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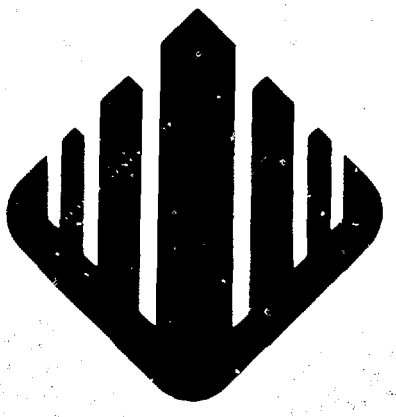

1987 INTERNATIONAL DECOMMISSIONING SYMPOSIUM

OCTOBER 4-8, 1987
D.L. LAWRENCE CONVENTION CENTER
PITTSBURGH, PENNSYLVANIA, USA

PROCEEDINGS

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PROCEEDINGS
of the

1987 INTERNATIONAL DECOMMISSIONING SYMPOSIUM

Pittsburgh, Pennsylvania
October 4-8, 1987

Edited by
Gail A. Tarcza

Westinghouse Hanford Company
Richland, Washington

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DECOMMISSIONING PROJECT EXPERIENCE II

DECOMMISSIONING OF GENTILLY-1 - CANADA

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ABSTRACT

Canada's 250 MW(e) Gentilly-1 station went into service in 1971 in the Province of Québec, Canada, and produced electricity intermittently until 1979. Since April 1986, the station has been decommissioned to a "Static State" following a two-year, 25 million dollars (Canadian) project.

Access to the Reactor Building has been permanently shut off except for one airlock that can be made operable for periodic inspection. Systems inside the Reactor Building have been isolated and sealed. All radioactive materials and components inside the Service Building have been removed, the area decontaminated and released to the local utility Hydro-Québec for a full scale simulator training centre. Spent fuel has been put in interim on-site storage inside concrete canisters. The spent fuel bay has been decontaminated, and a slab poured on top of it.

INTRODUCTION

Gentilly-1, (G-1) a 250 MW(e), 833 MWth, CANDU (CANada Deuterium Uranium), type reactor is located in Québec, Canada. This plant was a natural uranium fuelled, heavy water moderated, boiling light water cooled reactor. G-1 was put in service in 1971 and operated intermittently until 1978. In 1980 the station was placed into a state of preservation which could have permitted a restart. In 1982 the Senior Management committee of Hydro-Québec and Atomic Energy of Canada Limited recommended against the rehabilitation of the plant on economic grounds. Parts of 1982 and 1983 were devoted to engineering and economic studies related to the decommissioning of Gentilly-1.

A wide range of decommissioning scenarios (stages) were considered. These varied from safe storage (of all radiological hazards inside the plant) with surveillance, to prompt dismantling of the total plant and disposal of all equipment and structures for ultimate release of the site for unrestricted use. Decommissioning to IAEA Stage 3 was ruled out because of G-1's proximity to the 600 MW(e) G-2 operating plant and also due to the lack of existence of a commercial radioactive waste disposal facility in Canada. After careful consideration, a decision was made to decommission the plant to a "static state" condition, which is a variant of IAEA's Stage 1 (Ref. 1).

OBJECTIVES

There were two major objectives involved in undertaking this project: to reduce yearly operations and maintenance cost of G-1 from nearly $10 million (Canadian) to $1 million, and to gain hands-on experience in decontamination and decommissioning.

SCOPE, SCHEDULE AND BUDGET

The following major activities were completed as the project scope:

- On-site interim dry storage of spent fuel.
- Putting the reactor building, and systems/equipment in it, in a static state condition.
- Removal of all components from the Service Building and parts of the Turbine Building, decontamination and transfer of these areas to the local utility Hydro-Québec for use as a simulator training centre.
- Decontamination of spent fuel bay.
- Development of procedures and the Safety Analysis report to obtain a decommissioning license (and convert the plant into a waste storage facility) from the Canadian regulatory authority, the Atomic Energy Control Board (AECB).

The schedule for the actual decommissioning work was two years (1984 April to 1986 March).

The budget was set at $25 million (Canadian) for the two year project.

DRY STORAGE OF SPENT FUEL

Generally speaking, fuel related work is not part of the scope of a decommissioning project. However in the case of G-1 the work scope of putting the entire facility into a static state required that the spent fuel pool be decommissioned. There being no national spent fuel repository yet in existence in Canada, an engineering solution to the problem was devised.

Some 3213 bundles of irradiated fuel and activated fuel hardware were stored in eleven cylindrical concrete canisters that were specially designed for the purpose. Each canister (6 metres high with an outside diameter of 2.6 metres) has a steel liner that serves as the storage cavity for eight specially designed stainless steel baskets. Each basket contains 38 fuel bundles, each of which is placed over one basket pin. (The bundles had been stored in the spent-fuel bay for a minimum of 7 years and the decay heat emitted averaged approximately 1.4 watts per bundle at the time the program was started.) Irradiated fuel is contained in 85 baskets, while two baskets contain flux suppressors and keys that were parts of the fuel string.

TYPICAL STORAGE OPERATION

A typical storage operation started by unstringing the bundles and removing the central structural tubes (CST) that contained cobalt-60. Each bundle number was then identified and each bundle was then individually loaded into a numbered basket. Both the basket and bundle numbers were permanently recorded in keeping with IAEA safeguards requirements. Once a basket was filled, a cover was put on and the basket assembly was raised into a shielded station by a grapple that engaged inside a central opening. All these operations were done underwater for shielding purposes.

Inside the shielded station, the basket cover was removed, the fuel assembly was dried, and the basket assembly was seal-welded along the top and bottom by remote semi-automatic welding. The basket was then moved laterally inside the shielded station to a position directly below the on-site transfer flask. The flask doors were then opened, a grapple inside the flask was lowered to raise the basket from the shielded station into the flask, and the flask doors were closed. The flask was then ready for transfer to the canister area.

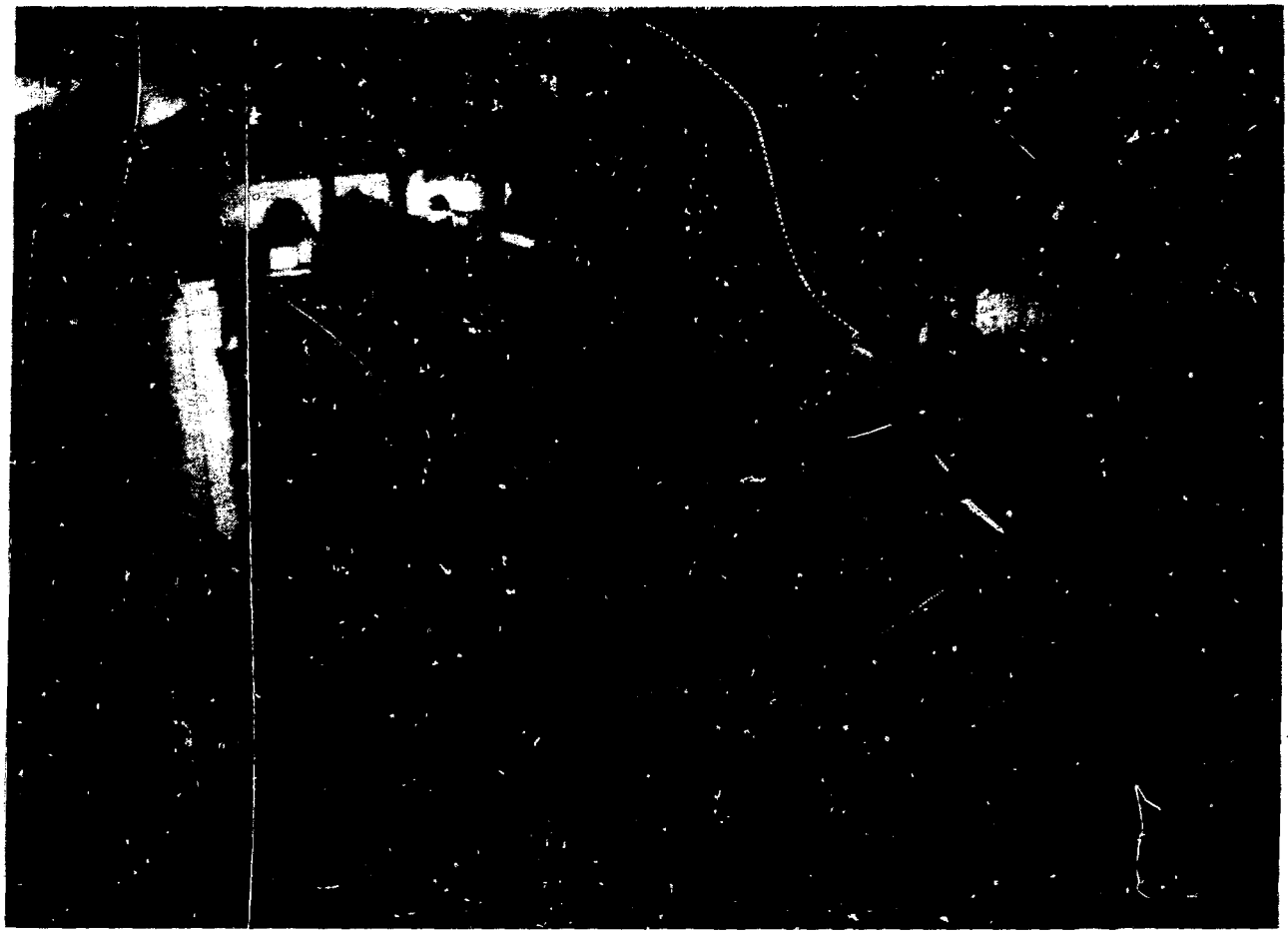
Inside the shielded station, radiation fields of up to 3500 rem were measured. However, outside the station (and the flask) radiation fields were always less than one millirem (on contact).

To transport the flask to the canister site, a specially designed trailer was used. The flask was then raised to the top of a canister by a 15-ton crane. The grapple assembly and a 3-ton crane lowered the basket from the flask into the canister. The top plug was then welded to the canister, and an IAEA safeguards seal was installed by Agency inspectors.

From start to finish, this operation took a crew of six an average of three hours. Since the process was repeated for all 11 canisters and the eight baskets each contained, the entire fuel-storage operation took six weeks, not including five weeks for construction of the canisters themselves.

The radiation level outside a loaded canister at the time the program was completed was about 0.6 millirem (on contact).

A surveillance program on-going since the project completion shows a radiation profile on contact outside 11 canisters to be between 0.4 to 0.6 millirem per hour.



Shielded station used in G-1 for fuel transfer operation.
Spent fuel bay is on the left.

THE REACTOR BUILDING

Most common pipes, cables, ducts, etc., between the reactor, turbine, and service buildings were cut and sealed to prevent the spread of radioactivity. All components in the reactor building were drained and dried, oil and other inflammable agents were removed; systems inside the reactor building were isolated and tagged, thus bringing it to a static state.

Access to the reactor building has been permanently shut off, except for one airlock that can be made operable for periodic inspection. The static state will be maintained for the reactor building for at least the next 40 or 50 years, with periodic inspection to ensure structural integrity. The delay will mean a significant reduction of radioactivity which will facilitate final dismantling.

SERVICE AND TURBINE BUILDINGS

All components including ventilation ducts and partition walls were removed, rendering all of the service building and parts of the turbine building into an empty shell consisting of only the structural slabs and exterior walls. After the fuel was removed, the spent fuel bay was decontaminated, and a new concrete slab was poured on top of the bay. All the areas in the service building and parts of the turbine building were transferred to Hydro-Québec who is presently installing a full scope training simulator for the adjacent 600 MWe Gentilly-2 reactor.

DECONTAMINATION WORK

Engineering and economic studies, and initial site experience had shown that large-scale system decontamination to release components for unrestricted use was neither time nor cost effective. Nonetheless, a significant decontamination programme was set up at the site to meet the criteria (Zone-1) to transfer areas for other uses, and to gain "hands-on" experience for determining future methods and estimating manpower and cost requirements.

Zone 1 criteria required no loose contamination, no beta fields above 1.0 millirem per hour and no gamma fields above 0.25 millirem per hour at 1 cm (Ref. 2).

Major experience was gained through decontamination of the feedwater and hydrazine dosing system, feedwater sampling system, various sizes of piping, fuel trays, new fuel inside the spent-fuel bay, several ventilating ducts and fans, and the spent fuel bay itself.

A Butterworth hydrolaser model 110 ET was used extensively (at pressures up to 6000 pounds per square inch) (Ref. 3).

RADIATION PROTECTION

All aspects of the Gentilly-1 decommissioning were regulated by a license from the AECB, which insisted on the maximum health and safety protection for workers and the public. To satisfy these requirements, and the ALARA principle, several documents were developed, specifically health and safety guidelines and radiation protection standards; a health and safety manual; and radiation protection procedures.

The health and safety group at site produced and distributed a computerized report that showed biweekly dose exposure of everyone working on the project. The individual doses were much lower than allowable (5 rem per year, 3 per quarter). The maximum recorded dose for a 12-month period for an individual was 225 millirem. However, the individual dose for most workers was less than 110 millirem.

PROJECT MANAGEMENT TEAM (PMT)

The PMT key members came from different divisions of AECL; namely, the AECL Research Company and CANDU Operations, as well as from other private companies involved in the nuclear energy field. The team worked very well as a unit, with each person fascinated by this challenge, resulting in an environment that was positive, productive and highly imaginative. The team developed a constructive philosophy and the following major goals towards the project implementation:

- This would be an "engineered dismantling" and not a wrecking job. AECL staff would lead all work involving active or contaminated components. Outside contractors would work in clean areas only. Workers' safety was paramount before any other consideration.

- A comprehensive training program was set up for all newly hired employees.

- Follow ALARA (as low as Reasonably Achievable) regarding personnel exposure.

The site organization was headed by a Station and Project Manager, and there were seven managers/supervisors responsible for Resident Engineering, Decontamination, Radiological Protection, Fuel Handling, Operations, Plant Services, and Security. To preserve their independence, the heads of Health and Safety and of Quality Assurance had reporting lines separate from that of the Station/Project Manager.

The total site staff averaged around 100-130 persons; 40 professional/technical and 50 craftspersons from AECL. Outside contractors had anywhere between 15 to 25 workers at one time.

Detail work plans and procedures including radiological review adhering to the ALARA, were prepared by the Engineering and Operations groups and were reviewed and approved by the Health and Safety groups before submitting to the AECB for their approval. Although this approval process somewhat slowed the work progress, it ensured that proper documents, drawings and procedures were prepared and records kept. This, clearly contributed to the fact that this two year project was completed without any significant radiological incident.

The total project scope was completed on schedule and under budget.

The project also met its objective of reducing the yearly operations and maintenance cost and providing hands-on experience in decommissioning.

Until 1984, about one hundred persons worked at G-1 on operation, maintenance and security activities. Since April 1986, the station has been totally unmanned. The canister room is equipped with cameras and alarms which are connected by microwave link to the nearby AECL La Prade heavy water facility.

There is a surveillance program whereby the air monitors and radiological profiles around the canisters are inspected and reported to AECB on a bi-annual basis. The reactor building is inspected once a year to confirm structural integrity, and to check air quality and for the presence of moisture. Hydro-Québec controls the access to G-1 through G-2 guardhouse, thus providing physical security.

Operations and maintenance cost have been reduced from $10 million to less than one million per year starting in April 1986. This cost saving alone will more than pay for this project.

From our observation, although many factors and processes contributed to the success of this job, the Project Team, being as imaginative, cohesive and talented as it was, was unquestionably the most important factor in this success.

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THE DECONTAMINATION AND DECOMMISSIONING OF THE CHEMICAL PROCESS CELL (CPC) AT THE WEST VALLEY REPROCESSING FACILITY

By
R. E. Lawrence and
R. A. Meigs

May 1987

ABSTRACT

To support interim storage of vitrified high level waste at the West Valley Demonstration project, the shielded, remotely operated Chemical Process Cell (CPC) was decommissioned and decontaminated. All equipment was removed, packaged and stored for future size reduction and decontamination. Floor debris was sampled, characterized, and vacuumed into remotely handled containers. The cell walls, ceiling, and floor were decontaminated. Three 20 Mg (22.5) ton concrete neutron absorber cores were cut with a high pressure water/abrasive jet cutting system and packaged for disposal. All operations were performed remotely using two overhead bridge cranes which included two 1.8 Mg (2-ton) hoists, one 14.5 Mg (16-ton) hoist, and an electro-mechanical manipulator. Additional operations were performed by an industrial robot mounted on a mobile platform. Initial general area dose rates in the cell ranged from 1 to 50 R/hr and were reduced significantly. Target levels of less than 10 mR/hr general area readings were established before decontamination and decommissioning was initiated; the paper will summarize the specific results obtained.

1.0 INTRODUCTION

The West Valley Demonstration Project is a Congressionally mandated activity conducted through the U. S. Department of Energy, Idaho Operations Office. The operator of the Project is West Valley Nuclear Services Co., Inc., a subsidiary of the Westinghouse Electric Corporation.

The objective of the project is the solidification of high level waste (HLW) generated by commercial nuclear fuel reprocessing which was conducted at the site from 1966 to 1972. Site operations became a U. S. Department of Energy responsibility in 1982.

To support the interim storage of solidified HLW, a large shielded cell is required. To minimize construction costs and maximize reuse of existing contaminated facilities, the Chemical Process Cell was

selected as the interim storage area. Hence, the decontamination and decommissioning of the cell became a contract master schedule milestone, well in advance of the anticipated solidification start date. This report discusses the work required to complete the decontamination and decommissioning work in the CPC.

2.0 DESCRIPTION OF THE CPC

The CPC is a remotely operated hot cell of the canyon type design. It is 28 m (93 ft) long, 6.7 m (22 ft) wide, and 13 m (43 ft) high. Overhead bridge cranes provide maintenance and remote handling capability. Four lead glass, oil immersed viewing windows allow control of the cranes from the operating aisle. There are two separate bridge cranes on two sets of crane rails. A 14.5 Mg/1.8 Mg bridge passes over top of a 1.8 Mg power manipulator combination bridge. The cell was originally built as a fuel dissolution and HLW evaporator area. Chopped fuel was handled in baskets using the 1.8 Mg cranes. Dissolution was performed in two dissolver vessels weighing 11.8 Mg each. Operating dose rates in excess of 10E5 R/hr required concrete shielding walls 1371 mm (54 in) thick.

Support cells include the Equipment Decontamination Room (EDR), the Chemical Crane Room (CCR), and the Scrap Removal Room (SRR). The EDR is connected by a tunnel with a shield door partition. A rail type car allowed transfer of equipment between the EDR and the CPC. The CCR allowed maintenance of the cranes behind a second shielding door in a low radiation field work area. The SRR was used to transfer leached cladding pieces (hulls) to a disposal area.

At the outset of the decontamination and decommissioning work in the CPC the dose rates in the CPC ranged from 12 to 56 R/hr as measured during a remote survey. Smear samples taken during the surveys indicated smearable contamination levels in excess of 10E9 dpm/100 cm^2 beta. A video survey showed debris and equipment strewn across the entire cell.

Indications of gross leakage of several active solution transfer lines were obvious. During plant operations, boiling over of the dissolver vessels was a well documented occurrence. Failed vessels were removed during reprocessing operations on three occasions with typical vessel contact dose rates of 100 R/hr.

