the research reactor spent fuel transport from the existing pools into the new storage facilities, including the fuel transition from wet storage to dry storage.

Considering the mutual utilization of a spent fuel storage facility for power reactor spent fuel and research reactor spent fuel, special interface equipment and systems applying remote technology will be necessary.

Transition of the spent fuel between different storage environments will make it necessary to provide more complex operations (detailed spent fuel examination, applications of special fuel canning before transfer, etc.) To provide the necessary protection for the operators during these manipulations and the reliability to avoid errors due to the large number of similar operating sequences, more and more remote equipment with higher and higher automatisation level should be used.

Further development in the remote technology including not only the nuclear industry, can promote the progress in research reactor spent fuel handling, with more reliable remote equipment, providing further occupational dose reduction as well.

Standardization at components and sub-systems level will help to integrate different configurations or systems.

Special remote systems designed for power reactors or for power rector spent fuel storage systems may be also used for research reactor spent fuel with necessary interface equipment or tools in the future.

5. CONCLUSION

Duo to different reasons, the remote technology utilization for research reactor fuel is not so widespread as it is for the power reactor fuels, however, the advantages of the application of such techniques are obvious.

Considering the increased number of research reactor spent fuel handling operations, some limited development in the area of remote technology application at research reactors are foreseen. However, due to the different nature of the power plants and research reactor facilities, utilization of the remote operations at research reactors will never be so usual and sophisticated as it is or will be at power reactor facilities.

Related to the construction of new research reactors and spent fuel storage facilities, application of the remote technology will be more and more common as the reliability, radiation tolerance and cost reduction of remote system and equipment increases with the development of the industry.

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REMOTE TECHNOLOGIES FOR HANDLING SPENT FUEL

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Abstract

The nuclear programme in India involves building and operating power and research reactors, production and use of isotopes, fabrication of reactor fuel, reprocessing of irradiated fuel, recovery of plutonium and uranium-233, fabrication of fuel containing plutonium-239, uranium-233, post-irradiation examination of fuel and hardware and handling solid and liquid radioactive wastes. Fuel that could be termed "spent" in thermal reactors is a source for second generation fuel (plutonium & uranium-233). Therefore, it is only logical to extend remote techniques beyond handling fuel from thermal reactors to fuel from fast reactors, post-irradiation examination etc. Fabrication of fuel containing plutonium and uranium-233 poses challenges in view of restriction on human exposure to radiation. Hence, automation will serve as a step towards remotisation. Automated systems, both rigid and flexible (using robots) need to be developed and implemented. Accounting of fissile material handled by robots in local area networks with appropriate access codes will be possible. While dealing with all these activities, it is essential to pay attention to maintenance and repair of the facilities. Remote techniques are essential here. There are a number of commonalities in these requirements and so development of modularized subsystems, and integration of different configurations should receive attention. On a long-term basis, activities like decontamination, decommissioning of facilities and handling of waste generated have to be addressed. While robotized remote systems have to be designed for existing facilities, future designs of facilities should take into account total operation with robotic remote systems.

1. INTRODUCTION

India's nuclear programme encompasses use of uranium (natural and enriched), reprocessing spent fuel and extraction of plutonium, fabricating fuel containing plutonium and uranium (mixed oxide or MOX and mixed carbide). The MOX fuel will be used in thermal reactors (pressurized heavy water reactors, PHWRs, and boiling water reactors, BWRs) and the mixed carbide fuel in the Fast Breeder Test Reactor (FBTR). Further, the programme includes irradiation of thorium, recovery of uranium-233 by reprocessing, fabrication of fuel with uranium-233 and use of uranium-233 fuel. Figure 1 gives an overview of the various fuel types developed and fabricated. Hence, the handling of spent fuel involves various activities including the fabrication of fuel containing plutonium-239 and uranium-233.

Handling spent fuel from research and power reactors is a continuous activity. The thermal research reactors (CIRUS and DHRUVA) use natural metallic uranium. Power reactors use oxides of enriched and natural uranium. Reprocessing of irradiated fuel is carried out in reprocessing plants, fabrication of plutonium bearing fuels is done in dedicated facilities. An important activity needing handling of irradiated fuel is post-irradiation examination of various types of fuel.

Effective closure of the "back-end" of the fuel cycle while highlighting reprocessing, will have to include handling waste generated. While dealing with waste (solid and liquid), a matter that could be kept in mind would be activities like decontamination and decommissioning of facilities.

In the above mentioned activities, it may be noted that there could be commonalities in technologies, and development of remote techniques will have impact on several areas of applications.

The two Research Reactors in Bhabha Atomic Research Centre (BARC), Trombay, Mumbai use metallic uranium as fuel. Fuel discharged from reactors is handled under water and underwater cutting of aluminum and components is carried out using special equipment designed for the purpose.

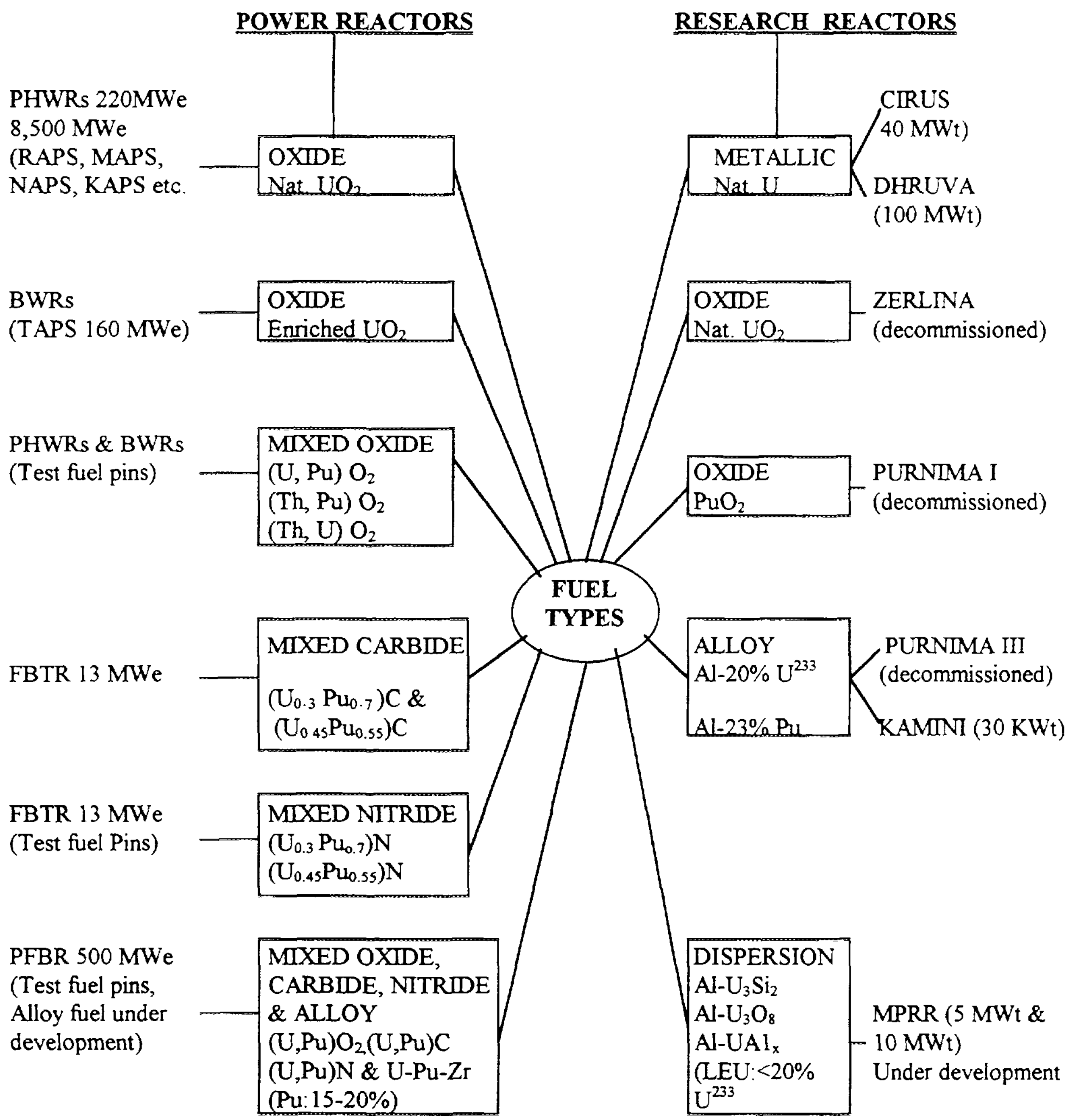

FIG. 1. Nuclear fuels developed and fabricated indigenously in India

The fuel assembly of the research reactor "Dhruva" is 9,256 mm long and weighs around 100 kg. The irradiated fuel assembly is brought vertically from the reactor building to the Spent Fuel Storage Bay and it is bisected into two pieces to get the fuel part separated from the main assembly. The fuel part is canned and loaded into the shipping cask which carries the fuel for reprocessing. A number of operations such as cutting, canning and loading into the shipping cask are carried out remotely in the bays under water 10 meters deep, with the help of a spent fuel handling and cutting system designed and built for this purpose (see Figures 2 and 3). A brief description of the operations and the machines used is given below.

■ DESCRIPTION OF OPERATION	■ EQUIPMENT USED
➤ LIFT THE FUEL BUNDLE ASSEMBLY (F.B.A.) FROM THE BUGGY AND PLACE IT IN THE HINGED BRACKET & SWING THE SAME TO BE POSITIONED IN THE CUTTING MACHINE.	➤ SELF CLOSING GRAPPLER WITH EOT CRANE.
➤ CLAMP THE F.B.A. ON V-CLAMPS BY OPERATING HAND WHEELS.	➤ HAND WHEEL OF THE CUTTING MACHINE.
➤ CUT THE F.B.A. IN TWO PIECES.	➤ SLITTING CUTTER MOUNTED ON SUBMERSIBLE MOTOR OF THE CUTTING MACHINE.
➤ REMOVE SEAL/SHIELD PLUGS AND PLACE IT IN THE RACKS.	➤ GRAPPLER AND EOT CRANE
➤ RETRACT THE CUTTER & GRIP THE FUEL BUNDLE IN MANIPULATOR JAWS.RELEASE UPPER AND LOWER CLAMPS OF THE CUTTING MACHINE.	➤ CUTTING MACHINE AND UNDER WATER MANIPULATOR.
➤ PLACE THE CAN (WITH PLUG) IN THE CANNING UNIT AND LOCK IT IN VERTICAL POSITION.	➤ CANNING UNIT AND UNDERWATER MANIPULATOR.
➤ SLIDE THE FUEL BUNDLE IN THE CAN AND PLUG IT FROM THE TOP.	➤ UNDER WATER MANIPULATOR.
➤ REMOVE THE CANNED FUEL BUNDLE & PLACE IN TILTER.	➤ UNDER WATER MANIPULATOR.
➤ REMOVE THE LOCK & ALLOW THE FUEL BUNDLE TO COME TO DESIRED POSITION.	➤ TILTER.
➤ TAKE OUT THE CANNED FUEL BUNDLE & KEEP IT IN THE RACK.	➤ DOUBLE JAW GRAPPLER.

FIG. 2. Spent fuel handling and cutting system installed in the water filled bays of the reactor DHRUVA

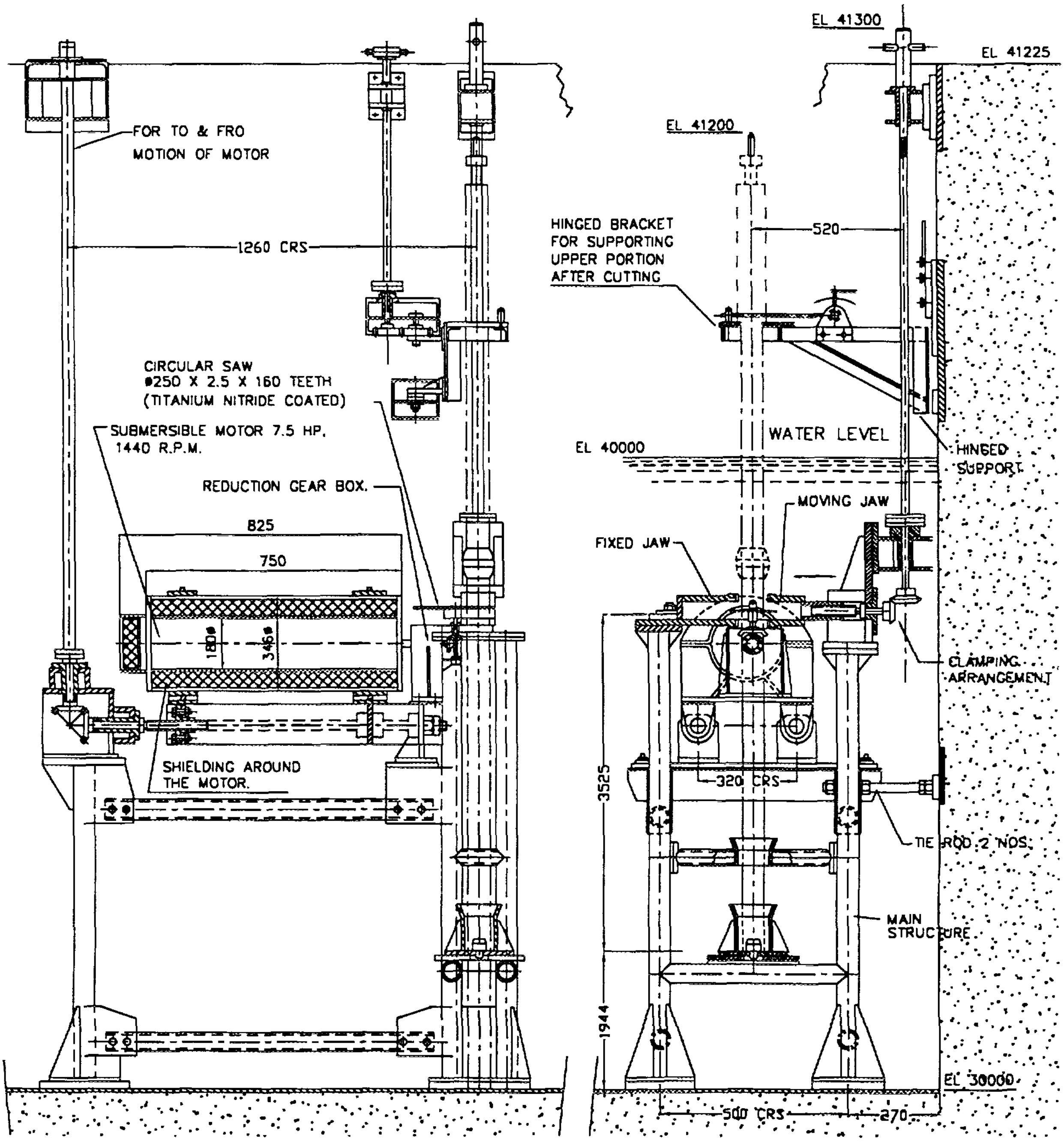

FIG. 3. General assembly of underwater cutting facility installed at DHRUVA, BARC

Operation of removal of fuel bundle from fuel buggy and placing it in the cutting machine is done, using a manually operated grappler and an Electrical Overhead Gantry crane. Bisection of fuel assembly is done with the help of a cutting machine. The machine consists of a main frame to support the fuel assembly vertically, a pair of V-clamps, a swinging bracket and a titanium carbide coated high speed steel circular saw, mounted on a shielded/canned submersible motor. Screw drives are provided for actuating the clamps and for feeding the saw forward. The drives are operated manually from the top of the pool through long shafts.

The upper part of the bisected fuel assembly containing the seal and shield plug is kept aside by swinging the bracket supporting the cut part. The lower part containing the irradiated fuel is held and transported to the canning station with the help of an under water manipulator The canned fuel assembly is made horizontal or set into any angle with the help of a tilting machine and is loaded into the cask with the help of a grappler having two pairs of jaws.

Reprocessing is carried out in shielded facilities with Master Slave Mechanical Manipulators Plans are being now made to use advanced remote systems that have been developed in the Division of Remote Handling and Robotics, BARC Mention should be made of the telemanipulators developed and manufactured totally indigenously These servo-telemanipulators use A/C induction motor servo systems developed in BARC (Figure 4) Accessibility of the parts of the plant will be greatly improved by using these systems Other special systems developed for remote operation in reprocessing facilities are remote pipe cutters and welding systems. Currently, experiments are being planned to use servo telemanipulators not only for handling irradiated fuel and other components but also for carrying out repair and maintenance in facilities. Operations like cutting and welding will be carried out on a robotic mode by using telemanipulators as computer aided telemanipulation systems Use of these manipulators is being expanded by mounting them as remote controlled mobile platforms for decontamination, decommissioning of facilities Experiments using sensory interfaces between the task environments and the manipulators are being done to improve transparency of operation and telepresence with improved sensitivity. Water jet techniques for extensive repair, decontamination and decommissioning are being actively pursued with help from Indian Institute of Technology, Madras.

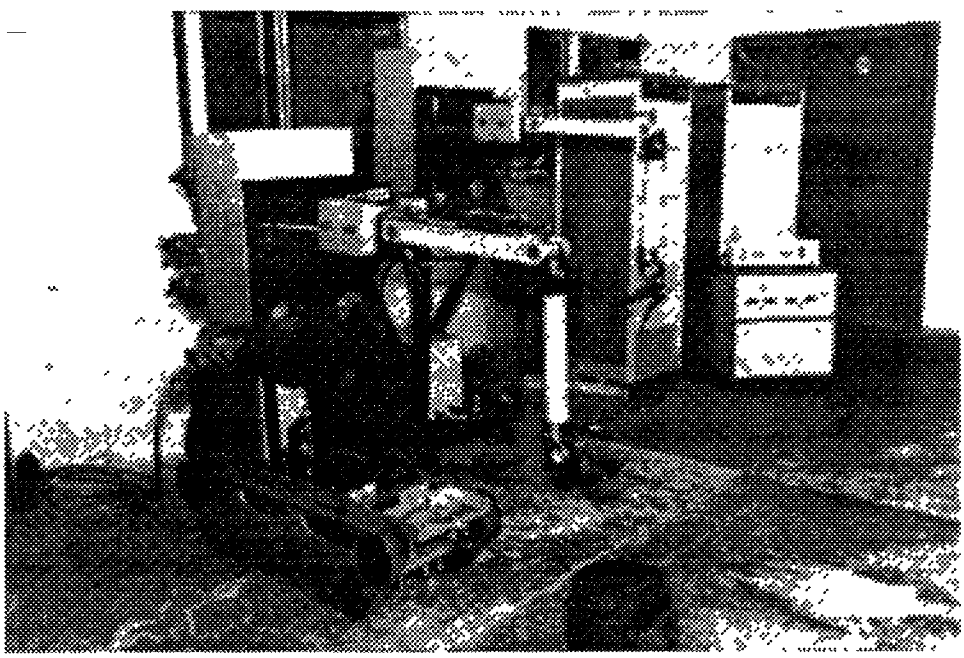

FIG 4 Servo manipulator mounted on a tracked mobile platform

Trials have been performed on unirradiated Zirconium clad PHWR fuel bundles (containing natural uranium oxide fuel) with objectives of dismantling and decladding, by using Nd-YAG (yttrium aluminum garnet) laser systems. The results of the trials are encouraging. This approach will greatly help to reduce the presence of zircaloy in irradiated fuel being reprocessed. Plans are being made to implement this technology in operating reprocessing plants. This technique will be friendly for efficient remote operation. Manipulation of the laser beam can be done by a robot or a numerically controlled table. Alternatively, the fuel bundle or element can be manipulated by these devices against a steady laser beam (see Table I and Figures 5 to 7).

TABLE I. PIE FACILITIES FOR IRRADIATED FUEL IN INDIA

	Hot cells BARC	Hot cells IGCAR
Type of cells Atmosphere Ventilation	βγ air once through	αβγ nitrogen recirculation
Type of fuel Burnup range	thermal reactor fuel Al clad metallic U Zr-2 clad UO_2 Zr clad MOX $(U+4\%Pu)O_2$ 20-25,000MWd/t	fast reactor fuel SS clad mixed carbide (U+55% Pu)C 2,000-25,000 MWd/t
Remote handling	master slave manipulators (MSMs)	MSMs power manipulator
Dismantling Pin section	bundle dismantling machine low speed cut off machine	CNC cutting machine low speed cut off machine
Viewing	shielded glass windows oil filled scanning wall periscope	shielded glass windows all solid scanning wall periscope
Testing equipment in hot cells	profilometry eddy current ultrasonic gamma scanning neutron radiography (APSARA) αβγ autoradiography remote microscopy clad ring tensile test	eddy current ultrasonic x-ray radiography neutron radiography neutron tomography replication facility remote microscopy

BARC - Bhabha Atomic Research Centre
IGCAR - Indira Gandhi Centre for Atomic Research

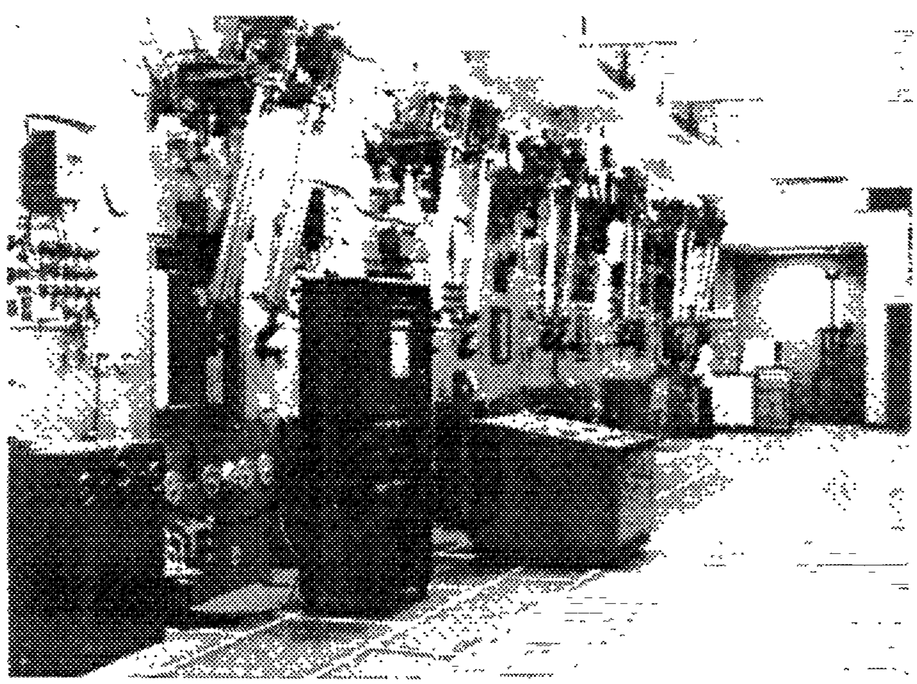

FIG. 5. Hot cell facilities for post irradiation examination in BARC

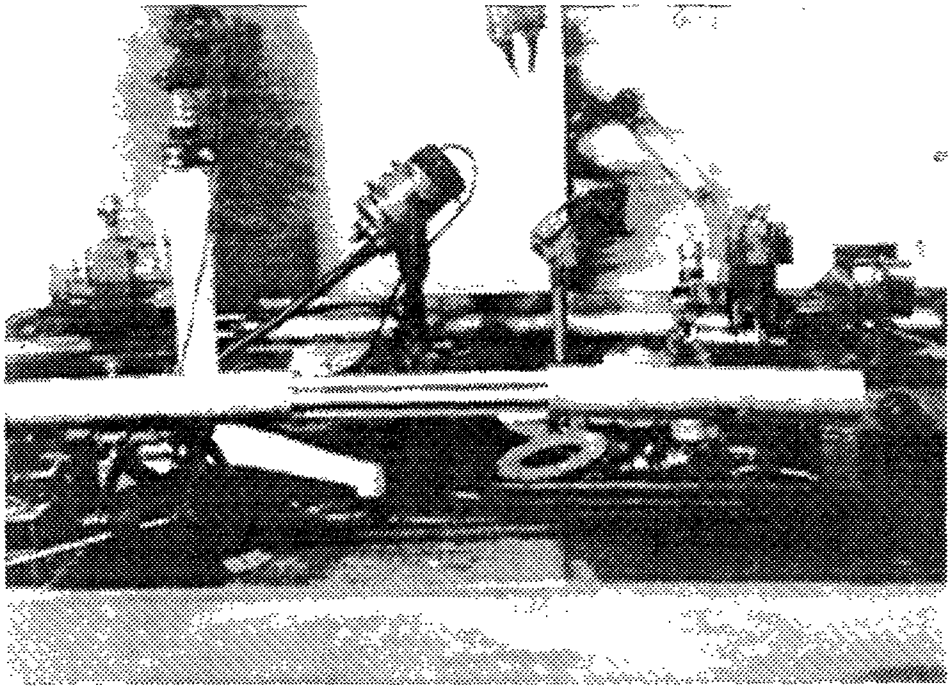

FIG. 6. Post irradiation examination equipment in BARC hot cells

FIG. 7. Post irradiation examination equipment in BARC hot cells

The Fast Breeder Test Reactor (FBTR) at Indira Gandhi Centre for Atomic Research at Kalpakkam, Madras is a reactor fueled by mixed carbides of plutonium and uranium (70% plutonium-carbide and 30% uranium-carbide) with sodium primary cooling. The cladding is stainless steel and fuel elements are in bundles. Each bundle contains 61 pins and is encased in a hexagonal sheath (see Figure 8). Irradiated fuel bundles of FBTR are brought to a hot cell for dismantling, post-irradiation examination and reprocessing (Table I). Machining operations are required during dismantling and are carried out with the help of a special purpose machine built and installed in the hot cell. Handling of fuel assembly or its components and operating the special purpose machine are done remotely with the help of various types of remote handling equipment installed in the cell. These are a pair of master slave manipulators (MSMs), a power manipulator and an incell crane. Viewing is done through the lead glass shielding window. The special purpose machine has been designed to perform linear motions in X,Y,Z direction and a rotation motion of the spindle. These motions are essential for positioning and feeding the cutting tool. Provision is made to hold the fuel assembly in chucks and rotate it around a vertical axis.

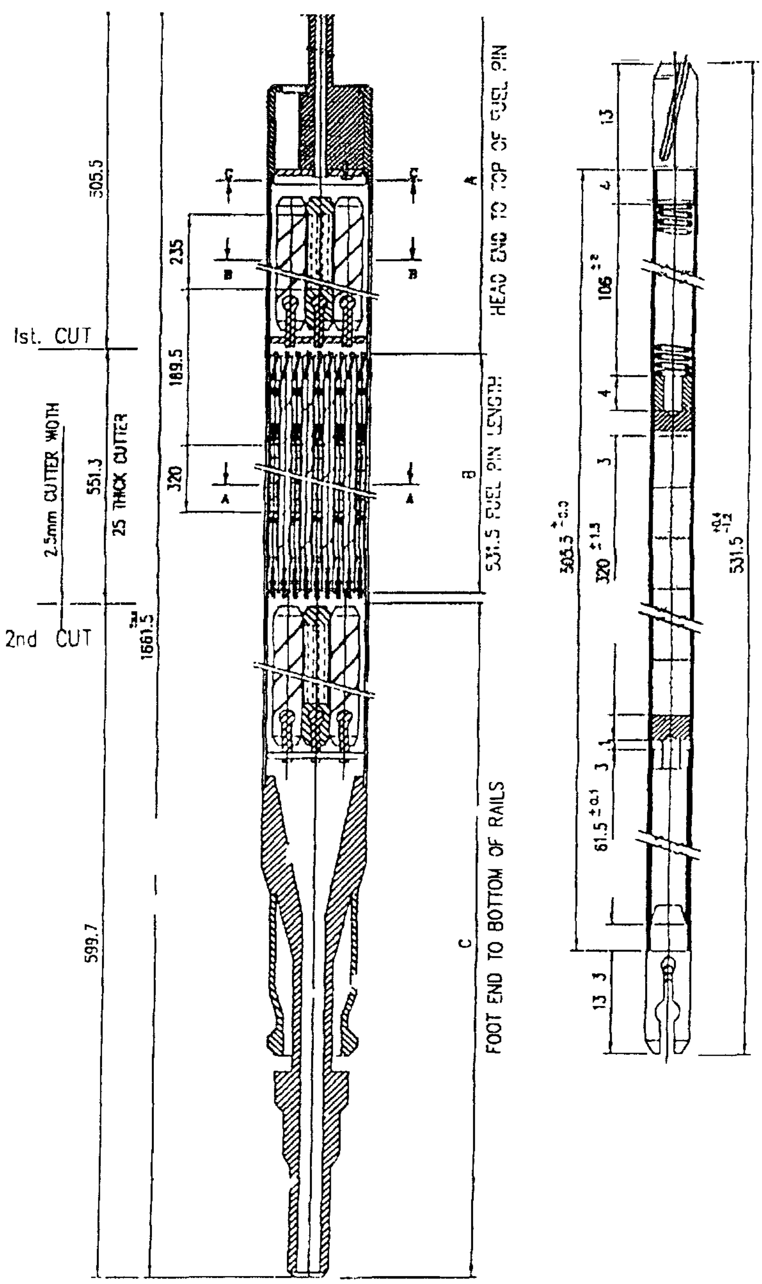

FIG. 8. FBTR fuel subassembly and fuel pin

The fuel bundle can be swung to enable loading and unloading from the machine. The fuel bundle to be dismantled is loaded vertically from top to go through the bores of the spindles carrying the pair of chucks. The X, Y & Z motions of the machine are 350 mm, 1,220 mm and 260 mm respectively. Though all motions are screw driven and are operated by geared electric motors, with push button control, provision has been made with additional design features to de-link all the motions from push button controls and perform motions including the tool changing operation manually but remotely with the help of manipulators (Figure 9).

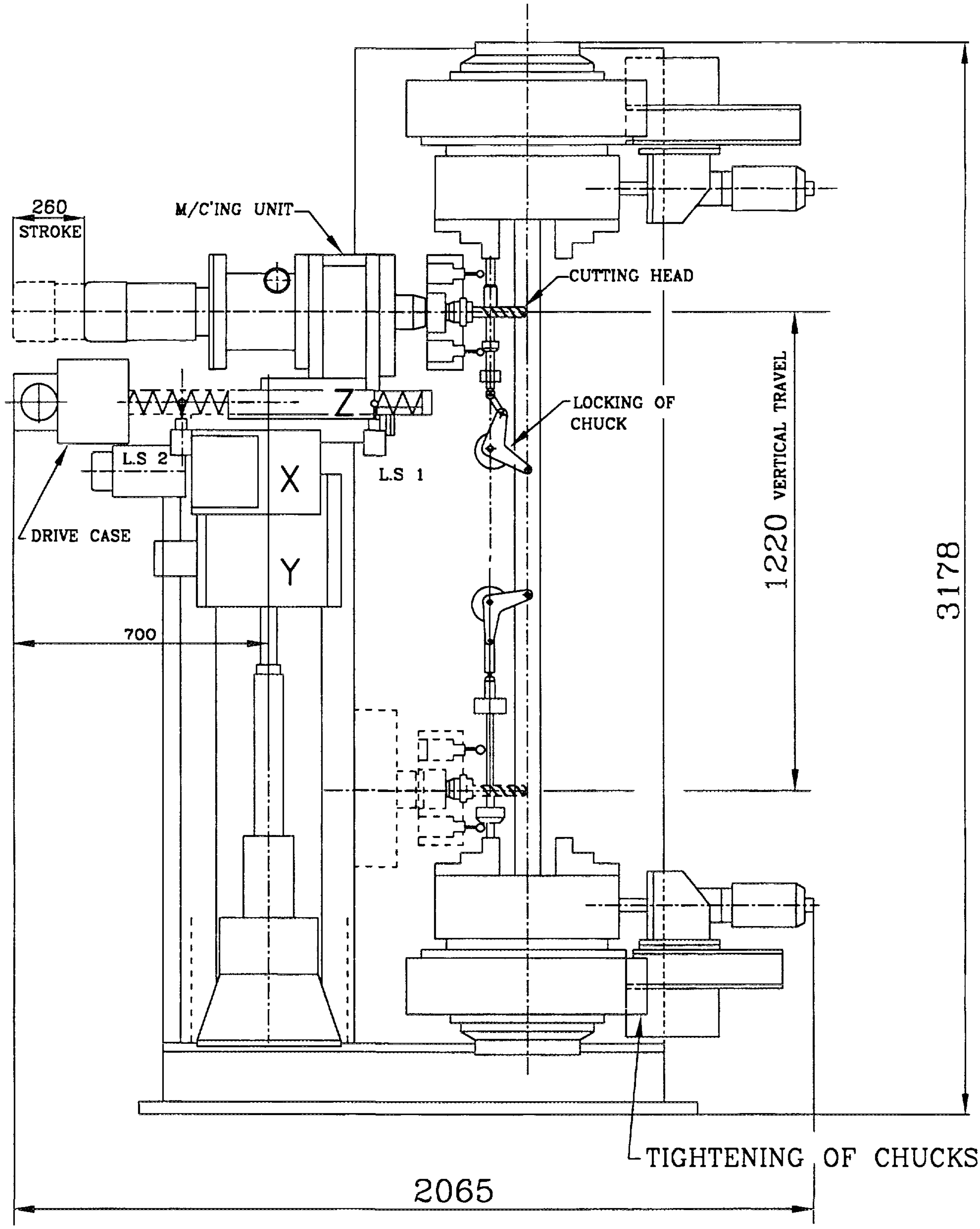

FIG. 9. Fuel dismantling machine for FBTR

Post-irradiation examinations have been carried out on a subassembly in hot cells at Indira Gandhi Centre for Atomic Research (IGCAR)'. Burn-up achieved was 25,000 MWd/t. The hot cells have been designed and constructed to handle plutonium based fuel. carbides of plutonium and uranium are highly pyrophoric and hence the hot cells are clad in stainless steel and operate under inert atmosphere of nitrogen under recirculation. A numerically controlled machine specially designed for the purpose is located in the hot cell and performs operations of dimensional inspection and dismantling of assemblies. Viewing is done through the lead glass shielding window. Other installed equipment like Master Slave Manipulators and Power Manipulators aid in handling fuel inside the hot cells. The entire operation of cleaning the assembly (of sodium), handling, inspection, dismantling and post-irradiation examination was carried out successfully. This has given the group great confidence to tackle problems connected with handling irradiated fuel in large numbers. Possibilities of reconstitution of fuel also are being considered.