3.0 PREPARATORY WORK

Prior work to support the decontamination and decommissioning of the CPC included decontamination and decommissioning of the EDR, the CCR, and the SRR. In addition, the CPC cranes were overhauled exten-

sively. Just prior to the start of the CPC work, the oil immersed shielding windows began to degrade noticeably. The cost and schedule considerations for removal and total refurbishment led to an engineered, in-situ repair procedure developed at West Valley. Since the windows were in good condition until only recently, it was reasoned that chemical or radiolytic damage to the glass in the windows was not likely and that the visibility problems were limited to cloudy oil. Cloudy oil is typically caused by oxidization of the oil in the presence of air and moisture. The uncontaminated oil was drained and the windows were briefly filled with Freon-113. The Freon was drained and found to be free of radioactive contaminants. The windows were purged with nitrogen to remove air, moisture, and Freon; and the windows were refilled with new oil. Optical clarity approached that of a new window and has remained excellent for two years. The procedure was attempted on windows in another cell to support other decontamination and decommissioning work. Since these windows had degraded much earlier, results were unsatisfactory, probably due to etching of the lead glass. Figure 1.0 shows the view into the CPC at the beginning of decontamination and decommissioning.

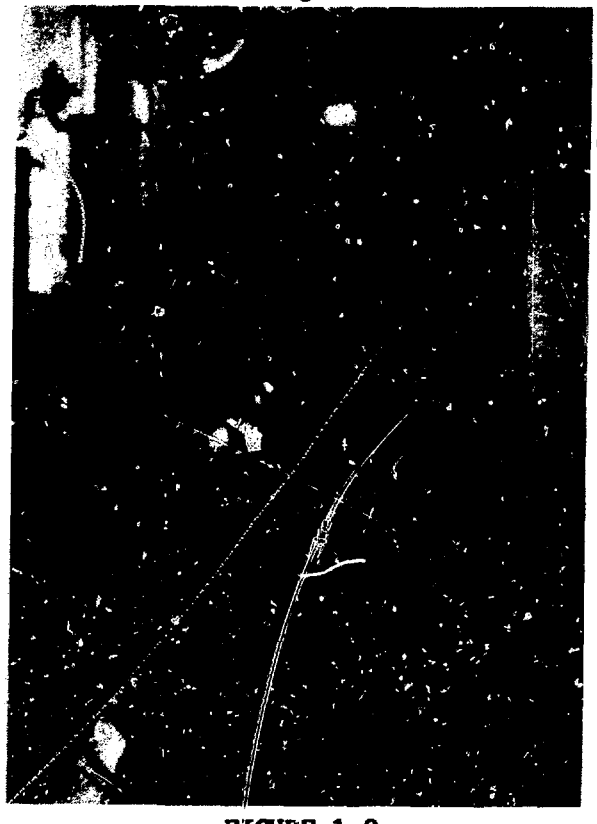

FIGURE 1.0
CPC At The Beginning of Decontamination and Decommissioning

4.0 REMOVAL OF EQUIPMENT

Due to the fast track nature of this decontamination and decommissioning work, an early decision was made to size reduce vessels at a dedicated size reduction facility rather than to attempt remote, in-place size reduction in the CPC. Two options were investigated for the removal of the vessels. The first included overpacking the vessels in shielded containers and storing them outside the facility.

The second option was packaging in strong-tight containers without shielding and transporting the packages to a shielded storage area. Shielded containers were eliminated from consideration due to the expense,

excessive weight, and the one-time-use nature of the containers. Instead, a multi-use shielded storage area concept was developed. Concrete storage modules were filled with low dose rate, high density waste in a compound surrounding the vessel package lay down area. Forty-five modules were required to surround the 6.7 m x 36.6 m (22 ft x 120 ft) area. The entire storage area was covered by a tensioned fabric structure which was mounted on wheels that ran in tracks on the ground. Overall dimensions for the structure are 57.3 m (188 ft) long, 15.2 m (50 ft) wide, and 7.6 m(25 ft) high at the peak. The structure is moved out of the way to allow crane handling of packages as large as 3.4 m (11 ft) x 3.4 m (11 ft) x 6.1 m (20 ft) with no overhead interference.

Remote removal of jumper piping connections was performed in a typical manner using a crane suspended impact wrench to loosen the Purex connectors. Jumpers were loaded into a container on the transfer car, the container was closed using the power manipulator, and then transferred to the EDR. Figure 2.0 shows a loaded container prior to movement to the EDR. In the EDR the container was over-packed into an outer, contamination free container. Pre-staged wrappers were folded into the container using the EDR cranes, the box lid was closed, and the box was sealed using a small crane handled impact wrench. A container radiation survey was performed using a probe on the EDR cranes and the sealed box was moved out of the plant to the storage area. All work was planned to keep personnel access requirements to a minimum. The closest approach by workers was a Radcon smear survey just prior to removal of the container. On several occasions

FIGURE 2

Container Loaded With Jumpers And Piping

decontamination of a loaded container was needed. Total group exposure for these decontamination efforts was as high as 0.5 Man-Rem. Size reduction of jumpers in the CPC was kept to a minimum by using large containers and preplanning as much as possible how to load long jumpers efficiently into the storage container. Contact dose rates on jumper containers was typically 2 R/hr with hot spots as high as 78 R/hr which was associated with a low spot in an HLW transfer jumper. Seven 1.8 m (6 ft) x 1.8 m (6 ft) x 3.6 m (12 ft) boxes were required to remove all jumpers.

After removal of jumpers and some miscellaneous equipment, preparations for vessel removal began. All vessel exterior surfaces were cleaned using 1.14 KPa 150 psi live steam from a wand in the power manipulator grip. Steam was used to circumvent a criticality concern from addition of rinse water to uncharacterized floor debris. After steam cleaning, the vessels received a clear fixative coating, the vessel internals were inspected using a crane suspended video camera, vessel heel dewatering was performed if needed with an air operated jet, and all cooling water nozzles were sealed with rubber plugs. Vessel inspections showed most vessels to be relatively clean inside. The exceptions were the recycle evaporator and the low level waste accountability tank which both had a layer of sludge about 0.3 m (1 ft) thick. Dissolvers were free of any accumulation of cladding.

Vessels were lifted using the original remote handling strong-backs and were tipped over onto the transfer car. Figure 3.0 shows a dissolver being lowered onto the transfer cart. The vessels were

FIGURE 3.0

Lowering Dissolver Onto the Transfer Cart.

moved to the EDR and packaged in a manner similar to jumper containers. Ten vessels were loaded into nine boxes. Due to difficulties encountered with differences between as-built drawings and interpretations of vessel vendor drawings, rework of all the specially built vessel containers was required. Contact dose rates on the packaged vessels ranged between 100 mR/hr and 110 R/hr.

After transfer of all jumper and vessel boxes to the storage area, temporary shielding was placed over the boxes with highest dose rate to reduce external dose rates. General area dose rates below 10 mR/hr were achieved outside the tent structure.

After the vessels were removed, three additional 1.8 m x 1.8 m x 3.6 m boxes were loaded with cut-up racks, non-jumpered piping, and miscellaneous equipment which was found to have fallen under the vessels. Size reduction was performed with an ordinary industrial abrasive chop saw modified to fit the power manipulator. Blade changes were performed manually in the crane maintenance area. Several saws were set up to allow cutting to continue after blade failure. Chop saw power cables were routed through the crane room into the cell and required constant attention to prevent tangling on in-cell equipment. A cable handling system would have increased productivity significantly.

5.0 SAMPLING AND VACUUMING OF FLOOR DEBRIS

Once the vessels and excess piping were removed, an industrial robot mounted on a mobile platform was moved into the cell to support sampling and floor vacuuming work. Floor debris was sampled using a vacuum cleaner motor in a specially fabricated housing. The vacuum cleaner was handled by the robot and was used to pick up a sample cartridge from an ordered rack. The cartridges were made from Lucite 25 mm air sampling filter housings. The pressure differential through the filters was sufficient to hold the cartridge onto the vacuum cleaner housing without any further mechanical device. The robot moved the sample cartridge to the floor and obtained a debris sample by a preprogrammed routine. It was quickly found that the variations in the debris and floor levels interfered with totally automated sampling, so the program was modified to include a pause for the operator to manually obtain a sample. Analytical results indicated typical Uranium concentrations were 0.1% by weight. One sample obtained near the dissolvers was as high as 8.0% Uranium. Isotope ratios indicated that the debris were primarily spent fuel in nature.

To vacuum the debris remotely, a 22.4 KW (30 HP), positive displacement vacuum system was used in conjunction with specially designed, remotely handled vacuum containers. The containers were

559 mm (22 in) OD and 1067 mm (42 in) high with 6.35 mm (1/4 in) thick carbon steel walls. A Goretex cartridge air filter with 0.74 M^2 (8 sq ft) of filter area was mounted inside each container.

An air pulse system was used to back pulse the filter to maintain low pressure differentials under high dust loading conditions. Neither the robot nor the power manipulator was capable of handling the pick-up nozzle in the tight confinement near vessel mounting pads. In addition, it was found that the clear fixative which was used on the vessels had formed an extremely tough crust layer with the floor debris. After a week of minimal progress, it was decided that vacuuming would be delayed until all vessel mounting pads were removed and the fixative crust layer could be broken up.

Two weeks of cutting operations were required to remove the 30 vessel mount pads by cutting through the 38 mm (1.5 in) dia. stainless steel mounting studs with the abrasive saw. After packaging the floor mounts in a 1.8 m x 1.8 m x 3.6 m container and treating the fixative with a 1.0 molar NaOH solution, vacuuming was resumed. A significant increase in productivity was noted and the operation was completed successfully, however, much development work is needed before remote vacuuming is optimized.

The debris collected weighed 180 Kg (398 lb). The original estimate for the quantity of floor debris was 1800 Kg (4000 lb). The difficulty in estimating the mass of debris lies in the loose, fluffy nature of the atmospherically deposited dust. An estimated 5000 Ci of total activity was collected in the three containers used. Dose rate measurements on the containers indicated 15 R/hr at 1.5 m (5 ft). The debris was transferred to the General Purpose Cell (GPC) which is located directly below the CPC where it awaits future processing.

6.0 INITIAL DECONTAMINATION

After floor vacuuming was completed, all surfaces in the cell were decontaminated using foamed alkaline detergents. The foam was applied with a wand on the power manipulator or from nozzles attached to the 14.5 Mg trolley to clean the ceiling. The foam was allowed to work for 15 to 30 minutes and was then rinsed with 4900 K Pa (700 psi) water at 138° C. Several passes were made using the alkaline detergents before the chemicals were switched to 0.1 M HNO_3. During foaming of the cell, immediate visible results were obvious. In cell light levels increased and white streaks were seen as the rinse solutions ran down the walls. A single decon pass of the cell took two shifts and generated 3785 l of spent decon solution. Decontamination factors were measured using a shielded probe. Decon results are shown in Table I.

TABLE I

RESULTS OF FOAM DECONTAMINATION IN THE CPC

LOC.	INITIAL	AFTER FOAM	DF	AFTER ACID FOAM	DF	DF (TOTAL)
1	585	350	1.67	152	2.30	3.85
2	667	355	1.88	195	1.82	3.42
3	643	307	2.09	210	1.46	3.06
4	641	512	1.25	380	1.35	1.69
5	690	437	1.58	569	.77	1.21
6	855	531	1.61	629	.84	2.60
7	977	668	1.46	680	.98	1.44
8	1260	708	1.77	631	1.12	2.00
9	822	564	1.46	438	1.29	1.88
Ave.	793	492	1.62	432	1.33	2.35

All measurements are in mR/hr, gamma only, as measured by a high intensity GM tube with an Eberline ESP-1 instrument. The probe was shielded from the CPC floor by a 100 mm x 100 mm (4 in x 4 in) lead cube. Locations were randomly selected, but repeatable positions throughout the cell. Some measurements may have been affected by the location of some high dose rate equipment (vacuum cleaner hoses) at the time of the survey.

7.0 DISSOLVER CORE REMOVAL

The fuel dissolver vessels in the CPC were designed as annular ring tanks with six fuel basket ports arranged on a 2.1 m (7 ft) dia. To provide neutron interaction control, a borated concrete core was built as a cast-in-place structure under each dissolver. There were two full size cores weighing 20 Mg each and one 4.5 Mg short base which was provided for a future dissolver. To provide sufficient storage space in the CPC, removal of the cores was necessitated. Removal of these cores presented two major engineering challenges. First, unlike the remainder of the equipment in CPC, the cast-in-place cores were not designed for remote removal. In addition, the 20 Mg weight of the cores exceeded the CPC cranes lifting capacity. Since the West Valley Demonstration Project has significant experience in high pressure water abrasive jet cutting operations,

this cutting method was evaluated and finally selected for removal of the cores.

A tracking device was designed to allow tracking of the 240 MPa (35,000 psi) abrasive/water jet nozzle around the base of the cores. The tracking device was remotely positioned over the core and each core was cut off at its base. Each cut required 8 hrs. of cutting time, used 10,220 l of water, and consumed 450 Kg of garnet abrasive grit.

A hydraulic cylinder system was built to fracture any remaining concrete in the central portion of the core. The two cylinder, 90 Mg (100 ton) capacity jack was suspended on a crane hook and positioned between the CPC wall and each core. The high pressure water supply from the abrasive jet cutting system was used to activate the cylinders. The cores were easily cracked and tipped up slightly before releasing the water pressure.

Each core was then decontaminated by removing the painted concrete surface layer using a manipulator held high pressure water rotary lance. After decontamination, a tipping fixture was positioned over each core. Clamp bolts were tightened using a canyon impact wrench normally used for Purex connector removal. Once the fixture was tightened, the 14.5 Mg crane was used to tip the cores onto a specially designed high capacity transfer car. Tipping the cores was required due to the 14.5 Mg weight limitation on the crane in the CPC. By tipping the cores rather than lifting them directly, the load to the cranes was limited to half the weight of the core.

The design of the tipping fixture was intended to take advantage of some guide plates which were shown on the construction drawings of the cores. When the fixture was placed on the first core, it was noted that the guide plates had been omitted in the field. A decision was made to try the lift depending on a friction clamp only. During the initial lift, the fixture shifted up approximately 100 mm before binding caused the fixture to grip tightly. The lift was continued; however, the downward shift in the center of gravity in the fixture prevented the completion of the tipping since the center of gravity was behind the plane of the hoist at the top dead center of the lift. Repeated attempts were made to complete the tipping, but no combination of trolley and hoist movement was successful. A 1.8 Mg hook was attached to the rearmost part of the tipping fixture in an effort to boost the core over center. The location of the hook caused the clamps of the tipping fixture to loosen allowing the core to slip half out of the fixture and come to a rest on the floor at a 30° angle. The core remained half way in the tipping fixture with the 14.5 Mg crane preventing the fixture from shifting.

Recovery efforts were concentrated on core drilling a hole into the wall of the cell at the center line of the core so that a hydraulic jack could be used to push the core over into the fixture and onto the transfer car. A two week schedule delay resulted from the recovery operations. After completing the removal of the first core, the clamping band was replaced on the tipping fixture and the second core was tipped without incident. The third smaller core was picked up directly using a fixture designed specifically for it.

Each core was moved into the EDR and loaded into a heavy duty container. Contact dose rates were 10 to 50 mR/hr. The container was sealed, moved out of the EDR, loaded onto a trailer, and moved to a hardstand area where it was stored, surrounded by other waste containers to reduce the exposed radiation levels.

The garnet abrasive on the floor of the CPC was wet vacuumed into canisters using a nozzle with a spray tip, initially, and later with a shrouded high pressure water decon tool suspended from a 1.8 Mg crane hook. While vacuuming abrasive, a twisting motion on the power manipulator evidently caused one of its telescoping tubes to jump its guide track. This condition was impossible to detect from outside the cell. When the manipulator was raised, several guide blocks inside the tubes jammed against each other, causing the telescoping tube hoist cable to fail, allowing the manipulator to fall to the floor. The manipulator was recovered with the 1.8 Mg hoists and moved on the transfer car to the EDR for repairs. A one-month delay to the decontamination and decommissioning schedule resulted from the recovery work.

8.0 FINAL DECONTAMINATION

After removing the abrasive from the CPC, a 1.8 m x 1.8 m x 3.6 m container was filled with miscellaneous debris which had accumulated during the previous operations. A steady buildup of small parts was observed throughout the decontamination and decommissioning work. This was primarily due to the large amount of time required to pick up the small parts. To allow this slow work to go on independent of the loading of large containers, two baskets made from expanded carbon steel plate were sent into the cell. Small parts were loaded into the baskets whenever a manipulator operator was available and no other work was planned in the cell. Once the baskets were filled, they were decontaminated with foam and rinse water to decrease the possibility of a hot spot on the container they were loaded into. This concept is currently being evaluated as a primary handling method for the equipment that will be removed from the remainder of the smaller head-end fuel handling cells.

A dose rate survey after removal of the last container of debris showed that the general area gamma exposure had decreased to 300 to 900 mR/hr. A decontamination test was performed using high pressure water with a manipulator held rotary lance. A water flow of 11.4 l/min (3 GPM) and 240 MPa (35,000 psi) was used to remove the paint and an estimated 3 mm inch of fine cement. Dose rate measurements were taken at three test locations using a shielded high intensity GM tube both before and after the decon passes. The decon factor (DF) was between 1.1 and 1.3 for the tests performed. The results of the decon test were much lower than desired or what was required to reach the original target goal of 10 mR/hr general area dose rates throughout the CPC. A further investigation was performed by drill sampling the wall to determine the depth of penetration of contaminants. Holes were drilled into the wall allowing the drilling dust to slide down a funnel into a vertical vinyl tube. The dust was thus stratified by depth of origin. The analytical lab analysis showed that the contamination was largely confined to the first 13 mm (1/2 in) layer. Comparison of the specific activity of the first 13 mm layer with the specific activity of some paint chip samples from the CPC showed that most of the contamination found in the 13 mm layer could be attributable to the paint contamination alone. The analytical results of the sampling are presented in Table II.

TABLE II

DEPTH OF CONTAMINATION PENETRATION IN CPC WALL SAMPLES

SAMPLE	0-10 mm	10 mm-20 mm	20 mm-30 mm	30 mm-40 mm	40 mm-50 mm
Sample 1					
Cs-137	3.64E-2	2.19E-4	<3.50E-5	<4.46E-5	1.18E-5
AM-241	5.25E-4	<4.54E-6	<3.97E-6	<4.20E-6	<1.06E-5
Sample 2					
Cs-137	1.37E-2	1.61E-3	4.71E-4	<3.18E-5	1.02E-4
Am-241	1.78E-4	<5.43E-6	<3.36E-6	<3.13E-6	<3.05E-4
Sample 3					
Cs-137	1.47E-1	1.67E-3	1.34E-3	1.58E-3	3.19E-3
Am-241	4.43E-4	<4.81E-6	<7.96E-6	<6.27E-6	2.91E-5

All numbers are in microcuries per gram.

Since these results did not backup the results of the in-cell decon test, it is presumed that the shielding configuration of the shielded GM tube was somehow altered generating misleading results.

Final wall and ceiling decontamination was then performed using a rotary lance 240 MPa (35,000 psi) spray system. Special fixtures were designed and built to allow the 14.5 Mg trolley to move the spray system to cover the ceiling and upper walls. Steady motion must be provided at all times to prevent excessive erosion of the wall surfaces. The lower wall surfaces were decontaminated using a crane hook handled, shrouded sprayer which allowed vacuum containment of the water and all removed debris. Hot spots and difficult to reach spots were decontaminated using a manipulator held rotary spray lance. All the rotary spray lances used hydraulically driven high pressure rotary unions to supply water to a manifold of sapphire jewel nozzles. Water usage ranges from 5.7 l/min (1.5 gpm) to 15 l/min (4 gpm) depending upon which rotary lance is in use. Figure 4.0 shows the CPC near completion of decontamination and decommissioning. The manipulator is using the rotary spray lance to move debris across the floor.

FIGURE 4.0

CPC Near Completion of Decontamination and Decommissioning

The final decontamination effort used 27,630 l of water and generated 2120 l (75 Ft^3) of Class C waste which may be TRU waste in some of the containers. The waste is presently stored for future stabilization processing. Following the final decon work, a radiation survey was performed, the results are summarized in Table III.

9.0 WORK GROUP ORGANIZATION AND SCHEDULE

The work force dedicated to the decontamination and decommissioning of the CPC was organized into an engineering group consisting of two to four engineers depending upon the work load and an operations group typically consisting of two supervisors and ten decontamination technicians. Engineering work began approximately six months before the planned start of decommissioning work. The organization of the operations group changed depending upon the type of work being done

TABLE III

RESULTS OF HIGH PRESSURE WATER
DECONTAMINATION IN THE CPC

LOCATION	BEFORE DECON	AFTER DECON	DF
1	152	67	2.27
2	195	94	2.07
3	210	102	2.06
4	380	110	3.45
5	569	140	4.06
6	629	186	3.38
7	680	257	2.65
8	631	410	1.54
9	438	261	1.68
AVERAGE	432	181	2.57

All measurements are in mR/hr, gamma only, as measured by a high intensity GM tube with an Eberline ESP-1 instrument. The probe was shielded from the CPC floor by a 100 mm x 100 mm lead cube. Locations were randomly selected, but repeatable positions throughout the cell.

in the CPC. During remote operations, with no ongoing waste removal work, two shifts of five people were used. A three shift schedule was tried briefly, but it was found that work actually slowed down since only day shift could make entries into the CCR and the EDR to support the remote operations. When waste containers were scheduled to be moved, the group was organized into one shift to support the man-power needs of container handling.

The master schedule for the project allowed 24 months for the completion of decontamination and decommissioning work in the CPC. The start date was to be the 1st of January, 1985. Due to delays in the deliveries of waste containers and the sprung structure storage area, the actual start date was delayed until the end of March, 1985. During the decontamination and decommissioning work in the CPC, there were events which resulted in lost time.

For a project with a scheduled duration of 110 weeks, a cumulative lost time of 35.5 weeks is obviously significiant. Through work-arounds and production rate improvements, the anticipated completion was only delayed by 10 weeks.

Table IV is a summary of schedule delay problems encountered during the CPC decontamination and decommissioning work:

TABLE IV

SUMMARY OF EVENTS WHICH RESULTED IN LOST TIME

	ITEM	LOST TIME (WEEKS)
	Delay in delivery of jumper containers	16 weeks
	Repair of vessel containers to provide clearances	4 weeks
*	Failure, recovery, and repair of the transfer car	2 weeks
*	Failure, recovery, and repair of the power manipulator	7 weeks
*	Failure, recovery, and repair of the 16-ton crane	1.5 weeks
	Inability to vacuum floor debris around obstacles	3 weeks
	Recovery operations for the 22.5-ton core	2 weeks
	Replacement of hinged rails between CPC and EDR	1 week
	TOTAL	**35.5 weeks**

* Involved more than one incident.

10.0 LESSONS LEARNED

- Tool power cabling and hoses were one of the biggest hindrances to work. A cable and hose handling system could have saved many manned entries into the CCR and the EDR to repair broken cables and hoses. A cable carrier system is under evaluation for use in other remote cell decontamination and decommissioning work at West Valley.

- A decon pass prior to equipment removal would have reduced exposure during waste handling and decreased the contamination control concerns significantly. Sampling of high pH decon solutions which had come in contact with fissile materials indicates there is no criticality concern as long as slab geometries are maintained until the solids have settled.

- An on bridge video system would be extremely helpful during remote decontamination and decommissioning work.

- A mock-up electro-mechanical manipulator could help to minimize rework of equipment in contaminated areas, thereby reducing exposure and saving schedule time.

- Total replacement, rather than refurbishment of the cranes, manipulator, transfer car, and hinged rails could have potentially saved as much as 11.5 weeks of schedule delays.

- An abrasive saw has proven to be a useful manipulator held tool.

- The addition of one set of master-slave manipulators could have saved a significant amount of time and exposure to perform tasks that the electro-mechanical manipulator could not perform such as replacement of saw blades.

- A transfer port for small equipment would also reduce manned entry requirements.

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The entire Maintenance Department
The entire Plant Systems Operations Department

USE OF FOAM CHEMICALS FOR DECONTAMINATION

by:

R. A. Meigs

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ABSTRACT

A means of rapid, gross smearable contamination removal is being used at the West Valley Demonstration Project. Decontamination chemicals are made into a foam using foam stabilizing surfactants and blown with compressed air onto surfaces to be decontaminated. The foam is left to work for 15 to 30 min. and is rinsed with 3500 to 7000 KPa (500 to 1000 psi) water sprays. Spent decontamination solution generated ranges from 0.03 l/M^2 to 0.14 l/M^2 (0.08 to 0.40 gal/sq. ft.). Decontamination factors for smearable contamination up to 10 are routinely achieved.

1.0 INTRODUCTION

The West Valley Demonstration Project is a Congressionally mandated activity conducted through the U. S. Department of Energy, Idaho Operations Office. The operator of the Project is West Valley Nuclear Services Co., Inc., a subsidiary of the Westinghouse Electric Corporation.

The objective of the project is the solidification of High Level Waste (HLW) generated by commercial nuclear fuel reprocessing which was conducted at the site from 1966 to 1972. Site operations became a U. S. Department of Energy responsibility in 1982.

To support the solidification of HLW, a large portion of the former reprocessing facility will be decontaminated for reuse. Solidification support systems will be installed in decontaminated, shielded cells to minimize construction of new facilities. This work involves the decommissioning of all old processing equipment and decontamination of the remaining cell surfaces. It is often desirable to decrease contamination levels, reduce smearable contamination and lower airborne contamination levels during the course of the decommissioning work to reduce personnel exposure levels and decrease the likelihood of contamination spread.

Since the West Valley facility has had no operable evaporators since the mid 1970's, contaminated water generated must be kept at an absolute minimum. This has forced the Decommissioning Department at the West Valley Project to develop capabilities which effect a significant decontamination factor with a minimum of water use and personnel radiation exposure to perform the decon work.

2.0 DESCRIPTION OF THE FOAMING OPERATION

The foams used at West Valley are generated by blowing compressed air into a mixture of the foam stabilizing chemical and the decontamination chemical which is being used. Two different chemicals have been used to date. The first is an alkaline detergent blend called AE-3003, which is commercially available from the DuBois Chemical Company. The second chemical used is 0.1 M HNO_3 (as mixed and applied) which is mixed on-site from the bulk HNO_3 tank. The alkaline detergent blend is comprised of 1w% NaOH, 1/2w% Isopropyl Alcohol, and a proprietary mixture of surfactants. The foam stabilizing chemical is Foam-Add, which is also available through DuBois Chemical Company, and consists of a proprietary blend of anionic surfactants that give rise to extremely stable foams. Batches of foaming chemicals are mixed in the following ratios:

FOR ALKALINE FOAM	**FOR ACID FOAM**
1 part Foam-Add	1 part Foam-Add
2 parts AE-3003	10 parts water
10 parts water	HNO_3 to give 0.1 M
total	

The compressed air and foam chemicals are mixed in a tee in a piping manifold and are pushed by the compressed air through the hose. Typical foam generating systems are shown in Figures 1 and 2. The pneumatic pump pressure system uses a piston pump to draw chemicals out of a drum allowing 190 litre (50 gallon) batches to be used. The pressure pot system uses a 19 litre (5 gallon) stainless steel pressure vessel under direct pneumatic pressure to supply the chemicals. The piston pump system is used for larger jobs where chemical mixing time must be reduced. Foam can easily be applied to surfaces 6 m (20 ft) away making it easy for personnel to work from low exposure areas if required. Foam quality can easily be judged by how slowly the foam "sags" or flows down a vertical surface. When the foam consistency is proper, the foam moves very slowly down a surface. Foam will cling to ceilings and undersides of equipment making it one of the few viable decon methods that works on all surfaces.

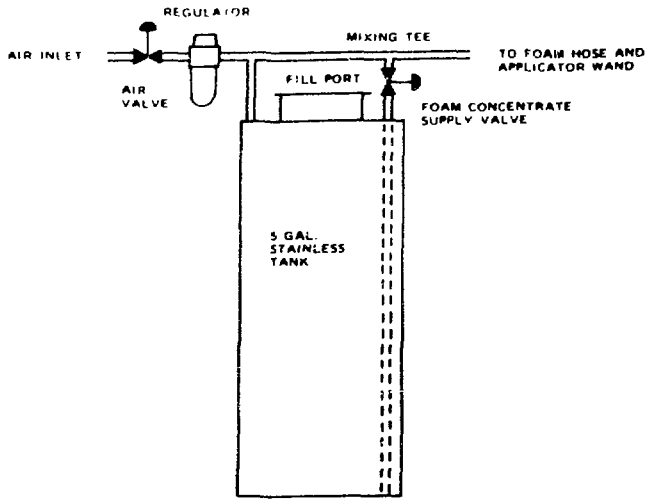

FIGURE 1.0
Pressure Pot Foam Applicator

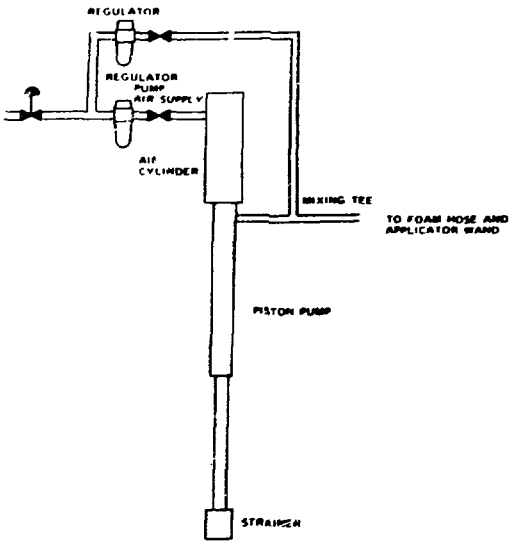

FIGURE 2.0
Piston Pump Foam Applicator

3.0 *TYPICAL FOAM DECON SEQUENCE*

The foaming and rinsing systems are located as close as possible outside of the area to be decontaminated, however, as much as 30 M (100 ft) of hose has been used with no difficulty. Hoses are routed into the work area either through the doorways and airlocks or through a utility penetration, if one is available. Filtered water and compressed air are used to prevent damage to pumps. An airline lubricator must be used when the pneumatic pump pressure system is used.

Chemicals are mixed in a plastic drum for the pneumatic pump pressure system and in a steel pail for the pressure pot systems. The foam is applied as quickly as possible, using the entire mixed chemical batch. The operator will typically wait in an airlock area until the foam has worked for 15 to 30 minutes. The foam is then rinsed using a 3500 KPa to 7000 KPa (500 to 1000 psi) sprayer using a flow rate of 7.6 l/min. to 11.4 l/min. (2 to 3 gpm). If water splash or leakage out of the contained area is of concern, plant utility water is used instead of the pressure washer to rinse down the foam, similar to a garden hose rinse.

Alkaline foam is used initially to remove the heavy soil and grease layers that are encountered in almost every area of a 20 year old

plant. If surveys indicate additional decon work is needed more passes using alkaline foam are done. If the equipment in the area is either non-functioning or is corrosion resistant then an HNO_3 foam pass is performed if additional decon work is needed after the alkaline foam no longer appears to help. Extremely heavy accumulations of grease such as bearing lubricant must be removed by other means such as Freon-113. Light lubrication films and grease drips are easily dispersed by the alkaline foams.

4.0 SPECIFIC DECONTAMINATION OPERATIONS AND RESULTS

4.1 The Extraction Cell Three (XC-3)

The XC-3 was the first cell at West Valley to be decontaminated using foam. The cell is a 4.9 m x 4.6 m x 17.4 m (16 ft x 15 ft x 57 ft) tall cell which contained chemical extraction columns. Wall surfaces are phenolic painted concrete with a stainless steel floor liner that reaches 460 mm (18 in) up the walls. After all piping and equipment were removed, the final decontamination was performed using foam. Work was performed manually from a cable suspended work platform suspended from above. All work was staged and supported from a containment located above the XC-3. Results from all decontamination work described here are shown in Table 1.

4.2 The Product Purification Cell (PPC)

The PPC was a cell similar to the XC-3 which contained the final product purification steps including Plutonium ion exchange columns and Uranium silica absorber beds. The decontamination operations were performed in the same manner as the work in the XC-3.

4.3 The Uranium Product Cell (UPC)

The UPC is a cell located adjacent to the bottom of the PPC. The cell is 14 m long x 7.9 m wide x 4.0 m high (46 ft x 26 ft. x 13 ft) and contains two 25,000 litre (6600 gal) Uranium product storage vessels. Wall surfaces are phenolic painted concrete with a stainless steel floor liner that reaches 460 mm (18 in) up the walls. The cell was decontaminated by entry through the man-way door working from floor level. Decontamination operations for the UPC were staged from the containment airlock tent located at the man-way doors.

4.4 The Chemical Process Cell (CPC)

The CPC is a remotely operated canyon type cell which was used for chopped fuel dissolution and concentration of HLW prior to storage. The cell is 28 m long x 6.7 m wide x 13 m high (93 ft. x 22 ft x 43 ft) and is serviced by two overhead bridge cranes which include an electro-mechanical manipulator. Wall surfaces are phenolic painted concrete with a stainless steel floor liner that reaches 460 mm (18 in) up the walls. After equipment removal and vacuuming the floor, the cell was decontaminated using a foamer/sprayer assembly held by the manipulator. Hoses were fed through a wall penetration at midcell and all support operations were performed in the operating aisle adjacent to the CPC. A remote switch system controlled solenoid valves which turned foam and rinse water on and off from a cell viewing window. This arrangement allowed nearly continuous cleaning operations using the pheumatic piston foam generator and 55 gallon batches for the first time.

4.5 The CPC/Equipment Decontamination Room (EDR) Shield Door Recess

The CPC/EDR shield door recess is a confined area which is only accessed from a roof hatch on the EDR roof. The recess area is 9.8 m x 2.4 m x 7.9 m high (32 ft x 8 ft x 26 ft) and contains a lateral traveling shield door which is 1.2 m x 4.9 m x 4.9 m high (4 ft x 16 ft x 16 ft). Wall surfaces are phenolic painted concrete with a bare concrete floor. The door is phenolic painted steel filled with concrete. A containment tent was set up above the roof hatch to allow removal of the 900 Kg (1-ton) hatch plug and to support decontamination operations in the door recess area. Initial decon work was performed from the tent using telescoping tube sprayer and foamer wands. This allowed the floor area to be cleaned without any workers being lowered down to the floor level. This saved a considerable amount of set-up work for entry into such an inaccessible location. Final cleanup of the area was performed with hand held tools, with workers standing on top of the door. The foaming equipment was staged on a roof one level above the EDR roof so that the same system could be used with a different set of hoses to decontaminate the CPC/Chemical Crane Room (CCR) door penthouse.

4.6 The CPC/CCR Shielding Door Penthouse

The CPC/CCR shielding door penthouse is an enclosure located above the CPC and the CCR which contains the shielding door hoists and the shield door when it is in its up position. The enclosure is 2.4 m x 4.1 m x 7.6 m high (8 ft x 30 ft x 25 ft)

and is comprised of a steel structure surrounded by painted concrete block. Decon operations were staged through two containment tents which accessed both upper and lower decks. This work was performed at the same time as work in the CPC/EDR door recess.

5.0 Discussion of Decontamination Results

Specific results obtained for a particular decontamination operation depend on a number of variables. The primary consideration is the type of contamination to be removed. If an area has gone without decontamination for an extended period, a build-up of greasy soils will make decontamination more difficult and will make the advantages of foam decontamination more obvious. For example, in the CPC no decontamination work had been performed on the walls of the cell in its entire lifetime. A spray-down with live 1140 KPa (150 psi) steam had little visible effect.

Rinsing down with utility water also showed no noticeable change in the dirt on the walls. During the foaming operations in the CPC, immediate results were seen as clean, white streaks were left behind on the walls as foam streamed down the walls. The light level in the cell increased dramatically as cleaning progressed.

A further consideration is whether scrubbing can be done on the surfaces. Although good results can be obtained without scrubbing, better results are seen when scrubbing is performed with deck brushes or brooms. It has also been observed that during warmer weather or if the chemicals and rinse water are heated better cleaning results. There is no radiological data to support better DFs under these conditions, however.

The final factor affecting foam decontamination results is the condition of the surface being decontaminated. Stainless steel surfaces give better DFs than painted carbon steel or painted concrete as might be expected. Surfaces near warm equipment such as motors, gearboxes, and bearings are usually more contaminated than their surroundings and do not respond as well to foam decontamination. This may be related in part to the lubricants used on these pieces of equipment. Additional studies are being performed at West Valley to determine the affects of paint quality on decontamination factors. It is expected that paint which is improperly applied and cured will make decontamination of the painted surfaces nearly impossible.

TABLE 1

RESULTS OF SPECIFIC FOAM DECONTAMINATION OPERATIONS

AT THE WEST VALLEY DEMONSTRATION PROJECT

JOB	AREA DECONNED (M^2)	WATER (Litre)	WATER (Litres/M^2)	TIME REQ. (Days)	SURVEY TYPE	INITIAL CONTAMINATION	FINAL CONTAMINATION	DF
XC3	430	1500	0.04	4	C	1100 Kdpm/100cm^2-alpha	85 Kdpm/100cm^2-alpha	13
PPC	370	1495	0.03	4	C	56 Kdpm/100cm^2-alpha 332 Kdpm/100cm^2-alpha	17 Kdpm/100cm^2-alpha 20 Kdpm/100cm^2-alpha	3.5 (walls) 16.6 (floor)
UPC	180	950	0.04	2	C	20 Kdpm/100cm^2-alpha	5 Kdpm/100cm^2-alpha	4.0
CPC	1450	31,000	0.18	10	R	714 mR/hr-gamma	388 mR/hr-gamma	1.88[2]
EDR Door	320	1135	0.03	3	C	10 Kdpm/100cm^2-beta	1 Kdpm/100cm^2-beta	10.0
CCR Door	390	1500	0.03	3	C	30 Kdpm/100cm^2-beta	10 Kdpm/100cm^2-beta	3.0
CCR/ Cranes	370	1210	0.027	3	R	5000 mR/hr-beta	1000 mR/hr-beta	5.0[2]

(1) Water volume estimated from spraying time with 3 gpm.

(2) Due to the high levels of contamination, no smear surveys were taken. Estimated DF for smearable contamination is 1000. This is due to the extremely soiled surfaces before decon. Smearable contamination surveys are indicated by a "C" under survey type. Radiation surveys are indicated by a "R" under survey type.

6.0 ADVANTAGES OF FOAM DECONTAMINATION

The advantages of foam decontamination are best summarized by comparison with more typical methods such as dilute detergent scrubbing and high pressure washing as in the following table:

TABLE 2

COMPARISON OF FOAM DECONTAMINATION WITH OTHER METHODS

PROBLEM	FOAM DECON	DETERGENTS	HIGH P. WATER
Time required	low unless scrubbing	longer due to scrubbing	longest, precision positioning needed
Worker exposure	low unless scrubbing	higher due to scrubbing	highest, precision positioning needed
Remoteability	extremely easy	scrubbing difficult	reaction forces are high
Water volumes	0.03 - 0.14 $1/M^2$	0.17 - 0.70 $1/M^2$	0.17 - 0.70 $1/M^2$
Chemical use	low	5 to 10 times higher than foam	none
Redeposition of soils	low	low	high without chemicals
Set-up time	1/2 day	1/2 day	2 days

7.0 SUMMARY

Foam decontamination is an excellent means of reducing high levels of smearable contamination. Water and decontamination chemical usage are minimized. Equipment and operating expenses are very low and radiation exposures to personnel can be reduced to very low levels if not eliminated completely by performing work remotely. Foam decontamination is highly recommended where a quick reduction in smearable and airborne contamination is needed prior to removal of equipment or maintenance operations. It is also recommended for use in routine hot cell operations on a periodic basis to reduce future decontamination difficulties from aging of the contamination layers.