Tomography offers possibilities of examination of assemblies without dismantling. With the KAMINI reactor going critical at IGCAR, work on neutron tomography of irradiated fast reactor fuel assemblies will be taken up. A 5-axis robot to handle and manipulate assemblies, in front of the neutron beam from KAMINI, has been developed. Extensive trials using this robot on fuel bundles with metallic uranium fuel have been carried out. The robot is driven by stepping motors, has three linear degrees of freedom for positioning assemblies, and two more degrees of freedom for orientation. While the initial trials were carried out using a gamma-beam, work is in progress to interface the system with a neutron beam from KAMINI (Figure 10).

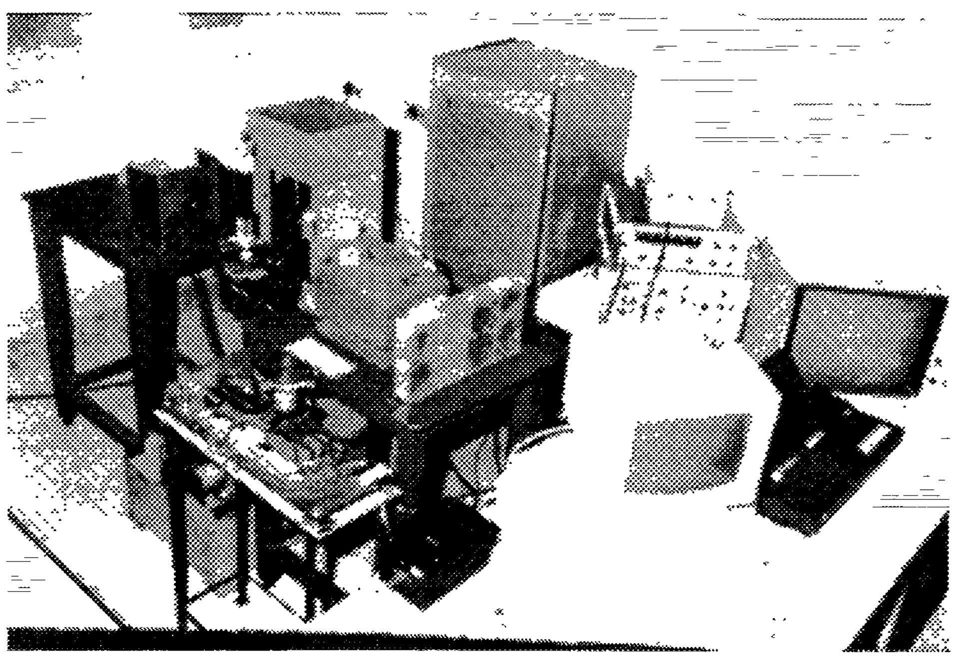

FIG. 10. Robotic system for neutron tomography of irradiated fuel

Automation is a way of achieving objectives of remote operation. This can reduce human operator exposure drastically. As mentioned before, our programmes include the use of plutonium and uranium-233 in our reactors.

Carbides of plutonium and uranium constituting FBTR fuel are highly pyrophoric. Therefore, the fuel fabricating train has to be enclosed in glove boxes under inert atmosphere with adequate shielding to eliminate radioactive exposure to operators. Handling of plutonium for fuel fabrication needs both rigid and flexible automation. A special automated inspection system for inspection of carbide fuel pellets has been developed. This system has been in successful operation for more than 10 years. The sintered pellets of mixed carbides are 4 mm wide and 8 mm long. The system measures diameter, length and weight. The linear density is calculated by a computer. The measured diameter and density are compared with acceptance parameters after which the pellet is accepted or rejected Accepted pellets are automatically stacked in columns of pre-determined length. All the data, including the position of pellets in the column, are recorded and can be accessed when needed (Figure 11)

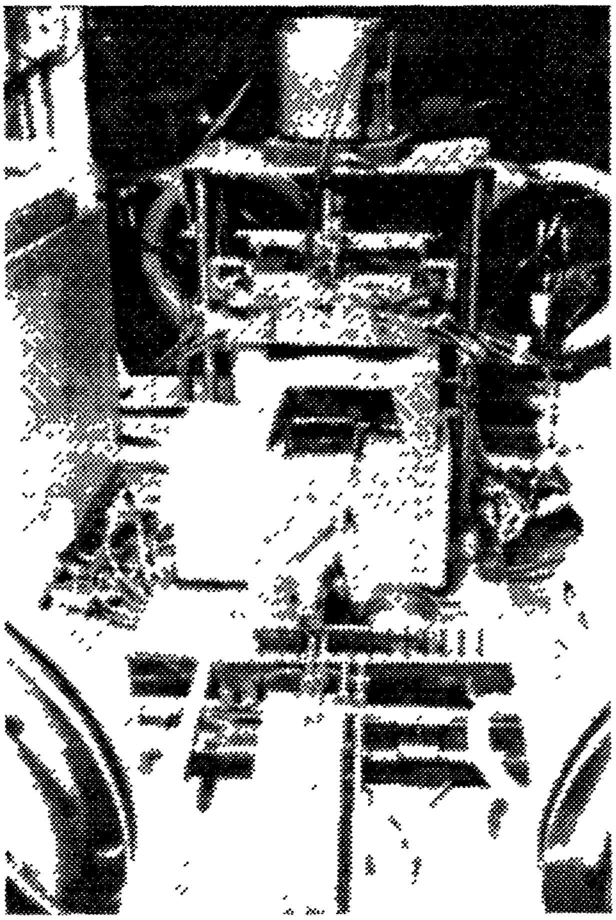

FIG 11 Computerized automated inspection system for mixed carbide fuel pellets (for FBTR)

Mixed oxides (MOX) of plutonium and uranium are now being used in our thermal reactors also. One interesting area of automation is the fuel fabricating train which deals with handling of compacted green mixed oxide pellets. A special system has been developed to unload green compacts from a hydraulic press and convey them down on a special inclined conveyor There are sensors that count four pellets at a time when the conveyer is stopped. A 5-axis articulated robot picks the pellets and stacks them in a molybdenum tray for the next step, i.e. sintering. The pellets are handled gently to avoid damage. The number of rows, arrays and layers of stacking can be programmed. The manufacturing cell consisting of the press, conveyer and robot is controlled by a personal computer which will be a part of a local area network. The entire system has been so developed that it can be dismantled and installed in active glove boxes housing the fuel fabricating train Subsequent maintenance of the robot is ensured as the system is a standardized configuration with modular subsystems that can be inserted through transfer ports of glove boxes (Figure 12)

The robot mentioned above has been made into a standard configuration which can be applied in other operations too. After sintering, pellets have to be unloaded from the molybdenum trays. This will be done by a robot, equipped with tactile sensors, which will search and pick up the pellets and stack them on a stand for the next operation, which is centreless grinding Owing to the configuration of the trays and poor color contrast between pellets and trays, vision systems will not be effective.

With tactile sensors on the end effector, the robot 'blindly' searches for the pellets and picks them up. The programmes for the robot control are such that the end effector of the robot can move on to the pellet whether in a cluster, standing alone or in a row. Extensive cold trails have been carried out and the system is now being readied for actual use in the MOX fuel plant at Tarapur, near Mumbai (Figure 13).

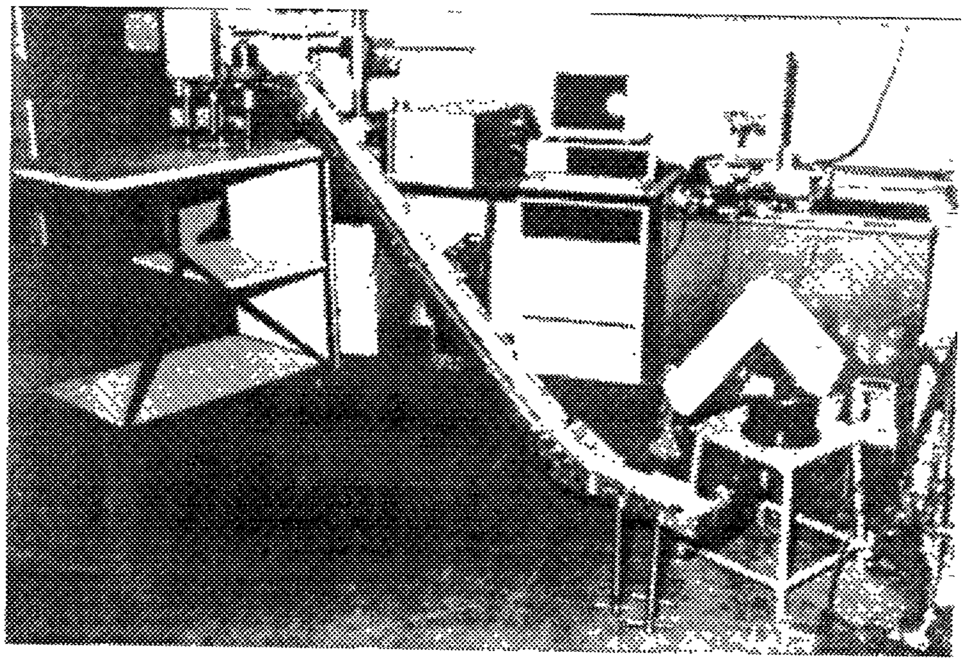

FIG. 12. Robotized manufacturing cell for handling compact green mixed oxide fuel pellets

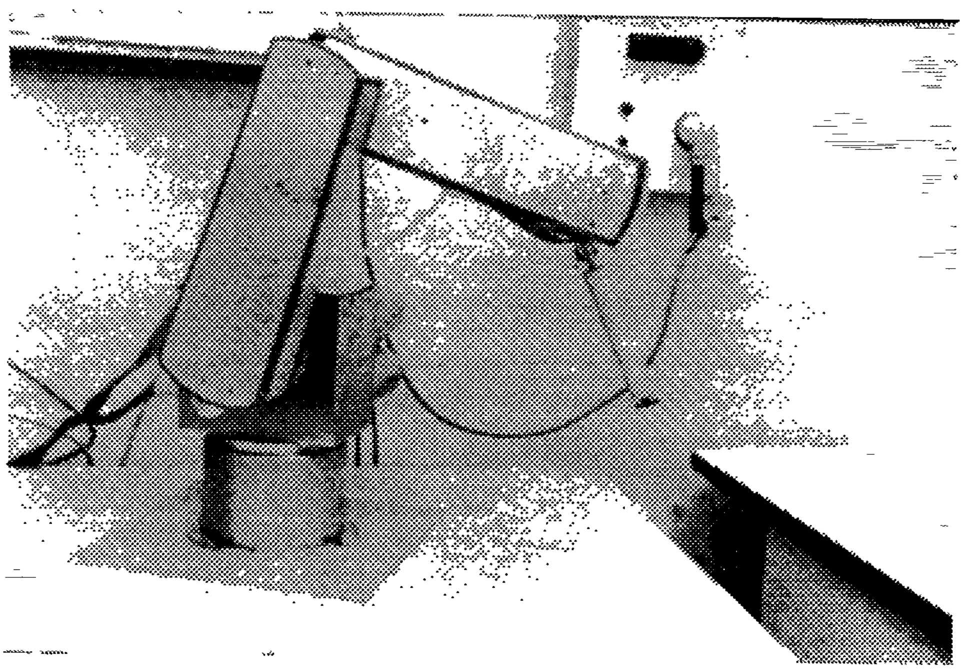

FIG. 13. Articulated robot with tactile sensors for searching, picking up and stacking MOX fuel pellets.

Many other experiments like intelligent stacking of pellets into column lengths, inspection of diameter and surface morphology etc. have been done to develop system resulting in flexible automation eliminating operator exposure in fabrication of fuel containing plutonium.

With a programme to build power reactors using natural uranium and thorium reserves in the country we have to plan for three generations of reactors. While FBTR represents the plutonium fueled second generation, KAMINI with uranium-233 belongs to the third generation. A 500 MW fast reactor, namely Prototype Fast Breeder Reactor, is in an advanced stage of design and planning. Fuel fabrication, reprocessing of irradiated fuel, fabrication of fuel with plutonium-239 & uranium-233 and waste management will become large scale activities. Remote technologies, automation and robotics which will be the main support for the whole programme, are being given strong emphasis. The emphasis of handling large quantities of irradiated fuel will have to be at an international level. If designs of reactors, fuels, fuel assemblies take into account activities connected with fuel fabrication, remote inspection (of reactors, fuel assemblies etc.) facility operation and spent fuel handling which need robotic remote devices, all the systems can be made safe and transparent for remote operation and management.

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KOREAN INTERIM STORAGE ISSUES AND R&D ACTIVITIES ON SPENT FUEL MANAGEMENT

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Abstract

Korea has witnessed over a decade of vicissitudes in the issue of radioactive waste management due mainly to the problem of site acquisition. As the major mission of the nation at radioactive waste management programme was to provide a center for disposal of low-level radwaste and for interim storage of spent nuclear fuel from nuclear power plants, the question of site securing has had a big impact on the implement action of overall programme. The site problem has resulted in, as an aftermath, restructuring of the overall programme for radioactive waste management. Missions of NEMAC (Nuclear Environment Management Center), originally established as a subsidiary of Korea Atomic Energy Research Institute (KAERI), for the national programme was dissolved as of the end of last year. Beginning of this year, a new entity NETEC (Nuclear Environment Technology Center) as a subsidiary of KEPCO (Korea Electric Power Co.) has taken over major tasks of the past NEMAC, while the R&D work associated with the past task of NEMAC is transferred back to KAERI. This paper gives a review on the past background which has driven the radioactive waste management in Korea to the current state of the affairs, with special emphasis on R&D activities associated with spent nuclear fuel transportation, handling, and storage.

1. HISTORICAL INTRODUCTION

1.1. Background of Korean radwaste management programme

Some research activities associated with radioactive waste management in Korea trace back to the seventies when KAERI became a national institute. Systematic R&D on the subject was initiated, however, in the mid-eighties when radioactive waste and spent fuel had begun to accumulate at increasing number at nuclear power plants.

Conscious of the radwaste management in the future, the Korean authority KAEC (Korea Atomic Energy Commission) amended in 1986 the Atomic Energy Law to set forth a programme for long-term management of radioactive waste and of spent fuel. Key projects of the national programme are:

- to provide a repository for disposal of low-and intermediate radioactive wastes from various sources in the nation;
- to provide an interim storage facility for centralized management of spent fuel that are stored temporarily at various reactor sites;
- to provide technical expertise by R&D activities on support of those projects.

Plans to implement the programme had been subsequently developed and approved by the AEC, i.e. major facilities to be built by 1997 on a collocation site to be selected. Until that time, the management of already accumulated wastes or spent fuel at reactor (AR) sites have to be taken care of by the utility itself through appropriate methods.

Institutional arrangements were also made by designating KAERI as the responsible body to implement the programme under government supervision (Ministry of Science & Technology). The necessary fund was provided by collecting a waste fee from KEPCO, the national utility, based on the "Polluters Pay" principle.

The legal arrangements of the programme had been finalized in 1989 when the enactment of the amended law was promulgated by Presidential Decree.

1.2. Public perception problems in search of site

When the Korean programme was set up in the mid-eighties, nobody was much conscious about the social problem that may interfere with the waste management programme, especially in the affairs of site search. The NEMAC team in preparation of the programme implementation began with technical assessment of a suitable site for the waste management center by survey of potential candidate sites all over the country and screened out them to several promising ones on the basis of selection criteria which were mostly technical but little social considerations. This negligence in social factors had to turn out to be a grave mistake as to be witnessed later.

At initial phase of the site survey, the site inspection team did not expect strong protest against the survey activities itself. The naive access to potential candidate sites, without enough prior understanding of the local inhabitants about the assurance of the project, seemed to have spurred them furious oppositions to the survey activities itself. One site after another, the survey team encountered barring oppositions from the local inhabitants instigated by anti-nuclear group of ecologists. These protests had been emphasized by the media which had taken part in promoting the negative aspects of everything about "garbage" disposal as a social issue. It was also a period of Korean society when social oppositions to governmental authority were often confounded with a sort of democratization against political oppression. These series of troubles at several sites was culminated when a serious incident of crash between the protesters and police forces happened in the winter of 1989 in Anmyon Island in west coast of the Peninsula. This mishap was marked as a political fiasco in the governmental measure in radioactive waste management policy confronted with negative perception of the affected public. An extensive revision of the governmental attitude on the problem was inevitable as an aftermath of the Anmyon Incident. The governmental authority of the programme had to be reorganized and a new approach to the problem was looked for a final breakthrough. Financial compensation to the affected inhabitants was one of such approach which was legalized and publicized.

The last coup de grace in the site acquisition problem came in nineties when the newly organized authority designated the Guleop Island, also in the Yellow Sea near Inchon Harbor, as the final resort of the site problem. As the island was not populated with any inhabitant, it was believed that the social protest of NIMBY (not in my back yard) could be avoided or mitigated, at least. This was not the case. By that time, the radwaste disposal had been regarded as something very dangerous not only in the site itself but also in the vicinity around the site. People around the Island, including the Inchon Harbor manifested such concern, in alliance with ecologist group of the country. The polemics was rather technical this time, in comparison with previous ones, because national consensus was more or less converging by that time in such way to approving the necessity of a site and of the justifiable alternative that had to be found on legal ground. The polemics has finally led to a detailed examination of geological structure by a national panel of experts who have finally concluded that the site is not adequate due to active faults found from the study. This conclusion which was announced by the authority in 1995 marked a wrapping up of the decade old project in search of a site for radioactive waste disposal and interim storage of spent fuel.

With the last failure in site acquisition, the government decided a general redirection in the state of the affairs by institutional rearrangements.

2. STATUS

2.1. General

With the last failure with the site acquisition effort, the government decided in 1996 a total redirection in the approach to the national programme for radioactive waste management by institutional rearrangement of the related missions.

First of all, the responsible organ, NEMAC, had to be dissolved and its missions newly rearranged under different scheme:

- The site acquisition task to be taken over by a new entity NETEC (Nuclear Environment Technology Center) by transfer of most of associated personnel of the former NEMAC to work for KEPCO under the policy management of the MOTIE (Ministry of Trade, Industry and Energy Resources). The NETEC inherits also the former NEMAC mission to build and operate the repository for low- and medium-level wastes and the interim storage facility for spent fuel (SF);
- The remaining mission of R&D associated with waste management of the former NEMAC is to be taken back by KAERI as a part of the overall nuclear research programme.

On legal side of the rearrangement, the provisions in the amended Atomic Energy Law is removed to be integrated into the “Electricity Enterprise Act”. "The Act for Promoting the Radwaste Management Project and Financial Support for Local Community" is abolished. Instead, the provisions for the local community support is added to "the Act of Support for Local Communities". The provisions for the securing of a disposal site is inserted in "the Special Act Relating to Development of Electric Power Resources". "The Radwaste Management Fund" founded in 1986 is abolished and "the Nuclear Research and Development Fund" is established to secure nuclear research and development expenses. Radwaste producers such as KEPCO and hospitals are to bear the cost for radwaste management.

The Korean government will re-assess public feeling and the social climate related to radioactive waste, and will redraft the radwaste project plan to safely and efficiently manage radioactive waste from generation, transportation, storage, treatment, to disposal. For this purpose, the government will act as follows : Firstly, it will review the types of disposal methods as well as the size of disposal facilities to find the most suitable ones to actual circumstances. Secondly, it will improve public understanding of the safety of radwaste management by reinforcing public acceptance programmes and maintain transparency in policy-making of radwaste project. Also, it will strengthen the local community support system to induce resident's invitation of the disposal facility. Thirdly, it will minimize the volume of radwaste by developing state-of-art volume reducing technology in addition to current volume contraction technology. Also, Korea will improve the relationships with advanced countries in nuclear technologies and international organizations in order to enhance the exchange of advanced technologies and information, and will elevate the level of radwaste disposal technology by participating in international joint research programmes on radwaste.

Currently, KEPCO is surveying new site for radwaste disposal and SF interim storage. Detailed prospects has not set up yet and any progress can not be found. KEPCO is considering the extension of AR facilities for the LLW and SF rather than construction of centralized storage facility.

2.2. R&D activities related to spent fuel management

While KAERI were conducting the national radwaste management project, it’s research effort was concentrated on developing the technology which is directly applicable to design and construction of the interim spent fuel storage facility (ISFSF). The list of these R&D topics is as follows:

- Study on long-term integrity of SF and development of integrity evaluation code, SIECO (Systematic Integrity Evaluation Code);
- Study on behavior of PWR SF in wet storage conditions;
- Development of SF storage container with oxygen scavenger;
- Development of remote handling and automation technology;
- Structural behavior of base-isolated pool structure;
- Pool water purification technology;
- Development of a large capacity SF transportation cask.

Besides these R&D activities, the SF technology development facility was conceptually designed for the purpose of developing the technology required for long-term dry storage or disposal of SF, and developing the remote handling technology of spent. Functions of this facility are SF cask handling, SF assembly inspection, storage in dry pit, fuel rod consolidation and packaging, integrity test of sample rods, and remote maintenance of equipment.

Since the national radwaste management programme is now reserved for further consideration, the current R&D activities are focused on the technology development related with the dry storage of SF. These are:

- Study on the oxidation behaviors of UO_2 and zircaloy under the air environment;
- Development of rod consolidation technology;
- Development of remote handling and maintenance technology.

Also, the study on direct use of PWR SF into CANDU reactor (DUPIC) project, one of the most challenging topics of KAERI, is on going.

3. INTERIM SPENT FUEL STORAGE FACILITY (ISFSF) CONSTRUCTION PROJECT

3.1. Project overview

The final decision on the ultimate management of SF has not been made in Korea. The 220th AEC held in July 1988 resolved the basic principle that the SF would be managed at the ISFSF in the national radwaste management complex site by government until further decision later. At 221st AEC the specified decision was made on the ISFSF. The decision was to construct the wet storage facility by Dec. 1997 whose capacity is initially 3,000 tU. Due to consecutive delay of site securing project, this decision was canceled at the 237th AEC held in Dec. 1995. Right now NETEC is studying the necessity of construction of interim storage including the future management plan of SF.

3.2. AR storage facility expansion programme

The country's first nuclear power plant (NPP), Kori-1, a 587 MWe pressurized water reactor (PWR), went into commercial operation in 1978, opening a new era of nuclear power generation in the country. The nation's nuclear power programme has continuously expanded since then. Today, Korea has twelve NPPs (10 PWRs and 2 CANDUs) in operation with generating capacity of 10,331 MWe. In addition, sixteen more reactors are planned to be constructed by 2010 as shown in Table I.

With the increase of NPPs, the amount of SF rapidly increases as shown in Table II. In accordance with 220th AEC's decision on spent fuel management that "The KEPCO will store SF at each NPP site until the ISFSF is operated," KEPCO has continuously expanded the storage facilities as shown in Table III [1]. In Uljin site, Uljin unit 2 storage pool was expanded in 1990 by reracking with high density (HD) storage rack before the first cycle SF discharge. In Kori site, Kori unit 3 pool capacity was expanded 1994 by adding the HD storage rack into the reserved space of the existing storage pool. In addition, 156 assemblies of Kori-1 fuel were transshipped to Kori-3 pool in 1991, and another 156 fuel assemblies are being transshipped to the pool of Kori-3 and Kori-4. For CANDU SF in Wolsong site, 60 units of dry concrete canister were installed in 1992 to accommodate 32,400 SF bundles. Currently, the dry storage facility is being designed for KORI NPP and more concrete canisters for CANDU SF are being constructed at Wolsung NPP. After these expansions, as shown in Table III, the storage capacity of AR facility is anticipated to endure until year 2006.

TABLE I. LONG-TERM PLAN OF NPP CONSTRUCTION BY 2010

	Unit	Reactor Type	Capacity (MWe)	Cumulative Capacity (MWe)	Start Operation (year)
In operation	Kori 1	PWR	587	587	78. 4.
	Wolsung 1	CANDU	679	1266	83. 4.
	Kori 2	PWR	659	1916	83. 7.
	Kori 3	PWR	950	2866	85. 9.
	Kori 4	PWR	950	3816	86. 4
	Yonggwang 1	PWR	950	4766	86. 8.
	Yonggwang 2	PWR	950	5716	87. 6.
	Uljin 1	PWR	950	6666	88. 9.
	Uljin 2	PWR	950	7616	89. 9.
	Yonggwang 3	PWR	1000	8616	95. 3.
	Yonggwang 4	PWR	1000	9616	96. 1.
	Wolsung 2	CANDU	715	10331	97. 7.
Under construction	Uljin 3	PWR	1000	11331	98. 8.
	Wolsung 3	CANDU	700	12031	98. 6.
	Uljin 4	PWR	1000	13031	99. 6.
	Wolsung 4	CANDU	700	13731	99. 6.
	Yonggwang 5	PWR	1000	14731	01. 12.
	Yonggwang 6	PWR	1000	15731	02. 12.
Planned	Uljin 5	PWR	1000	16731	03. 6.
	Uljin 6	PWR	1000	17731	04. 6.
	KNU 21	PWR	1000	18731	05. 6.
	KNU 22	PWR	1000	19731	06. 6.
	KNU 23	PWR	1000	20731	06. 6.
	KNU 24	PWR	1000	21731	07.
	KNU 25	PWR	1300	23031	08.
	KNU 26	PWR	1300	23744	09.
	KNU 27	PWR	1300	25044	10.
	KNU 28	PWR	1300	26344	10.

TABLE II. SPENT FUEL INVENTORY

tU

year	PWR		CANDU		Total inventory
	annual arising	cumulative inventory	annual arising	cumulative inventory	
1994	117	1,288	96	1,093	2,381
1997	204	1,843	95	1,378	3,221
2000	242	2,512	380	2,233	4,745
2005	318	3,912	380	4,133	8,045
2010	339	5,645	475	6,413	12,607

TABLE III. STATUS OF AR STORAGE POOL

Unit	Initial Capacity Full (year)	Expansion (Step 1, Finished)	Capacity (FAs)	Inventory (FAs) *as of April 1997	Anticipated Capacity Full after Step 1 (year)	Expansion (Step 2, plan)	Anticipated Capacity Full after Step 2 (year)
Kori 1 Kori 2 Kori 3 Kori 4	1991 2001 1997 1997	Transfer 156 FAs to Kori #3,4 Add HD rack (1994, 455 FAs) Add HD rack (1994, 455 FAs)	3,271	2,042	2002	Construct dry storage facility (2000, 400 FAs)	2006
Yonggwang 1 Yonggwang 2	1997 1998	ADD HD rack (1997, 400 FAs) ADD HD rack (1997, 400 FAs)	2,992	900	2006		2006
Uljin 1 Uljin 2	1994 1995	Reinstall HD rack (1991, 421 FAs)	1,675	573	2007		2007
Wolsung 1	1991	Add concrete canister (1992, 60 canister)	71,616	62,772	1997	Add concrete canister (1997, 80 canister)	2006

FAs = Fuel Assemblies

3.3. Interim spent fuel storage facility design

In 1983, a two-year study with Battelle Pacific Northwest Laboratory of USA concluded that AFR storage option would be suitable for the medium-and long-term spent fuel management in Korea. In 1986, joint studies with Sweden and Federal Republic of Germany were separately carried out to compare two methods of storing SF, wet storage and dry storage. Based on the results of these studies, the Atomic Energy Commission selected wet type for the first AFR interim spent fuel storage facility (ISFSF) to be completed by 1997. Subsequent expansions with a dry storage option will be followed as-needed basis. KAERI has carried out the site independent conceptual design of the ISFSF with Korea Power Engineering Company (KOPEC) as a domestic contractor and SGN in France as a foreign contractor. Also in 1992, KAERI and BNFL in the United Kingdom jointly reviewed the feasibility of Multi Encapsulated Baskets (MEB) for the prospective Korean facility and in 1993, KAERI and Ebasco in the USA have jointly conducted the comparative study between dry and wet storage methods.

3.3.1. Conceptual design of the ISFSF

KAERI has finished the conceptual design of the ISFSF in 1990. For this design, the design base such as storage method, capacity, transportation method and characteristics of SF has been set up (Table IV). Also the design requirements have been determined [2,3].

TABLE IV. DESIGN BASE OF THE ISFSF

Item	Design Base
O Storage type	• Wet unloading and wet storage
O Storage capacity	• 3,000 tU (Modular expansion)
O Reception capacity	• 650 tU/year
O Spent fuel characteristics	• all types of spent fuel generated form Korean NPP minimum 5 years cooling at AR pool • design enrichment and burn up – PWR : 4% enriched uranium, 40,000 MWd/tU burnup – CANDU : natural uranium, 7,500 MWd/tU
O Transportation cask (assumption)	• PWR – TN-17/MK2 (for capacity calculation) – TN-12/2 (for design of handling equipment) • CANDU : KAERI design
O Transportation method	• Sea transportation by ship
O Utility supply	• Centralized utility facility out side main building (only cooling water pump : inside main building)
O Electricity	• Centralized supply
O Secondary radwaste	• Liquid waste : disposal after solidification • Solid waste : disposal after encapsulation
O Layout	• Consider operability, maintainability and safety factor
O Lifetime	• 60 year

3.3.3.1. Design base

3.3.3.1.1. Storage method

Various storage method have been considered including the wet storage, metal cask, concrete silo, air cooled vault and dry well. The safety aspects was emphasized compared with technological and economical aspects. The result was that wet storage method was deemed to be more appropriate in consideration of various factors, especially licensing and schedules, among others.

3.3.1.1.2. Storage capacity

With the usual assumption that the life time of NPPs and the ISFSF are 30 years and 60 years respectively, the following result is obtained. The capacity of receiving area of the ISFSF should be enough to receive maximum anticipated amount of SF in the life time, with the possibility of expanding the storage area in the future. On the basis of this result, the optimum capacity of the ISFSF was decided to be initially 3,000 tU.

3.3.1.1.3. Transportation of spent fuel

Various transportation methods have been analyzed including the land transportation by truck or rail train, and the sea transportation by transportation ship. Since the maximum weight of loaded transportation vehicle is limited up to 40 ton, the truck transportation method is proven to be unacceptable. Also, the railroad transportation has no advantage if the high costs of new railroad construction is taken into account. Finally, the sea transportation is selected as the best option considering that all NPPs in Korea are located along the coast line and that this method can avoid traffic passage through highly populated area.

3.3.1.2. Design requirements

In determining the design requirements, various factors are considered such as NPP construction plan, storage capacity of AR storage facility, future policy of spent fuel management, current available technology, and safety margin etc. As a result the design requirements shown in Table 4 was obtained.