REMOTE CUTTING AND REMOVAL OF CHEMICAL PROCESSING CELL'S CONCRETE PEDESTAL BASES

by:
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and
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ABSTRACT

As part of the decommissioning of the Chemical Process Cell (CPC) at the West Valley Demonstration Project (WVDP), three 2.24 m (7 ft -4 in) diameter blocks of steel reinforced concrete had to be removed from the floor of the CPC. These blocks, which were cast into the CPC floor and covered with stainless steel as part of the floor liner, had to be cut free from the floor prior to removal in one piece. Tooling was developed to remotely cut the pedestal within 13 mm (1/2 in) of the floor with a 240 M Pa (35,000 psi) abrasive water-jet cutting system. Two of the pedestals weighed 20 Mg (22½ tons) each with only a 14.5 Mg (16-ton) overhead hoist available for handling. Special fixtures were developed which allowed the pedestals to be tipped over onto a transfer car. The transfer car was also a WVNS design, specifically built for pedestal removal. The pedestals were ultimately transferred into an outer room for placement in metal boxes for disposal.

1.0 Introduction

The West Valley Demonstration Project is a Department of Energy (DOE) project to solidify the liquid wastes remaining at the West Valley site from the reprocessing of commercial nuclear fuel assemblies. West Valley Nuclear Service Company, Inc., (WVNS), a subsidiary of Westinghouse Electric Corporation, is the prime contractor for the DOE in the operation of the site.

The waste, which is stored in underground tanks, will be solidified into a borosilicate glass for eventual storage at a federal repository. Until such time that the glass logs will be shipped from the West Valley site, they will be stored on site in one of the shielded cells within the existing facility. This cell, referred to as the Chemical Process Cell (CPC), is the largest cell at the West Valley site. The CPC is equipped with overhead hoists (one 14.5 Mg and two 1.8 Mg) and an electro-mechanical arm (PAR).

In order for the cell to be used for storing the glass logs, the entire cell had to be cleaned out. Since the cell was designed for remote operations, the vessels, piping, and other hardware were removed utilizing the overhead cranes and PAR.

With the vessels removed, the only major items that remained to be removed were three 2.24 m (7 ft -4 in) diameter borated concrete blocks referred to as pedestals (see Figure 1). These pedestals were used as neutron absorbers for the dissolver vessels during dissolution of the spent fuel. They were permanently cast in place with no capability for removal. Their 20 Mg (45,000 lb) weight exceeded the in-cell hoist's 14.5 Mg capacity. These two factors combined to make their remote removal a unique engineering challenge.

Two of the pedestals were 2.65 m (8 ft - 8½ in) tall and weighed 20 Mg. The third pedestal was 460 mm (18 in) tall and weighed 4.5 Mg (10,000 lb). The pedestals were not of a uniform cylinderical shape but had 6 equally spaced vertical Vee notches that were 0.5 m (20 in) wide by 0.25 m (10 in) deep. The Vee notches were lined with 4.8 mm (3/16 in) stainless steel the full height of the pedestals. The lower 460 mm of the pedestals were covered with 10 gauge stainless steel as part of the CPC floor liner.

Figure 1
Concrete Pedestals in CPC

Due to the radiation and contamination levels in the CPC, any method chosen for the pedestal removal would have to be done remotely. After giving consideration to methods that would break the pedestals into pieces, it was decided to remove the pedestals in one piece by cutting them off at the floor with the ultra high pressure 240 M Pa (35,000 psi) abrasive water-jet cutting system (ABJET).

The following removal concept was developed to remove the pedestals from the CPC:

- Cut the pedestals at the base, including the stainless steel skin around the pedestal and the steel rebar holding the pedestal to the floor at a construction joint with the ABJET cutting system;
- crack the pedestal loose from the floor at the construction joint by pushing between the side of the pedestal and the cell wall with a hydraulic jacking system;
- tip the pedestal over onto a specially designed transfer car using the overhead hoist;
- move the pedestal into the adjacent Equipment Decontamination Room (EDR) with the transfer car;
- lift the pedestal from the transfer car with the two overhead hoists available in the EDR and place in a steel box for contamination containment;
- move the boxed pedestal from the EDR and load onto a flatbed trailer for transport to a temporary storage area until it could be disposed of as radioactive materials in a disposal area.

2.0 Equipment Description

The pedestal cutting fixture (see Figure 2) design utilized the ABJET cutting system and positioning tracker. It was rolled steel plate ring with a circumferential guide for the hydraulically operated tracker to ride along. The ring was designed to fit around the pedestal and sit on the larger diameter pedestal base. It had a lifting bail assembly that allowed the fixture to be set in position with the overhead hoist.

Figure 2
Pedestal Cutting Fixture in Place on Full Scale Mock-Up

The tracker was held to the ring with open link chain that went around the ring and passed over a hydraulically driven sprocket on the tracker. As the hydraulic motor to the sprocket was operated, it pulled the tracker around the ring. Variable tracking speeds and direction could be achieved by controlling flow to the hydraulic motor.

A mount for the ABJET nozzle to the tracker was developed such that the nozzle could be remotely installed and removed from the tracker with the PAR. The mount design was such that when the nozzle was removed for carbide tube replacement on the ABJET nozzle, it could be positioned back onto the tracker without having to realign the nozzle with the previously started cut. The mount was also equipped with mechanical adjusters that allowed adjustment of the nozzle up and down and in and out when the cutting fixture was first installed around the pedestal. These adjusters were designed to be operable with the rotating wrist of the PAR.

The pedestal cracking device (see Figure 3) was designed to push on the top edge of the pedestals with two 45 Mg (50-ton) capacity hydraulic cylinders that operated with a maximum hydraulic pressure of 69 M Pa (10,000 psi). The cracker would be positioned between the pedestal and the 1.75 m (5 ft - 9 in) thick CPC steel reinforced concrete wall. Since one of the pedestal's Vee notches was aligned with the wall, a Vee shaped pusher head was designed for the end of the cylinders.

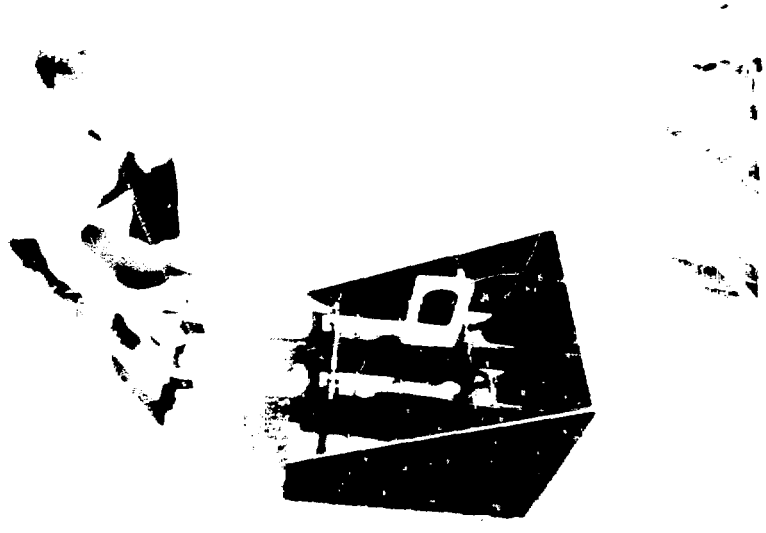

Figure 3
Pedestal Cracking Fixture

To tip the 20 Mg pedestal onto the transfer car, a two-piece frame work was designed (see Figure 4) that would be placed over the pedestal and clamped in place with four 32 mm (1¼ in) diameter bolts tightened remotely with an impact wrench hung from the overhead hoist. The framework provided a lifting bail for the overhead 14.5 Mg hoist to be attached to lift on one side of the pedestal for tipping. The position of the bail was such that it would only require 10 Mg (12½-tons) of lift by the hoist to tip the pedestal on its side. The bail was positioned such that as the pedestal is being tipped and the bail is in direct alignment over the wheels, the pedestal's center of gravity (c.g.) is located just in front of this line such that the c.g. will cause the pedestal to tip over.

Figure 4
Pedestal Tipping Fixture

Before the pedestal was to be tipped over, it had to be moved toward the transfer car approximately 460 mm (18 in). This was required for the pedestal to be centered on the transfer car for even weight distribution and for centering in the tunnel connecting the CPC to the EDR. With the pedestal in the horizontal position, there was less than 300 mm (12 in) of clearance on each side of the pedestal as it passed through the tunnel. To provide this lateral movement, wheels were incorporated in the tipping fixture on the bottom edge of the frame opposite the lifting bail. As the pedestal was tipped, the wheels supported the pedestal and allowed it to be rolled. Due to the irregular slope of the floor, a leveling track plate for the wheels was developed (see Figure 5). This plate was positioned under the wheels and was supported by beams that bridged the sloped portion of the floor. This track plate had side rails for each wheel to accurately guide the pedestal toward the transfer car. Also, wheel stops were built into the plate that positioned the pedestal at the exact location prior to tipping it onto the cart such that the pedestal was centered on the cart.

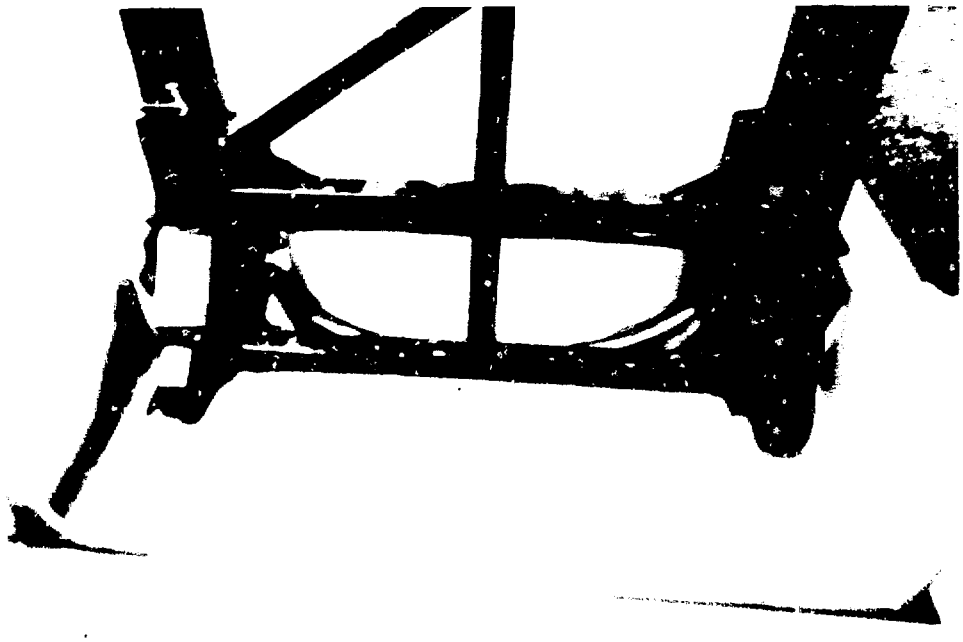

Figure 5
Tipping Fixture Track Plate

A new cart was designed for the pedestal removal and for use in completing the remaining decontamination effort in the CPC. The transfer cart design was patterned after conventional railroad cars with a main low slung bed and two pivoting end trucks. One of the end trucks has four wheel drive, driven by an electric motor.

To remove the 4.5 Mg short pedestal from the CPC, a lifting fixture was designed that was set over the pedestal and clamped in place with two bolts like those on the tipping fixture (see Figure 6). This allowed the pedestal to be lifted with the 14.5 Mg hoist and placed on the transfer cart for removal to the EDR.

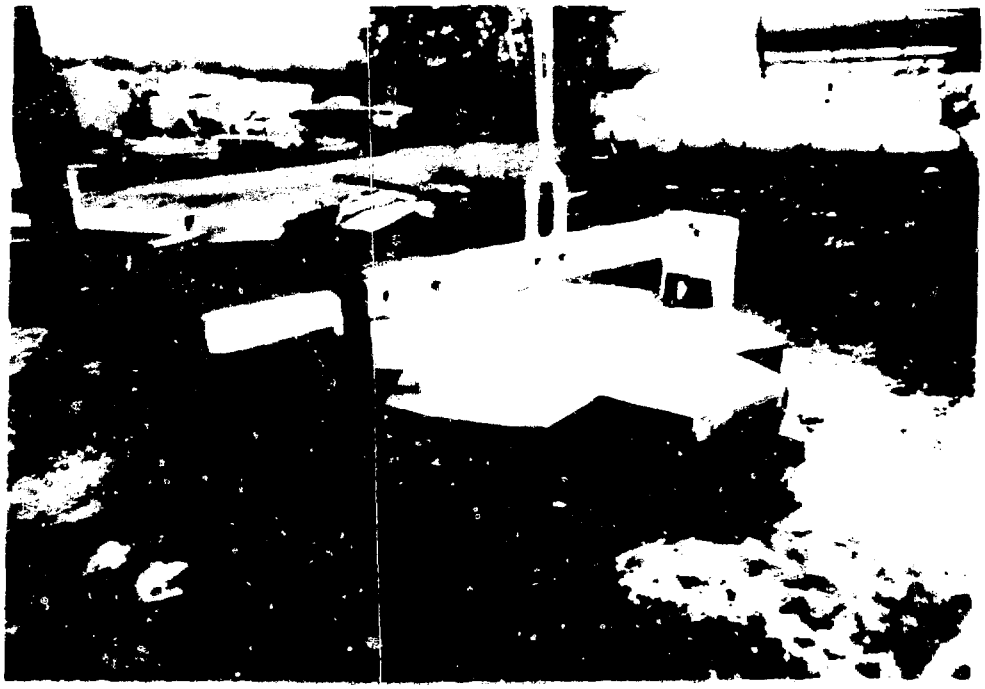

Figure 6
Short Pedestal Lifting Fixture

3.0 Equipment Testing

A full-scale mock-up of the 20 Mg pedestal was designed along with a support frame to simulate the CPC floor, wall, and transfer cart tracks (see Figure 7). The mock-up was used to train the selected team of operators on the use of the equipment and to make a functional checkout of the equipment. The cutting fixture was tested for fit by remotely setting it into place on the full-scale mock-up with a mobile crane and completely cutting the mock-up off with the ABJET nozzle. The hydraulic cracker was tested with the water from the ABJET power unit and found to work well. The transfer car was installed on the support frame tracks and temporarily wired so that it could be operated. The tipping fixture track plate and support beams were positioned around the pedestal in preparation for placing the tipping fixture on the pedestal. The tipping fixture was assembled and positioned around the pedestal. It was tightened to the pedestal with the use of a hand held impact wrench. The fitup of the framework to the pedestal was checked and found to be very good. A 54 Mg (60-ton) mobile hydraulic crane was positioned alongside of the pedestal in preparation for tipping. The pedestal was slowly and smoothly tipped onto the transfer cart (see Figure 8). All parts of the operation went as was planned during the design of the equipment.

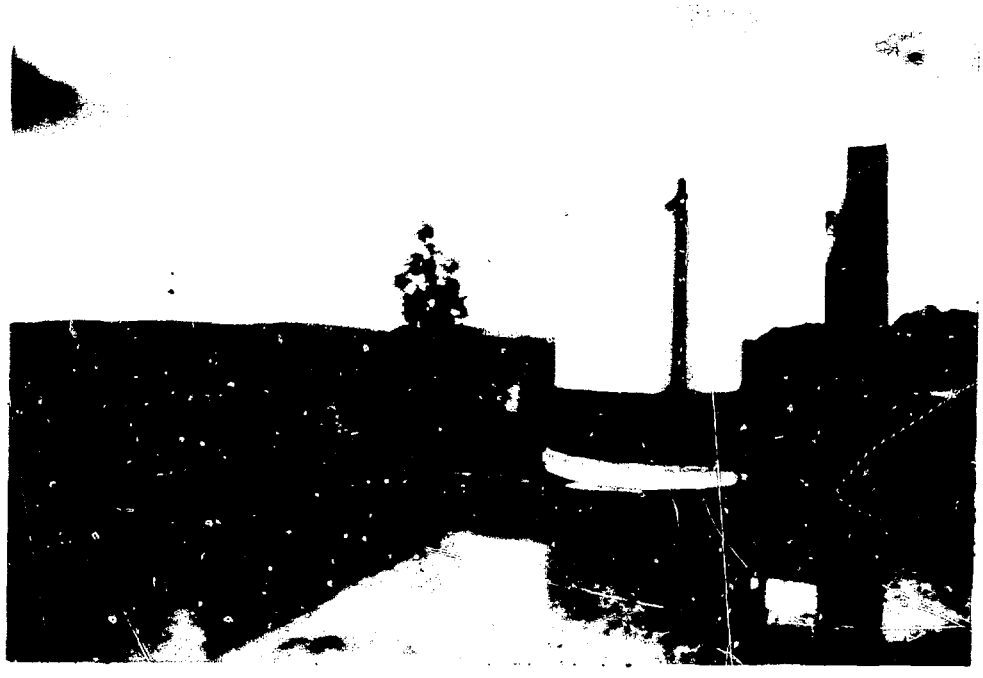

Figure 7
Full Scale Pedestal Mock-up on Support Frame

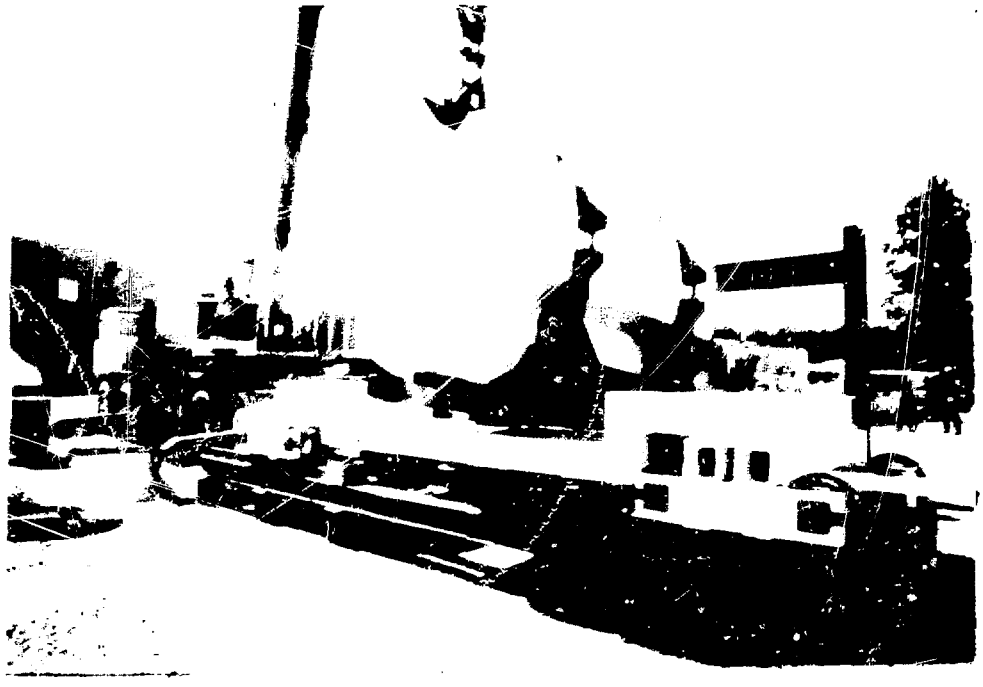

Figure 8
Tipping Pedestal Mock-up onto Transfer Cart

With the pedestal on the cart, the track plate and beams were moved with the crane as would be done in the CPC. The transfer cart was then operated with the 20 Mg load. This operation was also very smooth both starting and stopping. Weight measurements were taken on the pedestal with a dynamometer to measure the actual force required to tip the pedestal over 11,300 Kg (25,000 lbs.), to stand it back up 9525 Kg (21,000 lb.), and its total weight 20,000 Kg (45,000 lbs.).

Mock-up equipment also included a separate 0.3 m (1 ft) section of a pedestal. This short model was used for miscellaneous ABJET testing but proved to be most useful testing the short pedestal lifting frame. The lifting fixture was positioned on the pedestal and tightened in place with an impact wrench just as was done on the tipping fixture. Additional weight was added to the pedestal so that the test would be done on a pedestal heavier than the one in the CPC. The fixture was lifted with the mobile crane and held with no sign of slippage.

4.0 Pedestal Removal in Cell

At this point, the mock-up testing was completed. The equipment was disassembled and moved to the EDR for movement into the CPC. The new transfer cart was installed first so that it could be used for equipment transfer. ABJET water and hydraulic hoses were passed through a core drilled hole in the EDR outer wall for connection to the cutting fixture prior to its transfer into the CPC.

The cutting fixture and hydraulic cracker were taken into the CPC on the transfer cart. The cutting fixture was lifted from the transfer cart with the 1.8 Mg hoist, positioned over the first pedestal to be cut and lowered in place. The ABJET nozzle was then installed on the tracker. The PAR was used to adjust the nozzle 13 mm (½ in) below the top edge of the pedestal base and 25 mm (1 in) away from the side of the base. It took approximately 7 hours to complete the cut around the pedestal to an estimated depth of 380 mm (15 in).

When the cutting was completed on this pedestal, the nozzle was removed and taken to the crane room along with the hydraulic cracker. The water hose from the nozzle was removed and installed on the cracker. The cracker was then taken back into the CPC and positioned between the pedestal and the wall with the 1.8 Mg hoist. The pedestal was successfully cracked with less than 18 Mg of force exerted by the cracker. The cracker was then returned to the CPC crane room.

The cutting and cracking operation was repeated on the two remaining pedestals. About 1360 Kg (3000 lb) (0.6 m^3 (21.5 cubic ft) of grit and 26,500 l (7000 gallons) of water were used for cutting all three pedestals. The water was pumped from the floor sumps to the plant's liquid waste tank during the cutting operation. The location of the sump from the cutting area was far enough that the grit settled out on the floor and did not affect the operation of the sump pump.

A remotely handled rotary decon nozzle which operated with 69 M Pa (10,000 psi) water from the ABJET power unit, was used to clean the pedestals prior to installing the tipping fixture. This step was included to reduce the contamination as well as the radiation levels on the pedestals before they were transferred into EDR. It also cleaned the floor where the support beams for the guide tray were to be installed.

The tipping equipment was brought into the CPC and set in place around the first 20 Mg pedestal using the 1.8 Mg hoist (see Figure 9). Detail inspection with the portable CCTV revealed that wedge plates on the pedestal Vee notches which were shown on the drawings did not exist. The design of the fixture only considered these plates as stops to prevent the pedestal from sliding out of the fixture if slippage were to occur. They were not part of the primary attachment of the fixture to the pedestal. The in-cell impact wrench suspended from a hoist was used to tighten the bolts on the tipping fixture. Three of the four bolts were tightened satisfactorily. The fourth bolt would not tighten; it was later found that a spacer nut on the bolt moved and locked the bolt such that it would not rotate. Since it would have taken several work shifts to remove the fixture and make the necessary repairs and the fixture appeared to be clamped adequately, it was decided to try to tip the pedestal with the set-up as it was. When the 14.5 Mg hoist started to lift the pedestal, the top band slipped several inches and then bound tightly around the top of the pedestal so the decision was made to continue with the tipping operation. The pedestal was tipped onto the wheels and moved to the stops on the track tray. The pedestal continued to be tipped with the hoists until it reached the balance point with the crane hook in line with the wheels. The slippage of the fixture moved the center of gravity of the pedestal relative to the fixture such that it would not tip over by itself. After several other unsuccessful efforts to tip the pedestal over, one of the 1.8 Mg hoists was attached to the bottom of the fixture. It had been observed that during the initial tipping attempts the pedestal would break over center but the stretch on the 14.5 Mg hoist cables would pull it back. It was felt that if the 1.8 Mg hoist could hold the pedestal in the "over-center" position until the slow speed 14.5 Mg hoist could be lowered, the tipping operation could be completed. Unfortunately, as the 1.8 Mg hoist pulled on the fixture, it loosened the hold the fixture had on the pedestal and allowed the pedestal to slip out of the fixture and onto the floor. The bottom end of the pedestal ended up resting on the floor with the top end still in the tipping fixture resting on the cart. Initial inspections did not find any damage to the equipment.

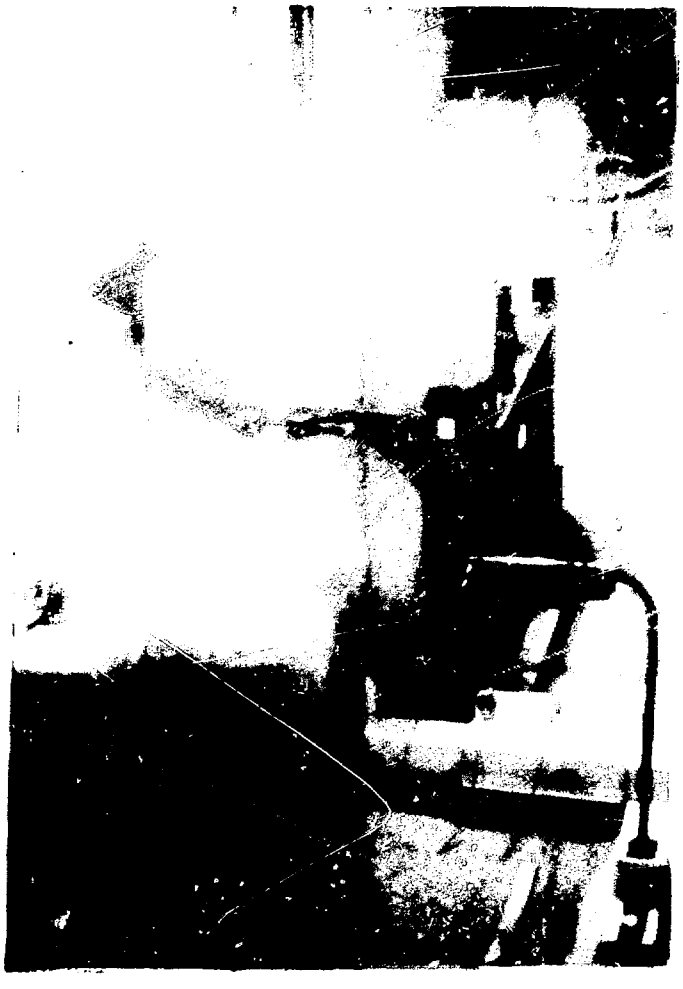

Figure 9
Tipping Fixture on Pedestal in CPC

A recovery plan was developed that would put the pedestal back onto the cart. With the top end of the pedestal still positioned over the cart, it was felt that the bottom of the pedestal could be rigged with a sling and lifted with the 14.5 Mg hoist to a horizontal position. The pedestal could be pushed onto the cart. A sling was modified with a push pole that would allow the PAR to place the sling under the pedestal. A 200 mm (8 in) diameter hole was core drilled through the1.75 m (5 ft - 9 in) thick wall of the CPC in line with the center of the bottom of the pedestal. The location of this hole was in an operating aisle between the CPC and another cell with 1.5 m (5 ft) thick walls. A 6 m (20 ft) length of 152 mm (6 in) diameter pipe was inserted in the hole. With the pedestal lifted to a horizontal position, a hydraulic jack was placed between the pipe and the adjacent cell wall such that the pipe would push on the base of the pedestal (see Figure 10). This method proved successful in completing the pedestal tipping operation and transfer of the pedestal onto the cart for removal from the CPC. This entire recovery effort took two weeks to develop and complete.

Figure 10
Slipped Pedestal Recovered to Transfer Car

The pedestal was moved to the EDR where it was removed from the cart with slings, spreader beams and the two overhead hoists that had enough capacity to lift the pedestal. The pedestal was placed in a steel disposal box, that was sealed, checked for external contamination, then transferred to a temporary storage area outside until such time that it can be disposed as radioactive waste.

Close inspection of the tipping fixture after its removal from the first pedestal found that bending had occurred in the bands that wrapped around the pedestal. Also, the clamping bolts needed to be replaced. New parts were ordered and assembled on the fixture which resulted in another week delay in the schedule. It was taken back into the CPC where it was installed on the second pedestal. This time all bolts were tightened, no slippage was noted and the pedestal was tipped just as it was originally planned. This pedestal was also transferred into the EDR, placed in a disposal box, and moved outdoors for storage.

The short pedestal lifting fixture was taken into the CPC where it was installed on the pedestal. This pedestal was lifted by the 14.5 Mg hoist and positioned on the transfer cart and taken to the EDR. This pedestal was also placed in a disposal box along with it's lifting fixture. This box was taken to storage outside with the other pedestals.

5.0 Lessons Learned

In reviewing this project and the problems encountered, the tipping fixture bolt and eventual slippage of the pedestal onto the CPC floor was the only major problem encountered that impacted the schedule. When the pedestal cutting started in the CPC, the CPC operations were 35 days behind schedule. Even with the 3 week delay caused by the slipped pedestal and its recovery, the final removal of the pedestals from the CPC ended up to be only 6 days behind the original schedule, recovering 29 days of the CPC schedule. This schedule recovery was due to the equipment operating in a relatively trouble free manner, extended shifts worked by the operators, engineering coverage on shift and the skillful remote operation of the equipment by the operators.

This completed a project that was initially considered by some to be too difficult to attempt. Due to an aggressive schedule for completion, an intensive WVNS team effort had to be put forth. Three engineers, 1 draftsman, and 2 summer students (with only drafting class background) developed the concepts, designed, ordered, and followed fabrication of all the equipment. A select team of operators worked with the engineers during the equipment checkout and mock-up testing. This was followed by the engineers working with the operators on shift during the actual pedestal removal. During the recovery of the slipped pedestal, Engineering, Operations and Radiation and Safety worked as a team to minimize the recovery time and put the project back on the planned production schedule.

DECONTAMINATION AND DECOMMISSIONING OF A FUEL REPROCESSING PILOT PLANT, A PROGRESS REPORT*

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ABSTRACT

The Strontium Semiworks complex on the Hanford Site operated as a pilot plant from 1949 to 1967 to develop two different methods for fuels reprocessing and a method for separating strontium from high level liquid waste. The complex was maintained in safe storage from 1967 until 1983 when the decision was made to decontaminate and decommission the plant. Alternatives considered were: 1) earthen entombment without demolition; 2) partial dismantlement/entombment; 3) razing of all above ground structures; 4) total dismantlement; and 5) no action. An Environmental Assessment was prepared for the project in 1984 and a Finding of No Significant Impact was issued on May 15, 1985. The approved method of decommissioning was Alternative 2 based upon stability of the end product, cost of the project, and projected impacts. As of July 1987 three buildings have been decontaminated, two buildings have been completely dismantled, and the main process building has been partially dismantled and entombed. Remaining work includes cleaning out a 190,000-L (50,000 gal.) tank, decontaminating and demolishing a 61-m (200-ft) stack, entombing an exhaust filter system, and placing an engineered barrier over the entombed facilities.

INTRODUCTION

The Strontium Semiworks complex, located in the 200 East Area of the Hanford Site, was built in 1949 as a pilot plant for the Redox fuels separation process. In 1954 the complex was converted into a pilot plant for the plutonium-uranium (PUREX) process. It continued in this capacity until it was shut down in 1956.

After extensive decontamination, the complex was modified and put back into service in 1961 for the recovery of strontium from fission product waste. The last processing operation performed was the purification of one batch of americium, curium, and promethium. The facility was retired in 1967 and maintained in a safe storage mode until the start of decommissioning in 1983.

The complex originally consisted of a five-cell chemical processing building (figure 1) and its support facilities. The major complex constituents were as shown in figure 2.

* This work was performed for the U.S. Department of Energy under Contract DE-AC06-77RL0-1030.

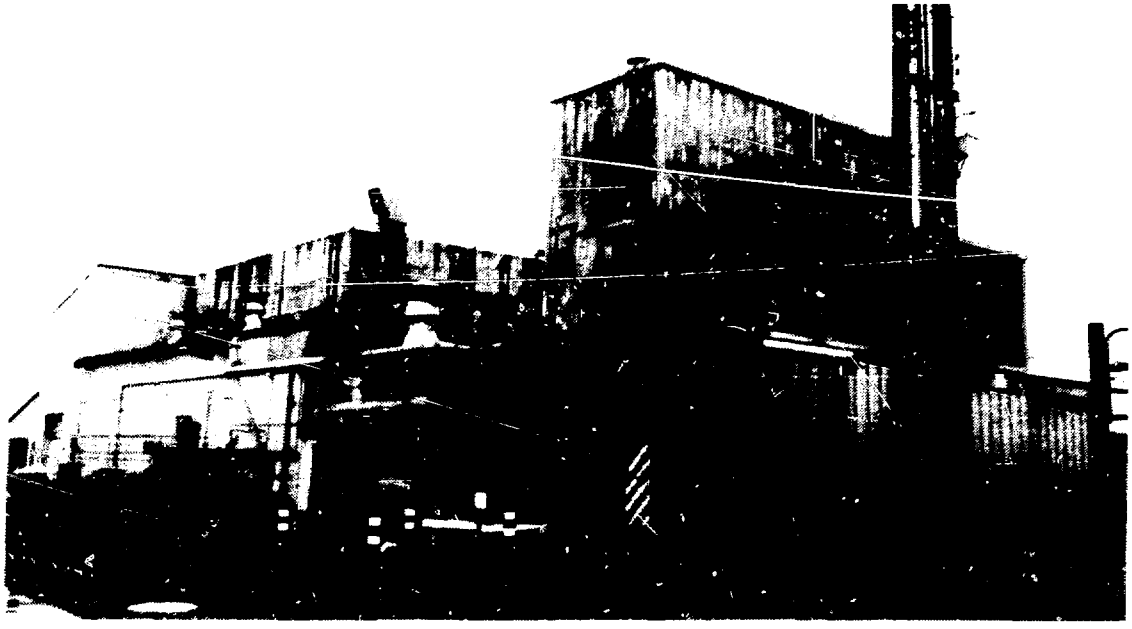

Figure 1. 201-C Process Building and 271-C Aqueous Make-up and Control Building

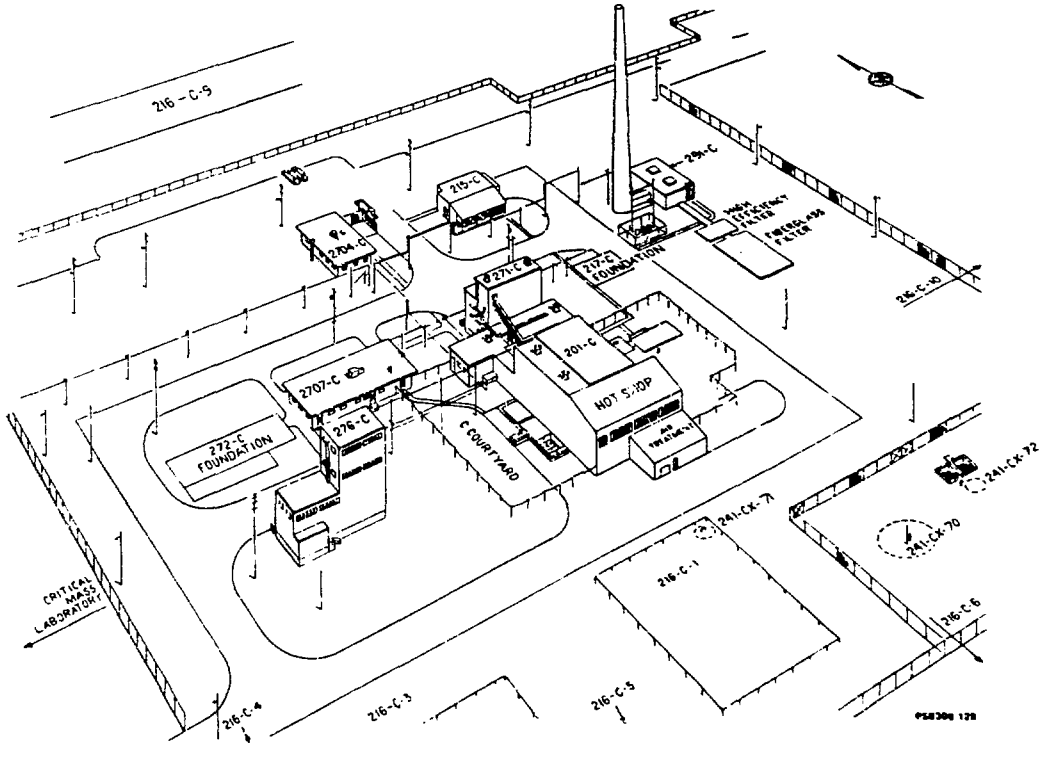

Figure 2. Strontium Semiworks Complex

INTRODUCTION (Continued)

In 1982, an engineering study was prepared to evaluate the feasibility of decommissioning the complex based upon several alternatives and the projected costs of each. A decision was made in 1983 to proceed. From 1983 to 1985 the pre-project documentation was prepared, which consisted of a detailed Project Plan, an Environmental Assessment, and a Safety Analysis Report. The consensus of these documents was that the project could be carried out at reasonable cost and with minimal risk to personnel or the environment. The worker radiation exposure was estimated to be from 0.11 to 0.21 man-Sv (110 to 210 man-rem) and the projected cost was $4.9 million in 1984 funds with no contingency. The hypothetical worst case doses from a project accident are shown in Table I and the calculated long term doses to an on-site intruder one hundred years following completion of the project are shown in Table II.

Table I. Doses Resulting from a Worst Case Accident

	Work Force (man-Rem)*		Public (man-Rem)	
	1-year dose	50-year dose	1-year dose	50-year dose
Total body	12	170	2.5	200
Bone	190	3100	25	1000
Lung	53	53	5.9	5.9
Liver	26	780	2.9	87

* man-Sv = man-Rem/100

Table II. Calculated Doses to Intruders

	Salvage Digger (Rem)*		Homesteader (Rem)	
	1-year dose	50-year dose	1-year dose	50-year dose
Total Body	0.013	0.55	0	3.6 E-4
Bone	0.25	7.5	3.0 E-5	8.5 E-3
Lung	700	7.3	7.1 E-5	8.7 E-5

* Sv = Rem/100

OPERATIONAL ACCOMPLISHMENTS

Accomplishments to date include the decontamination of the 215-C Building, the 276-C Building, and the 2707-C Building. The 271-C Building has been dismantled. The 201-C Building has been dismantled and demolished to a height of 3 m (10 ft) above grade and the remaining portions entombed in concrete. Decommissioning details on each building follow.

215-C Gas Preparation Building Decommissioning

The 215-C Building is a two-room, one-level concrete structure. The main equipment room is 8 x 3 m (27 x 10 ft) and housed the compressor, air dryers, and controls for the 201-C process air system. The other room is 3.6 x 1.8 m (12 x 6 ft) and contained fused switches for operating the compressor and air dryer. On the south side of the building is a lean-to which protected three compressed air tanks. The building was contaminated with low levels of beta- emitting radioisotopes.

Decommissioning began by removing all of the equipment. All equipment was buried in a low level waste burial ground, except for the compressed air tanks which were not contaminated and were sold to the public as excess government property. The building structure was decontaminated by wiping down and is currently being used as non-contaminated storage.