3.3.1.3. Technical features of the ISFSF

Based on the design base and design requirements, the conceptual design of the ISFSF has been conducted and its outlined feature is as follows. As shown in Fig.1 the ISFSF consists of a truck bay, a cooling pit, an unloading pool, a service pool for preparation of storage, and a storage pool. The storage pool is divided into two parts, one for PWR and another CANDU SF, respectively. Pool water is cooled by heat exchanger and cleaned by deep bed ion exchanger both outside the pool. Sea water is used as an ultimate heat sink when the facility is located at seashore. Inside of the pool wall is lined with stainless steel to assure leak tightness. Secondary waste arising from the facility is packaged and/or immobilized by cementation. Achieving high standards of nuclear safety is the major concern in designing this facility. Design goals are to limit the environmental release of radioactivity so that a member of the public in the regional vicinity group receive no more than 0.5msv/y, and to provide adequate biological protection so that the maximum average man-rem dose commitment to operating personnel is less than 0.5 rem/y.

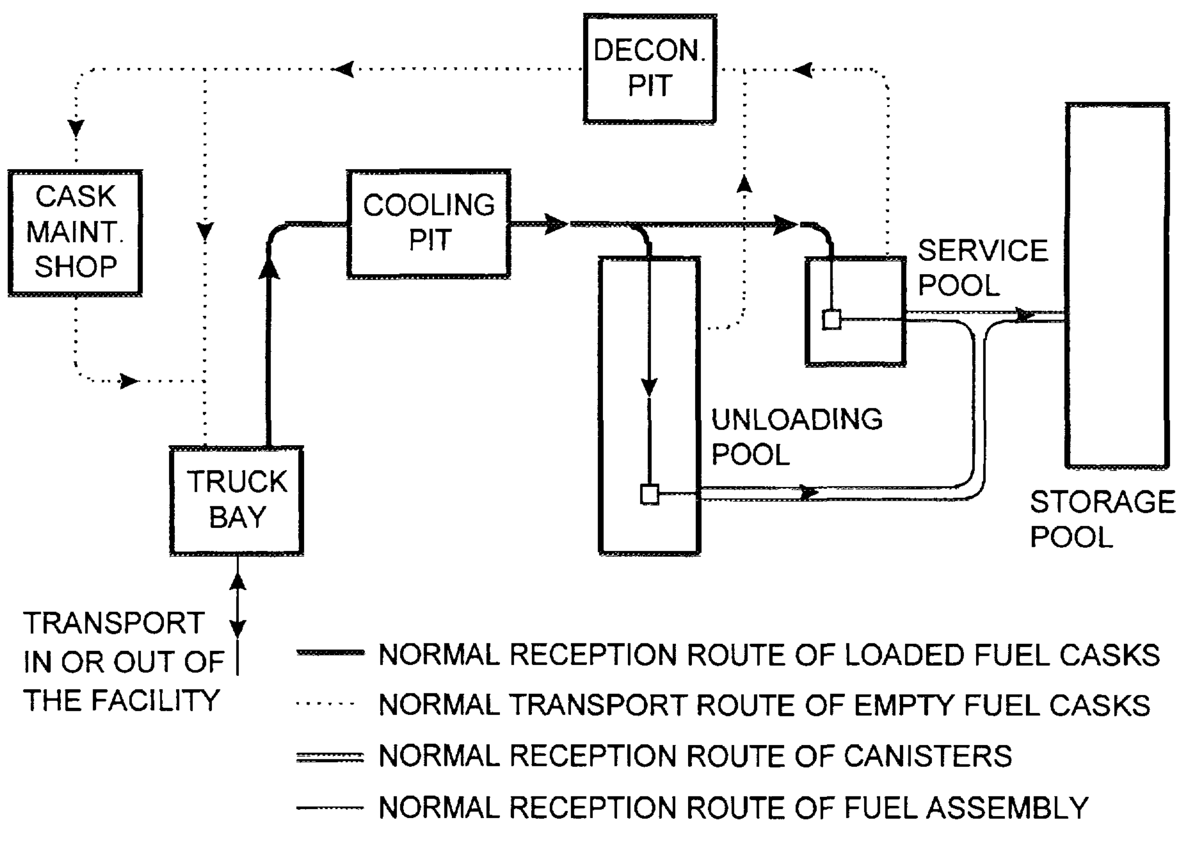

FIG. 1. Handling sequence diagram of spent fuel

3.3.2. Containerized storage option

While the site securing project had been delaying, the optimization study of the conceptual design had been conducted by KAERI with BNFL of UK [4]. In this study, the containerized storage method, developed by BNFL and applied in the THORP reprocessing facility, was examined. This method adopts the multi element bottle (MEB) which contains seven PWR assembly as shown in Fig. 2. The MEB can be used for dual purpose, transportation and storage. The designed lifetime of MEB is 40 years and is made by borate stainless steel. KAERI analyzed the impact on the conceptual design by introducing the MEB and compared the applicability to the ISFSF of containerized storage method with the non-containerized method in terms of technological and economical aspects. The technological comparison result is given in Table V. In economical aspects, the containerized method offers marginal cost savings over non-containerized one.

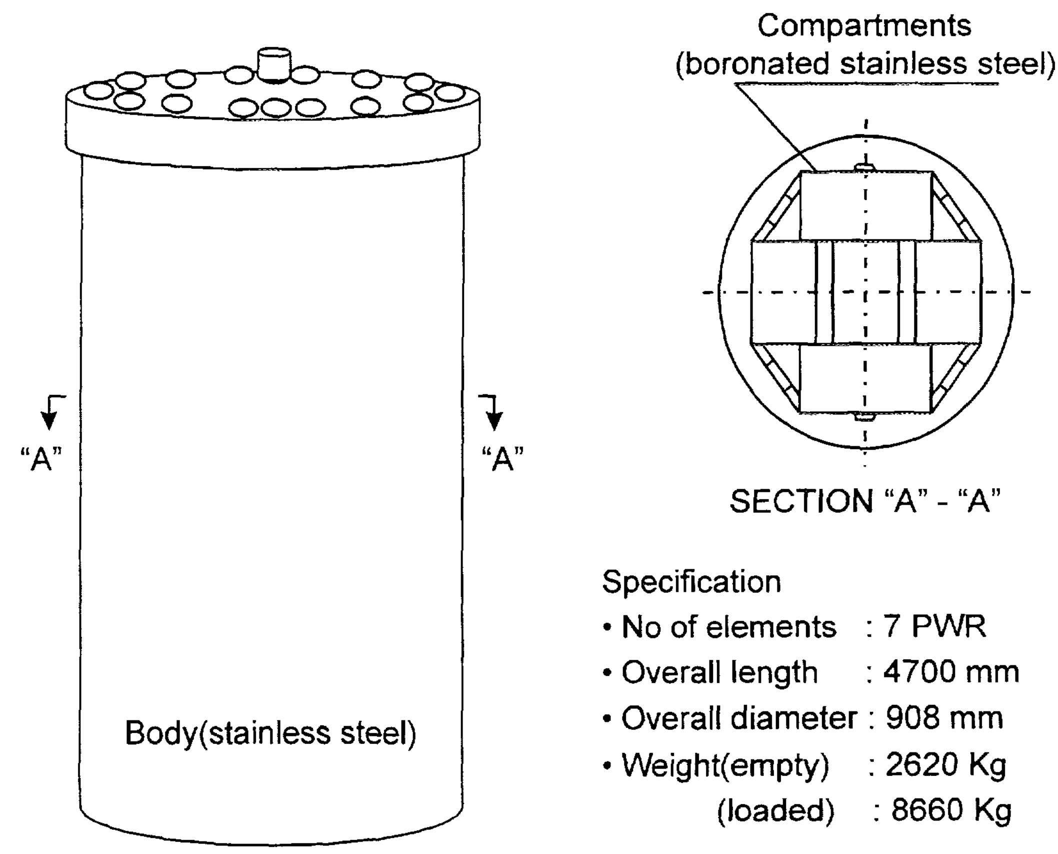

FIG. 2. Schematics of MEB (Multi Element Bottle)

3.3.3. Dry storage option

The dry storage method had been reviewed by KAERI to select the best option for SF storage in 1993. KOPEC and Ebasco in the USA participated in the comparative study on storage method has been conducted for PWR and CANDU SF.

TABLE V. RESULT OF ANALYSIS ON TECHNOLOGICAL ASPECT

Items	Result	
	Non-Containerized	Containerized
Difficulty in process operation	□	Δ
Availability of technology	O	Δ
Required Storage Volume	O	O
Cooling	Δ	O
Secondary waste generation	□	O
Simplicity of supplementary equipment and process	□	O
Construction period	Δ	O
Maintenance	□	O
Applicability at NPP	O	Δ
Possibility of extension	Δ	O
Spent fuel integrity at storage pool	O	Δ
Radiological safety	Δ	O
Nuclear safeguards	O	Δ

O Good Δ Slightly poor □ Poor

3.3.3.1. Comparative study on PWR spent fuel storage option

In this study, several alternatives such as Modular Vault Dry Storage (MVDS), Nutech Horizontal Modular Storage (NUHOMS), and water pool storage types were selected, and each of the alternatives were analyzed by preliminary conceptual design [5] for comparative assessment in terms of technical and economical aspects. The results of this assessment indicated that the water pool type was more favorable in consideration of such factors as safety and licensing, concept maturity, socioeconomic impacts, siting requirement and the flexibility to the future national spent fuel management policy. It also indicated that the dry types were more favorable in such factors as environmental impacts, ease of operation and maintenance, and the construction schedule. A quantitative evaluation was performed by assigning weighting factors and subsequent scoring on each criteria. The results showed that the water pool type had marginal advantage over the dry types in some technical aspects.

The results of the investment costs evaluation have shown that, in case of 4% of discount rate, the unit cost for water pool type and NUHOMS concept were nearly similar, but the unit cost for MVDS concept was the highest as $155-142/kgU for water pool type, $179-162/kgU for MVDS concept and $154-138/kgU for NUHOMS. On the other hand, in case of 8-10% of discount rate, the costs for water pool type and MVDS concept were nearly similar, but the unit cost for NUHOMS concept with high modulability was the smallest as $160-175/kgU for water pool type, $168-175/kgU for MVDS concept and $144-151/kgU for NUHOMS concept.

It has been concluded from these results that the water pool type was preferred in terms of technical aspects while the dry types were marginally preferred for investment point of view.

3.3.3.2. Comparative study on CANDU spent fuel storage option

For spent CANDU fuel, four options were selected as follows:

- storing CANDU SF together with PWR SF in a wet interim storage facility;
- storing CANDU SF separately;
- storing CANDU SF AR and storing PWR SF in the ISFSF;
- storing CANDU SF away from reactor and storing PWR SF in the ISFSF.

These options were assessed in technical and economical aspects. In technical aspects, the option of storing CANDU SF together with PWR SF in a wet interim storage facility requires more complex structure, system configuration and equipment capacity compared with the case of storing only PWR fuel due to the additional design requirements. In economic aspects, assessment was made by the comparison of life cycle costs and levelized unit cost. It was shown that storing CANDU SF and PWR SF separately (PWR fuel in interim wet storage, CANDU fuel in dry storage at reactor or away-form reactor) had advantage over storing both PWR and CANDU SF in the same wet interim storage facility.

It is concluded from these results that CANDU SF may better be stored separately from PWR SF in consideration of long-term management, and that appropriate method be applied for storage of CANDU SF.

4. SPENT FUEL TRANSPORTATION PROGRAMME

4.1. Transportation scenario

As previously described (Section 3.3.1.3), KAERI has analyzed the transportation method of SF from NPPs to the ISFSF [6]. In this study the sea transportation method has been chosen as the best option, since all NPPs are located along the coastline and the land transportation would require passing through highly populated areas.

Also, based on the amount of SF and radwaste, the transportation scenario has been analyzed. In this analysis, the amount of SF and low level waste to be transported was assumed as cumulative amount generated from 23 NPPs counting both the existing NPPs and NPPs to be constructed by 2006. The analytical result shows that transportation ship of 3,000 ton scale should transport 576 PWR assemblies, 18,360 CANDU fuel assemblies and up to 3,196 drums of radioactive waste on annual basis during the first operation stage.

4.2. Transportation cask

Two casks models, i.e. KSC-1 (1986) and KSC-4 (1990), have been designed by KAERI and fabricated by domestic heavy industrial companies [7]. The design parameters have been determined by computer-aided calculations as well as mock-up tests. In the course of design and fabrication stages, licensing procedures have been implemented in compliance with the rules and regulations as ordained by the Atomic Energy Law of Korea.

The wet-type KSC-1, which weighs 28 tons, is capable of loading one PWR SF assembly with a burnup of 40,000 MWd/tU and a cooling time of one year. Major shielding materials used for the cask were lead against gamma-rays and water against neutrons. The transportation system including the KSC-1 cask has been used to transport seven SF assemblies and one basket with 46 failed fuel rods from PWR nuclear power plants (NPPs) to the post-irradiation examination (PIE) facility of KAERI for hot cell examination under strict rules and regulations for nuclear material safeguards.

The KSC-4 weighs 37 tons and is capable of loading four PWR SF assemblies with a burnup of 38,000 MWd/tU and a cooling time of three years [5]. It can be used in both dry and wet conditions. *Major shielding material against gamma-rays is lead and that against neutrons is hydrogen-rich resin.* The design specifications are summarized in Table VI. Using the KSC-4 casks, a total of 312 PWR SF assemblies has been transshipped to date from Kori-1 to Kori-3 and Kori-4 NPPs.

No technical problems have so far been encountered with the KSC-1 and KSC-4 casks during loading, transportation and unloading of the SF. Design of the KSC-7 cask for PWR SF has recently been completed (1995) and its specification is given on Table VII.

TABLE VI. DESIGN SPECIFICATIONS OF KSC-4 CASK

Items	Specifications
Type of package	• B(U) type, fissile class III
Loaded weight	• 37 ton
Size (m)	
• outside diameter	• 1.35
• height	• 4.82
Materials	
• shells	• SA 240 type 304
• fuel basket	• borax/borate SS
• gamma-rays shield	• lead casting
• neutron shield	• hydrogenous resin
Fuel	
• type	• PWR assembly
• max. burnup	• 38,000 MWd/tU
• cooling time	• 3 years
• radioactive decay heat	• 7 kW
• radioactivity	• 70.3 PBq

TABLE VII. DESIGN SPECIFICATIONS OF KSC-7 CASK

Items	Specifications
Type of package	• B(U) type, fissile class III
Loaded weight	• 70 ton
Size (m)	
• outside diameter	• 1.86
• height	• 5.40
Materials	
• shells	• stainless steel, type 304
• fuel basket	• borax/borate SS
• gamma-rays shield	• lead
• neutron shield	• silicone mixture
Fuel	
• type	• PWR assembly
• max. burnup	• 50,000 MWd/tU
• cooling time	• 1.5 years
• radioactive decay heat	• 32.3 kW
• radiation	• 2.53E+17 photons/sec • 3.12E+09 neutrons/sec

4.3. Cask handling devices

To apply the KSC-1 and KSC-7 to transshipment of SF between NPPs and from NPP to PIEF of KAERI, several devices were designed and fabricated [8]. They are a handling tool of SF, a lifting yoke and a lid handling device. Also, cask maintenance tools such as an internal basket decontamination system, a surface decontamination system, a leak test system and a internal cavity drying system, etc. have been developed. Since these devices are manually operated, there is potential risk of operator exposure to ionizing radiation. However, no severe problem has been encountered during unloading and loading operation. Major devices developed for the cask operation are as follows.

4.3.1. Handling tool of spent fuel assembly

As shown in Fig. 3, a handling tool is used for grappling and transferring one spent FA from storage pool to load into cask or for reverse operation. The handling tool 8.9m long is made of stainless steel. This tool is hooked on the chain of crane and can be manually operated outside the pool.

4.3.2. Lifting yoke

A lifting yoke shown in Fig. 4 is used for grappling the cask when loading or unloading it from transportation truck and the loading pool. This device consists of a main frame, an arm assembly which grapples the cask trunnion, and a guide plate which is used for guide of yoke arm to the trunnion of cask. Two arms are opened by a hydraulic cylinder simultaneously. Two eccentrical holes are provided at the end of each arm, a larger one for insertion and a smaller one for holding and lifting trunnions after the grapples are safely inserted. The lifting yoke weighs one ton and has a maximum lifting capacity over 100 ton which is almost three times that of cask.

4.3.3. Lid handling tool

A cask lid weighs one ton and has an eye bolt at its center. When lifting the lid, the lid handling tool shown in Fig. 5 is employed. This tool has a 3 ton capacity of lifting weight which is three times that of the cask lid. A gripper driven by a hydraulic cylinder is installed at the hook of the lid handling tool and its motion is remotely controlled by manipulating a handle at the other end of the lid handling tool.

4.3.4. Internal basket decontamination system

An internal basket decontamination system has been developed. This system consists of a decontamination brush, a high pressure water pump and filter system as shown in Fig. 6. The decontamination brush with high pressure (180 kg/cm^2) water nozzle, which is attached at the decontamination equipment housing, cleans the internal surface of a basket while rotating and moving into the vertical direction inside basket. This sequence is repeated until four baskets are decontaminated. The contaminated particles are sucked into the water tank through the filter bed and the purified water is recirculated by a high pressure water pump. The flow rate of pump is about 16.7 lpm. The contaminated filter assembly is replaced by using the specially designed filter handling cask. The performance of this system has been investigated by applying the contaminated KSC-4 which has been employed in transshipment operation of 156 spent FAs between NPPs. Table 8 show the amount radioactivity reduction inside the basket as a result of decontamination. As shown in Table 8 the radioactivity is reduced to 0.175 $\mu Ci/100cm^2$ after decontamination. As compared with the level before decontamination (2.38 $\mu Ci/100cm^2$), the performance of this system is proven to be affordable.

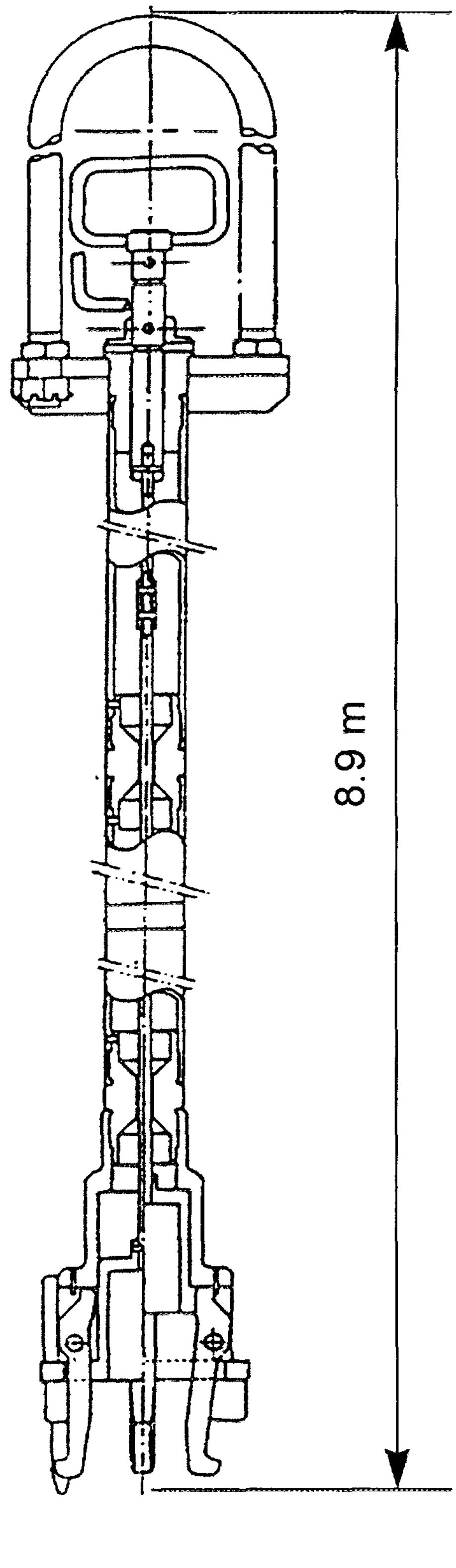

FIG. 3. Spent fuel assembly handling tool

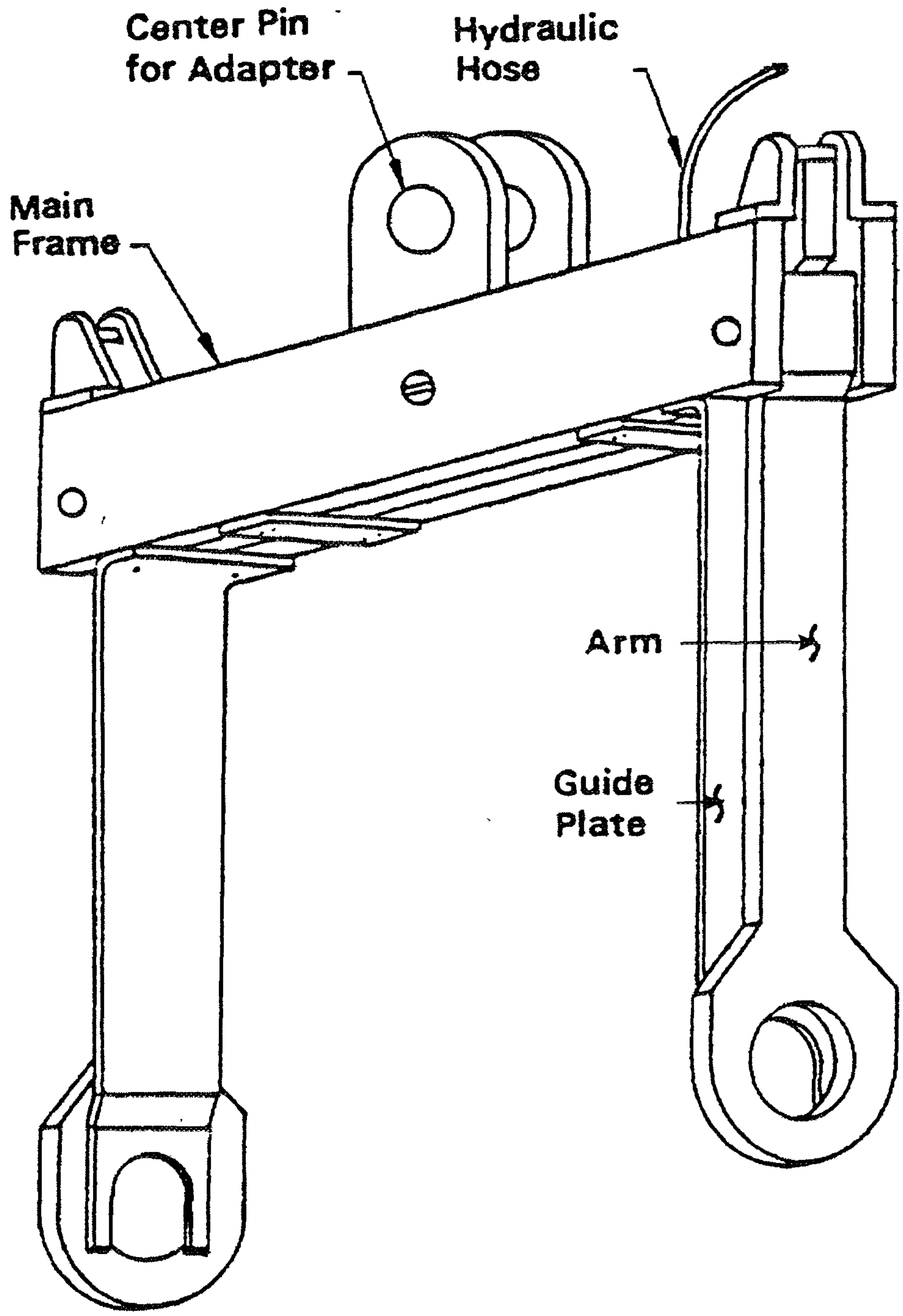

FIG. 4. Cask lifting yoke

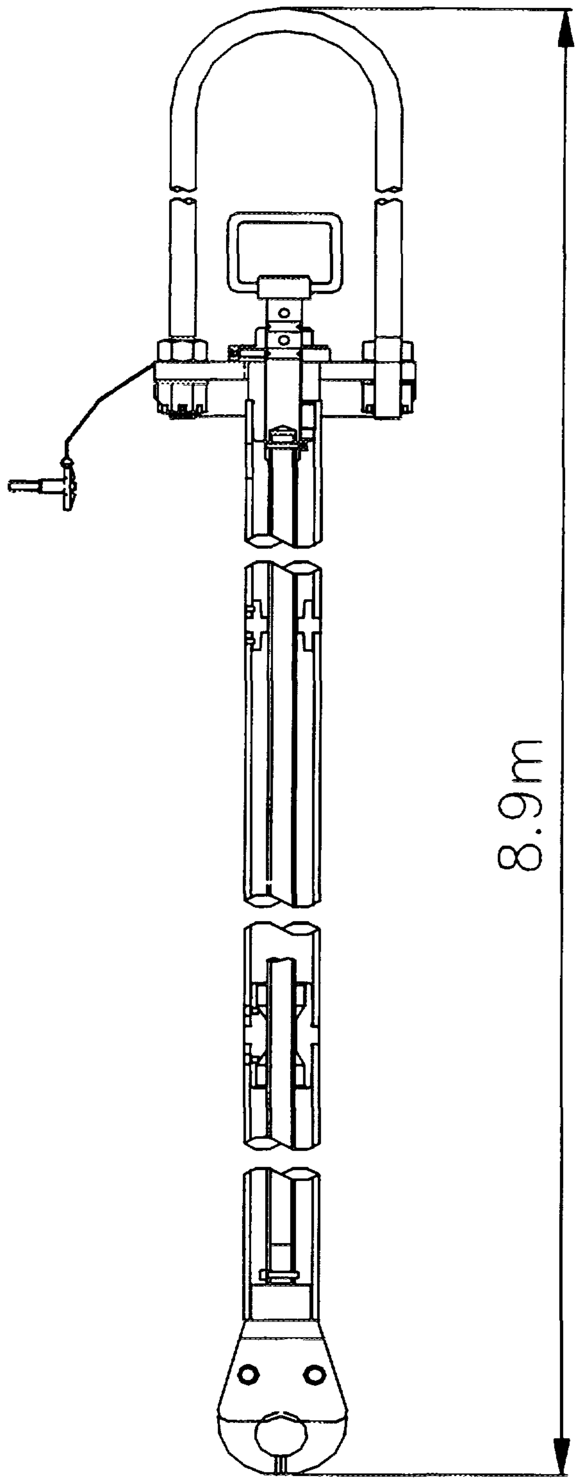

FIG. 5. Cask lid handling tool

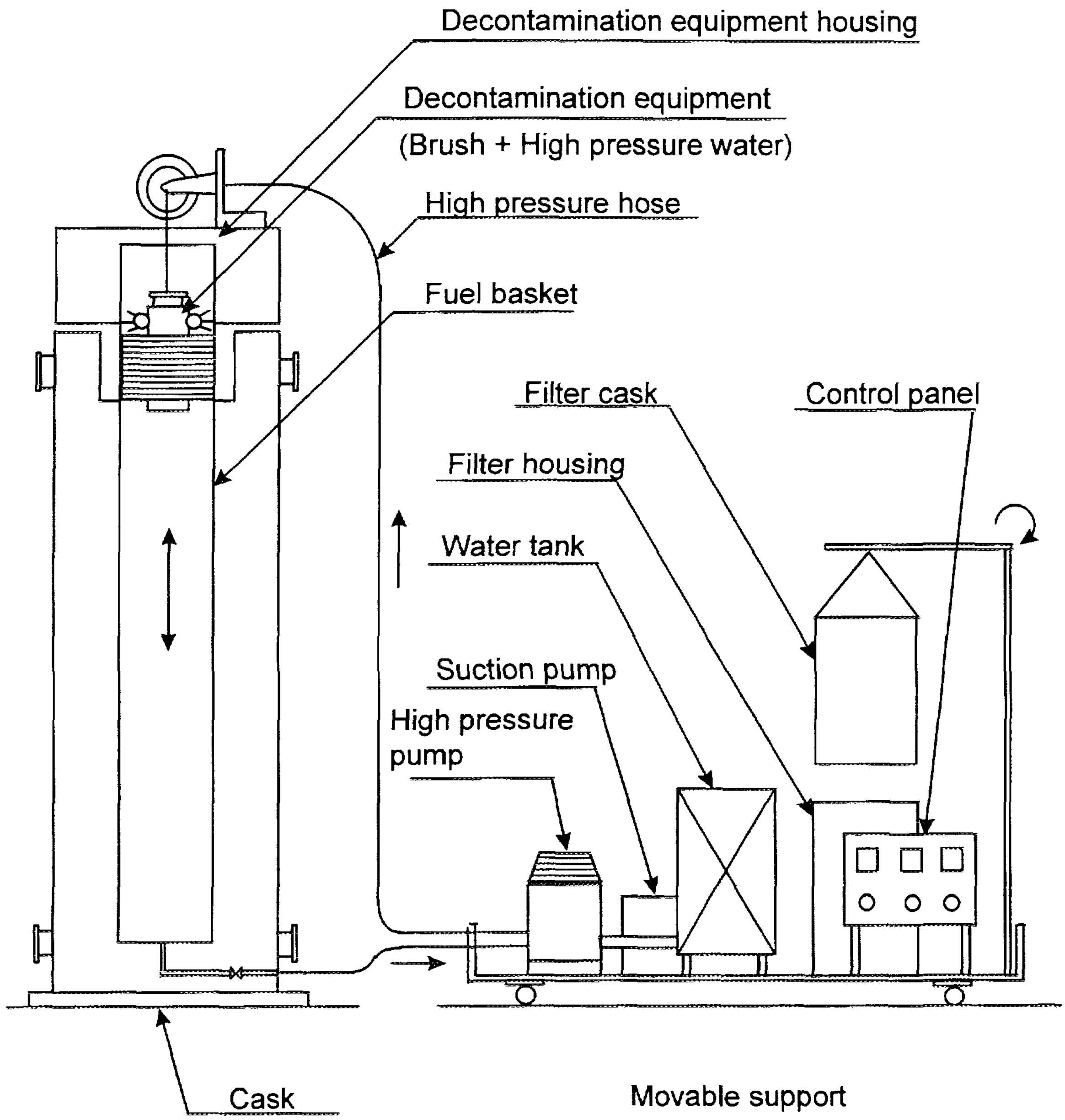

FIG. 6. Internal basket decontamination system

TABLE VIII. COMPARISON OF RADIOACTIVITY OF BASKET BEFORE AND AFTER DECONTAMINATION

Nuclides	Radioactivity ($\mu Ci/100cm^2$)	
	Before Decontamination	Post Decontamination
AM - 241	2.762 E - 03	0
CE - 144	8.667 E - 03	1.094 E - 03
CO - 60	2.303 E - 00	1.718 E - 03
CS - 137	1.313 E - 02	0
MN - 54	7.497 E - 03	0
RU - 106	2.292 E - 02	0
SB - 125	2.089 E - 02	2.177 E - 03
Total	2.381	0.175
D.F. (Decontamination Factor) : 13.6		

4.4. R&D activities for remote cask handling devices

A wide variety of sophisticated handling devices for both spent FAs and casks have been developed for minimizing the radiation exposure to the operators. The devices developed are an anti-swing crane and a Remote Cask Grappling and Lid Unbolting Device (RECGUD). The state of art technology has been adopted and these devices can be employed in NPPs for remote cask handling in near future.

4.4 1. Anti-swing crane

In order to remove the load swinging motion that could hinder operational safety as well as efficiency, an anti-swing crane has been developed. As shown in Fig. 7, the anti-swing crane system consists of a conventional crane, a swing angle measuring device and a control system [9,10]. The swing angle measuring device [11] shown in Fig. 8 works upon two point laser distance sensors attached on the rope and reflection plate attached on the trolley. As the rope swings, the laser sensor also swings in such a way to measure the linear distance changes between laser source and reflection plate. This changes can be translated into the swing angle by the use of geometric calculation. Fig. 9 presents the schematics of control system. Two analog to digital converters are used for capturing the swing angle signal from two laser sensors. Three pulse generators are used for the control commands to each motors and seven digital input device are used for the limit switches signal. Also, three sets of digital counters and frequency to voltage converters are used for the position and velocity input of trolley in each directions. All of these devices were installed in a 386 PC.