The roof of the building remains contaminated with low levels of radioisotopes.

2707-C Storage and Change House

The 2707-C Building is a one-level wood frame structure, 18 x 7.3 m (60 x 24 ft). The building was constructed to provide maintenance and instrument shops and hot and cold locker rooms. A personnel decontamination room located in one corner of the facility contained a sink, cabinets, and supplies.

The building was decontaminated by removing the decon sink and all contaminated materials in the area. The roof remains contaminated with low levels of radioisotopes.

276-C Solvent Handling Facility

The 276-C Building has a steel framework, insulated metal siding, concrete floors and roof. Overall dimensions are 5.5 m wide by 15 m long (18 x 49 ft) with the east half rising four floors above grade (14 m). The building contained equipment and tanks for the treatment and storage of process solvents used in 201-C operations. A heating and ventilation (HVAC) unit was located on the second floor.

Decommissioning was performed by dismantling the HVAC system and all piping and conduit. Tanks were removed from the upper floors through removable wall panels. The building surfaces were decontaminated by wiping.

271-C Aqueous Makeup and Control Building

The 271-C Building, a three story structure on a concrete foundation, was constructed of steel framing with insulated metal siding and a steel deck roof. Floors were constructed of 12.5-cm-(5-in.-) thick concrete slabs.

The building contained piping, pumps, control instrumentation for the 201-C Building, a heating and ventilation unit and twenty-one aqueous chemical tanks. The tanks ranged in size from 113 to 3030 L (30 to 800 gal.).

The building was decommissioned via dismantlement. All piping, conduit, instrument lines, and small equipment were cut out and removed once they were identified as to potential hazards (acids, radioactivity, etc.). Large equipment and tanks were temporarily left in place. All loose radioactive contamination in the building was removed by wiping or was stabilized by painting.

Each section of siding was unbolted from the building and lowered to the ground with a crane. The building superstructure was then dismantled by rigging onto a section with a crane and using an oxyacetylene torch to cut the section loose. The large tanks on each floor were removed as that floor level became accessible by crane.

201-C Process Building

The 201-C facility consists of three integrated cells (A, B, and C), seven process galleries, a gallery exhaust system, a hot shop, and an air treatment room. In addition, two below-grade cells (D and E) are connected to the east side of the facility. Layout of the cells is shown in figure 3 and a cross section of the building as it stood prior to decommissioning is shown in figure 4.

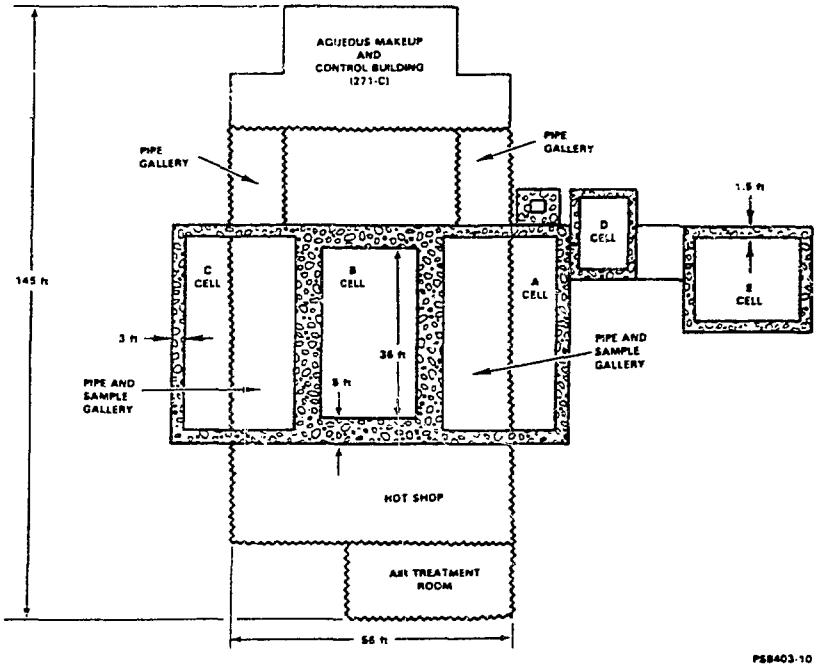

Figure 3. Plot Plan of the 201-C and 271-C Buildings

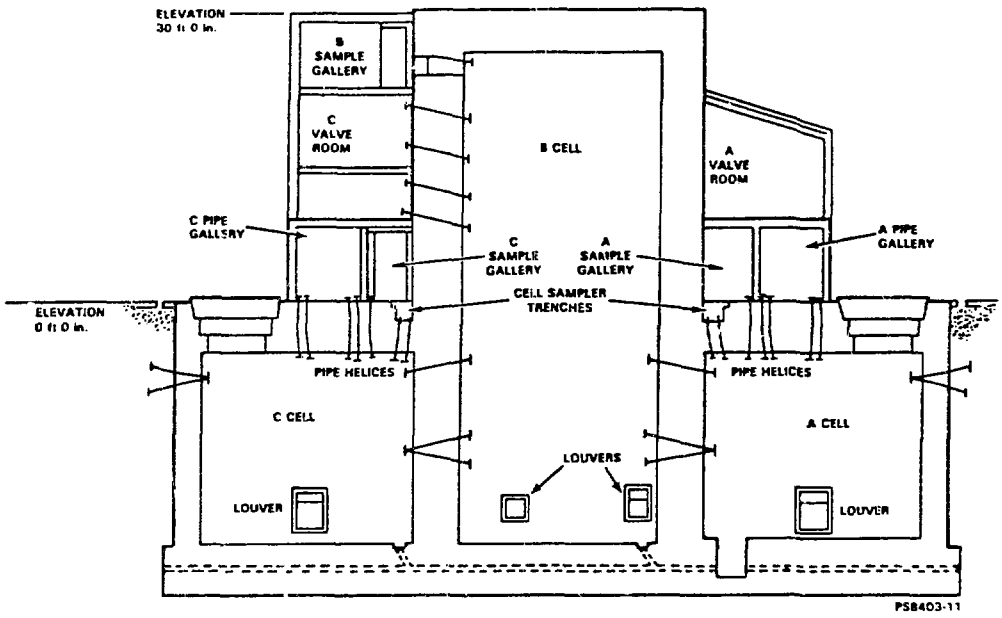

Figure 4. Cross Section View of the 201-C Building

This building housed all of the process tanks, piping, and columns. Although the building was designed for contact maintenance, very few, if any, entries were ever made into A and C cells. All of the valves, pumps, and columns were located in B cell away from the process tanks.

Decommissioning of 201-C was carried out in four phases: 1) gallery, hot shop, and air treatment room dismantlement; 2) B cell decontamination; 3) cell and gallery entombment; and 4) B cell demolition.

Gallery, Hot Shop and Air Treatment Room Dismantlement. The B sample gallery contained process samplers, several of which had leaked during operation. The northernmost sampler had leaked an acidic solution in sufficient quantities to contaminate the wall in a triangular pattern down to grade level. The contamination was etched several centimeters into the concrete. The A and C valve rooms contained asbestos-lagged piping and valves. These galleries were decommissioned by removing contaminated equipment. Smearable contamination was wiped off or painted over. A large wall area below the leaking sampler was chipped out to a depth up to 8 cm. Concrete saws scored the wall before hydraulically operated chisels chipped out the concrete.

The upper gallery structures were dismantled much like the 271-C Building by removing the roof and siding and cutting the superstructure.

The hot shop and the air treatment room were decontaminated and dismantled using the same methods used on the galleries.

B Cell Decontamination. Since the Project Plan called for the demolition of the upper 6 m (20 ft) of this cell, the interior had to be decontaminated and all equipment and piping removed from the upper portions. This task proved to be the most demanding of any performed to-date because of the high levels of contamination and the high radiation dose rates present. The initial dose rates encountered were several Gy/hr from the floor with one hot spot measuring 50 Gy/hr (5,000 rad/hr). The first effort was to pour a 0.3-m- (1-ft-) thick cap of concrete on the floor to seal in the contamination and reduce dose rates to a tolerable level. Concrete with a superplasticiser additive was pumped onto the floor and smoothed out with rakes.

An electrically driven scaffold was then suspended from the ceiling along the centerline of the cell to allow access to the upper portions for decontamination. Initial radiation readings in the areas where piping and equipment had to be removed were approximately 1 to 50 mGy/hr (0.1 to 5 rad/hr) at 0.6 m (2 ft) from the wall.

Heated solutions of potassium permanganate and sodium hydroxide were sprayed onto the walls to wash the contamination off. This was generally successful, as dose rates were lowered to 0.1 to 1.0 mGy/hr (10 to 100 mrad/hr) at 0.6 m (2 ft).

The scaffold was then relocated in turn to each of the wall areas and the piping and equipment were removed. Pipe was removed by first assuring that each was drained by tapping it with an explosively actuated pipe tee. Any residual liquids were drained into a container and disposed of. The piping and equipment were removed down to 0.6 m (2 ft) above grade and placed on the cell floor where concrete was poured to entomb it.

After the piping and equipment were removed, the cell interior was sandblasted with a water injection sandblaster (figure 5) to remove residual contamination. This system deconned the walls to generally less than 0.1 mGy/hr (10 mrad/hr). Spots greater than 0.1 mGy/hr were limited to hairline cracks and other discontinuities in the concrete. These were removed by chipping them out with a small ram-hoe. The west wall of the cell was not deconned to this level since it was to be treated separately during the demolition phase due to its general interior and exterior contamination status.

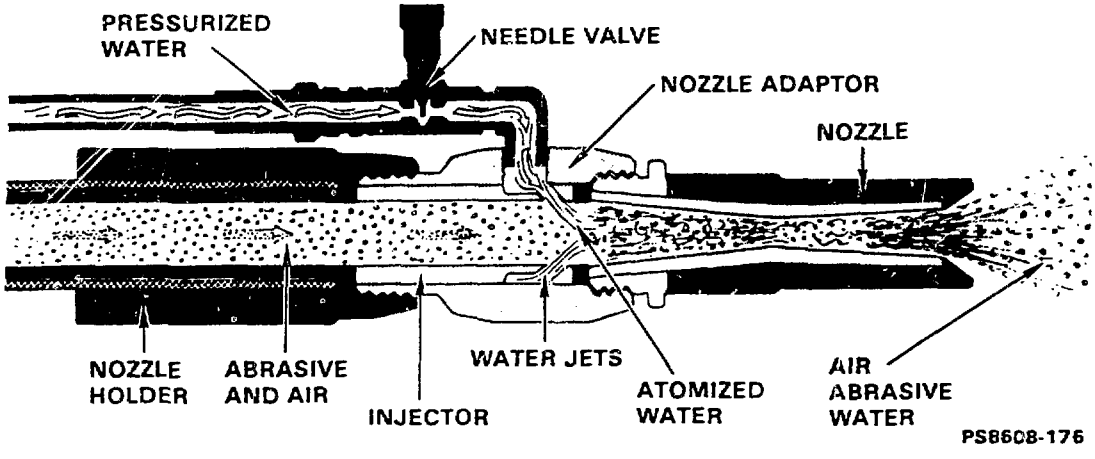

Figure 5. Water Injection Sandblast Head

Entombments. The remaining cells and the lower galleries were then filled with a bulk grade of concrete which was formulated to be essentially self-leveling.

However, before the cells could be filled, there were ten large tanks (greater than 1900-L [500-gal.] capacity) which had to be filled to prevent them from breaking loose and floating and to prevent large void areas in the entombed cells. Filling these tanks was a difficult task. There were no readily accessible fill pipes and high radiation dose rates in the cells prevented access to the tanks. Several alternative methods were evaluated including remotely cutting holes in the tops of the tanks with an arc torch. The method chosen was to pump a thin slurry of grout into each tank via its liquid level measurement tube (0.6-cm [1/4-in.] inside diameter). Testing found that slurry of 50 % fly ash and 50 % cement mixed with water could be pumped and would cure to a dry form. Thirteen tanks were filled with slurry, including three which were included for operational convenience.

After a temporary exhaust duct had been connected to a different cover block opening, the cells were filled by pumping concrete through the cover block openings in their tops. The temporary exhaust connection was necessary since the normal exhaust port was located at the bottom of the cells. Concrete was pumped at rates of approximately 60 m^3/day (80 yd^3/day) during the filling by using a stationary pumper truck and feeding it with 7.6-m^3 (10-yd) delivery trucks.

The galleries were filled by pumping through holes in their tops. Concrete was poured in lifts of 1.2 m (4 ft) and allowed to cure for 48 hours before the next pour so that the gallery walls would act as their own forms. Even so, one section of wall broke through and had to be formed up with plywood.

B Cell Demolition. The demolition of the upper 6 m (20 ft) of the 1.5-m-(5-ft-) thick walls and roof of B Cell was performed by drilling vertical 5-cm-(2-in.-) diameter holes in a staggered pattern with one hole for each 0.2 m^2 (2 ft^2) of surface area. The holes were filled with an expanding grout (S-Mite^R) and allowed to cure for at least one week. The expanding grout produced a general cracking and stress in the structure which facilitated its demolition with a ram-hoe rated at 6000-J (4420-ft/lb) breaker force.

The west wall of the cell was demolished by felling it into the cell cavity. The steel reinforcing bars in the wall were scored with 10-cm-(4-in.-) deep cuts on the inside and outside surfaces. Slanting holes were drilled into the base of the wall at the cut point and filled with S-Mite to provide a fracture zone. The two D8-L caterpillars were attached to lifting eyes installed near the top of the walls to pull the wall over into the cell.

After felling, the wall section was capped with concrete to seal it into the monolith (figure 6).

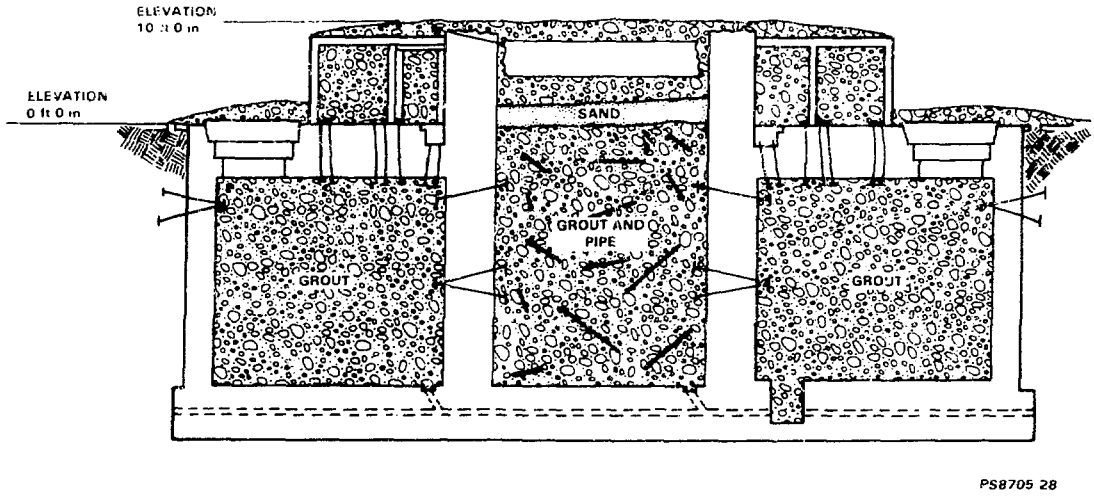

Figure 6. Cross Section of 201-C Entombment

REMAINING ACTIVITIES

Activities which remain to be accomplished include the cleaning out of a waste tank (241-CX-70); the demolition of the 2707-C Building and the 291-C Building; decontamination and demolition of the 61-m (200-ft) tall exhaust stack; entombment of the exhaust filter system; and construction of an engineered earthen barrier over the entombed facilities.

Tank 241-CX-70 Cleaning

Tank 241-CX-70 is a 190,000-L (50,000-gal.) capacity waste tank located southeast of 201-C. The steel-lined concrete tank is 6 m (20 ft) in diameter and 4.5 m (15 ft) high and its top is located 3.3 m (11 ft) below grade.

The tank contains 39,000 L (10,300 gal.) of residue waste with 300 Ci of strontium-90, 51 Ci of cesium-137, and 10 Ci of plutonium-239, -240. The Project Plan requires that this waste be removed to the high-level waste storage and processing system in the 200 East Area.

The tank will be cleaned by installing a rotary tank sluicer and pump assembly in a caisson located in the ground and immediately over an access riser. Due to the radiological restrictions of this task, the sluicing will be semi-remote. The waste solution will be pumped overground in a 5-cm (2-in.) diameter pipeline for 33 m (110 ft) where it will connect to an existing below-ground waste transfer pipeline. The light, fluffy nature of the waste material readily lends itself to pumping and the total pumped volume is anticipated not to exceed 378,000 L (100,000 gal.).

2707-C Building Demolition

The 2707-C Building must be demolished to allow construction of the engineered earthen barrier. Its demolition will be carried out using standard dismantlement and demolition techniques.

291-C-1 Stack Decontamination and Demolition

The 291-C-1 stack, 61 m (200 ft) tall, is built as a stack within a stack. The outer stack is constructed of concrete and steel and tapers from 4.3 m (14 ft) in diameter at the base to 2.3 m (7.5 ft) in diameter at the top. The inner stack is constructed of brick and mortar and tapers from 2.7 m (9 ft) in diameter at the base to 1.5 m (5 ft) in diameter at the top. The inner and outer stacks are separated by a free annular space. An annular shaped sheet metal cap on the top seals the inner and outer stacks together.

The interior of the inner stack is contaminated with high levels of strontium and very low levels of plutonium. A radiation exposure profile taken in September of 1986 showed levels of 2.5 to 3.5 mGy/hr (250 to 350 mrad/hr) measured along the centerline from the top down to about 12 m (40 ft) from the bottom. At that point, the exposure rate climbs to exceed 90 mGy/hr (9 rad/hr) near the bottom. Contamination smears taken in the annular area near the bottom show only a few hundred counts per minute on a pancake-style GM counter.

Decontamination is planned via remote sandblasting (figure 7). The rotary blast head will be lowered down the stack with a winch at a rate of 25 cm/min (10 in./min). Reverse stack ventilation will draw the dust down the stack and into the large fiberglass and high-efficiency particulate air (HEPA) filter system. A pneumatic sand transfer system will move spent sand from the stack base to tank 241-CX-70 where it will be covered with concrete at a later date. The stack will be radiologically surveyed as cleaning progresses to assess decontamination. The goal is to achieve residual exposure rates of 0.05 mGy/hr (5 mrad/hr) or less. Following sandblasting, the stack and annulus will be filled with grout to the 2.1-m (7-ft) level (grade). The inner stack and annulus will be painted with remote rotary spray paint heads as necessary to cover minor remaining contamination.

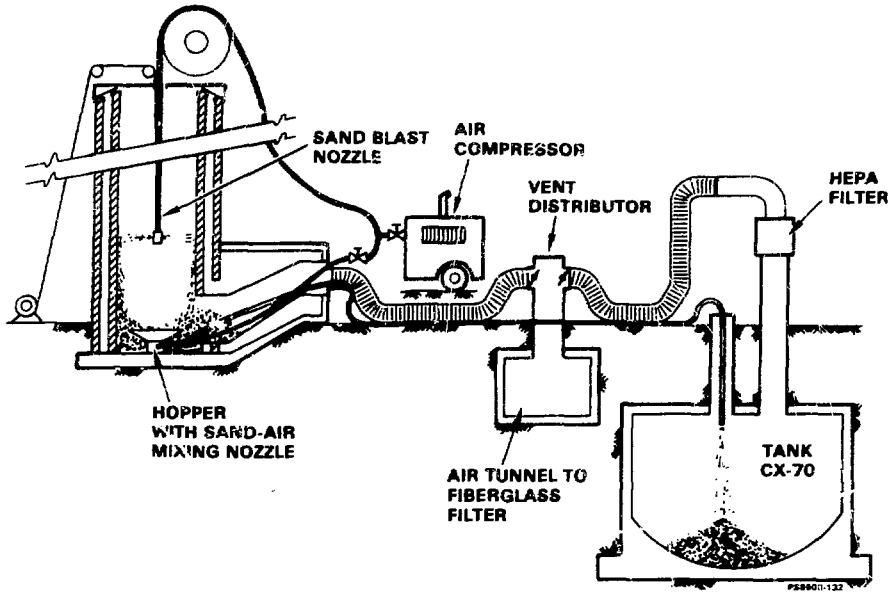

Figure 7. Remote Sandblasting Equipment for the 291-C-1 Stack

Following decontamination, the stack above grade level will be felled with explosives using the vee cut technique. The stack will fall into a trench excavated in the earthen barrier material directly south of its base. A water mist spray will be set up along the trench to control dust and to capture any residual materials which may become airborne when the stack hits the ground. After felling, the stack remnants will be broken up and covered over with an earthen barrier.

Exhaust Filter System Entombment

The exhaust filter system consists of a fiberglass filter unit and a HEPA filter unit. The fiberglass filter unit contains 40 deep-bed fiberglass filters which are 1.5 x 1.5 x 1.2 m thick (5 x 5 x 4 ft). The filters are housed in an underground concrete cell that is 15.8 m long by 8.2 m wide by 2.4 m high (52 x 27 x 8 ft). This filter system contains a calculated 600 Ci of strontium and cesium and 5 Ci of curium and plutonium.

The HEPA filter unit, constructed of steel, is 4 m long by 1.9 m wide by 2 m tall (13 x 6 x 6.5 ft) and houses 22 filters. The HEPA filter system contains an estimated 900 Ci of strontium and cesium.

Decommissioning of these filter banks includes high pressure grouting to fill all internal voids and capping with concrete.

291-C Building Dismantlement

The 291-C Building is the exhaust system fan house for the 201-C Building. The structure, 11 x 7.3 x 3.4 m high (36 x 24 x 11 ft), is a wooden frame building with a 70-hp electric fan and a 70-hp steam turbine fan.

Decommissioning will involve removal of the fans and duct work, removing or painting all loose contamination, and dismantling the building with techniques similar to those used for the 2707-C Building. Removal of the fans and duct work will require negative ventilation to prevent the release of contamination.

Engineered Barrier Placement

An engineered earthen barrier as shown in figures 8 and 9 will be placed over all entombed waste. This barrier will be constructed of bottom ash from the 200 East Area Steam Plant, sand, topsoil and a geotextile (fiberglass cloth). Rock armoring is to be placed on all side slopes.

The design of this barrier is based upon engineering performed by Rockwell for long-term multilayer barriers suitable for placement over high-level transuranic (TRU) waste sites at Hanford. The high-level/TRU barriers for Hanford are being designed to last thousands of years without active maintenance. The Strontium Semiworks barrier incorporates several of the features of a high-level/TRU waste barrier. However, since the entombed Strontium Semiworks site will be low-level waste, its barrier need not last nearly that long without maintenance.

DECOMMISSIONING INNOVATIONS

During the course of this project thus far several innovative techniques have been used. Motorized scaffolding allowed safe access to the upper reaches of the 15-m- (50-ft-) high B Cell without requiring people to climb on fixed scaffolding or walk over dismantlement debris on the floor of the cell.

Explosively activated pipe tees were used to safely open old pipelines to assure that they were empty of residual liquids before cutting. These tees could be set up to safely drain liquids into a catchment for later disposal.

A bulk fill grade of concrete allowed the pouring of over 500 m^3 (700 yds^3) of concrete without vibrators or other means to achieve a reasonably level surface. The bulk fill grade was also less expensive than standard at a delivered 100-k (60-mi) round trip cost of only \$56/m^3 (\$43/yd).

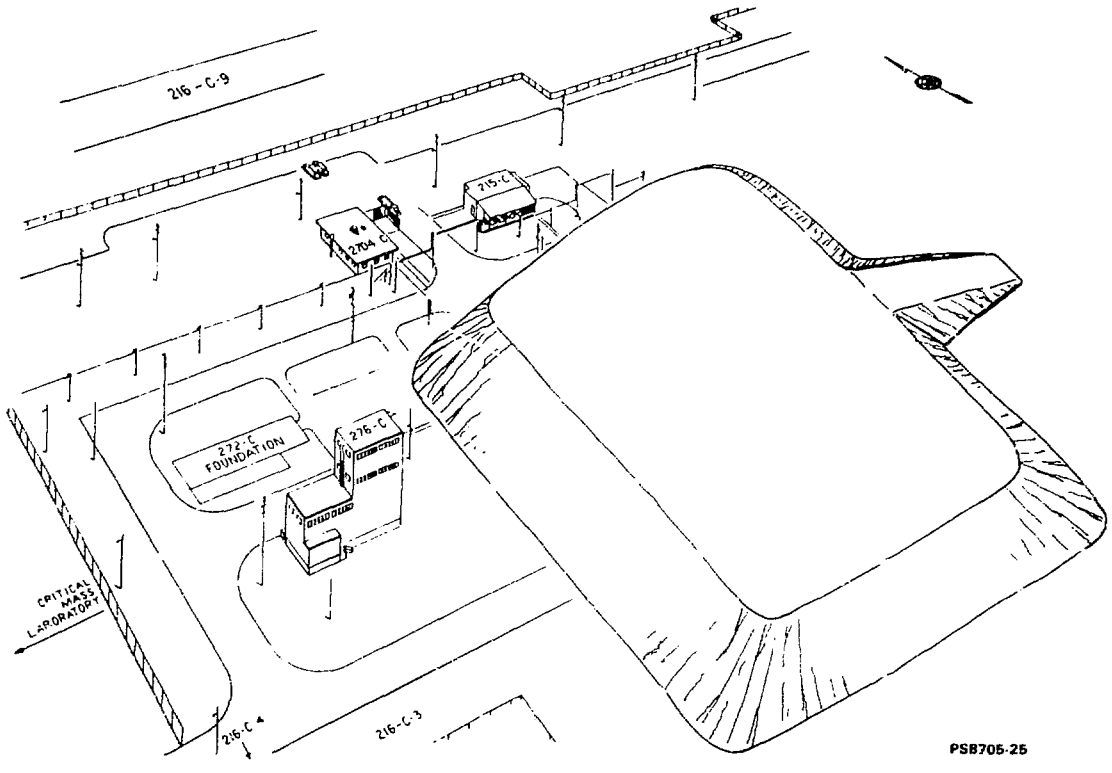

Figure 8. Area Drawing of Semiworks Earthen Barrier

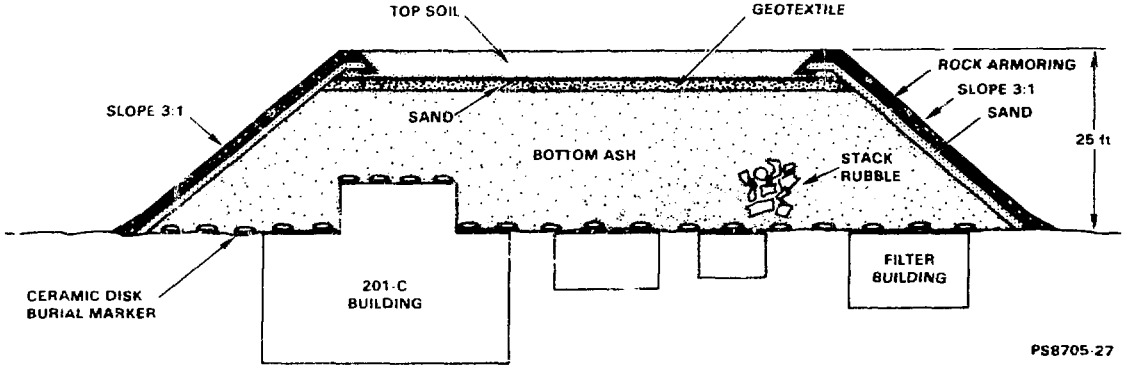

Figure 9. Earthen Barrier Design Cross Section

A slurry grout was used to fill tanks through existing small-diameter pipelines. This prevented the need for direct tank access for their filling.

A water injection sandblast technique was used to clean the walls of a hot cell while preventing the recontamination of already cleaned areas. The injection system successfully prevented airborne contamination.

Other equipment used to enhance the operational safety of the project includes the use of Gore-Tex ^{R} clothing for protection from skin contamination in a wet, hot environment. A controlled descent device was also used when working at heights above 3 m (10 ft). These devices were anchored overhead and would automatically lower workers at a rate of 0.6 m/min. (2 ft/min.) in the event of a fall or in the event that an emergency escape was necessary.

SUMMARY

The decontamination and decommissioning of the Strontium Semiworks Plant has provided numerous opportunities to use existing and newly acquired skills and techniques.

Performance has met or nearly met the established schedule; however, unanticipated changes in the overall scope or difficulty have driven the project anticipated cost up to a current total of $6.7 million. The total expended personnel radiation dose is lower than originally anticipated due to aggressive management of ALARA techniques. The current total dose is 0.06 man-Sv (6 man-rem) and remaining tasks are low-exposure work.

SPENT FUEL POOL CLEANUP AND STABILIZATION

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ABSTRACT

Each of the plutonium production reactors at Hanford had a large water-filled spent fuel pool to provide interim storage of irradiated fuel while awaiting shipment to the separation facilities. After cessation of reactor operations the fuel was removed from the pools and the water levels were drawn down to a 5- to 10-foot depth. The pools were maintained with the water to provide shielding and radiological control.

What appeared to be a straight forward project to process the water, remove the sediments from the basin, and stabilize the contamination on the floors and walls became a very complex and time consuming operation. The sediment characteristics varied from pool to pool, the ion exchange system required modification, areas of hard-pack sediments were discovered on the floors, special arrangements to handle and package high dose rate items for shipment were required, and contract problems ensued with the subcontractor.

The original schedule to complete the project from preliminary engineering to final stabilization of the pools was 15 months. The actual time required was about 25 months. The original cost estimate to perform the work was $2,651,000. The actual cost of the project was $5,120,000, which included $150,000 for payment of claims to the subcontractor.

This paper summarizes the experiences associated with the cleanup and radiological stabilization of the 100-B, -C, -D, and -DR spent fuel pools, and discusses a number of lessons learned items.

INTRODUCTION

There are eight fuel storage basins at retired production facilities in the 100 Areas at Hanford. Two of these basins were modified and are used for N-Reactor fuel storage (105-KE and 105-KW). Two other basins (105-F and 105-H) were stabilized by filling with soil following the reactors' shutdown. The remaining four basins (105-B, 105-C, 105-D, and 105-DR) contained contaminated water, sediment, materials, and equipment left in the basins at the time the reactors were shut down between 1964 and 1969.

As part of the safe storage responsibilities for the retired reactor areas, DOE-RL requested that UNC proceed with the project to clean up and stabilize the 105-B, -C, -D and -DR fuel storage basins. The project included the removal of the contaminated water, sediment, materials and equipment in order to reduce the potential for a loss of radiological control and to minimize surveillance and maintenance efforts for these facilities while waiting for final decommissioning. One of the objectives of this interim stabilization project was to leave the basins in a condition that would not affect the decommissioning options to be considered through the National Environmental Policy Act (NEPA) process for the final disposition of the shutdown reactors. The project work was originally planned in nine phases. The first phase, preliminary engineering and project planning, was completed in January 1984. The ninth phase is project closeout, which included a final project report.

The other seven phases were divided into the work tasks to be done by UNC and tasks to be performed by a subcontractor.

UNC

- Small material/equipment removal (Phase II),
- Concrete surface cleaning (Phase III),
- Large material/equipment removal (Phase IV), and
- High dose rate item removal (Phase V).

Subcontractor

- Sediment removal and disposal (Phase VI),
- Contaminated water processing (Phase VII), and
- Final concrete sealing (Phase VIII).

These seven phases were to be completed sequentially for the D, DR, B, and C basins. The sediment removal and disposal and processing of the contaminated water by the subcontractor for all four basins was to be completed by September 30, 1984.

The actual work did not follow the planned sequence. The concrete surface cleaning (Phase III) was left until the final concrete sealing (Phase VIII) with both phases performed by UNC.

DISCUSSION

General

The fuel storage basins are located in the 105-B, -C, -D, and -DR reactor buildings which are in the 100-B/C and 100-D/DR dual reactor areas. These areas are located along the south side of the Columbia River where it traverses the northern part of the Hanford Site in south-central Washington State. The reactor areas are located approximately 30 miles from the city of Richland (Figure 1).

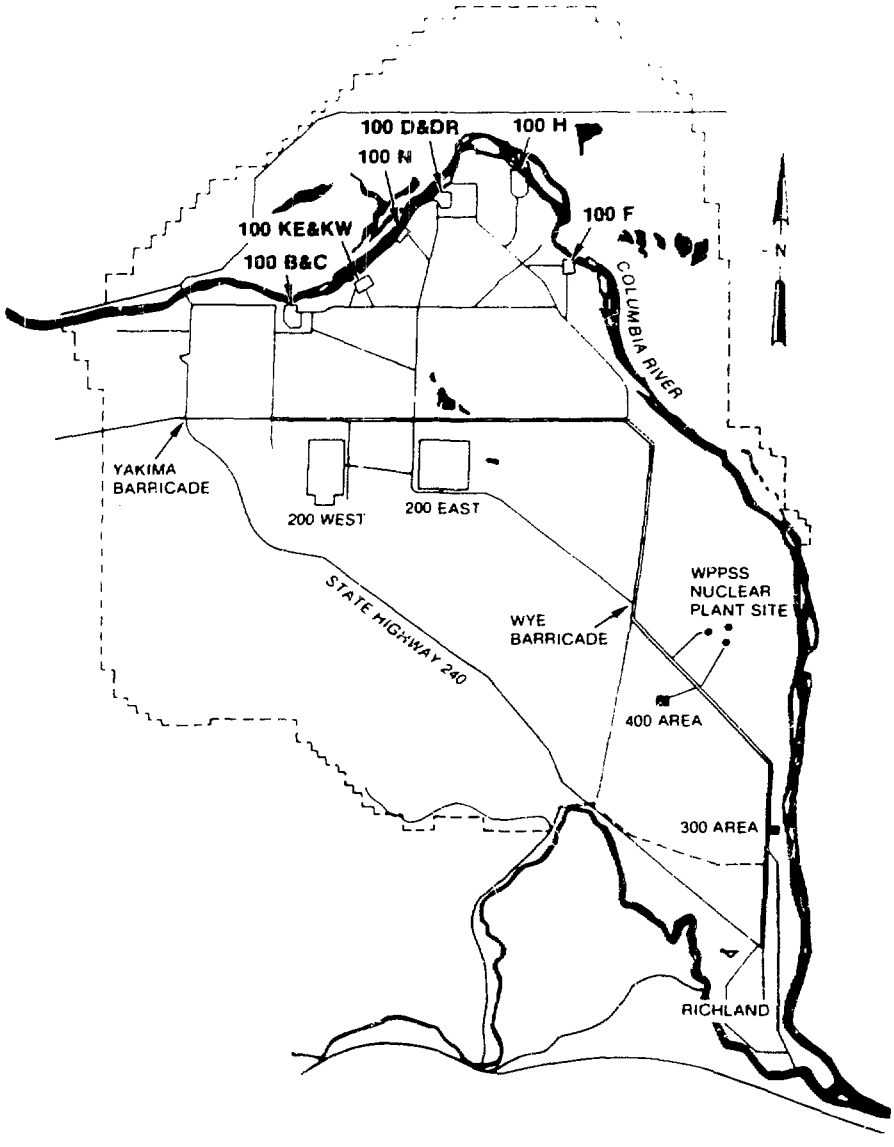

Figure 1. Hanford Site

The fuel storage basins are located at the rear of the reactors (Figure 2). The concrete basin area served as a collection, storage, and transfer facility for the irradiated fuel elements discharged from the reactor. The water in the basins served both as coolant and as shielding. Although the arrangement of the 105-C basin is slightly different, each reactor fuel storage basin consists of a discharge chute and fuel element pickup area, a storage area, a transfer area, and a wash pad area. The total floor area for these components averages between 7,000 and 10,000 ft^2.

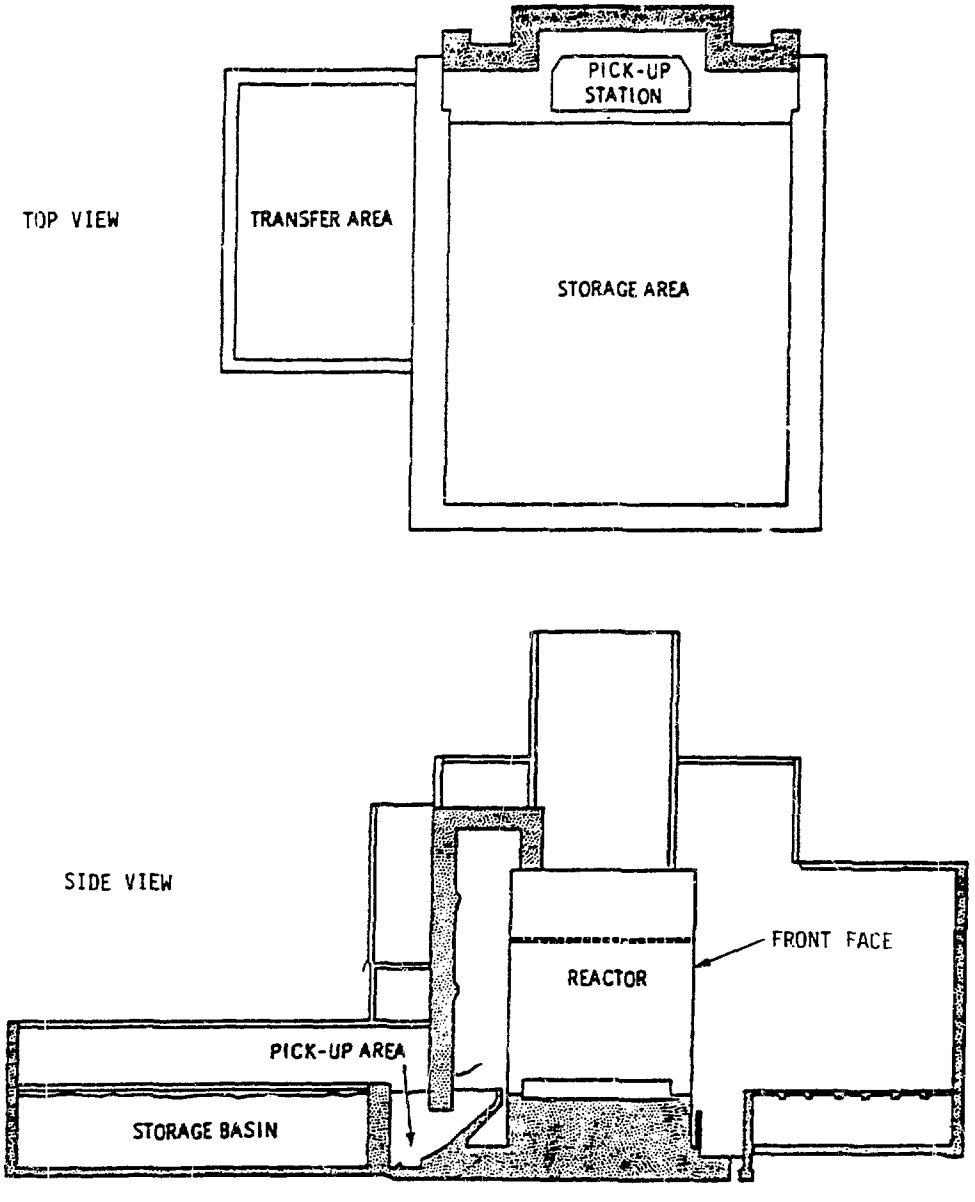

Figure 2. Layout of Fuel Storage Basin within Reactor Building

The transfer area pits which measure 6 ft-4 in. x 9 ft are located at one corner of the fuel storage basins and at the inner end of the fuel transfer areas. The transfer pits are 5 ft deeper than the basins and are connected to the basins by a canal and a monorail system (See Figure 3). Here the irradiated fuel was loaded into casks, then raised by an overhead crane and placed in special railroad cars for shipment to the chemical reprocessing facilities in the Hanford 200 Area.