Several anti-swing control algorithms using open-loop and closed-loop approaches have been developed and implemented using a one ton scale anti-swing crane system. These algorithms are an acceleration profile planning [12], a pre-programmed velocity feedback controller [12], a fuzzy controller [13], and the hybrid anti-swing/position controller [14,15]. The performance of these control algorithms is verified through a series of simulations and experiments. The results show that the swinging and position errors of load are greatly reduced as compared with those of conventional crane KAERI has designed a commercialized version of the above-mentioned controllers by integrating various functions of these controllers into a dedicated firmware and has transferred the anti-swing technology to the industrial sectors Currently, a much large scale anti-swing crane is now being developed for application in the factory automation process.

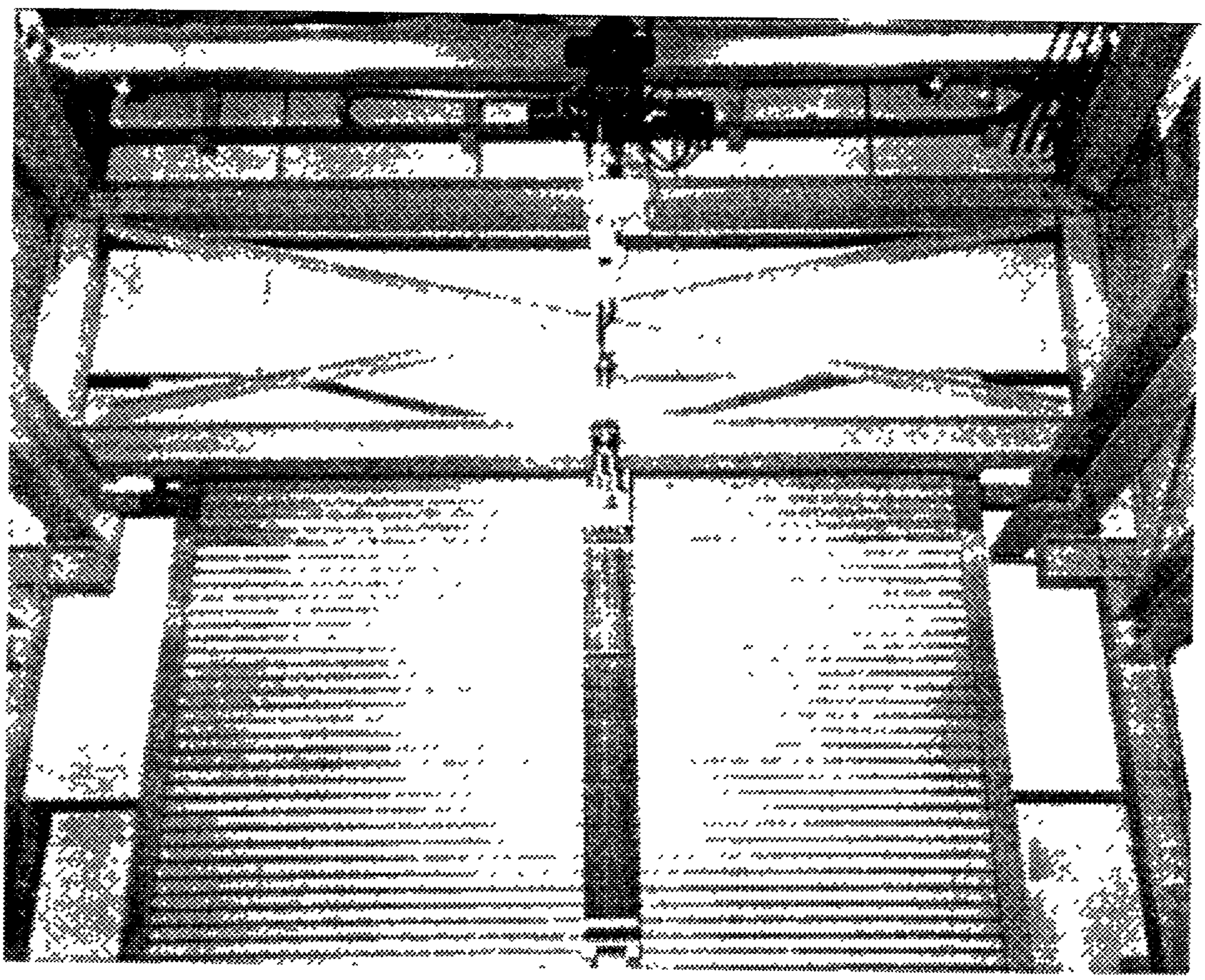

FIG 7 Anti-swing crane transporting fuel assembly

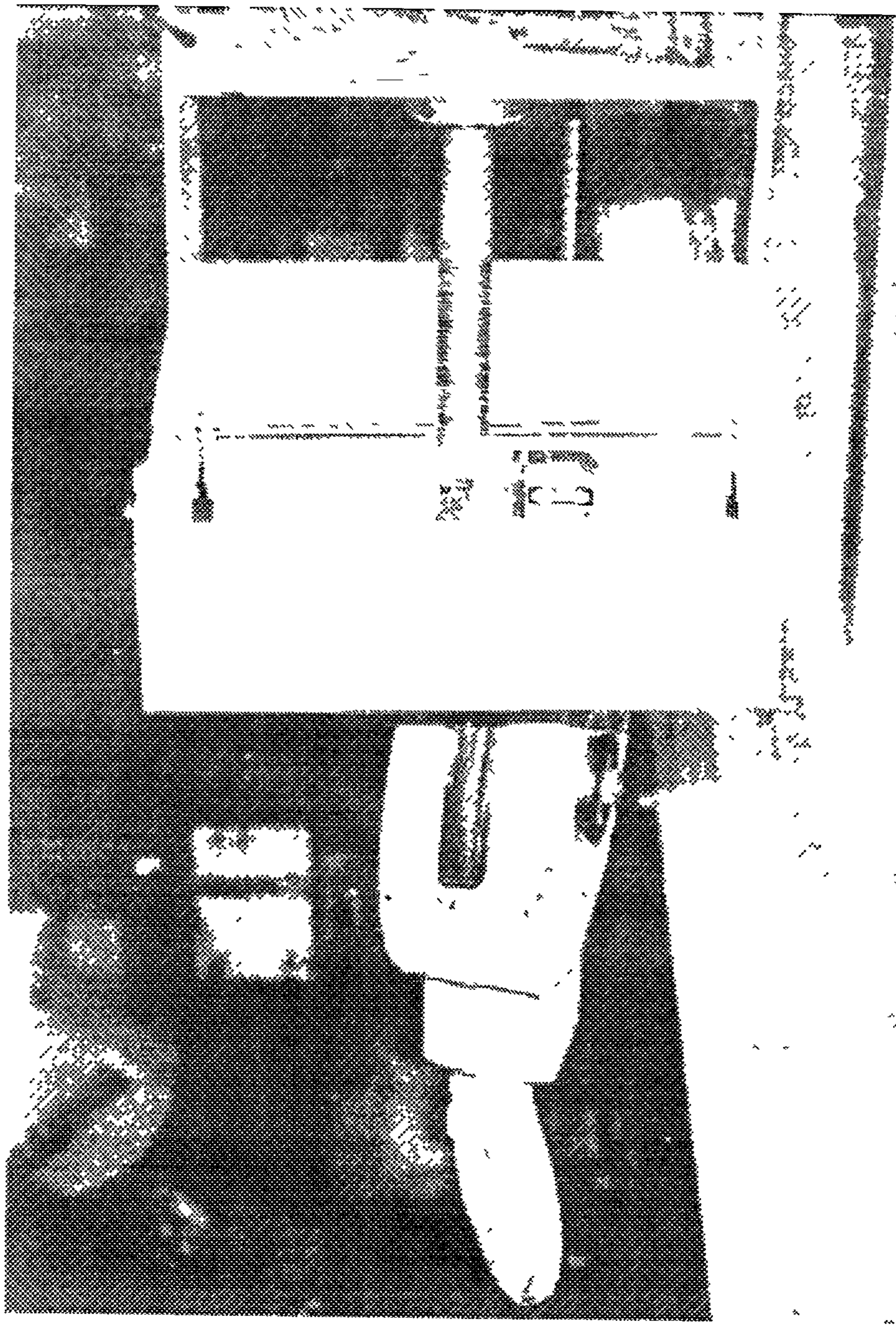

FIG 8 Swing angle measuring device using laser displacement sensor

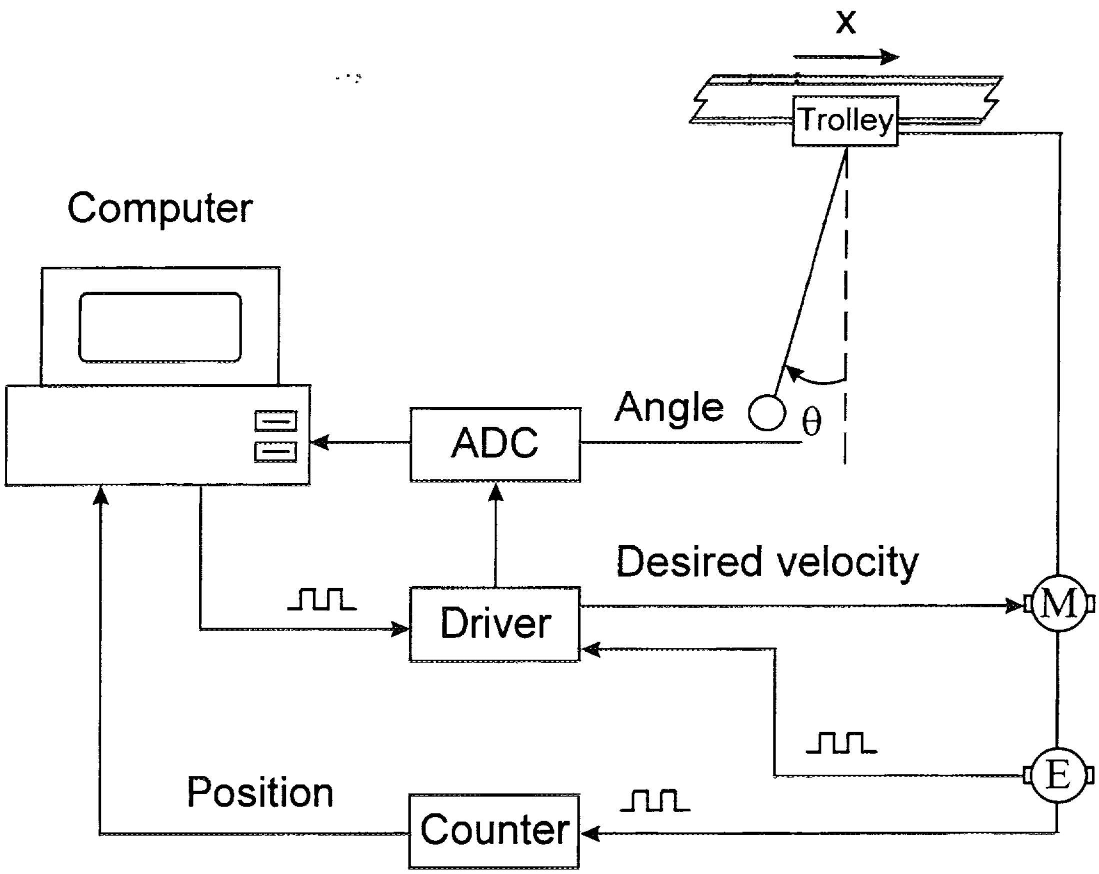

FIG 9. Schematic diagram of control system

4.4.2. Remote cask grappling and lid unbolting devices

Among various handling tasks associated with cask handling operation, moving the cask and unbolting the cask lid are the most tedious and thereby the most significant contributor to the occupational exposure to the radiation. These tasks were selected as a R&D item to which robotics technologies are applicable. Therefore, along with the anti-swing crane, a Remote Cask Grappling and Lid Unbolting Devices (RECGUD) has been developed as a dedicated device capable of precisely grappling the cask trunnion and unbolting the lid bolt on the cask lid in a fully remote and automatic manner while it is suspended by the anti-swing crane.

4.4.2.1. Design considerations

The RECGUD is designed in such a way that its application requires no special modification on neither the cask nor the cask handling facility. Therefore, the device can be adopted to the current cask handling circumstances without much technological refurbishment. The lid of KSC-4 is bolted by 16 stud bolts with 50 kgf.m torque. The model cask is shorter than the actual one and it's weight is 400 kg. But, the mechanical and geometrical features of its lid are identical to the KSC-4. The prototype of RECGUD is designed to have a 1 ton payload capacity to carry the model cask of KSC-4 and a resolution of 2 mm on all axes for accurate grappling and fine positioning of the end effector. Also it's torque wrench module is designed to have over 50 kgf.m torque.

4.4.2.2. Mechanical structure

As shown in Figures 10 and 11 the RECGUD consists of a main body, two grapples, a wrench guide, a torque wrench module, and a bolt tray. The main body, the frame for the device, is suspended horizontally at the rotation axis to a anti swing crane. At both ends of the body, two grapples are installed vertically. On the same rotation axis, the wrench guide is installed parallel to and below the body. At the end of rotation axis, a CCD vision camera, an ultrasonic sensor, a laser sensor, and a bolt tray are mounted.

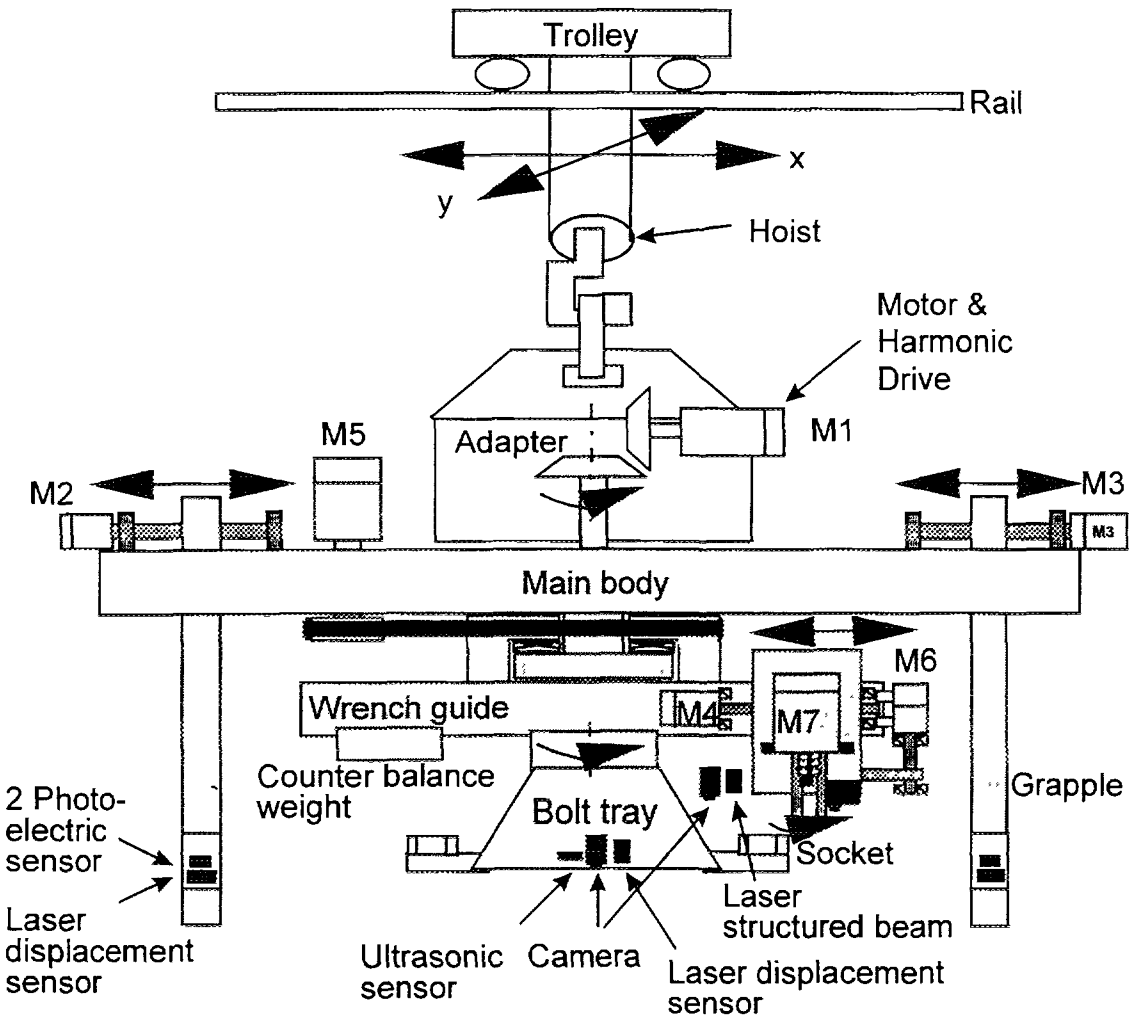

FIG. 10. Structure of RECGUD

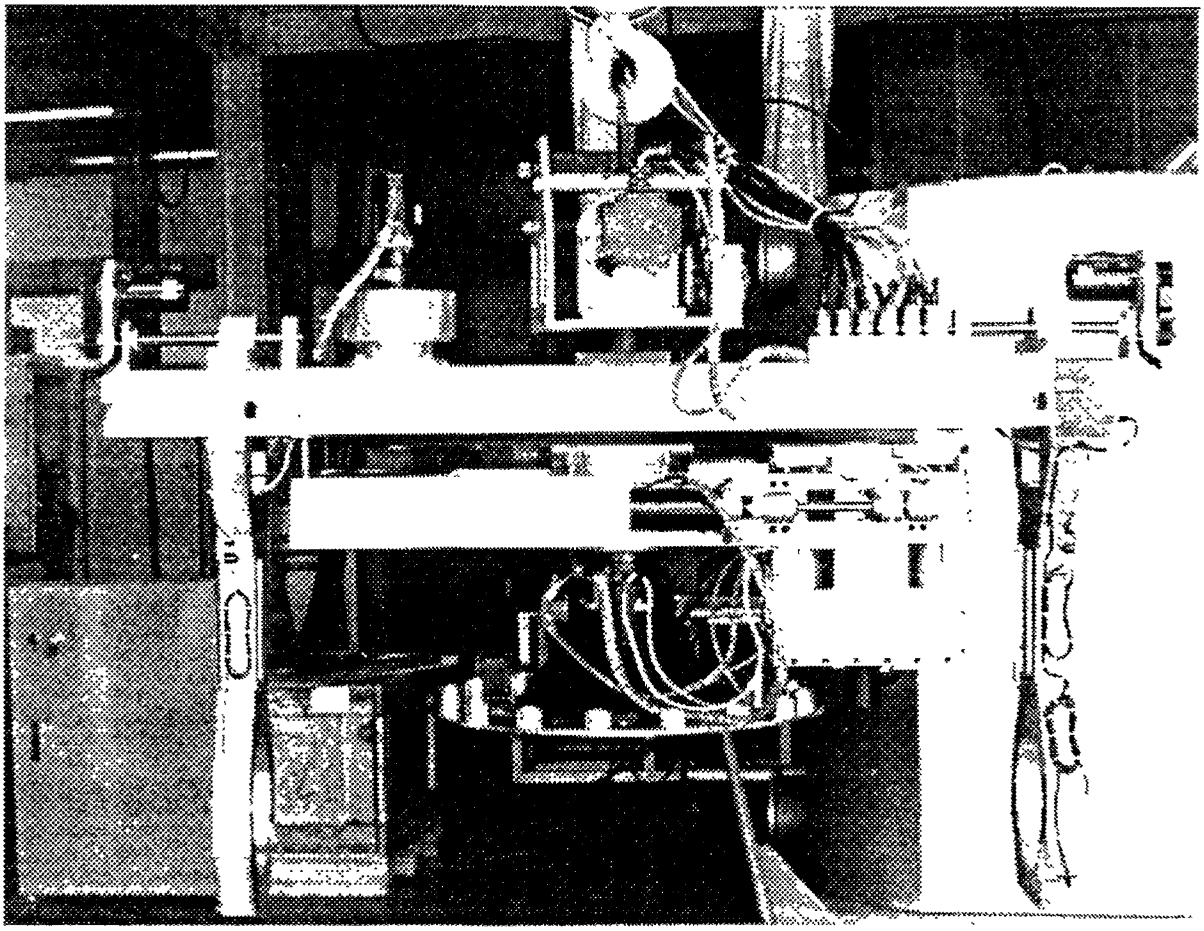

FIG. 11. Photography of RECGUD

The grapples have a grapple hole at their lower part. The grapple motors actuate each grapple independently along the longitudinal axis of body until the grapple hole is inserted over the trunnion. At one end of the longitudinal axis of wrench guide, the torque wrench module is installed. The guide is mounted with an ac servo motor for translating the torque wrench module along the longitudinal axis of guide. The wrench guide rotating motor actuates the guide to rotate about the rotation axis of body. The rotation of guide allows the torque wrench module to move along the circumference of cask lid. At the other end of the longitudinal axis of guide, a balancing weight is mounted. The weight plays the role of balancing the device while the device is carried by the crane.

For adopting a socket wrench, an adapter is mounted to the spindle of the motor. The socket is twelve-facet which makes the mating of the socket to the bolts easier. The depth of socket is 90 mm which is deep enough to accommodate both a loosened bolt and a tool for holding the loosened bolt. As shown in Fig. 12, the tool consists of a spring and an electric magnet. The spring is mounted in the socket and the electric magnet for holding the loosened bolt is attached at the end of spring.

The bolt tray with 16 bolt holes where the loosened bolts are collected in order is mounted at the end of rotation axis of body. As the socket approaches the bolt tray, the overall positioning imprecision must be accommodated by the taper of bolt hole.

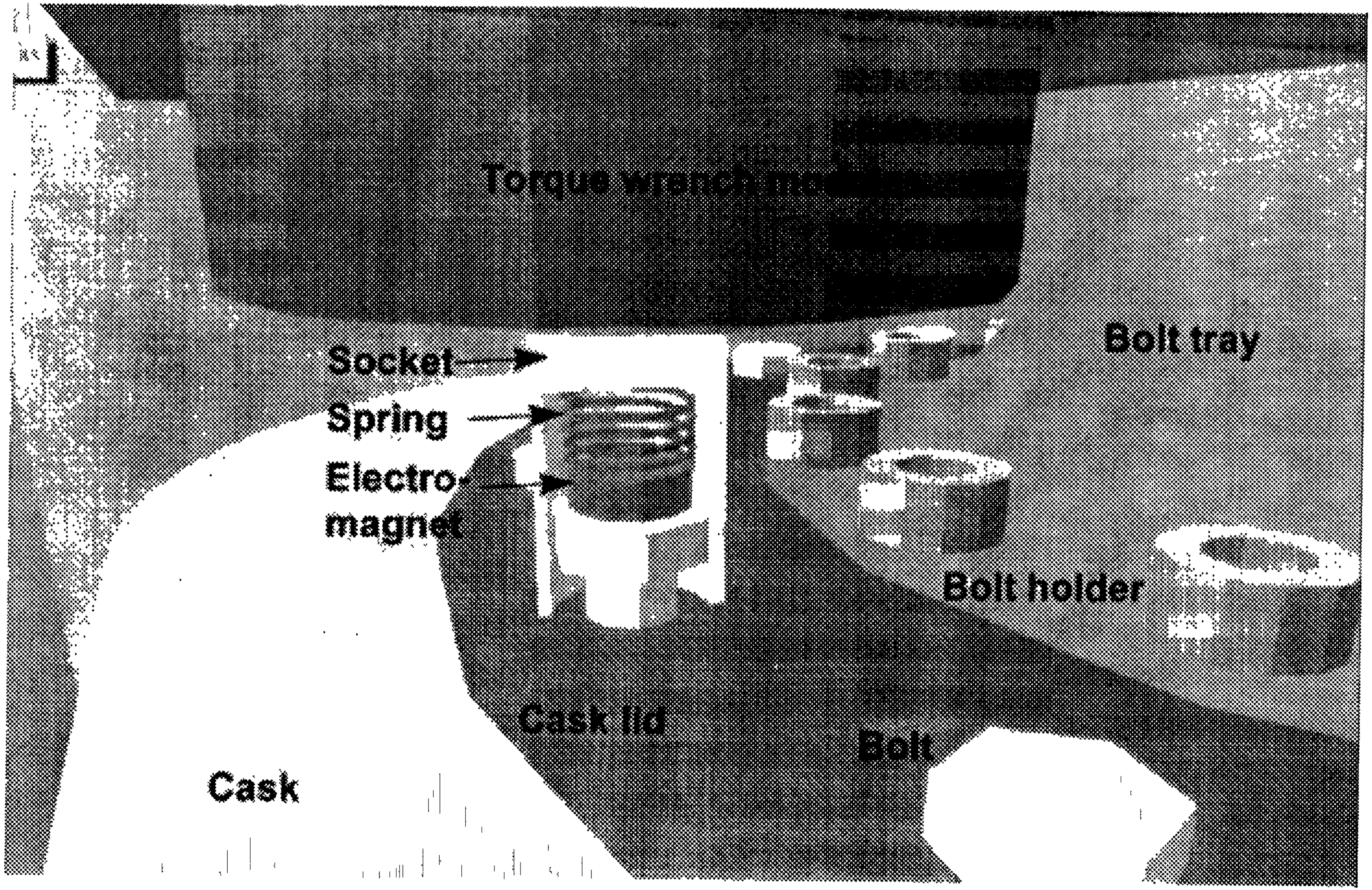

FIG. 12. Electro-magnet and spring inside socket

4.4.2.3. Sensors

The RECGUD is equipped with various sensors for the information about the position of cask and lid bolts and the orientation of cask. The device uses an ultrasonic sensor, five laser displacement sensors, four photoelectric sensors, and two CCD vision cameras.

The ultrasonic sensor in used to calculate the vertical distance between the device and the cask lid until the distance is short enough such that a laser displacement sensor can be used to measure a shorter distance.

Once the RECGUD stops at the exact position, the axis through the centers of two grappling holes and the axis through the centers of two cask trunnions are on the same horizontal plane. Then, the grapples can insert the trunnions, once the orientation of the device is corrected with respect to that of cask. The other laser sensors are for measuring horizontal distance between the grapple and the cask so that the grapple can stop precisely inserting into the cask trunnion

The photoelectric sensors are for measuring the position and orientation errors of the grapples to the trunnion. The output signals from the sensors which are mounted around the grappling holes toward trunnion are compared to calculate these errors by adopting neural network algorithm.

Two CCD vision cameras are used for the device. The first camera, the device camera, mounted at the end of rotation axis of body, is used to continuously identify the planar location of the cask. Based on this information, the crane is guided to place the device over the cask such that the center of device is aligned with that of cask lid. Then, the orientation of cask is identified. The second camera, the torque wrench camera, mounted on the casing of torque wrench module, is used to identify the location of the bolt and to position the wrench module over the bolt head. For the camera, a strip of laser beam is projected over the top surface and the location of the bolt head is identified by analyzing the distorted pattern perceived by the camera.

4.4.2.4. Control system

As shown in Fig. 13 the control system consists of an actuator subsystem, a sensor subsystem, and a control subsystem. In the actuator subsystem, the rotational axes of device are driven by ac servo motors coupled with speed reducers and the translational axes of device are driven by ac servo motors coupled with ball-screws. The sensor subsystem comprises various sensors and cameras which are used for positioning each part of the device. The control subsystem is designed to be controlled either by a computer keyboard or by a hand controller. The command signals from the hand controller are transmitted through PC-bus to Programmable Mullet-Axes Controller (PMAC). Signals sent by sensors are collected and analyzed in PMAC. Then, the control computer runs on a programme to manipulate the data and output it to the device through PMAC which uses a proportional and integral controller to compensate the control loop. In addition, the desired movement of RECGUD can be performed by entering the absolute coordinate of each axis of device through the keyboard. The communication between the control computer of the device and the crane controller is coordinated via RS-232 serial link.

4.4.2.5. Control algorithms

Various control algorithm have been developed to facilitate the operation of RECGUD such as: the image processing algorithm for the identifying the location of the cask and bolt, the neural network for recognizing the offset position between the hole of grapple and the cask trunnion, and the rotational control algorithms for reducing the operation time of positioning the grapple to the cask trunnion.

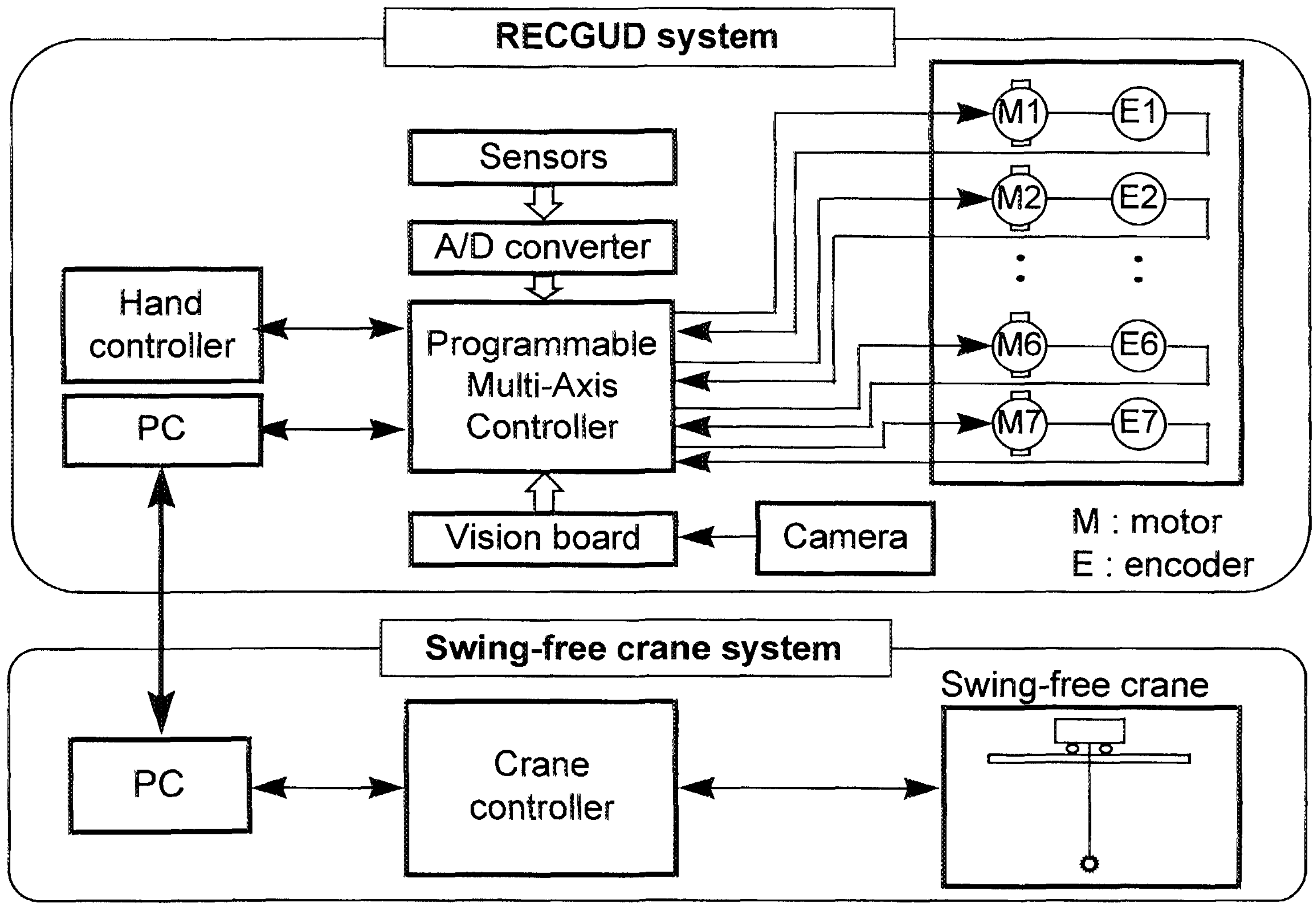

FIG. 13. Schematic diagram of RECGUD control system

The planar location of cask is identified by processing the image from a CCD camera mounted at the end of rotation axis of main body [16]. The conventional image processing technology is modified by introducing the pattern recognition algorithm. This algorithm is used mainly to improve the identification resolution in practical environment where the light is non-uniform. The identification results is affordable such that the accuracy of position error between two centers of RECGUD and the cask is proven to be within 1 cm at a condition that the distance between them is 4 m.

A grapple device, a component of RECGUD, has to be inserted over two trunnions anchored diagonally on the cask wall in order to lift the cask for transportation and unbolting/fastening the cask lid. However, an insertion of a grapple over trunnions is difficult due to position and orientation errors of the center position of the grapple device caused by incongruity with the center of the cask. To deal with this problem, a neural network is used to predict position errors of the grapple device using photoelectric sensors installed on the grapples [17]. Neural network training is performed to infer a mapping between sensor values and position and orientation errors. These estimated errors are to provide control inputs to correct the center position of the grapple device. Data is obtained by using a half scale apparatus that simulates the grapple device of RECGUD and trunnions of a cask. Results show that the trained neural network is able to estimate the position and orientation errors of the grapple device's center with accuracy of below 0.2 cm when presented with untrained sensor inputs, i.e., new locations of the grapple device.

Since RECGUD is suspended to an overhead crane, its body should undergo a prolonged oscillation upon actuation in rotational direction and it becomes difficult to achieve precise grappling of the cask. To suppress the rotational oscillation of the body, an open loop input shaping technique has been developed [18]. This method can rapidly suppress the rotational oscillation within one cycle of oscillation period even though the system dynamics is not precisely modeled.

5. CONCLUDING REMARKS

Korean spent fuel management programme is reviewed including the management policy. KAERI designated to a national radwaste management organization by government has tried to secure the site for LLW disposal facility and the ISFSF. It has finished the conceptual design of the ISFSF and updated the design to make it a lot simpler to operate and cheaper to construct and operate over its full life cycle by conducting the optimization and comparative studies. Also, KAERI developed various technology which are required for the ISFSF design such as transportation and wet storage technology. After the radwaste management programme was transferred from KAERI to KEPCO. KEPCO is reviewing the feasibility of site and the ISFSF.

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REMOTE TECHNOLOGY IN RBMK-1000 SPENT FUEL MANAGEMENT AT NPP SITE

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Abstract

The report describes the remote technologies employed in the nuclear power plant with RBMK-1000 type. Spent fuel transfer and handling operations at reactor (AR) and away from reactor (AFR) on reactor site (RS) facilities are illustrated by the example of the Leningradskaya NPP and are typical for all NPPs with RBMK-1000. The current approach to spent fuel management at NPP sites is also presented.

1. INTRODUCTION

There are 11 commercial power reactors of RBMK-1000 type now operating in the Russian Federation: 4 power units at the Leningradskaya NPP, 3 units at the Smolenskaya NPP and 4 units at the Kurskaya NPP (commissioning of the 5th unit is also planned). Spent fuel (SF) arisings amount to 550 tU/yr. and the total of 7,700 tU are currently stored at the on-site facilities of the power plants.

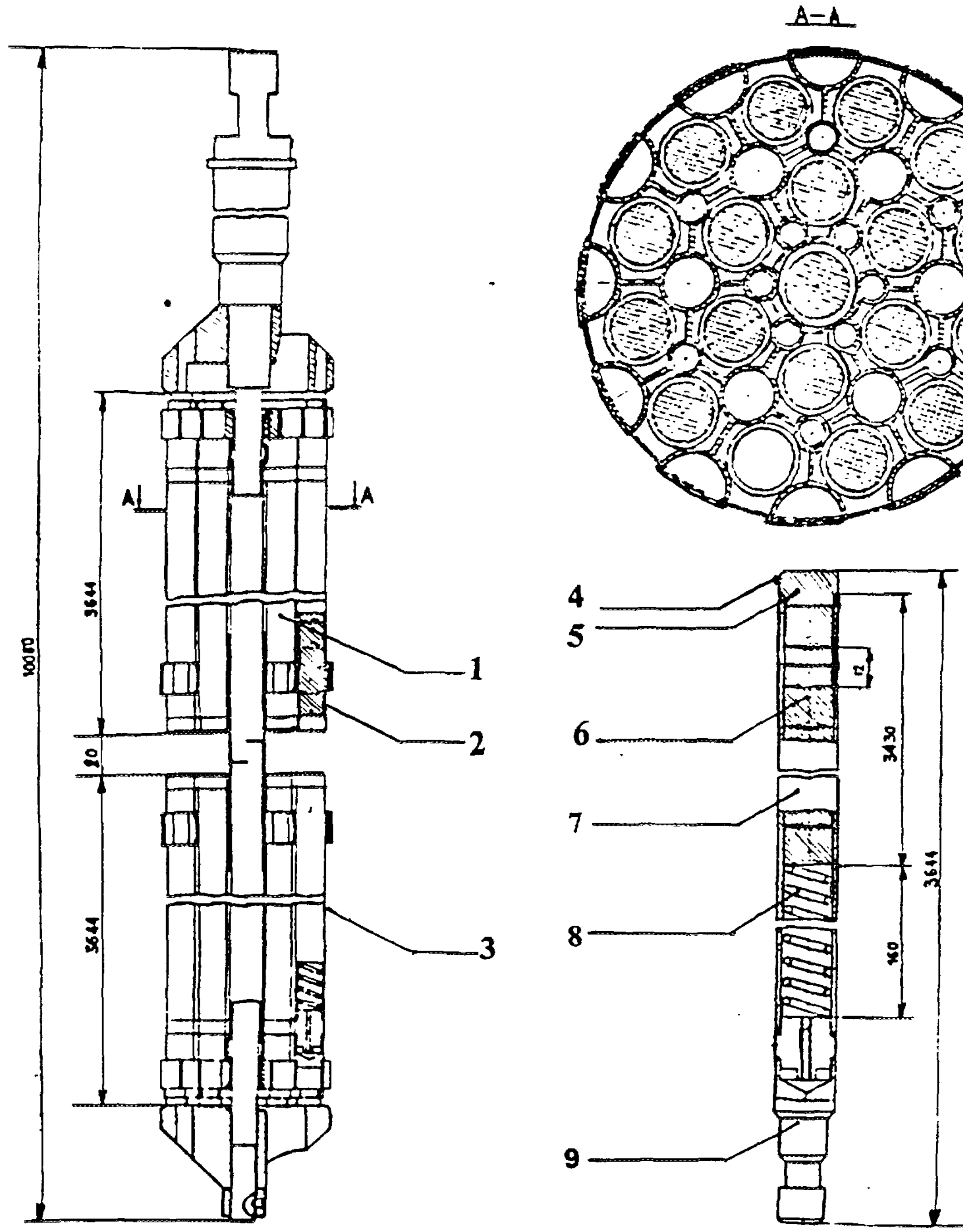
Nuclear fuel of RBMK-1000 type is fabricated from UO_2 in the form of pellets in a leak-tight Zr+1%Nb cladding. The RBMK-1000 fuel assembly consists of 2 bundles of fuel rods (18 rods per bundle) spaced by grids and enclosed into a leak-tight Zr+1%Nb hull. The active zone of the assembly is about 3.5 m (Fig. 1).

The features of RBMK-1000 fuel assemblies (FAs) and fuel rods bundles are listed in Table I. The main characteristics of RBMK-1000 reactors are listed in Table II. The choice of equipment and conditions for SF management largely depend on the characteristics of spent fuel discharged (fuel activity, residual heat release, defectiveness). The fuel discharged from reactors is transferred to AR pools for cooling and reducing the radioactivity level.

The current practice of fuel integrity monitoring (FIM) at the NPP with RBMK-1000 reactors involves a three-step procedure. This approach provides operative monitoring of the core conditions and rejection of failed fuel. The 1st step involves the monitoring of fuel rods behavior in the core of the operating reactors. FIM is performed with the use of regular equipment for a fuel rods batch or a reactor channel control. The 2nd step is aimed at the detection, if necessary, of leaking FAs and is performed at the reactor shut down and cooled. The 3rd step involves the monitoring of the fuel condition in water-filled storage pools (monitoring of water radiochemistry in pool water and cans with FAs). Leaking and damaged FAs discharged from the reactor are stored at the cooling pools in sealed cans.

2. SPENT FUEL TRANSFER AND HANDLING OPERATIONS

Spent fuel transfer and handling operations at AR and AFR-RS facilities are illustrated by the example of the Leningradskaya NPP and are typical for all NPPs with RBMK-1000 reactors.



1. fuel assembly
2. upper bundle of fuel rods
3. lower bundle of fuel rods

4. fuel rod
5. drown
6. Fuel pellet
7. cladding
8. spring
9. end fitting

FIG. 1. RBMK-1000 fuel assembly

TABLE I. FEATURE OF FAs AND FUEL ROD BUNDLES DESIGN

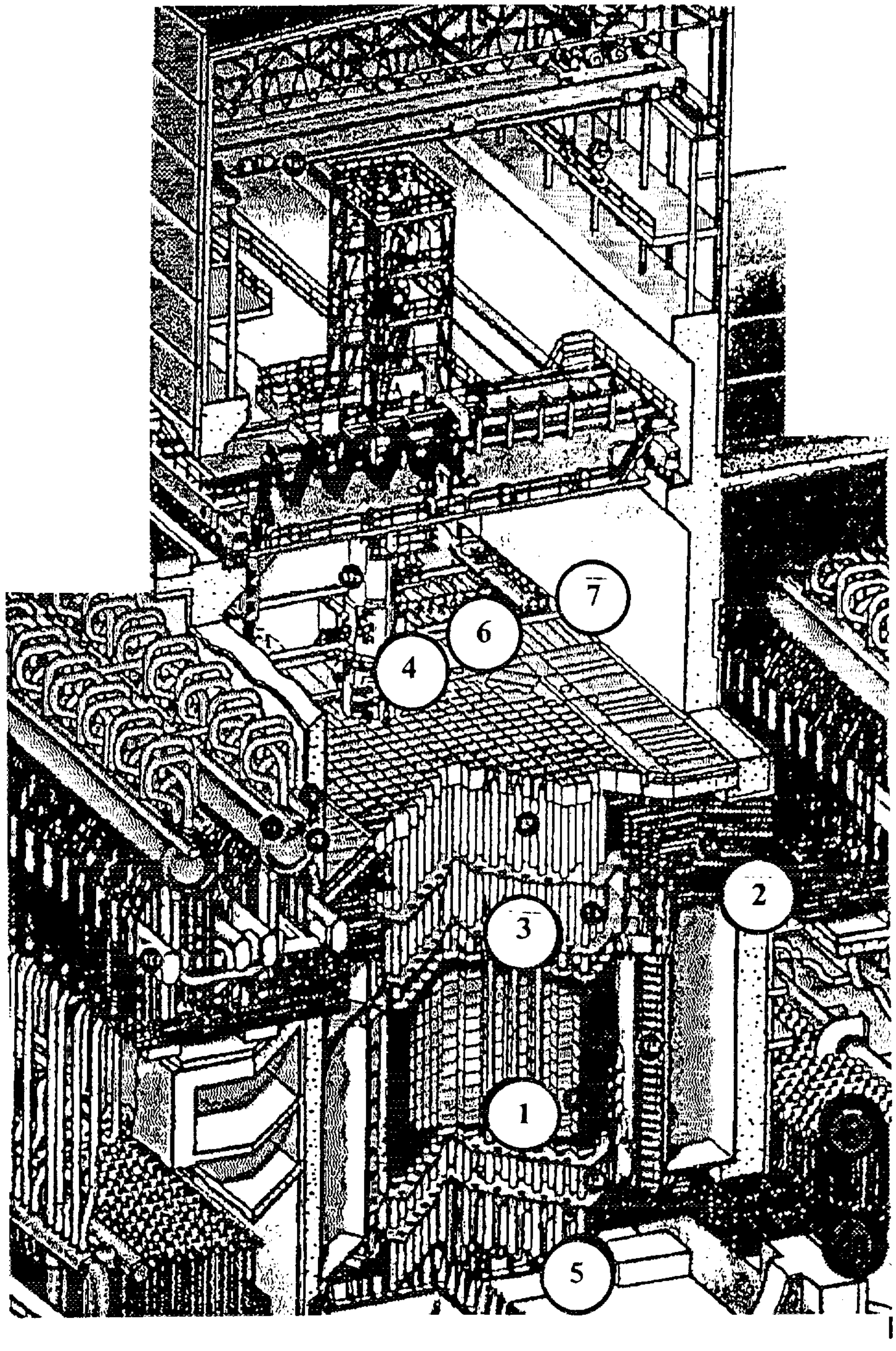
Parameter	Unit	Value
Length of FA	mm	<10,037
Diameter	mm	<79.2
Mass of FA	kg	190.0
Mass of U	kg	114.7 ± 1.6
Number of fuel rods in fuel rod bundle:		18
- at diameter 32 mm		6
- at diameter 62 mm		12
Outer diameter of fuel cladding	mm	13.6
Minimum thickness of fuel cladding	mm	0.825
Diameter of fuel pellet	mm	11.5
Fuel mass in fuel rod	kg	3.6
Length of heat releasing part of fuel rod	mm	3,432
Fuel composition		UO_2
Cladding material		Zr alloy with 1% Nb (alloy 110)

TABLE II. MAIN CHARACTERISTICS OF RBMK-1000 REACTOR

Characteristics	Unit	Value
Capacity, (heat)	MW	3,200
Initial charge,	tU	180
Initial enrichment, U^{235}	%	2.4
Average burnup	MWd/kg	15.5 - 22.3
Yearly discharge	tU	50
Number of FAs in core		1,604
Activity after 3-yr cooling	mCi/tU	0.5
Heat release after 3-yr cooling	kW/tU	2.5

2.1. AR pool

Spent fuel is discharged from the reactor with the refueling machine (RM), either with the reactor on power (main refueling) or with the reactor shut down (Fig. 2).



1. reactor core
2. steam lines
3. upper biological shield
4. reloading machine
5. lower biological shield
6. water pool
7. transfer cart

FIG. 2. RBMK-1000 reactor

2.1.1. On-power refueling

The 50/10 t crane of the central hall removes the closure plug from a refueled channel. A special program verifies the functioning of RM gears and systems on a training stand. The RM is installed over the channel simulator of the stand and is loaded with a fresh FA with its suspension. Then RM moves toward the reactor, interfaces the refueled channel, retracts a spent fuel assembly, gauges the channel route and loads the channel with the fresh FA. On completing the operation the RM with the spent FA moves toward the fuel canning area of the AR pool, interfaces the head of an empty can (previously installed there with a floor 1 t crane) and loads this with the spent FA with the suspension. The can is capped and the floor crane transfers it along a slit in the floor of the storage area to its storage position in one of the storage pools. The operation is carried out under a shielding water layer. The transportation slits are provided with hinged hatch covers to prevent steam and gases escape from the water surface in the central hall. The operation of the RM is remotely controlled from a special control panel. In emergencies the RM can be manually operated. It is equipped with a shield and special cabin for 2 operators with a system of manual drives.

2.1.2. Refueling with reactor shut down

These refueling operations are similar to the on-load refueling, but at the lower working parameters. The loading of "fresh" FAs into the reactor is performed with the 50/10 t bridge crane of the central hall. Specified conditions for wet spent fuel storage are sustained by support systems for pool water cooling and purification, technological and special control. Spent fuel is transferred from the power unit by an on-site TK-8 container, accommodating a 9-element transfer canister. Defective fuel is not transported from the AR pools. The equipment for fuel handling and transfer is housed in the main hall, railway wagon depot, fuel overloading pit, room of control panel, rooms of support systems (Fig. 3).

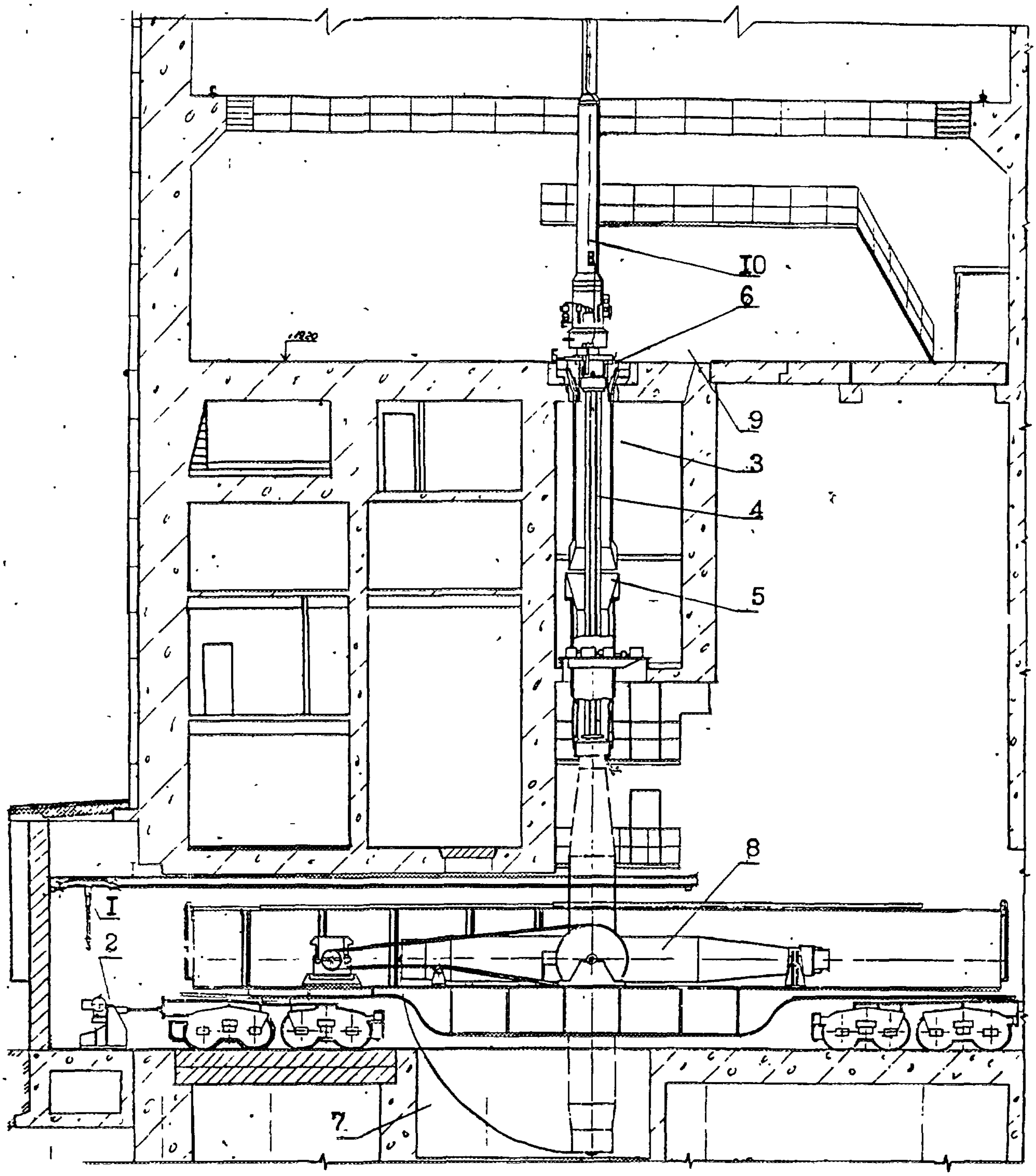
2.2. AFR-RS storage facility

This type of facilities has been designed for the interim storage of RBMK spent fuel. The construction and equipment assemblage were scheduled in 2 phases. The 1st phase involved the construction of rooms for SF reception, unloading and storage (Fig. 4). At present the 1st phase of the storage facility has been put to operation at all NPPs with RBMK-1000 reactors. The 2nd phase should include the construction of a series of hot cells for cutting FAs in two fuel rod bundles, loading and storing them in the transfer canisters with subsequent shipping to a reprocessing plant. The facility includes:

- a SF reception and dispatch area consisting of a transport entrance, main hall (for fuel overloading), support systems rooms (operators', cable-trolley rooms).The main hall has concrete wall and a metallic floor, thus providing personnel safety during remotely handled operations;

- a storage area comprising 5 identical water-filled pools situated in a pool hall. The pools are interconnected by a canyon. Four pools are operating and the 5th is stand-by and can be used in the event of repair in one of the operating pools. When drained the pools are cut off from the canyon by gates. The SF assemblies with their suspensions are stored in cans (one item/can) protecting FAs from unintentional damage and pool water from contamination.

The total design capacity of the storage area is 17,520 FAs. The FAs are moved toward their storage positions and seated on the beams of the metal floor. The operation is performed with a 1-t crane which has a limited lifting height and provides water shielding above the active part of the fuel. The AR pools utilize water cooling and purification systems. The water cooling plant is activated when the water temperature has reached 50^0C.



1. 5 t crane
2. levelling mechanism
3. guidıng pit
4. transfer canister for 9 FAs
5. guiding sleeve
6. loading device
7. areaway
8. TK-8 container
9. central hall
10. protecting transfer (unloading) device

FIG 3. Placing loaded transfer canister with FA in TK-8 cask

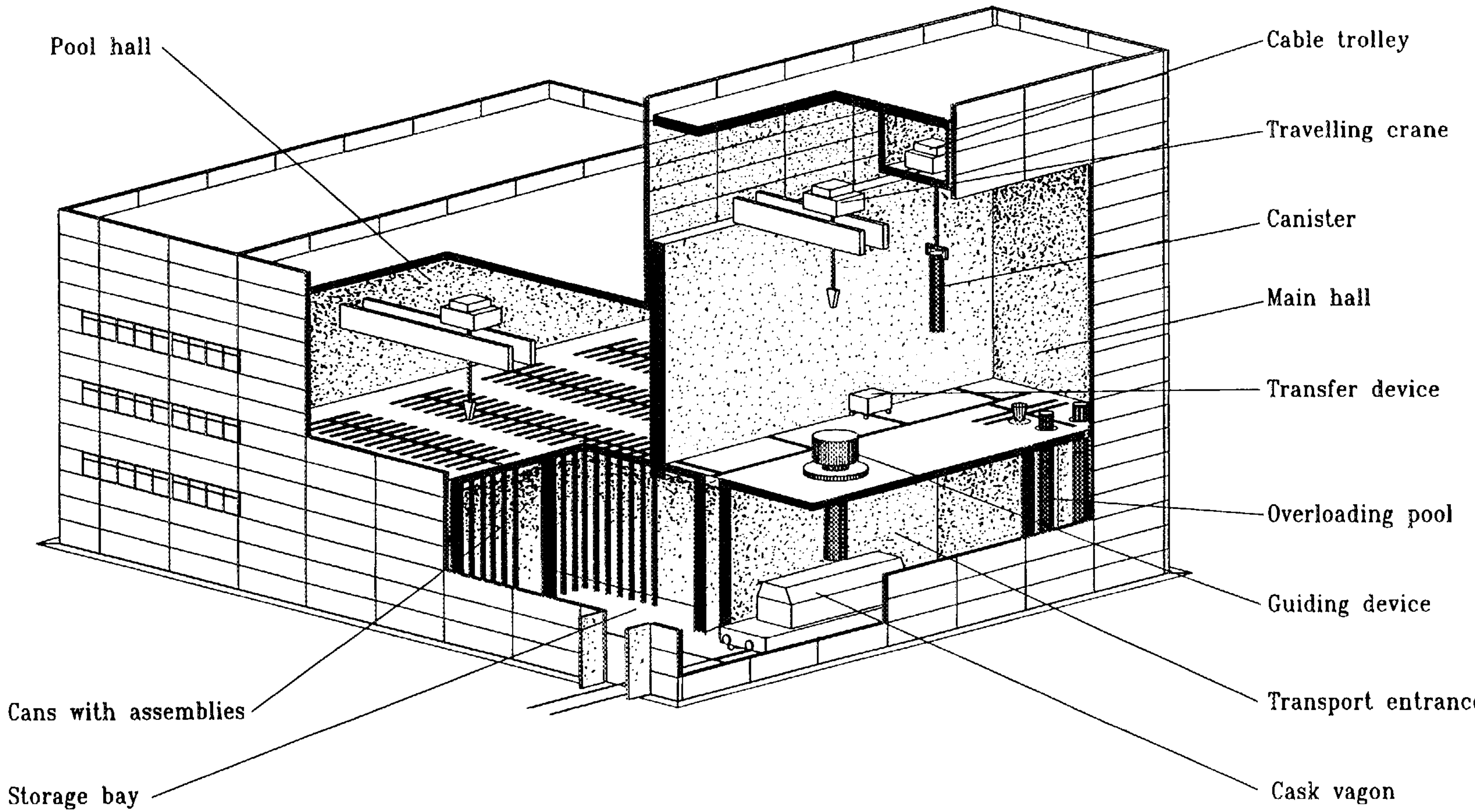


FIG. 4. Interim storage building for RBMK spent fuel

2.2.1. Reception and unloading of transfer canister from TK-8 cask

The TK-8 cask arrives from the reactor unit and is transported by a rail motor car or diesel engine to the transport entrance, where it is fixed and leveled lengthwise (Fig. 5). The wagon roof doors open and wagon mechanisms connect to power. The cask raises to an upright position and gets free of the hood. The movable shield of the guiding sleeve slides on the cask neck. The transfer canister with 9 FAs is retracted from the cask via the guiding device and landed into the water-filled pit of the overloading pool. The operations are performed by a special 15-t cable and 5-t grapple and are controlled via the remote control panel. Then the trolley returns to its starting position. The empty transfer canister is removed from the pit and loaded into the cask by a 20/5 t bridge crane. The movable shield of the guiding device goes to its upper position. The hood is placed atop the cask and fastened to it. The cask neck is decontaminated, if necessary. Then the cask is turned to a horizontal position, the wagon roof doors close and the cask is disconnected from the leveling mechanism and power. The cask returns to the reactor unit for the next loading.

2.2.2. Transfer canister - to - can overloading

The plugs are removed from the canister nests and replaced by adapters of a corresponding type (the adapters being seated on the FA heads). The grapple is hooked to the cable trolley and latched onto the FA head with a mooring rope. The trolley retracts the FA from the canister and places it into the can. The operation is remotely handled. The adapter on the FA is unlatched and replaced by a yoke. The 20/5 t crane transfers the can with the FA to its storage position (Fig. 6). The operation is performed under water shielding. On completing the unloading of the 9th FA the crane retracts the canister from the pit and places it on the brackets of the metal floor of the room for storing transfer canisters.

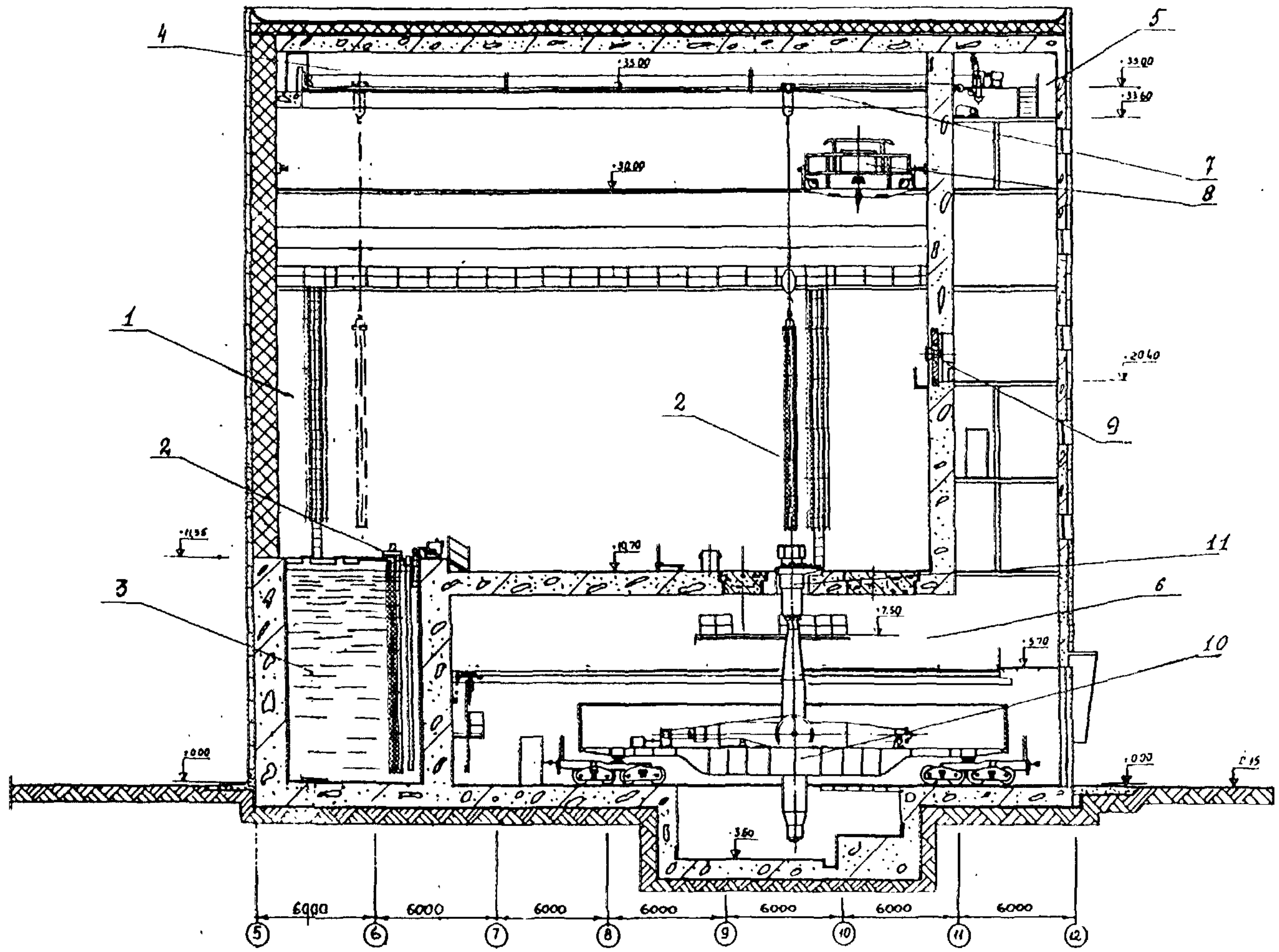
2.2.3. Transferring loaded cans to storage pools

The 20/5 t crane of the main hall transfers the loaded can from the slitted floor of the canyon and places it into the adapter of the corresponding storage pool. The 1 t crane of the pool hall removes the can from the adapter and transfers it along the slit in the metal floor to its storage position (the operation is performed under a shielding water layer). Then the yoke is removed from the can and returned to the can loading area in the main hall. The opened area of the slitted floor is covered again to prevent steam and gas escape from the water surface to the pool hall. The 1 t crane and other overloading operations are controlled by an operator staying in the pool hall.

3. CURRENT APPROACH TO SF MANAGEMENT AT NPP SITES

The approach considers changing over to the dry storage mode after the long-term (10 years) storage in water pools. It was projected to start the construction of a regional facility for storing spent fuel from all RBMK-1000 reactors. The construction was delayed for indefinite periods. Of dry alternatives most suitable is an on-site cask storage facility projected for 10 to 30 yr. service life. Such facility is most efficient from the viewpoint of safe storage conditions and fuel retrievability and complies with the current national rules for SF management. Dual purpose metal-concrete casks are being developed for this purpose (Fig. 7 and Table III). The cost of the cask storage facility comprises the costs of cask, cask loading operation and facility operation, including the costs of hoisting and transfer equipment and a building for casks accommodation. The casks loading costs are currently estimated at 30 to 50 % of the casks costs. Hence, the development of an efficient mode for cask loading is most important.

The RBMK-1000 FAs are distinguished for their size (> 10 m). Before loading into the cask the assembly must be separated into 2 fuel rod bundles and the suspension. The bundles are placed in casks for storage while the suspension goes for decontamination and remelting as radwaste.



1. main hall
2. transfer canister capacity 9 FAs
3. storage of transfer canisters
4. room for crane trolley
5. machine room
6. railway wagon depot
7. crane trolley
8. main hall crane
9. control board N1 for main hall crane
10. cask wagon
11. movable shield

FIG. 5. Technical area of AFR -RS: transport and technical equipment

Preparation for reloading

Installation of transport canister

Installation of empty can

Installation of loading funnel

Removing plugs from transfer canisters nests

FA reloading

Remotely handled retraction of FA with use of cable trolley

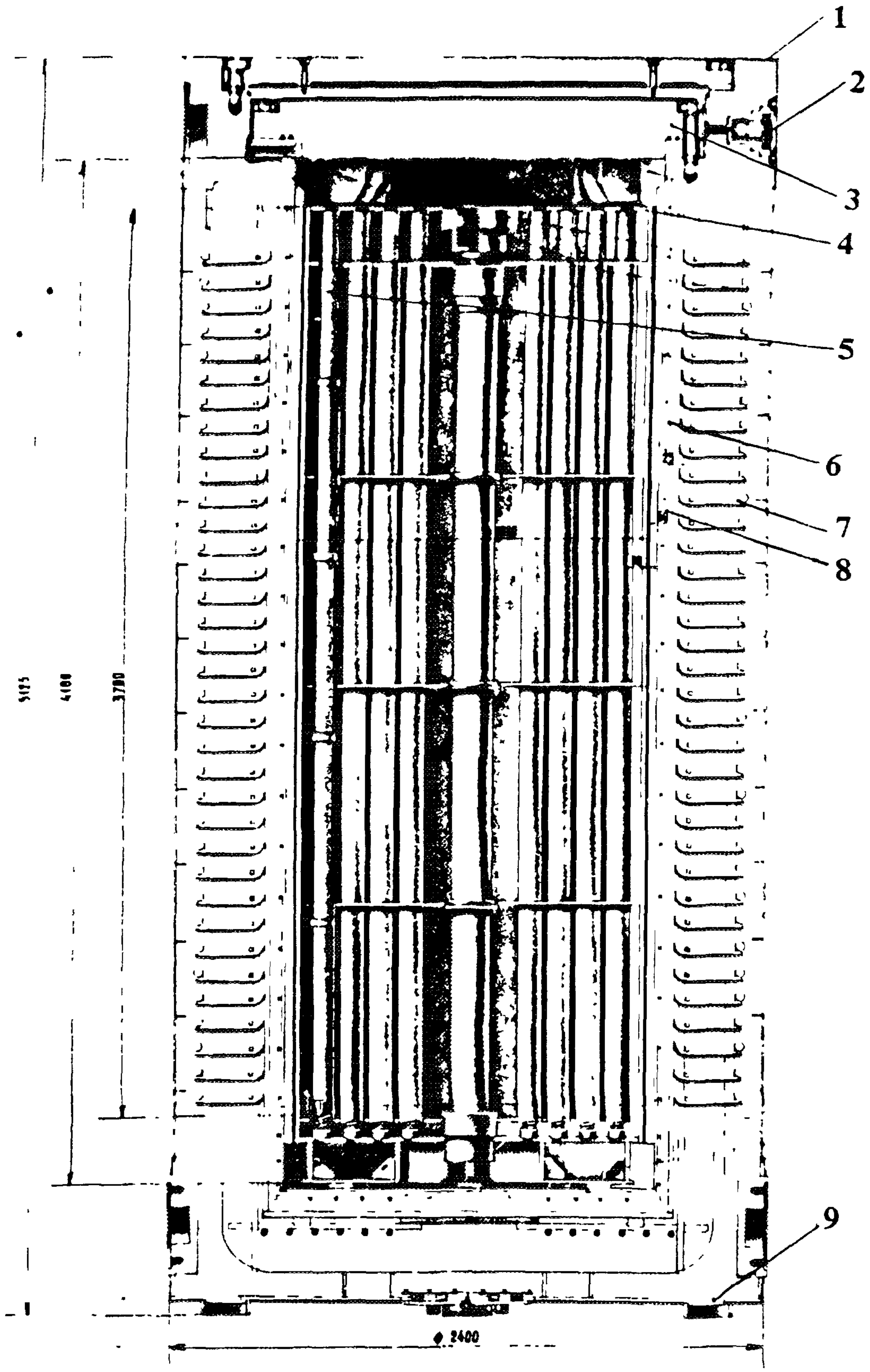
Lowering FA into empty can

Preparing can with FA for transfer to storage position

Removal of loading funnel

Can-yoke latching

FIG 6 Flowchart of transport handling operations in FA remote reloading from a transport canister to can



1. sealing plate
2. valve for controlling inter-lid space leak-tightness
3. shielding cover
4. basket
5. fuel bundles
6. reinforcing casing of the inner shell
7. reinforcing casing for strengthening shell
8 modified reinforced concrete
9 perforation for concrete

FIG. 7 Metal concrete cask (MCC) for fuel rod bundle storage

TABLE III. TECHNICAL CHARACTERISTICS OF MCC

Characteristic	Unit	Value
Number of fuel bundles		114
Mass of MCC with the basket and FA	t	95
Basket mass	t	4.8
Empty MCC mass	t	70
including:		
• reinforced B-100	t	4.5
• shielding concrete B-90	t	30
• reinforcement	t	7.7
• steel 092C	t	7.4
• SS 12X18H10T	t	5.0
• steel forging 12X18H10T/09H2MFA-A	t	0.7/14.5
• fastenings	t	0.5

From safety considerations each fuel bundle should be enclosed into an ampoule made of corrosion-resistant steel. The ampoule serves as an additional protective barrier against fuel damage in the case of fuel drop during the cask loading with fuel. Moreover, the ampoule is expected to provide safe unloading of fuel from the cask after 50 year storage. All fuel cutting and loading operations will be carried out in a hot cell and remotely controlled. According to the project being currently developed such cells will adjoin the AFR-RS facility on the plant site. The technology of FA separation in the cells will be as follows:

- Cans with fuel are transferred from the storage pools to a special overloading pool and placed on a transfer cart. The cart has the load capacity of 8 FAs, which corresponds to the cell capacity per working shift.
- The cart transfers the loaded cans via the transport canyon to the cell. Here a cantilever crane places the FAs into a drying stand.
- After drying, each FA is placed in the ampoule by the crane, the lower fuel bundle is cut off by a milling cutter and the upper bundle is transferred to the second ampoule. Here, the suspension is cut off and removed for utilization. The ampoules are covered with lids. This technology has a minimum of hoisting and transfer operations and provides maximum efficiency of the hot cell.
- The ampoules with fuel are placed into a transfer canister (item by item), which is loaded through a sluice into a cask.
- The cell's design involves the use of unified equipment applicable with any type of casks now in use for transportation and storage of the RBMK-1000 fuel (metal-concrete container, TK-11, Constor).
- Upon loading the cask with fuel and setting the protective lid the cask is inspected for leakage; moisture content in the gas medium is measured and the cask cavity dried with hot air, its outer surface decontaminated, if required.

The designed capacity of the cell equipment is 3,500 FAs/yr. With such capacity a period of 9–10 years is required to prepare for dry storage all the fuel now stored in the wet AFR of the Leningradskaya NPP, thus providing the conditions for normal plant operation during the whole service life.

SWEDISH CONCEPT FOR FINAL CONDITIONING OF SPENT LWR FUEL FOR DISPOSAL IN HARD ROCK

XA9948982

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Abstract

The Swedish Nuclear Fuel and Waste Management Co, SKB, currently owns and manages a final repository for radioactive operational waste (SFR), a central interim storage facility for spent fuel (CLAB) and a transport system. The remaining parts of the Swedish nuclear waste management system are an Encapsulation Plant and a Deep Repository for spent fuel and other long-lived wastes.

The disposal canisters, which the spent fuel will be encapsulated in, have an inner cast insert for mechanical strength and an outer copper canister for corrosion resistance. At present, SKB is conducting research and testing of manufacturing methods for both the cast insert and the copper canister. So far full-size canisters have been manufactured using two different methods: extrusion and forming from rolled plate.

Encapsulation of spent fuel in disposal canisters will take place in an Encapsulation Plant, which is planned to be built as an extension to CLAB. In the new plant the fuel is first identified and measured in pools. The fuel assemblies are then brought up, out of the water, to a handling cell where they are dried and placed in a disposal canister. Next, the canister is transferred to another station where the air in the cast insert is exchanged with argon and the steel lid is bolted to the insert. At a welding station, a copper lid is welded to the copper canister with electron beam welding. In the next station the weld is inspected and machined. When a canister has passed the non-destructive testing it is transferred to a station for monitoring and, if necessary, decontamination. Finally, the canister is brought to a buffer store for filled canisters where it awaits shipment to the Deep Repository.

SKB is presently building a Canister Laboratory in order to test the very crucial sealing operations.

1. INTRODUCTION

In Sweden approximately 50 % of the electricity is produced by nuclear power from 12 reactors located at 4 sites. Within the Swedish nuclear program, these reactors will generate about 8,000 tons of spent fuel and 200,000 m^3 of radioactive waste. A comprehensive system for radioactive waste management has been developed during the last two decades. The system is owned and managed by the Swedish Nuclear Fuel and Waste Management Co, SKB, which in turn is owned by the Swedish nuclear power utilities.

At present, the system consists of a final repository for radioactive operational waste (SFR), a central interim storage facility for spent fuel (CLAB) and a transport system. The remaining parts, an Encapsulation Plant and a Deep Repository for spent fuel and other long-lived wastes, are now being planned. The Encapsulation Plant Project started in 1993. In this paper, the current status of the project is presented.

1.1. Existing facilities

1.1.1. Final Repository for Radioactive Operational Waste, SFR

The final repository for radioactive operational waste, SFR, is the central disposal facility for most of the short-lived low and intermediate level waste from the operation of the nuclear power plants. SFR, which is located near the Forsmark Nuclear Power Plant, is built in bedrock at a depth of about 50 meters underneath the bottom of the Baltic Sea. Currently, there are four rock caverns and one silo with a total capacity of 60,000 m^3. Since the start of operation in 1988, approximately 20,000 m^3 of waste has been disposed of in SFR. For future waste from decommissioning of the nuclear reactors, there is a planned expansion of SFR for an additional 100,000 m^3 of waste.

1.1.2. Central Interim Storage Facility for Spent Fuel, CLAB

The central interim storage facility for spent fuel, CLAB, is located close to the Oskarshamn Nuclear Power Plant and consists of a receiving building at ground level and a storage building in a rock cavern, approximately 25 meters below ground. The fuel assemblies are stored in water pools in the storage building, in special storage canisters. Core components are also stored in CLAB in a similar manner. After 30-40 years of interim storage, the spent fuel and core components will be encapsulated before transfer to the Deep Repository.

Today, CLAB has a storage capacity for 5,000 tons of spent fuel. Since the start of operation in 1985, approximately 2,600 tons of spent fuel has been received. In the future, CLAB will be expanded with another storage building, parallel with the existing one, for an additional 3,000 tons of spent fuel. Construction of the second storage building is planned to start in 1998 and be finished in 2004.

1.1.3. Transport System

Since all the nuclear power plants in Sweden are situated on the coast, all transports from the nuclear power plants to CLAB and SFR are made using the purpose-built ship M/S Sigyn. The spent fuel is transported in casks and the operational waste in shielded containers. Special terminal vehicles are used for transferring the casks and containers to and from the ship.

2. DISPOSAL CANISTER

In the Swedish system for handling radioactive waste, the spent fuel will be encapsulated in corrosion resistant canisters which will be placed in a Deep Repository, approximately 500 meters down in the Swedish bedrock. In order to prevent ground water flow around the canisters, they will be surrounded by bentonite clay. Throughout the more than 10 years of research and development of the disposal canister, several different canister alternatives have been studied.

The environment which is prevailing 500 meters down in the Swedish bedrock is oxygen-free. Under these reducing conditions copper is the most suitable canister material for corrosion resistance. To improve the mechanical strength, the copper canister contains a stronger component of another material.

In the Swedish deep repository concept, the canister is an important barrier which will prevent radioactive isotopes from being released to the biosphere. It is crucial that the canister remains intact for a very long time but there are no regulatory leakage requirements specified on the canister welds. However, in the safety analyses made by SKB it is conservatively assumed that 0.1 % of the canisters have welds with defects which will lead to early leakage.

Instead of having specified requirements on the different subsystems, the regulatory body puts guidelines on the complete system. The guidelines state that the releases from a nuclear site shall give an annual dose of no more than 0.1 mSv to individuals in a critical group.

2.1. Current canister design

In the current design, the disposal canister consists of an outer copper canister for corrosion resistance and an inner cast insert for mechanical strength [1], see Fig. 1. The cast insert, which has channels for the fuel assemblies, has a minimum wall-thickness of 50 mm and is manufactured from cast steel or cast iron. The material which will be used in the final design depends mainly on the results from the manufacturing trials. The cast insert is sealed by bolting a steel lid to the insert.

The outer copper canister has a wall-thickness of 50 mm. Electron beam welding is used for welding a copper lid to the canister. Development of the welding techniques is made in co-operation with TWI in Cambridge, England.

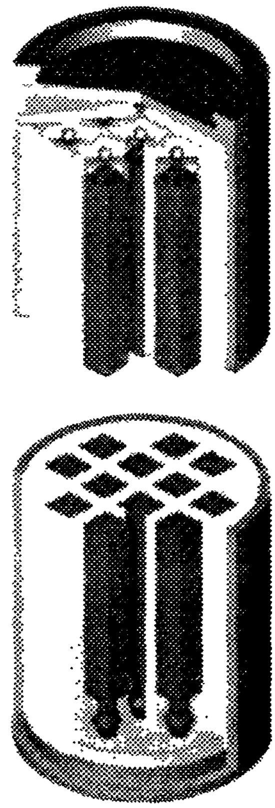


FIG. 1. Current design of a copper canister with a cast insert for BWR fuel

A disposal canister can hold either 12 BWR or 4 PWR fuel assemblies. The diameter of the canister is 1,050 mm and the length 4,850 mm. With spent fuel, a canister weighs 25-30 tons, depending on the cast insert material and if it is a BWR or PWR type canister. For identification, each disposal canister is marked with a unique indication.

2.2. Manufacturing trials

SKB has been conducting research and development of manufacturing methods for the disposal canister for more than 10 years. At first, manufacturing trials were made in a smaller scale but in February 1996, the first complete full-size copper canister with a steel container ever manufactured was delivered to CLAB, see Fig. 2.

The first canister was manufactured using extrusion. The copper cylinder which resulted from that was then machined internally and externally. A copper bottom was welded on before a steel container was fitted into the copper shell. In order to seal the disposal canister, a copper lid was finally welded to the copper canister. Later in 1996 a second full-size canister was delivered. This copper canister was manufactured by forming from rolled plate and then welding the two halves together. Electron beam welding was used for all copper welds on both canisters. When manufacturing of these first two canisters started, the design of the disposal canister was a copper canister containing a cylindrical steel container with a void in the center. Since then, the reference canister has changed to the current design with a cast insert. As soon as the new design was developed, SKB started manufacturing trials of cast inserts. So far, full-length inserts have been manufactured from both cast steel and cast iron. Several other cast steel and iron inserts have been ordered and will be cast in the near future.

In parallel with the manufacturing trials, methods for non-destructive testing of the welds are being developed. The current plan is to use both ultra sonic and X-ray testing of all welds but other non-destructive testing methods are also being studied at the moment.

The manufacturing trials so far have shown that it is feasible to produce full-scale disposal canisters according to specifications. However, the development work on manufacturing methods continues throughout the coming years.

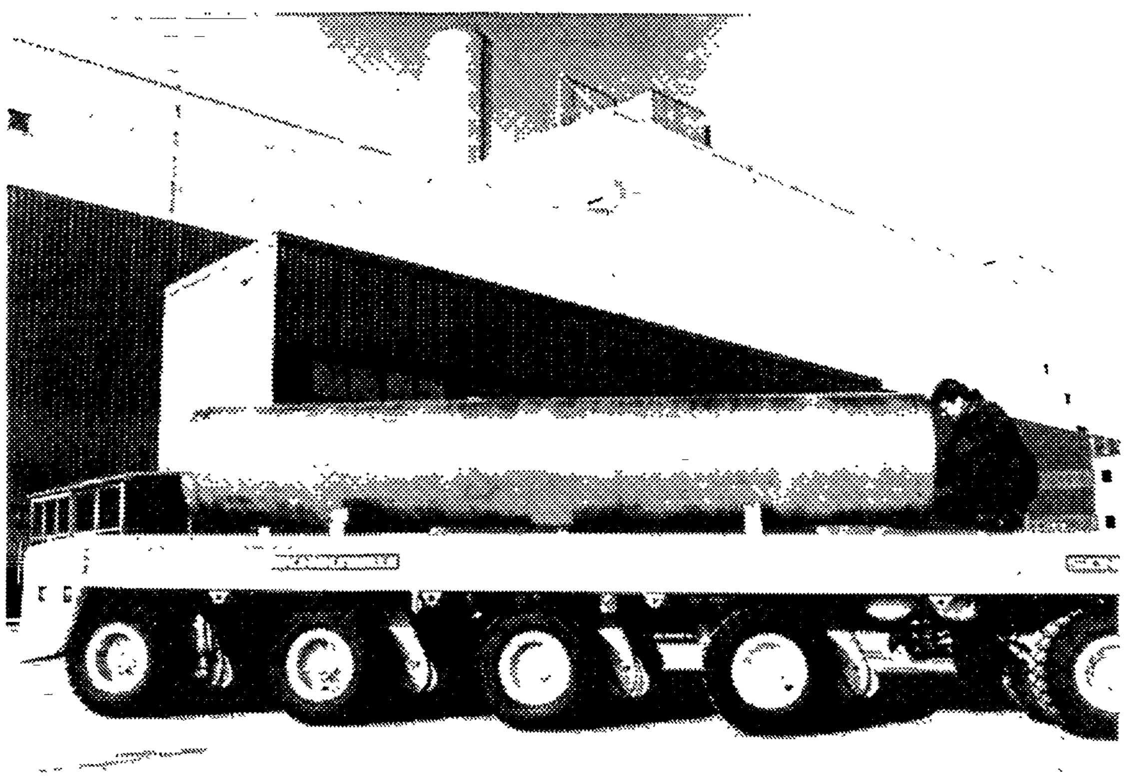


FIG. 2. The first trial full-size disposal canister

3. ENCAPSULATION OF SPENT FUEL

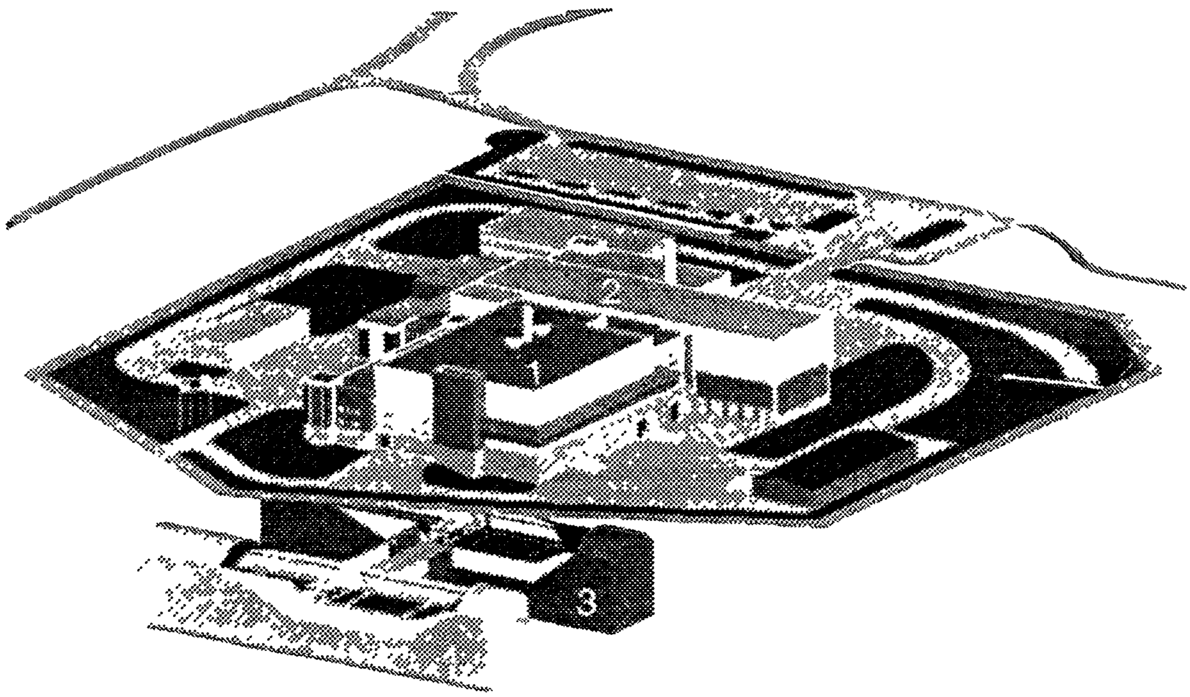
3.1. Encapsulation plant

Since the spent fuel is stored in the CLAB interim storage facility and the location of the future Deep Repository is not yet decided, the Encapsulation Plant is planned to be built as an extension to CLAB, as shown in Fig. 3. This location provides possibilities to extend several existing functions into the Encapsulation Plant. These functions include the fuel elevator, cooling systems, water purification systems and electrical power supply. At a first stage only spent fuel will be encapsulated but preparations are made for the later addition of equipment for treating core components.

In 1994 design work of the Encapsulation Plant was started and in 1996 the work had resulted in a Basic Design, which will form the basis for the application for construction of the plant. BNFL Engineering Ltd. did all work involving the encapsulation process and ABB Atom AB designed the auxiliary systems. The layout of the plant and co-ordination of the design work was performed by SKB.

The plant is designed for an annual output of approximately 210 disposal canisters per year, i.e. on the average one canister per workday. The operating staff shall be able to work both in the Encapsulation Plant and in CLAB. During normal operation, approximately 30 people will be working daytime with encapsulation and maintenance.

The Encapsulation Plant will be approximately 65 x 80 meters in size and about 25 meters high, which is equivalent to the height of the existing receiving building in CLAB. The cost of the plant is close to SEK 2 000 million (USD 260 million). When construction of the Encapsulation Plant can begin depends mainly on the progress of siting the Deep Repository.



1 Encapsulation Plant | 2 Receiving building (CLAB)
3 Storage building (CLAB) | 4 Future storage building (CLAB)

FIG 3 The Encapsulation Plant built as an extension to the existing CLAB facility

3.2. Encapsulation process

3 2 1 Receiving of Empty Disposal Canisters

The cast insert is assembled into the copper cylinder before transportation of the disposal canister to the Encapsulation Plant A canister is transported horizontally and is, after arrival, raised to a vertical position with a special tilting equipment All parts are inspected thoroughly in an inspection frame before the canister is placed in a shielded frame which is used for transfer of the canister within the plant The frame is lifted and transferred using a remote controlled air film transporter

3.2 2 Fuel Elevator

The fuel assemblies which are stored in CLAB have very different values of burn-up and residual power These properties give the heat output of the canisters, which is a restricting factor in the Deep Repository To minimize the total number of disposal canisters, the combination of fuel assemblies in a canister is optimized Based on the fuel data in CLAB, storage canisters with suitable fuel for encapsulation are chosen

For the transfer of fuel from the storage pools in CLAB to the Encapsulation Plant, the existing fuel elevator is used A storage canister is moved, using the handling machine in the storage building, to a water filled elevator cage. The elevator is then raised, turned by a turntable and lowered down into a transfer channel in the Encapsulation Plant, see Fig. 4. Today, there is similar handling when transferring storage canisters between the receiving building and the building pools in CLAB

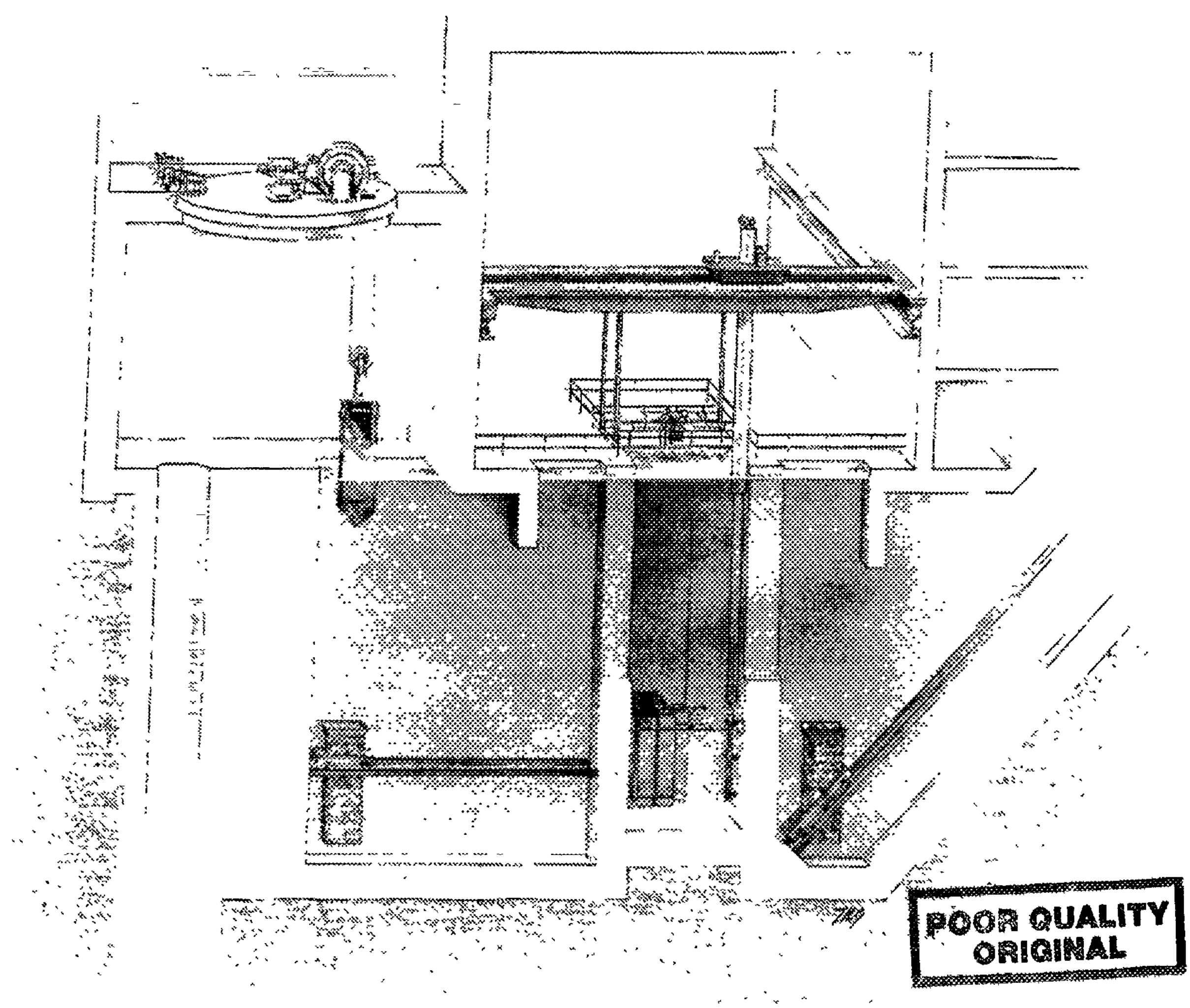


Fig. 4. Fuel elevator, fuel handling machine over the pools and ramp elevator

3.2.3. Handling Pool

The storage canister is moved from the transfer channel to the handling pool with a handling machine. In the handling pool it is placed in a rack which has 16 canister positions. In order to simplify visual inspection of operations, the handling machine is operated from a platform which is situated directly above the pool. Before any fuel is lifted out, the canister and the fuel assemblies are identified. The handling machine is similar to the existing fuel handling machines in CLAB.

After identification, one fuel assembly at a time is lifted out and transferred to a fuel monitoring station where gamma measurements are made in order to calculate, for example, burn-up and residual power. The results of these measurements are compared with the historical fuel data. In the next step, the assembly is transferred to a transfer canister which is positioned in the canister rack. A transfer canister is similar to a storage canister but can only hold 12 BWR or 4 PWR fuel assemblies, which is the same as the number of fuel channels in a disposal canister.

3.2.4. Ramp Elevator

When a transfer canister is filled, it is brought to a transfer bogie in a connection pool. The bogie with the canister is lifted up an inclined elevator, out of the water. The design with a ramp elevator ensures that high lifts are avoided. When the canister is over the water surface, it is allowed to drain before it is lifted to a handling cell.

The water in the pools acts as radiation shielding and gives the necessary cooling for the fuel The air in the elevator and the handling hall is separated by the transfer channel acting as a water seal In the same way, the air in the handling hall and the handling cell is separated by the connection pool All pools can, independently of each other, be emptied of water for maintenance

3 2 5 Handling Cell

The transfer canister is lifted, by a handling cell crane, to the handling cell where it is placed in one of two drying stations There, the fuel is dried with recirculating air with a temperature of about 120°C. A shielded frame with an empty disposal canister is docked from below to another part of the cell The connection between the disposal canister and the handling cell is tight so that the outside of the canister is not contaminated and the air in the cell does not escape and cause airborne activity in other parts of the plant

When the fuel assemblies are dry they are transferred, one by one, to the insert of the disposal canister The top of the disposal canister is protected so that the surface will not be damaged during handling When the canister is filled, the shielded frame with the disposal canister is moved away from the handling cell and is transferred to the next station The empty transfer canister is brought back, via the ramp elevator, to the handling pool

3 2 6 Inerting and Lidding Station

In the inerting and lidding station, the air in the cast insert is removed and exchanged with argon In order to achieve this, the insert is vacuumed down and then filled with argon a number of times. The connection between the canister and the inerting and lidding station is similar to the connection at the handling cell When the atmosphere in the insert is of the required quality, a steel lid is bolted to the cast insert The lid is tested for tightness before the canister is transferred to a welding station

3 2 7 Welding Station

At the welding station the disposal canister is docked to a welding chamber within the station Also here the connection is made in a similar manner as previous stations When the canister is connected, the chamber is vacuumed down and a copper lid is placed on the copper canister The lid is then sealed to the canister using electron beam welding During welding the canister is rotated and the welding equipment is fixed When the weld is completed, the disposal canister is transferred to the next station for testing

3 2 8 NDT and Machining Station

When the canister is docked to this station, a visual control of the weld is made before the weld area is machined Non-destructive testing is then performed using both ultra sonic and X-ray techniques If the weld contains defects which are repairable, the canister is brought back to the welding station for re-welding In case a weld has failed in a way that it can not be repaired, the copper lid is removed, the steel lid unbolted in the inerting and lidding station and the fuel, finally, unloaded in the handling cell

3 2 9 Monitoring and Decontamination Station

When a canister has passed the non-destructive testing it is lifted out of the shielded frame, using a remote controlled shielded handling machine, and is transferred to a monitoring and decontamination station The canister is lowered down into the station where smear tests are taken on the entire outer surface to monitor that it is clean The station is equipped with high pressure water which can be used if there is need for decontamination, after which new smear tests are taken. The surface dose rate is also measured before the canister is transferred to the buffer store

3.2.10. Buffer Store for Filled Canisters

The buffer store for filled canisters is situated under a radiation shielded floor with openings over the storage positions. Each opening is covered with a shield plug. A canister is transferred to an available position with the shielded handling machine. The plug is lifted and the canister is lowered down and is released from the machine. Before the handling machine leaves the position, the shield plug is replaced over the opening. The buffer store is cooled with air and has approximately 50 storage positions.

3.2.11. Transport Cask for Transfer to the Deep Repository

When a canister is to be delivered to the Deep Repository it is transferred, again using the shielded handling machine, from the buffer store to a loading position. A transport cask is docked underneath that position. The canister is lowered into the transport cask and the cask is fitted with a lid. The cask is then lifted, with an overhead crane, to a transport frame which the cask is lowered onto. The same type of handling is already used, routinely, in CLAB. With a transport vehicle, the cask with the disposal canister is moved out of the Encapsulation Plant for further transportation to the Deep Repository.

3.3. Canister laboratory

In order to test the very crucial sealing operations, SKB is building a Canister Laboratory in the town of Oskarshamn, where part of an old shipyard has been purchased for that purpose. The main parts of the laboratory are the electron beam welding and non-destructive testing operations. There are, however, plans to also test and demonstrate other parts of the encapsulation process at a later stage, e.g. fuel drying, transfer of fuel to the disposal canister and sealing of the cast insert.

The building has been refurbished and the welding and non-destructive testing equipment will be delivered in the near future. The equipment in the Canister Laboratory will be able to perform welds both horizontally and at a variable angle. The welding trials are planned to commence in 1998 and the results from those trials will be submitted to support the application for construction of the Encapsulation Plant.

REFERENCE

[1] SKB, RD&D-Programme 95, Treatment and final disposal of nuclear waste, Programme for encapsulation, deep geological disposal, and research, development and demonstration, Swedish Nuclear Fuel and Waste Management Co, SKB, Stockholm, Sweden (1995).

XA9948983

REMOTE TECHNOLOGY IN THE SPENT FUEL ROUTE IN THE UK

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Abstract

Remote technologies employed in front end (commercial) reprocessing operations of metallic and oxide fuel at Sellafield in the UK are described. An overview of the transportation, fuel receiving and preparation facilities are given together with the remote technology developments employed to improve operations. It is concluded that the facilities and remote technology used within them are mature and based upon simple and robust principles. Remote operations and maintenance in these facilities is often easier than in many facilities downstream of the dissolution stage.

Fuel design considerations for shearing and handling are described and it is concluded that advanced and higher burnup fuel can be accommodated by existing reprocessing and interim storage routes with current remote technologies.

Two different storage systems are available from UK companies which use existing spent fuel handling technology/equipment.

The pace of remote technology development is currently being set by the demands of other nuclear process areas such as decommissioning and plant clean out; these will spin-off into front end processes.

1. INTRODUCTION

In the UK there are three commercial reactor types, Magnox, CAGR and PWR. Additionally commercial reprocessing of various nuclear fuel from around the world takes place at British Nuclear Fuels plc (BNFL) Sellafield plant. Both reactor and reprocessing facilities utilise remote technology to handle and perform operations on the spent fuel during reactor discharge through to the start of the dissolution process. Not all spent fuel in the UK is reprocessed. Magnox Electric operate an interim dry store at Wylfa built by GEC Alsthom ESL. BNFL also offer dry storage systems to the world market (excluding the US). These interim storage systems utilise similar handling equipment to the more conventional reactor and wet storage handling facilities.

In this paper the remote technology applied to front end reprocessing operations in the UK (including preparation for reprocessing) will be described. Also included will be fuel design considerations for shearing/handling, advanced and higher burnup fuel, storage options and ongoing developments in these areas. BNFL operate two reprocessing plants at Sellafield, one taking Magnox metallic fuel and the second (and more recent plant) THORP taking all oxide fuel (PWR, SGHWR, AGR, BWR etc.).

Non commercial reactor systems and fuel treatments have been omitted from this review although there are some interesting technologies utilised which may find future application in the treatment of commercial spent fuel.

It should be noted that at the time of writing the nuclear industry is facing significant financial pressure not just in the UK but worldwide. Although nuclear generated electricity in the UK has exceeded the figures for coal and gas for the first time during the second quarter of 1997, there is no new nuclear capacity under construction whereas numerous gas fired stations are being built. Fuel cycle costs are under great pressure and whilst in the short term this may mean reprocessing is less favoured than interim storage, there is great optimism that the economic situation will turn and fuel cycle costs will be driven down to make reprocessing the favourite option.

2. REMOTE TECHNOLOGY IN FRONT END REPROCESSING OPERATIONS (INCLUDING PREPARATION FOR REPROCESSING) AT BNFL SELLAFIELD

2.1. Transportation to Sellafield

The fuel received from the UK, Europe or Japan is transported to Sellafield by road, rail and sea. Used fuel is contained within purpose-built flasks that protect the fuel and maintain it in a constant environment, as well as protecting the external environment from radioactive discharges. The shielded flasks are constructed of steel and lead, weighing up to 110 t and carrying up to 10 t of fuel. Flasks are designed and built to international standards and pose no hazard to people or the environment in normal or accident transport condition. The transport of used nuclear fuel has been taking place safely since the early 1950s, covering millions of miles without an incident involving any breach of containment.

In order to ensure high standards of safety in the transport of used nuclear fuel, the flasks are regularly inspected and maintained within a dedicated flask maintenance facility. The refurbishment commences with cleaning and inspection of the flask and its associated components, followed by grit blasting and painting, culminating in re-assembly and testing prior to dispatch.

2.2. Fuel receipts

2.2.1. Fuel Handling Plant

The fuel handling plant comprises three inlet cells, three linked storage ponds, three sub-ponds, two decanner cells for Magnox fuel and an AGR dismantler cell. Receipt capacity is typically two flasks per cell per day for Magnox fuel (5 t) and one flask per day per cell for AGR fuel (1 t).

Magnox and AGR fuel is received in the fuel handling facility inlet cells where the flask lids are unbolted manually to allow them to be removed remotely. The fuel skips are unloaded and then placed in a pond storage skip. These skips are sealed, purged and filled with water before being lowered into the transfer channel of the main storage pond on a ramp trolley. From the transfer channel they are transferred to their storage position by one of two skip-handling machines. These are remotely driven and their movements are computer controlled to ensure accurate placement and fuel movement records. Each pond has 220 grid positions and skips may be stacked up to three high. This gives each pond a maximum capacity of 800 t Magnox fuel or 1,680 t dismantled AGR fuel.

Two of the three sub ponds are used to purge Magnox fuel skips, then their lids are removed and the internal skip is placed in a rotary skip washing machine to remove any accumulated sludge. The third sub pond is used for a variety of maintenance tasks on the underwater equipment and it houses an end cropping machine for one type of Magnox fuel.

Magnox fuel remains in the pond for at least 100 days before preparation for reprocessing and AGR fuel for about three years. During this storage period the skip handler may move the fuel under computer control to positions in the pond where the fuel boxes may be sampled. The tooling used for such operations is simple and is generally pole mounted and operated manually.

2.2.2. THORP (Thermal Oxide Reprocessing Plant)

LWR and consolidated AGR fuel arrives by rail at the receipt and storage building which is the first part of THORP. Flasks are lifted by a heavy-duty crane into the receipt building for monitoring, flask venting and, if necessary, washing.

Following receipt inspection, the flask lid bolts are removed robotically (Fig. 1) and the flasks are lowered into the inlet pond which has two depths, eight metres and fourteen metres, in order to be able to accommodate the different sizes of flasks - AGR flasks are smaller than PWR/ BWR flasks.

The flask lids are remove under water, and the containers holding the fuel are lifted out. PWR/BWR fuel is held in long steel containers known as multi-element bottles (MEBs), while AGR fuel is stored in steel boxes. The transport flask is removed from the inlet pond, cleaned, dried and monitored. The MEBs are placed into storage racks while the boxes of AGR fuel are moved separately. A purpose-built crane known as the rack transfer machine, moves the racks and boxes of fuel along an underwater channel to predetermined positions in THORP's eight-metre deep storage pond.

The use of MEBs/boxes minimises the spread of contamination, expedites unloading and reduces radiation dose uptake to personnel to very low levels.

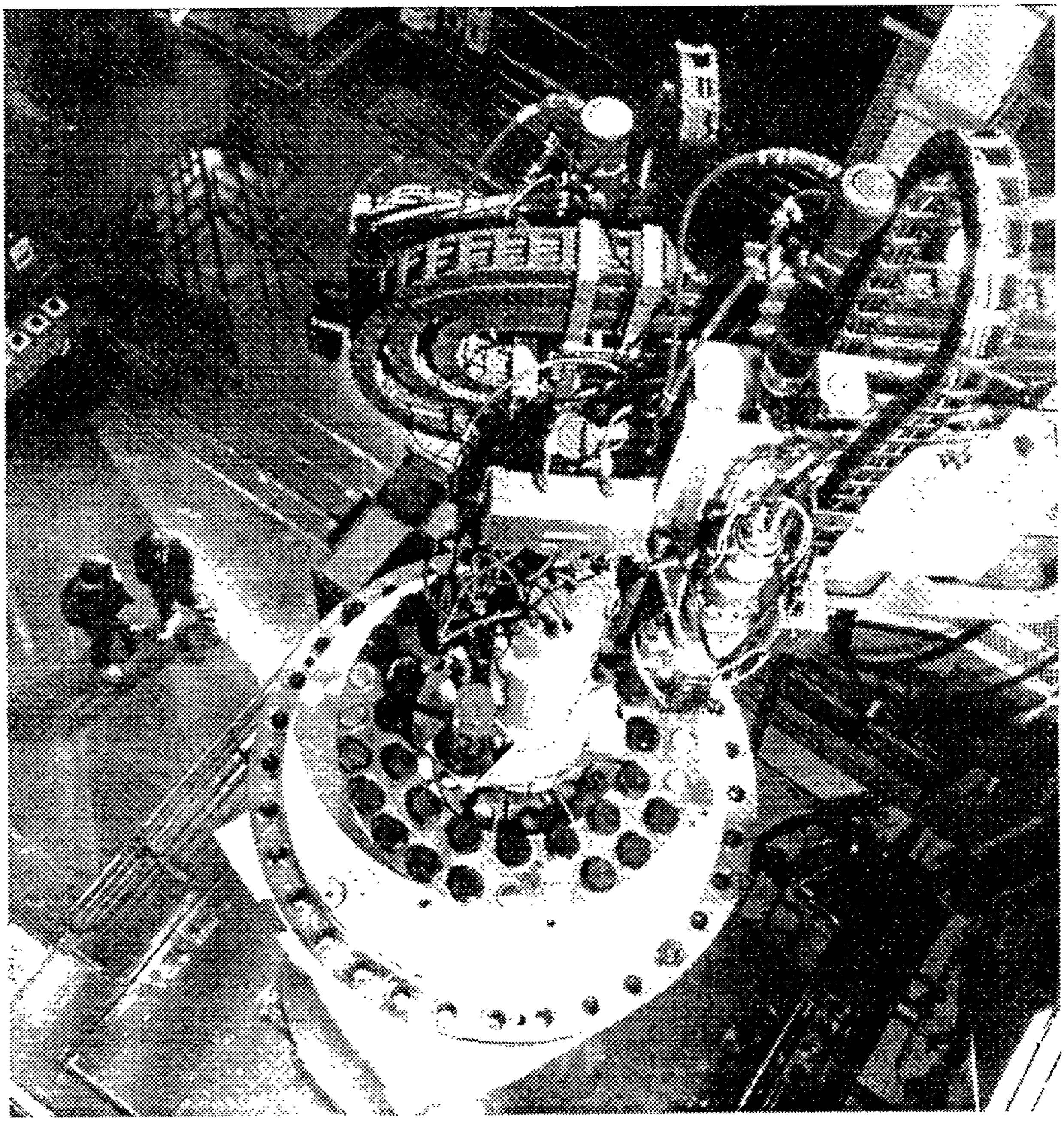

FIG. 1. THORP flask lid unbolting machine

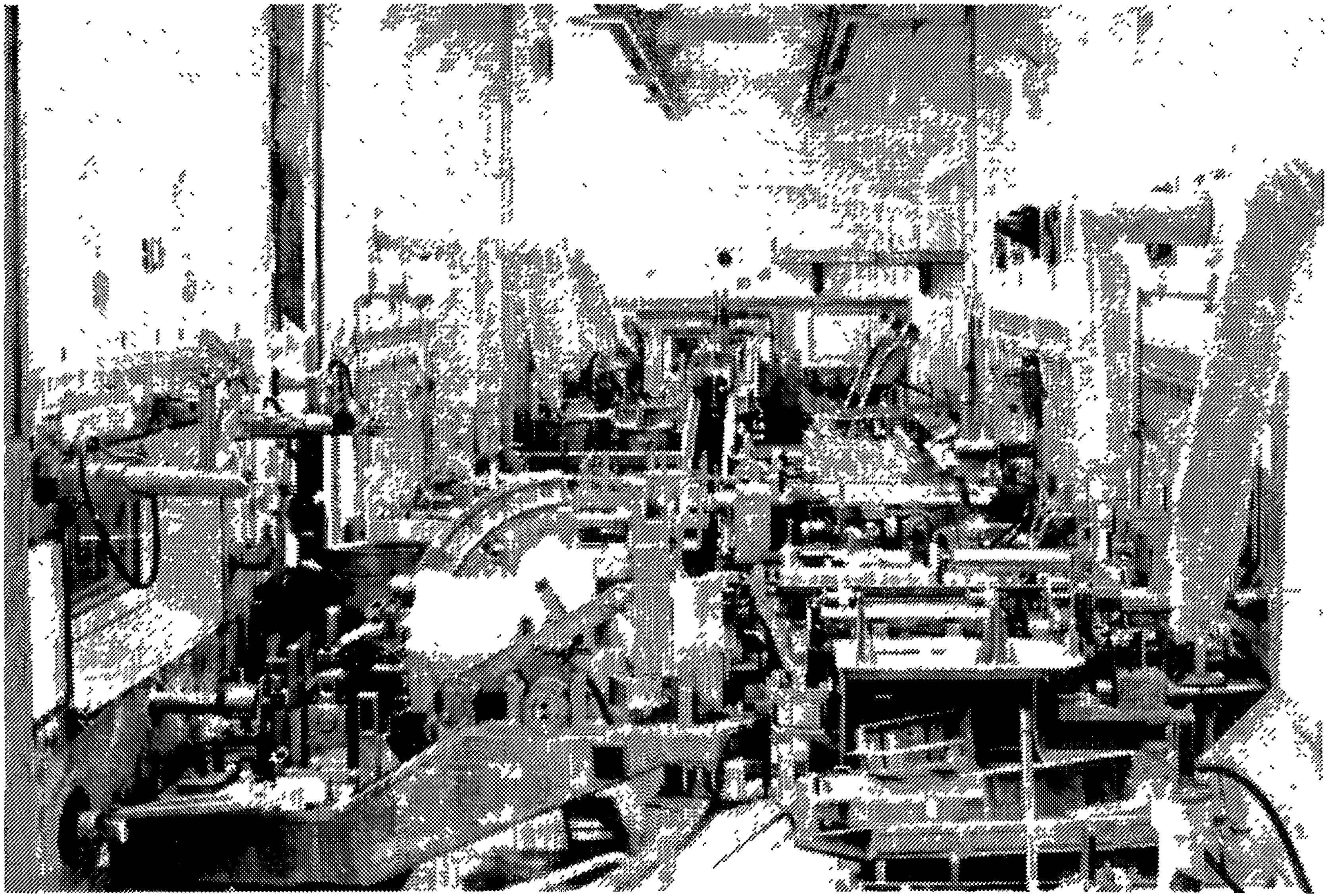

FIG 2 Fuel handlıng plant decanter cell

AGR fuel remains in the storage pond until at least three years after it has been discharged from the reactor, and PWR/BWR fuel for at least five years, to allow short-lived radioactivity to decay. The THORP complex has two connected ponds, which together are able to hold sufficient fuel, some 3,650 t, for over four years of reprocessing.

Other buildings on the Sellafield site provide further storage for the different types of oxide fuel that BNFL is contracted to handle. These smaller scale facilities provide the flexibility to deal with many fuel types and designs that often require specialist handling.

2.3. Preparation for reprocessing

2.3.1. Fuel handling Plant

Magnox and AGR fuel are transferred from the storage pond to their preparation cell via the transfer channel and ramp trolleys.

Magnox Fuel

Many different types of Magnox fuel are prepared in the decanning cell (Fig. 2). The processes performed here are common to all Magnox variants and result in the cladding being removed by a number of discrete operations:

- fuel elements lifted by hydraulic tong from the skips;
- desplittered (removal of external fins on certain designs of element by pushing through a forming die);
- end fittings cropped;
- decanned (removal of Magnox cladding by pushing through can slitter head);
- bare uranium bar loaded into Magazine flask for transfer to Magnox Reprocessing Plant;
- weighed.

All fuel element movements and preparation operations in the decanner cells are performed via fixed automation (driven by through wall shafts). The decanner cells were designed to be readily maintainable and have all drives, hydraulic seals, connections and the majority of electrical limit switches located outside the biological shield. Master Slave Manipulators (MSMs) and an overhead power manipulator/crane provide the means to maintain the equipment in the cell and to intervene where necessary.

AGR Fuel

From the storage pond AGR fuel elements are taken to the dismantling cell where the pins are extracted from the fuel elements and placed in stainless steel slotted cans. The pins from three elements (108 pins in total) can be consolidated into one slotted can, requiring less space for storage and providing a suitable density for shearing in the head end section of THORP. The grids and graphite sleeves remaining are size reduced and transferred to intermediate-level waste treatment plants. After being loaded with fuel, the slotted cans are stored in enclosed skips in the fuel handling plant before being transferred to THORP receipt and storage facility prior to reprocessing.

A crane rail mounted telehoist is the main handling tool for AGR elements and slotted cans. This unit can automatically load and unload a fuel skip once referenced to the position of the skip. The dismantler cell shares the design, operation and maintenance principles applied to the decanners.

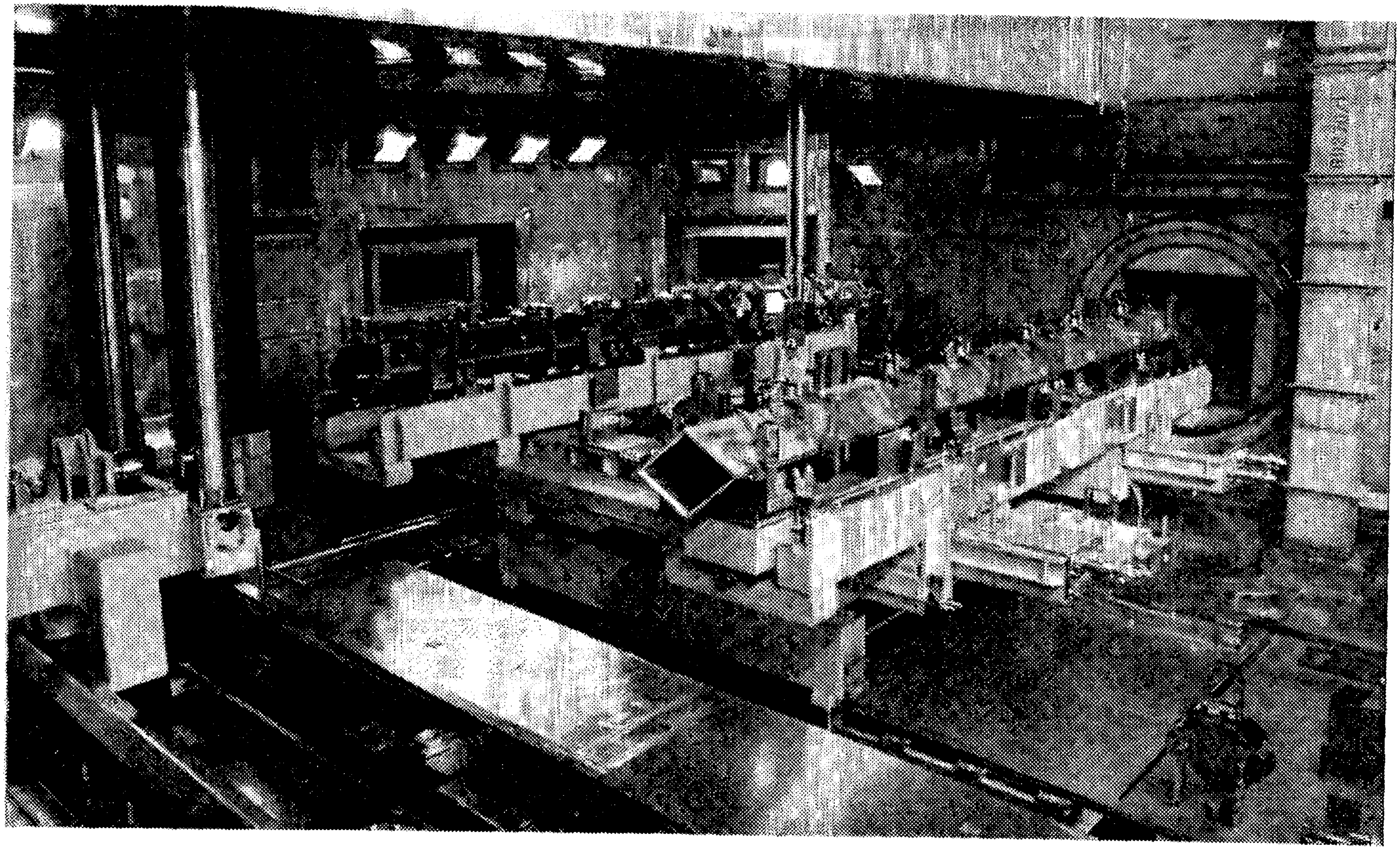

FIG.3. THORP shear cell

Fuel Handling Plant remote technology & developments

Apart from the technology used in the fuel preparation route the fuel handling plant has remote maintenance and decontamination facilities for magazines/flasks, decanner equipment, MSMs, long tools, and heavy equipment (e.g. ramp bogies).

The fuel handling plant has been operating since 1985 and is therefore 'mature'. Developments have largely been driven to reduce costs and waste arisings and some examples are:

- heavy duty MSMs to give greater life where heavy duty operations are performed;
- slitter heads to maximise packing density of Magnox swarf (i.e. small pieces resulting from machining processing, here from fuel cladding);
- recanning small pieces of Magnox fuel;
- compaction of splitter waste.

2.3.2. THORP

Feed Pond

Leaving the storage pond, racks and boxes of fuel are transferred to the feed pond through a connecting channel water by a computer-controlled rack/skip handler. In the feed pond, the containers of fuel are vented, purged and then measured to check they have completed the specified storage time. Purpose-built machines then open the bottles and boxes and remove the PWR/BWR fuel assemblies and cans of AGR fuel. These are measured for radiation to check that they meet the safety specification before being placed on a special carriage and elevated up an inclined chute into the shear cell. Maintenance of the elevator and other underwater equipment is generally performed using pole-mounted tools.

Shear Cell

LWR fuel assemblies and cans of AGR fuel are transferred from the elevator to the turntable (Fig. 3), which moves the fuel in line with the shear machine. A ram pushes the fuel into the machine where it is cut into approximately five centimetre (two inch) long pieces. These pieces fall down a chute into one of three steel vessels known as dissolvers. Maintenance of the shear pack is performed in specially equipped bulges which are separated from the shear cave by a shield door. Both the shear cell and the maintenance cells are served by an overhead rail mounted servo manipulator/crane. This handling machine is used to dismantle and assemble all other maintainable equipment in the shear cell. Units to be maintained (including the shear pack) are transported to the maintenance area by the crane.

THORP Remote Technology & Developments

Building on the experiences gained in designing and operating the fuel handling plant, THORP's remote operating equipment has been designed to be simple, robust, highly reliable and easily maintainable.

Certain contingency developments are taking place to ensure recovery options are available e.g. in the event of a fuel element lifting feature failure. Enhancements will be made to enable the plant to meet the business and regulatory requirements of future. At this moment in time, however, it is not anticipated that any enhancements will need to be made in the remote technology deployed in the front end area..

2.4. Remote technology development in other nuclear process areas

In many respects current developments in remote technology are being driven by the requirements of decommissioning projects and downstream waste handling plants. Examples include:

- Sonar/lidar (light detection and ranging device or laser radar) survey and guidance in pond and silo clean-out projects;
- Remote operation vehicles (ROVs) for dry and wet environments both for inspection, sampling and dismantling/cleanup;
- Large, sophisticated, remote dismantling machines for decommissioning redundant processing plants and reactors;
- Automated waste sorting and size reduction facilities associated with the above.

Developments from these applications will no doubt provide spin-off benefits to the front end reprocessing area.

3. FUEL DESIGN CONSIDERATIONS FOR SHEARING/HANDLING

In general shearing does not rank highly in fuel design priorities. In-reactor performance and corrosion resistance are paramount. The technology of handling both at reactor and reprocessing plant or interim store is mature with reliable grabs engaging with the fuel element or stringer end fittings and safely moving it to the desired location.

Most PWR fuel elements are designed to be part dismantled so that failed pins may be replaced and the element returned to service. There is little experience of this so far in the UK, but Sizewell B is equipped with facilities to enable these operations.

The slotted cans, in which AGR fuel is consolidated, are designed to improve the shearing and downstream waste processing. Considerations taken are:

- minimising can wall thickness and optimum material selection and physical state to ensure a clean shear rather than a fold;
- minimise waste piece size and fuel powder production;
- maximises shear blade life.

It some cases it is not economic to remove BWR fuel shrouds at the reactor (so they may be reused on new elements) in which case the elements may be sheared with the shroud in place. Demonstrating the capability and optimising the shearing of such fuel is carried out on a full scale inactive shear machine located at Sellafield. This machine also permits off-line shear pack improvements to be verified before introduction.

4. ADVANCED AND HIGHER BURNUP FUEL

LWR operators have chosen more aggressive core designs and operation strategies to improve capacity factors. The more demanding fuel duties include extended cycle length and fuel burnup, increased fuel peaking factor, plant thermal uprating, higher enrichment and burnable neutron absorbers, more sophisticated fuel designs, and new cladding materials. These have led to a demand for more innovative fuel assembly designs that offer greater economic returns and extended cycle length, while maintaining high fuel reliability.

Fuel cladding corrosion rates increase as fuel duties and burnup increases. This has led a number of fuel vendors to develop advanced claddings such as ARE's OPTIN, Westinghouse's ZIRLO, Siemens' ATRIUM & FOCUS, and Framatome's M4 & M5 [1].

The developments in claddings basically divide into two categories; modified Zr (outside of the ASTM specification) and duplex/triplex cladding.

Similarly assembly designs are becoming more complex in order to achieve higher safety margins to enable higher duty and burnup. For example, Siemens is developing advanced water channels as an option for their ATRIUM BWR 9x9 and 10x10 designs. There is also a lot of development in bottom nozzles/filters both for PWR and BWR. There appears to be a general move away from stainless steel and/or inconel in assembly construction. This includes both the bottom and the top nozzles.

Most of the utilities/vendors have extensive PIE work currently in progress in order to validate the margins for operation at high burnup and duty. Utilities are mainly concerned with ductility, creep, and growth whereas the principle properties that affect handling & shearing are oxide thickness and hydride content.

The developments in fuel assembly design and cladding materials should improve the final condition of high burnup fuel with respect to corrosion and distortion. This should improve the reliability of any handling and shearing process and also increase the possible for storage duration.

Development of advanced fuel for the UK's Magnox and AGR reactors is very limited. One fuel worth mentioning is MagRox which is oxide fuel (AGR fuel pins) contained in an external geometry compatible with the current operational and handling constraints imposed by the Magnox fuel route. The current programme is to undertake a pilot loading in BNFL's Calder Hall in 1998 with post irradiation examination in 1999 and 2000. This could be followed, if the business case is made, by large-scale loading in other Magnox reactors.

It is too early to comment in detail on the remote technology which may be required to accommodate this type of fuel in the reprocessing route.

5. STORAGE OPTIONS

For most nuclear utilities, spent fuel storage capacity is a key issue. Existing storage pools were never designed to hold lifetime levels of spent fuel. Yet without a safe solution, utilities could be forced to shut reactors down, with significant losses resulting.

With direct disposal technology unproven in terms of costs, technology and public acceptance, there are currently only two realistic solutions to the problem, reprocessing and interim storage. In the UK there are two interim storage solution available from different suppliers.

5.1. GEC Alsthom Engineering Ltd.

A special purpose facility has been in operation at Wylfa Nuclear Power Station to store spent Magnox fuel for over 25 years. This was engineered by GEC Alsthom Engineering Ltd. and forms the reference for its Modular Dry Vault Store (MDVS) [1] technology. More recent MDVSs have been built at Fort St. Vrain, USA and PAKS, Hungary.

Wylfa's Dry Store comprises three original cells and two new ones, cells 4 and 5. The capacity of cells 1,2 and 3 combined are 300,000 fuel elements that are loaded into the store directly from the fuel transfer machine.

Another remote technology development worthy of note and already in service at Wylfa is the GEC Alsthom Engineering Ltd. Helios manipulator for handling fuel. This electrically driven modular robotic device may be configured to accept payloads between 30 kg and 200 kg with 1.5 m to 5 m reach (Fig. 4). At Wylfa it has a reach of 4.1 m with a payload of 28 kg and is the principal remote handling tool in the diverse discharge facility This facility is required to safely discharge irradiated fuel from Wylfa's dry store - cells 4 and 5.

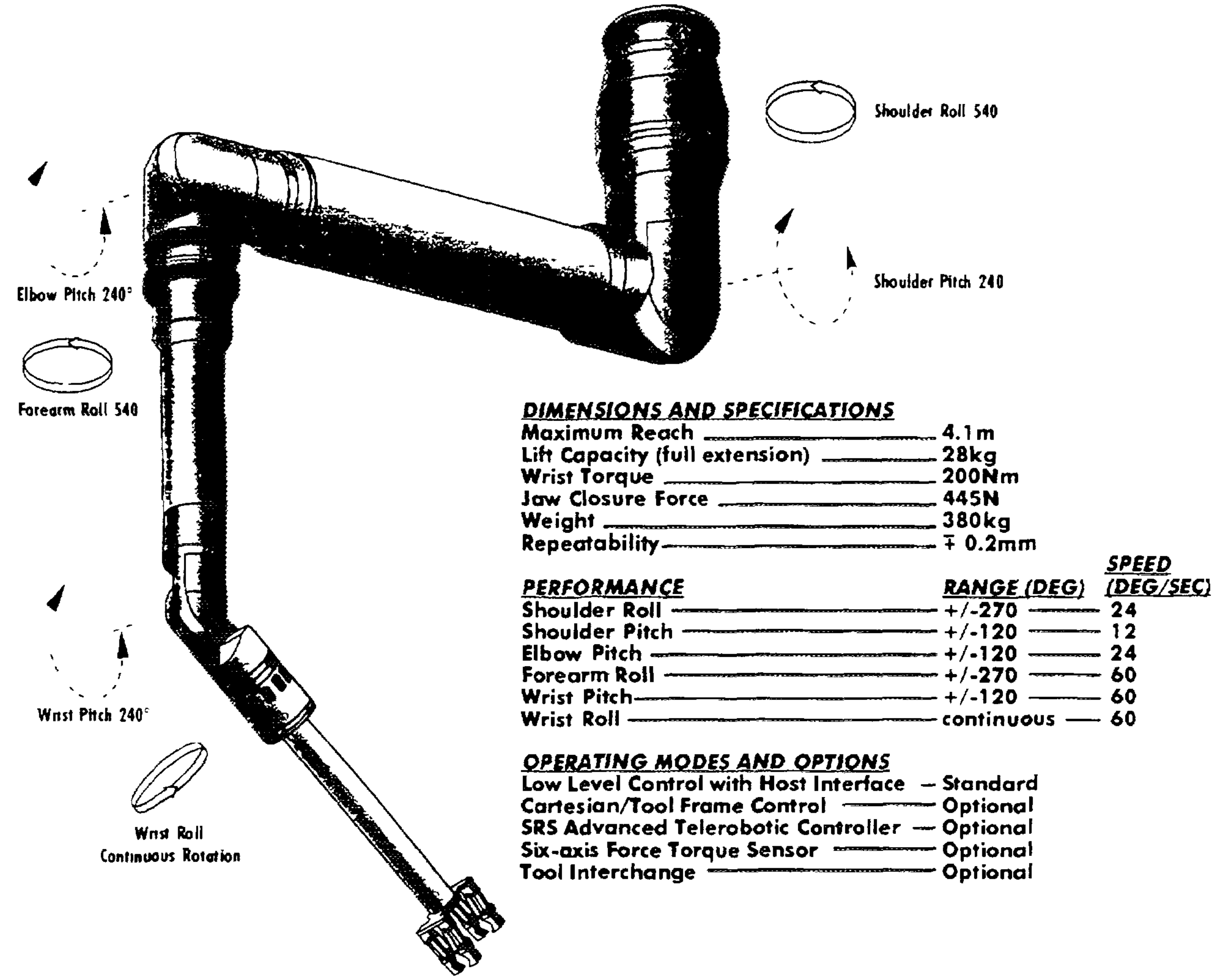

FIG 4 GEC Alsthom Engineering Ltd HELIOS modular advanced robotic arm

5.2. BNFL/Sierra Nuclear Corporation (SNC) TranStor™

BNFL has collaborated with Sierra Nuclear Corporation (SNC) of the USA to provide an advanced storage system [2]. TranStor™ to customers worldwide. This system can provide safe, low cost, high quality interim storage for over 50 years. TranStor™ (Figs. 5 & 6) is a development of the original and highly successful ventilated storage cask (VSC) system, which was designed in the late 1980's and is licensed by the Nuclear Regulatory Commission (NRC) of the United States. It is now in use at a number of reactor sites. Over 100 casks have been built and a further 300 are contracted, many of these being repeat orders.

TranStor™ is designed with simplicity in mind. Within the spent fuel pool the fuel is loaded into a steel canister (located within a shielded cask) and fitted with a shield lid. On removal from the pool, the shield lid is welded into place and a second structural lid is then welded on. The canister is drained of water; vacuum dried and back filled with helium. The shielded cask containing the canister of dry fuel is then placed on top of a reinforced concrete cask. The canister of dry fuel is lowered into the concrete cask through a gate in the base of the uppermost cask. A lid is then fitted to the concrete cask and it is then transferred to a simple reinforced concrete storage pad at the reactor site. Should there be a need to move the fuel away from the reactor site, a shipping cask has been designed to house the canister during transport. This cask meets IAEA standards.

TranStor™ offers superior flexibility in implementation, dry storage capacity and operational alternatives for a full range of fuel types including PWR, BWR, VVER 1000, VVER 440, Candu and RBMK. The flexibility in implementation stems from the modular design and short construction timescales, making it easy for utilities to add capacity as required with minimal lead times, to meet changing spent fuel management strategies

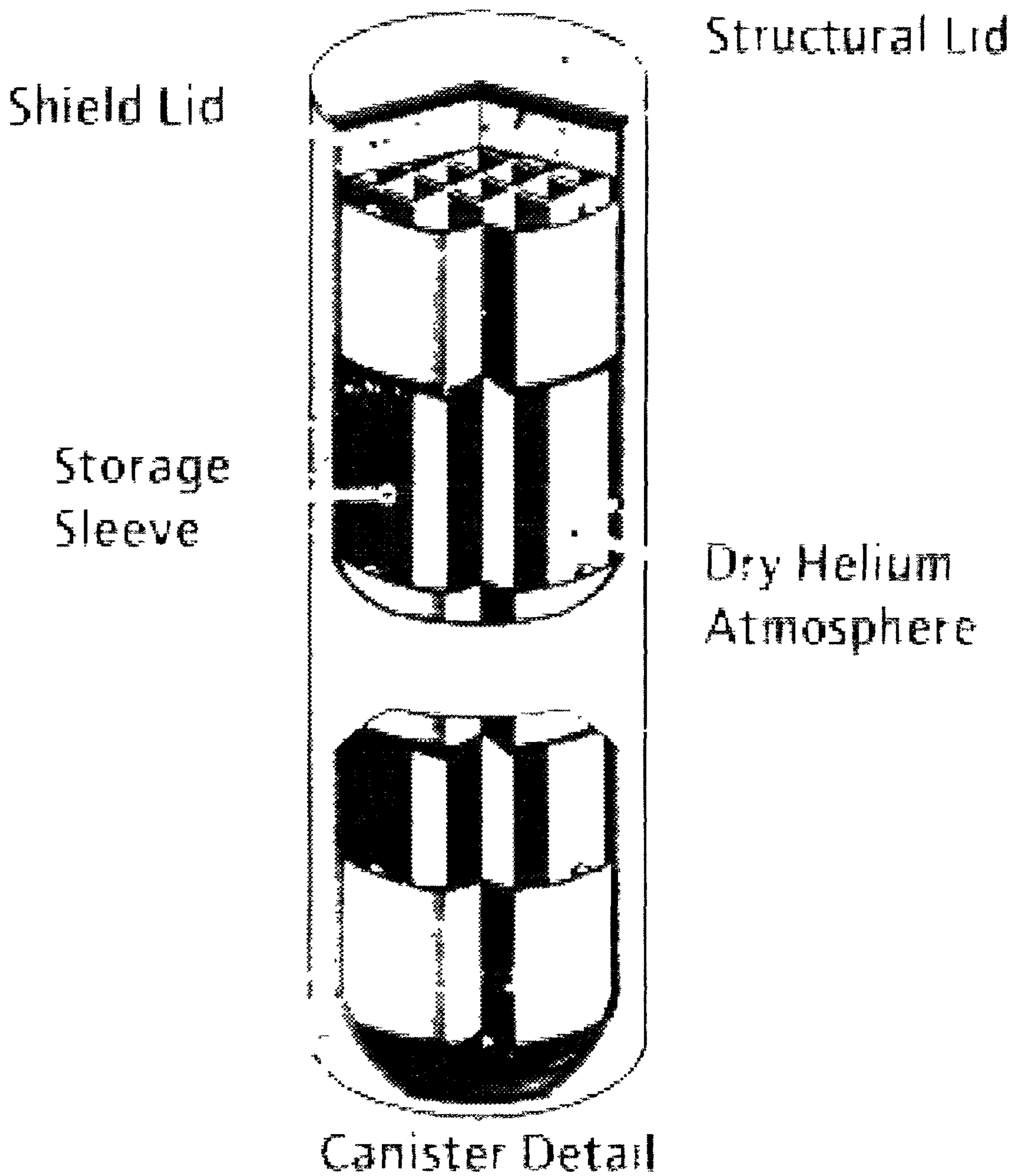

FIG. 5 TranStor™ canister

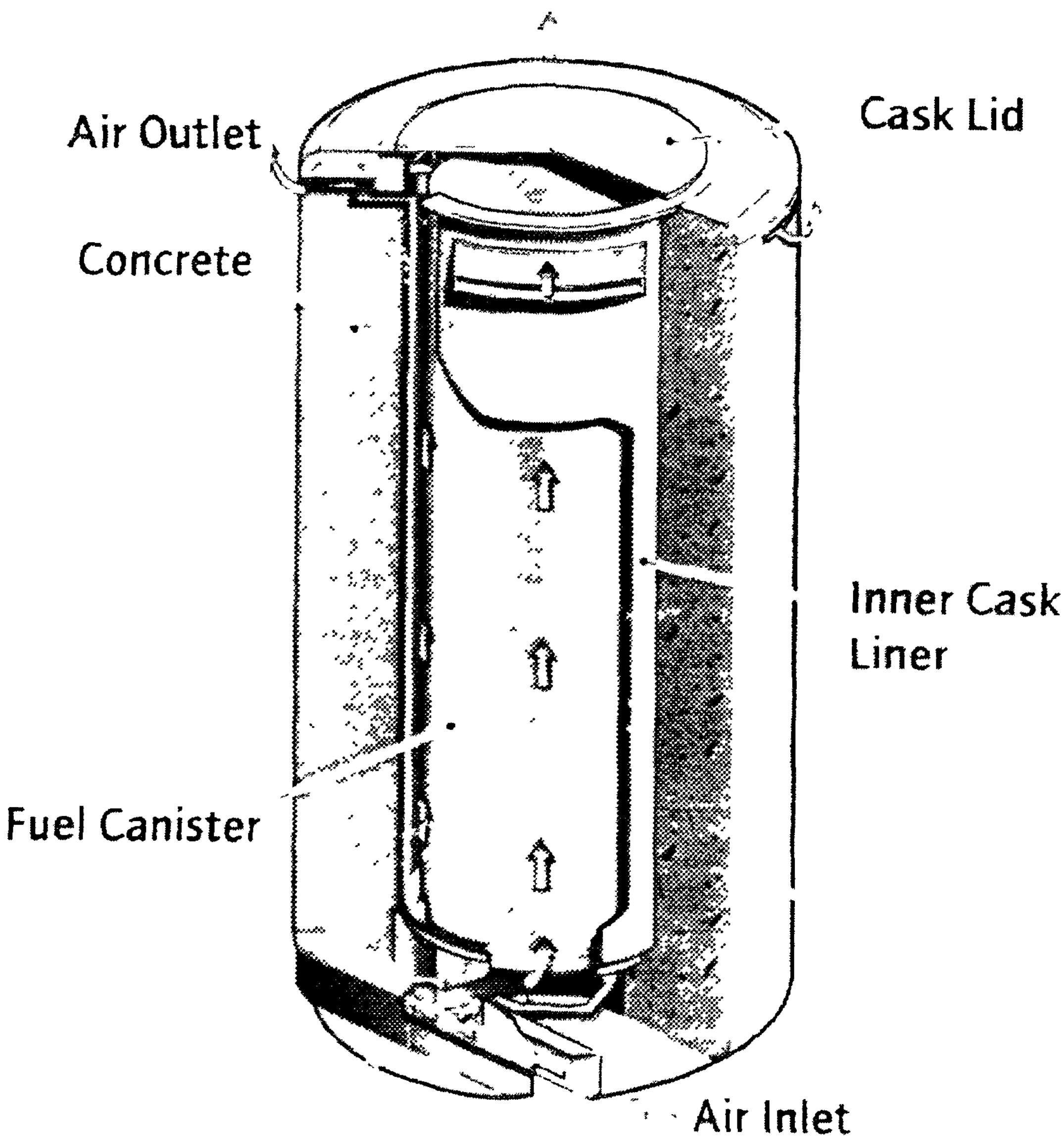

FIG 6 TranStor™ module

The cask and canisters can be adapted to store fuel with a wide range of physical and radiological parameters The system can accommodate the high burnup and MOX fuel designs planned for the future Canisters can also store failed fuel and solid fuel bearing wastes

The TranStor™ fuel canisters are transportable, allowing for fuel to be removed from the reactor site if required, possibly to a central storage site, for reprocessing or final disposal The TranStor™ system is therefore compatible with any future spent fuel policy that utilities may wish to adopt, it keeps the options open

6. CONCLUSIONS

This review of the remote technologies in the UK has confirmed:

- The facilities use mature remote technology;
- It is simple and robust wherever possible;
- Maintenance is more easily carried out than in facilities downstream of the shearing process;
- Higher burnup fuel can be accommodated within the reprocessing and interim storage routes with the same remote technologies as present day fuel;
- The pace of remote technology development is being set by downstream waste plants, decommissioning and clean-up projects; Front end processes will no doubt benefit from resultant spin-offs;
- Fuel vendors are aggressively pursuing developments that will enable higher burnup and higher performance; These developments should improve the final condition of high burnup fuel with respect to corrosion and distortion that in turn will improve the reliability of handling and shearing processes and also increase the maximum period for safe storage.

ACKNOWLEDGEMENTS

Details of Wylfa Dry Store and the Helios manipulator are included by kind permission of GEC Alsthom Engineering Ltd.

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CURRENT US STRATEGY AND TECHNOLOGIES FOR SPENT FUEL HANDLING

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Abstract

The United States Department of Energy has recently completed a topical safety analysis report outlining the design and operation of a Centralized Interim Storage Facility for spent commercial nuclear fuel. During the course of the design, dose assessments indicated the need for remote operation of many of the cask handling operations. Use of robotic equipment was identified as a desirable handling solution that is capable of automating many of the operations to maintain throughput, and sufficiently flexible to handle five or more different storage cask designs in varying numbers on a given day. This paper discusses the facility and the dose assessment leading to this choice, and reviews factors to be considered when choosing robotics or automation. Further, a new computer simulation tool to quantify dose to humans working in radiological environments, the Radiological Environment Modeling System (REMS), is introduced. REMS has been developed to produce a more accurate estimate of dose to radiation workers in new activities with radiological hazards.

1. INTRODUCTION

The United States Department of Energy has recently completed a topical safety analysis report outlining the design and operation of a Centralized Interim Storage Facility for spent commercial nuclear fuel[b]. During the course of the design, dose assessments indicated the need for remote operation of many of the cask handling operations. Use of robotic equipment was identified as a desirable handling solution that is capable of providing remote operations to maintain throughput, and sufficiently flexible to handle five or more different storage cask designs in varying numbers on a given day. This paper discusses the facility and the dose assessment leading to this choice, and reviews factors to be considered when choosing robotics or automation. Further, a new computer simulation tool to quantify dose to humans working in radiological environments, the Radiological Environment Modeling System (REMS), is introduced. REMS has been developed to produce a more accurate estimate of dose to radiation workers in new activities with radiological hazards.

2. THE CENTRALIZED INTERIM STORAGE FACILITY

The United States Department of Energy (DOE) has the responsibility to develop and operate a Civilian Radioactive Waste Management System (CRWMS) that will remove spent nuclear fuel (SNF) from commercial reactors in the United States and dispose of the fuel in a permanent geologic repository. Elements of the CRWMS include temporary storage facilities, transport capabilities, and the long-term repository facilities.

The Centralized Interim Storage Facility, or CISF, provides the temporary federal storage capability for Spent Nuclear Fuel (SNF) under the oversight of the DOE. The DOE Office of Civilian Radioactive Waste Management (OCRWM), and the CRWMS Management and Operating Contractor, recently completed a Topical Safety Analysis Report (TSAR) for the CISF. The TSAR

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[a] A United States Department of Energy Facility

[b] U.S. Nuclear Regulatory Commission Docket # 7221

was submitted to the U.S. Nuclear Regulatory Commission (NRC) on May 1, 1997 and is currently under review, having been docketed on June 10, 1997.

The purpose of the CISF is to provide safe temporary storage of commercial SNF. The CISF will receive, handle, and store SNF in a manner that protects the health and safety of the public and workers, and maintains the quality of the environment in compliance with federal regulations[c].

The storage of spent nuclear fuel at the CISF will be based on the use of cask systems certified by the NRC. These cask systems include transportable storage casks and dual-purpose canister-based storage and transport systems. Facility design capacity is 40,000 metric tons of uranium, translating to approximately 5,300 to 7,800 storage casks depending on the vendor systems. For the preparation of the CISF design, cask systems were utilized that were either docketed by the NRC or under development by the DOE as of June 1, 1996. These cask systems are:

- VECTRA MP187 System
- Holtec HI-STAR 100 System
- Sierra TranStor™ System
- Westinghouse Large/Small MPC System
- NAC STC System

3. CISF RADIATION DOSE ASSESSMENT AND REDUCTION

Initial design and operations of the CISF were based on experience at utility reactor facilities. Utilities perform manual operations for fuel transfer and storage system closure at nuclear power plants, but not on a continuous basis. As the CISF design and dose assessment progressed, it became apparent that the high volume of casks to be handled would result in undesirably high cumulative occupational radiation doses. Table I shows the estimated average annual individual exposure for each of the fuel storage systems. The preliminary values represent the doses expected using manual handling methods, based on information provided by each cask vendor's Safety Analysis Report and utility experience with independent spent fuel storage installations. Clearly traditional hands-on operational doses could be improved using remote-manual or automatic operations.

To maintain minimal radiation doses, the CISF design reflects consideration of the "As Low As Reasonably Achievable" (ALARA) principles given in NRC Regulatory Guide 8.8 and the applicable criteria of Title 10 of the US Code of Federal Regulations Part 72. To reach the ALARA goal of an average dose of 10 person-mSv per year or less, specific measures are adopted to improve CISF operations. These are:

- Design structures, systems, and components (SSCs) that require maintenance or repair such that maintenance frequency and personnel stay times in radiation areas are minimized.
- Utilize robotic and remotely operated equipment and video systems to the extent practical to minimize personnel exposure to radiation sources.
- Place operations personnel in shielded, remote operating stations.
- Use dedicated, shielded transporters for moving casks to the storage area.
- Place administrative, security and radiation protection activities away from radiation areas.
- Use permanent and temporary radiation shielding.

[c] Title 10 Code of Federal Regulations Part 72

- Monitor area radiation with local and remote readouts in the transfer facility.
- Monitor casks in the storage area continuously and remotely.
- Be capable of restricting access in radiation areas.
- Improve ventilation systems for the transfer facility radiation areas, including monitoring of all effluents and filtration systems to reduce possible human exposures and releases of radiation to the environment if accident-level events occur.
- Connect cask venting systems directly to the transfer facility ventilation system to reduce radiological release concentrations and to allow release monitoring.
- Improve decontamination facilities for transportation casks to reduce radiological contamination of other SSCs and personnel during cask handling.

After these measures were applied, a final dose assessment was made, indicating substantially lower average doses. Table I shows a comparison between the preliminary and final dose assessments for receipt and transfer operations for each occupation category, in terms of average milli-Sievert (mSv) *per person per year.*

TABLE I. AVERAGE ANNUAL INDIVIDUAL EXPOSURE ESTIMATE: PRELIMINARY AND FINAL ASSESSMENTS

(person-mSv/person-year)

Cask System	Preliminary			Final (ALARA)		
	Operators[1]	Rad. Prot.	Security	Operators[1]	Rad. Prot.	Security
Holtec HI-STAR 100	41	15	12	3	7	2
NAC STC	46	15	12	6	11	2
TranStor™	78	10	3	14	10	1
VECTRA MP187	83	09	5	27	8	1
Westinghouse Large MPC	75	11	7	6	5	1
Westinghouse Small MPC	73	11	7	6	5	1

[1] Crane operators and prime mover operators are included with general operations personnel.

Another goal of the ALARA review is to reduce the total occupational doses. Table II shows a comparison between the preliminary and final dose assessment in terms of total person-mSv per year. This table clearly shows that the dose reductions made in the final dose assessment are effective from an ALARA standpoint. From 20% to 50% of the reduction is attributable to the use of robotic manipulators specified in the TSAR.

In order to achieve ALARA design goals for the transfer facility, it is necessary to provide facility operations with handling equipment that is automated or remote-controlled. Up to half of the dose reductions in Table I and Table II are attributable to remote/automated equipment. Such handling equipment provides the ability to remotely manipulate objects and safely perform repetitive work operations, which minimizes the annual radioactive dose for operating personnel while maintaining cask throughput rates.

TABLE II. TOTAL ANNUAL OPERATIONS DOSE ASSESSMENT COMPARISON

(person-mSv per year)

Cask System	Preliminary Dose Assessment			Final (ALARA) Dose Assessment			ALARA Dose Reduction
	Receipt[1]	Maint.[2]	Total	Receipt[1]	Maint.[2]	Total	
Holtec HI-STAR 100	510	10	520	50	10	60	460
NAC STC	570	9360	9930	80	10	90	9840
TranStor™	1660	18860	20520	260	10	270	20250
VECTRA MP 187	1210	9460	10670	440	10	450	10220
Westinghouse Large MPC	1830	9460	11290	170	10	180	11110
Westinghouse Small MPC	1790	9460	11250	170	10	180	11070

[1] Based on receipt of 232 casks per year. Receipt includes all operations to receive transport casks, prepare these for transfer, transfer canisters to storage casks, and place storage casks in the storage yard.

[2] Based on 20,000 metric tons uranium (MTU) in storage area. Maintenance includes all inspections of storage systems in the storage yard required as per technical specifications

The improved handling procedures resulting in the final dose estimates of Tables I and II were the result of general concept specifications. The following describes remote and automated equipment assumed in the improved CISF design.

The CISF utilizes a gantry-mounted robot in the shipping/receiving area, with a support frame for the manipulators spanning three rail/truck lanes. The gantry frame can travel the width of the shipping/receiving area. Two robotic arms (manipulators) are supported on a platform to perform precise and accurate tasks. Cameras are used to aid operators in observing tasks performed by the gantry-mounted robot via closed-circuit television (CCTV) monitors located in the crane operating room. In addition, monitors are provided in a room below the crane operating room so that workers in the shipping/receiving area can both observe operations from a low radiation dose area, and have convenient access to the work area when hands-on operations are required.

Stationary-mounted manipulators are provided in the canister transfer area for performing activities on both the Westinghouse and TranStor™ transportation casks. The transportation casks are placed on indexed locations on the canister transfer area floor, such that the manipulators can access necessary areas of the casks. The manipulators are used in conjunction with cameras located in the canister transfer area. CCTV monitors are provided in the shielded remote console rooms located along the walkways at the end of the canister transfer area. Control consoles for operating the manipulators and other automated equipment are located in the remote console rooms.

As part of the automated operations for ALARA radiation dose minimization during transfer preparation, automated bolt/stud tensioners and alignment devices are included. These devices are similar to widely used, commercially available devices with high bolt torquing capability, and will be used to remove, retain, and reinstall the transportation cask trunnions, retainer blocks, lid bolts, and venting/sampling ports with their associated bolts or studs.

Automated alignment devices are also included for operations involving removing and installing impact limiters and personnel barriers, and aligning casks for canister transfer operations. Automated equipment can be remotely controlled from the monitoring room located below the crane operating room and the remote console rooms at the end of the canister transfer area.

After NRC approval of the CISF TSAR, additional work will be required to transform the general specifications of the TSAR into detailed equipment requirements and specifications, and to assure that interface requirements for the cask systems are met. As part of this process, further consideration will be given to equipment needs, and additional dose assessments will be conducted as equipment and procedural specifications evolve. The next two sections offer considerations regarding automation and robotics, and introduce a tool that could be applied for faster and more accurate dose assessments.

4. AUTOMATION AND ROBOTICS

Automation and robotics have been used in some operations and considered for many others in the nuclear industry to reduce the hazards, increase work quality and provide a more rapid response to developing needs. This section discusses the topics of robotics and automation in general terms, and relative advantages of each.

To *facilitate discussion, this definition of terms is offered.*

Automation - Automation may be defined as automatic control of a system by mechanical or electronic devices that replace human observation, effort and decision.

Hard or fixed automation - Non-programmable, fixed tooling which is designed and dedicated for specific operations. Hard automation is cost effective for a high production rate. It is typically not easily changed to accommodate new operations.

Robot - According to the Robotic Industries Association in the USA, "A robot is a reprogrammable, multi-functional manipulator designed to move material, parts, tools, or specialized devices through variable programmed motions for the performance of a variety of tasks."

Flexible automation - The ability to reprogram or multi-task an automated system. Robots are considered flexible because they are capable of redirection or being used for new purposes.

Teleoperation - The remote control of manipulators or other machinery by direct manual input. This is commonly seen in the nuclear industry as mechanical or electro-mechanical manipulators in hot cells. Teleoperated devices, not being programmable, are by definition not robots.

Telerobotics - The control of a manipulator by direct human input is augmented by computer control. This hybrid maintains the human decision capacity while relieving the operator of many details (such as joint positioning) and increasing sensor integration opportunities (such as obstacle avoidance).

When deciding upon an automated approach, a cost-benefit analysis (CBA) should be conducted. The CBA process generally consists of the following steps: problem definition, analysis design, data collection, and option analysis with respect to the costs and benefits. One example of CBA applied to nuclear fuel cask handling is given in [1]. Care should be taken to define the problem with the proper scope. For example, if a robot is compared to a long-handled tool to do a single task, the costs of the robot may overwhelm that of the tool, therefore precluding the use of the robot. However, the cost of the robotic system may be amortized over tens or hundreds of different operations making it a more attractive solution than it may seem.

Care should be taken to identify and consider all monetary, quantitative and qualitative non-monetary benefits. Monetary benefits include direct capital cost, operational cost, labor cost, and waste disposal costs. They also include indirect or derived costs, such as those due to throughput rates, waste generation, lawsuits, and design, approval and construction time factors. Non-monetary quantifiable benefits include the radiation dose. Some attempts have been made to assign a monetary value to a unit of dose. However, in many situations, simply stating dose units is necessary and sufficient to meet regulatory limits or design goals without a monetary transformation. Finally, non-monetary qualitative factors such as worker morale and social impact, can affect the quality of work or time to operation, and thus the technology choice.

Several differences between robots and hard automation may have an impact when deciding upon an automated approach. The case for hard automation can be made based upon throughput and simplicity. When large numbers of identical workpieces are to be manipulated at the highest possible speed, hard automation may be optimal. Costs associated with unique design can be quickly amortized. Hasegawa [2] maintains that to break even in a hard automation assembly process, approximately 200,000 products must be assembled. Special-purpose automation can also be optimized to minimize parts and subsystems. This could lead to greater reliability and lower maintenance costs.

When throughput is lower, or when a mix of products is being processed by the same line, robots may be more desirable. First, robots are reusable. Unlike hard automation, robots are multi-purpose and can be reprogrammed for many different tasks. A large portion of equipment and experience can be applied to new tasks. Because of this, the useful life of the robotic system may be three or more times longer than that of fixed automation devices. Second, reprogramming can result in greater utilization and higher equipment efficiency. Robots have the ability to rapidly adapt from one workpiece to another, such as different container types, and the throughput rates can be scaled up or down for the different types. Third, tooling costs for robotic systems tend to be lower. This is due to the machine's dexterity, giving it the ability to move around some physical constraints. Finally, production can often be started sooner due to fewer construction and tooling constraints.

Remote automation may be considered an advantage when dealing with several thousand transport containers or nuclear fuel assemblies per year. Fixed automation may be ideal for container opening, fuel handling, packing, and closure operations, provided that there is little or no variation in the workpieces. However, in the case of the CISF, no less than five different cask systems may be used for fuel storage, each with different sets of tools and operations. In this case, the CISF will benefit from a more flexible automated system, allowing virtually instant reprogramming to manipulate each of the anticipated cask systems, and any others that may be licensed in the future. Further, the robotic system may be modified to provide a telerobotic mode, which could be of particular benefit in the recovery from off-normal conditions where pre-programming does not apply.

5. THE RADIOLOGICAL ENVIRONMENT MODELING SYSTEM (REMS)

In the nuclear industry, the decision to use remote, automated or robotic equipment is typically driven by the need to reduce radiation doses to workers. In new facility designs, dose measurements are not always available and experience with particular operations may not yet exist. Therefore, estimates of anticipated doses must be made.

A relatively new tool to quantify dose to humans working in radiological environments is the Radiological Environment Modeling System (REMS) [3]. REMS utilizes commercially available graphical simulation products, augmented with custom C code and radiation transport codes to provide radiation exposure information to, and collect radiation dose information from, graphically animated workcell simulations.

To analyze the radiation doses likely to be imparted by a set of operations, the operations are first simulated using human models in graphical simulation. An example of this is shown in FIG. 1, where a neutron source is being reduced physically and chemically to separated elements. The simulation is then presented to knowledgeable operators for process validation.

REMS utilizes the IGRIP (Interactive Graphical Robot Instruction Program) simulation software and its ERGO human ergonomic assessment extension from Deneb Robotics, Inc. The human model has been modified by SNL to include 43 sensor locations at regulated and sensitive parts of the body as illustrated in Figure 2. Any combination of the sensor locations may be selected to meet the specific needs of the user.

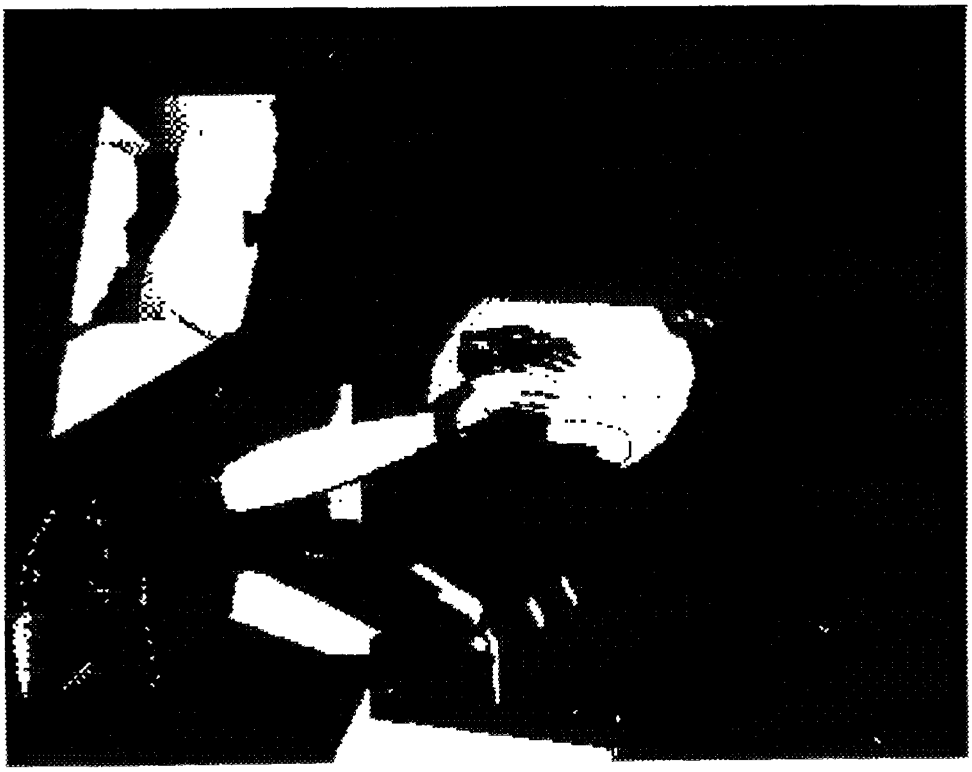

FIG. 1. REMS tracks radiation doses to workers in a neutron source dismantlement operation.

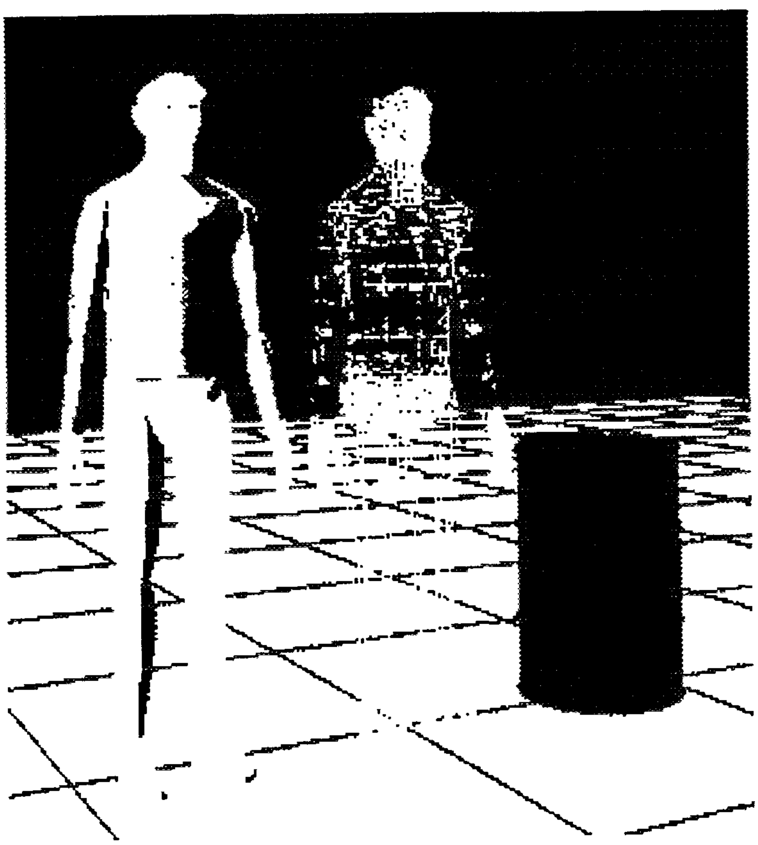

FIG. 2. REMS human model instrumented with dosimeters

Through the use of radiation transport codes or measured data, a radiation exposure input database may be formulated. The REMS suite of computer codes currently includes a 1-dimensional modeler and a material properties database to assist the operator in the creation of point-source geometries. Any shielding is taken into account at this stage. A transport code and selectable tissue damage databases are then used to map the radiation dose rates in the facility of interest.

The simulations utilize these maps to compute and accumulate doses to the human models operating around radiation sources. Process time, distances, shielding, and human/machine activity may be modeled accurately in the simulations. The accumulated dose is recorded in output files, and the user is able to process and view this output. The entire REMS capability can be operated from a single graphical user interface.

The REMS analytical tool provides several benefits beyond conventional spread-sheet analysis. First, the simulation is available for validation. Operators can verify that each simulated process is accurate by visual examination. Secondly, as the simulation executes, each body movement is accounted for in the distance calculations. This results in greater detail than is normally practical in spread-sheet accounting, and thus in greater accuracy of the integrated dose calculation. If measured dose rate data are used for the dose maps, REMS could result in the best dose estimation available. A third benefit of REMS is that dose calculations may be easier to defend to clients and regulatory agencies. The ability to demonstrate new processes visually lends confidence to the observer that the calculations are complete and accurate, while facilitating dialog and feed-back. Finally, REMS simulations can be used as training aids, familiarizing trainees with various equipment and processes.

To date, REMS has been used to analyze manual operations for radiation exposure, and to identify possible candidates for automation at three DOE nuclear material handling locations.

6. SUMMARY

Dose assessments of CISF operations based on traditional manual operation of spent nuclear fuel storage cask systems indicate the need for substantial dose reduction measures. Robotic systems are considered a critical component of these measures, contributing 20% to 50% of the dose reduction in the final CISF dose assessment improved ALARA measures.

Robotics and automation are both useful techniques. Hard automation may be simpler and is best used when many thousands of identical operations are to be performed at high speed. However, it is difficult to quickly modify and does not lend itself well to a mix of operations or workpieces. Robotic automation is more flexible, enabling rapid changeover from one type of operation to another. This flexibility is an advantage in the case of the CISF, where five or more different fuel storage systems may be utilized.

REMS is a new dose assessment tool now being used to analyze worker doses in nuclear material processing lines. REMS combines a source geometry modeler, radiation transport codes and dose conversion standards (or measured dose maps) with computer animation for improved dose assessment resolution. The graphical simulation facilitates operations validation and training. It also accounts for body movement with high sampling rates, improving integrated dose calculations. REMS has been used in three nuclear material handling dose analyses.

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