At the time of the basin cleanup project each of the basins had from 4 ft to 10 ft depth of water for radiological control. A scum layer consisting of accumulated dust and algae covered the surface of the water. Each of the basins contained significant quantities of miscellaneous debris and hardware remaining from reactor operations. This miscellaneous material consisted mainly of contaminated piping pieces, thermocouple wires, fuel storage buckets, tongs, reactor process tubes, and fuel element spacers. Covering the floor of each basin was varying depths of sediment.

The sediment appeared to be primarily iron oxide and silt with a depth of less than one inch at 105-C and up to six to twelve inches at 105-D and 105-DR.

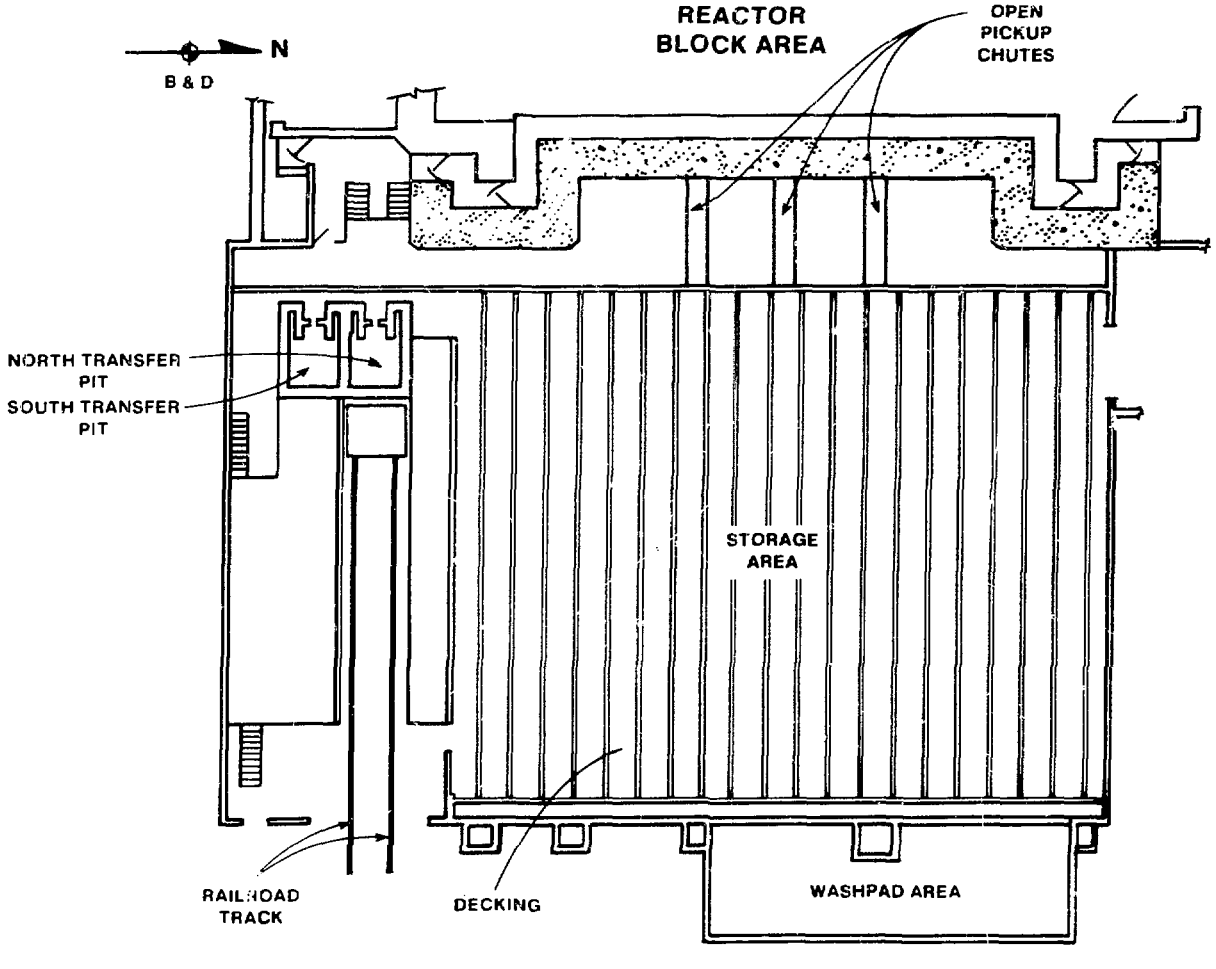

Figure 3. Typical Layout for Fuel Storage Basins

Radionuclide Inventory and Radiation Survey

Samples of the sediment and water were taken at each basin. Table 1 shows the concentration of each major radionuclide. The total mass of sediment at each basin was estimated at 50,000 kg. The average beta-gamma and plutonium-239/240 concentrations in the basin water were 2.2×10^{5} and 3.7×10^{1} p Ci/liter, respectively.

An underwater probe was used to measure the dose rates over the entire floor area of each basin. Dose rates ranged less than 100 mR/hr to hot spots of several R/hr at contact. The personnel working level above the basins was uniformly 1 mR/hr or less.

Project Plan and Objectives

The objective of the Fuel Storage Basin Cleanup and Stabilization Project was to remove the contaminated water and sediment from the 105-B, -C, -D, and -DR fuel storage basins and "fix" the remaining contamination in order to reduce the potential for a loss of radiological control and to minimize surveillance and maintenance of these facilities while waiting for final decommissioning disposition. The intent of this effort was to leave the basins in a condition that would not require any substantial subsequent decontamination to effect the final decommissioning mode.

The work would remove the water from each storage basin to preclude any possibility of leakage, and to stabilize the dry basins such that the potential for spread of contamination to the environment would be minimized. In order to do this the sediment and debris had to be removed from the basin floor. Further, the basin walls needed to be hosed down as the water level was lowered so that the additional sediment generated by the cleaning action could also be removed. When the water was completely drained, the basin walls were surveyed and when a potential existed for possible spread of contamination, the applicable areas were "stabilized" in a manner to minimize the possibility of losing contamination control.

The objectives for this project supported the overall Decommissioning Programs' main objective, which is to decommission all of the shut-down facilities in the 100 Areas of the Hanford site in the safest, most environmentally sound, and most cost-effective way possible.

Project Activities

Small Material/Equipment Removal. Small material/equipment removal began at 105-B basin in early 1984. Equipment and hardware that would fit into the fuel storage buckets and have a contact dose rate of less than 200 mR/hr were removed. The loaded buckets were placed into 10 mil plastic bags with industrial grade vermiculite as an absorbent and then placed into fiberboard boxes for disposal. Other small items with contact dose rates of greater than 200 mr/hr were removed and packaged in shielded containers for transport and disposal. Any items having high dose rate and having the dimensions of a fuel element were stored in small underwater shielded casks for later packaging and

TABLE 1
105 STORAGE BASIN SEDIMENT SAMPLES

Concentration (pCi/g)

Area	Location	Pu-238	Pu-239/240	Sr-90	H-3	Eu-155	Cs-137	Eu-154	Co-60	Eu-152	U	Ni-63
105B	Techview Pit	8.0×10^{2}	1.7×10^{4}	4.3×10^{4}	1.2×10^{2}	5.7×10^{4}	3.6×10^{5}	2.6×10^{5}	6.7×10^{5}	4.1×10^{3}		
105B	Dummy Ele. Chute	5.0×10^{1}	1.3×10^{4}	6.2×10^{5}	1.1×10^{2}	4.5×10^{4}	2.9×10^{5}	2.4×10^{5}	8.6×10^{5}	4.0×10^{4}		
105B	Pickup Chute	5.0×10^{3}	9.5×10^{4}	7.3×10^{5}	2.1×10^{3}	4.4×10^{4}	7.9×10^{5}	9.2×10^{4}	5.0×10^{5}	6.5×10^{4}		
105B	Transfer Area	1.9×10^{2}	4.0×10^{3}	2.2×10^{4}	6.1×10^{1}	1.8×10^{4}	1.1×10^{5}	1.6×10^{5}	9.5×10^{5}	7.8×10^{4}		1.2×10^{6}
Ave.		1.5×10^{3}	3.2×10^{4}	3.5×10^{5}	6.0×10^{2}	4.1×10^{4}	3.9×10^{5}	1.9×10^{5}	8.0×10^{5}	4.7×10^{4}		
105C	Dummy Ele. Chute	1.7×10^{3}	6.8×10^{4}	2.3×10^{4}	1.1×10^{2}	7.5×10^{4}	1.4×10^{5}	3.4×10^{5}	1.2×10^{6}	1.8×10^{5}		
105C	Techview Pit	7.9×10^{2}	3.3×10^{4}	1.8×10^{4}	2.0×10^{2}	2.1×10^{4}	2.5×10^{5}	1.0×10^{5}	6.4×10^{5}	4.8×10^{4}		
105C	Transfer Area	6.6×10^{2}	1.6×10^{4}	1.0×10^{4}	5.4×10^{1}	1.4×10^{5}	1.3×10^{5}	5.8×10^{5}	1.4×10^{6}	3.0×10^{5}		
105C	Pickup Chute	2.5×10^{2}	3.6×10^{3}	1.8×10^{4}	7.2×10^{1}	2.4×10^{4}	1.4×10^{5}	1.8×10^{5}	1.5×10^{6}	1.1×10^{4}		3.1×10^{5}
Ave.		8.5×10^{2}	3.0×10^{4}	1.7×10^{4}	1.1×10^{2}	6.5×10^{4}	1.6×10^{5}	3.0×10^{5}	1.2×10^{6}	1.4×10^{5}		
105D	Pickup Chute	9.0×10^{1}	2.1×10^{3}	4.6×10^{3}	3.1×10^{2}	1.4×10^{5}	5.4×10^{4}	3.7×10^{5}	1.0×10^{6}	2.8×10^{5}		
105D	Transfer Pit	1.8×10^{2}	1.1×10^{3}	2.2×10^{4}	3.6×10^{2}	5.1×10^{4}	6.3×10^{4}	2.6×10^{5}	1.0×10^{6}	1.9×10^{5}		
105D	Northwall	8.8×10^{1}	4.9×10^{2}	7.5×10^{3}	4.9×10^{2}	6.4×10^{4}	2.1×10^{4}	2.2×10^{5}	1.1×10^{6}	1.5×10^{4}		2.0×10^{5}
105D	Techview Pit	5.0×10^{1}	9.4×10^{2}	6.4×10^{3}	5.4×10^{2}	3.5×10^{4}	3.1×10^{4}	1.7×10^{4}	5.3×10^{5}	1.7×10^{5}		
Ave.		1.0×10^{2}	1.2×10^{3}	1.0×10^{4}	4.2×10^{2}	7.2×10^{4}	4.2×10^{4}	2.6×10^{5}	9.1×10^{5}	1.6×10^{5}		
105DR	Techview Pit	1.0×10^{2}	1.4×10^{3}	8.5×10^{3}	7.6×10^{2}	4.2×10^{4}	6.4×10^{4}	2.8×10^{5}	1.3×10^{6}	2.2×10^{5}		4.3×10^{5}
105DR	Wash Pad	3.6×10^{1}	7.1×10^{2}	3.1×10^{0}	4.5×10^{2}	1.3×10^{4}	5.0×10^{4}	6.7×10^{4}	4.8×10^{5}	3.6×10^{4}		
105DR	Pickup Chute.	1.0×10^{2}	3.0×10^{3}	6.3×10^{4}	3.2×10^{2}	2.3×10^{4}	3.1×10^{5}	1.3×10^{5}	6.4×10^{5}	5.5×10^{3}		
105DR	Transfer Area	5.3×10^{1}	1.1×10^{3}	1.8×10^{4}	2.0×10^{2}	3.3×10^{4}	5.3×10^{4}	2.2×10^{5}	9.9×10^{5}	5.3×10^{4}		
Ave.		7.2×10^{1}	1.6×10^{3}	2.2×10^{4}	4.3×10^{2}	2.8×10^{4}	1.2×10^{5}	1.7×10^{5}	8.5×10^{5}	7.9×10^{4}		

transport to 100-N Area for accurate identification. Remote handling techniques were used for removal of small items; consequently, personnel dose rates were usually less than 100 mrem/hr. The small material/equipment removal phase of the project proceeded as planned. Removal of all visible small items from 105-B basin was complete during the week of February 9, 1984, and completion of the other three basins shortly thereafter.

Large Material/Equipment Removal. Some special tooling was designed and fabricated to assist in the removal of large items. Large items were present in all the basins and ranged from pieces of large piping and structural steel to process tubing. The 105-C basin had the greatest quantity of large items mainly in the form of zirconium process tubing. The zirconium tubing required segmenting to piece sizes that would easily fit into shipping containers. Cutting of the longer pieces of tubing to lengths that would fit into the shipping containers was accomplished by use of a remote-controlled tubing cutter jig. Even though contact dose rates on some tubing sections were several R/hr, personnel dose rates averaged less than 100 mrem/hr during the packaging operation. Due to high dose rates, all zirconium tubing segments were packaged into special lead- or concrete-lined boxes (UNI-4476 containers).

When all visible hardware (small and large) was removed and packaged, the basin floors were raked to uncover any remaining items that may have been covered with sediment. A significant number of small items were removed as a result of the raking. The concrete walls and structures above the cover water in the basins were washed down with water to remove loose contamination.

Water and Sediment Removal. A subcontractor was obtained on a fixed-price contract to remove the water and sediments from the basins. The subcontractor began water and sediment removal at 105-D basin in July 1984. Subcontractor equipment consisted of a vacuum head to remove sediment, a filter press to remove sediments from the water, cation and anion exchange columns, final filters, holdup tanks to provide sample points before release of water, and associated piping and pumps to run the system.

The subcontractor processed about 620,000 gallons of water from the 105-D basin. Of this quantity, 120,000 gallons required reprocessing prior to discharge because the water would not meet the release levels[1]. The major difficulties experienced by the subcontractor during the liquid removal were related to plugging of the ion exchange columns, and failure of the filter press to remove particles. As a result of equipment problems, the holdup tanks became contaminated and required subsequent decontamination prior to resuming operations.

The subcontractor removed approximately 400 ft^2 of sediment from the 105-D storage basin. Two vacuum head designs were tried in unsuccessful attempts to collect the sediments on the filter press. Finally, the basin sediment was washed with a fire hose into the transfer pits and manually shoveled into 55-gal drums. The contact dose rate on the sediment in the pit was 1.5 R/hr. Personnel dose rates ranged from 200 to 350 mrem/hr while the sediments were shoveled into the drum.

During the washdown, it was discovered that what was thought to be the floor surface was actually compacted sediment. The compacted sediment ranged in thickness from less than an inch up to ten inches. During the removal of this compacted sediment, additional basin debris was uncovered which included two suspect fuel pieces. These suspect fuel pieces were shipped in shielded casks from 105-D to 105-C which served as a collection point until all suspect pieces could be transferred to 100-N for accurate identification.

The subcontractor completed removal of the water and sediment from 105-D basin after 90 working days at the end of September 1984 and began setting up at the 105-DR basin.

The subcontractor began processing water and sediment from the DR basin during October 1984. A bag filter had been added to their processing system at 105-D and was added at the pond discharge for 105-DR to protect against possibly contaminating the discharge pond. Approximately 70,000 gal had to be reprocessed from the holding tanks prior to discharge because the water failed to meet the release limits. The subcontractor again experienced problems with processing the basin water due to particulate material.

The subcontractor removed approximately 400 ft^3 of sediment from the DR storage basin. A new vacuum head design was tried in an unsuccessful attempt to collect the sediment on the filter press. As at 105-D basin, the sediment was finally removed by washing it into the transfer pits with a fire hose and manually shoveling it into 55-gal drums. Dose rates were similar to those experienced at the 105-D basin.

Fewer total items were found in the DR sediment, but of these items three were identified as suspect fuel elements. The suspect fuel elements were transferred to 105-C for examination, and were eventually sent to 100-N.

The subcontractor completed removal of the water and sediment from 105-DR basin after 30 working days at the beginning of November 1984 and began setting up at 105-B basin. An additional filter was added upstream of the ion exchange columns to prevent plugging of the columns by particulates.

The subcontractor processed approximately 100,000 gal of 105-B basin water, of which 60,000 gal were discharged following once-through processing to the 105-B discharge pond and approximately 40,000 gallons had to be reprocessed prior to discharge because the water failed to meet the release limits. Processing problems were traced to contamination by particulate material in the holdup tanks. As a result, basin cleanup work was again delayed while the tanks were decontaminated.

As the water was lowered to the 6-in. level, increased dose rates were encountered around the pickup chute. Attempts by the subcontractor to remove sediment from this area resulted in increased dose rates to personnel from the lowered water level and from particulates accumulated in the process equipment. As a result, UNC directed the subcontractor to raise the water level for shielding, to suspend operations at 105-B, and to mobilize his equipment to the 105-C fuel storage basin. The subcontractor had worked 30 days in 105-B. The final cleanout of the 105-B basin was completed later by UNC forces.

Water and sediment processing from the 105-C basin began in December 1984. When the cover water was processed down to about the 1-ft level, the dose rate above the water and at the work areas began increasing significantly. Work was then suspended at the direction of UNC after 15 working days.

The agreement with the subcontractor was renegotiated to allow completion of the subcontract with the current status of the basins. UNC purchased the subcontractor's processing equipment by exercising the options provided in the original contract. The subcontractor left the facilities in April 1985 and UNC resumed operations at the 105-C basin in May 1985.

Prior to resuming operations, the process equipment utilized by the subcontractor was evaluated and several modifications were made to both the equipment and operating procedures. The following is a brief summary outlining these changes.

1. The major cleanup problem was the inability of the subcontractor's equipment to process the particulates and water simultaneously. The particulate material caused low flow rates through the system due to plugging and process water contamination. To eliminate these problems, the following design and procedural modifications were made by UNC:

   A. Particulate bursts from the filter press were eliminated by recirculating the water through a 1300-gal mix tank during interruptions in system flow, which eliminated pressure and flow transients across the filter press. In addition, the filter press was precoated with diatomaceous earth and water clarity was checked prior to commencing operations.

   B. Two "CUNO" filter units were placed between the filter press and ion exchange columns to protect the inlet screens from becoming plugged with particulate material. The plugged screens were replaced with clean screens to ensure good system flow rates.

   C. An ion exchange column containing IRA-938 Anion Resin was included in the system to remove any colloidal material from the process water.

   D. A bag filter was placed downstream of the ion exchange columns to capture possible resin fines.

   E. Two 20,000-gal holdup tanks were thoroughly decontaminated prior to commencing operations. In addition, the tanks were coated with ALARA Coat to ensure that any residual contamination remained fixed.

2. Only three ion exchange columns were utilized: two 25-ft^3, sodium-form, cation exchange resin columns and one IRA-938 anion exchange resin column.

3. A new vacuum head was designed by UNC. A bronze, swimming pool-type, vacuum head was utilized. In order to maintain the sludge in suspension, several high pressure water nozzles were added to the head (Figure 4).

4. A new diaphragm pump was ordered. The 2-in. flap valve design could pump material as large as 1-3/4 in. in diameter.

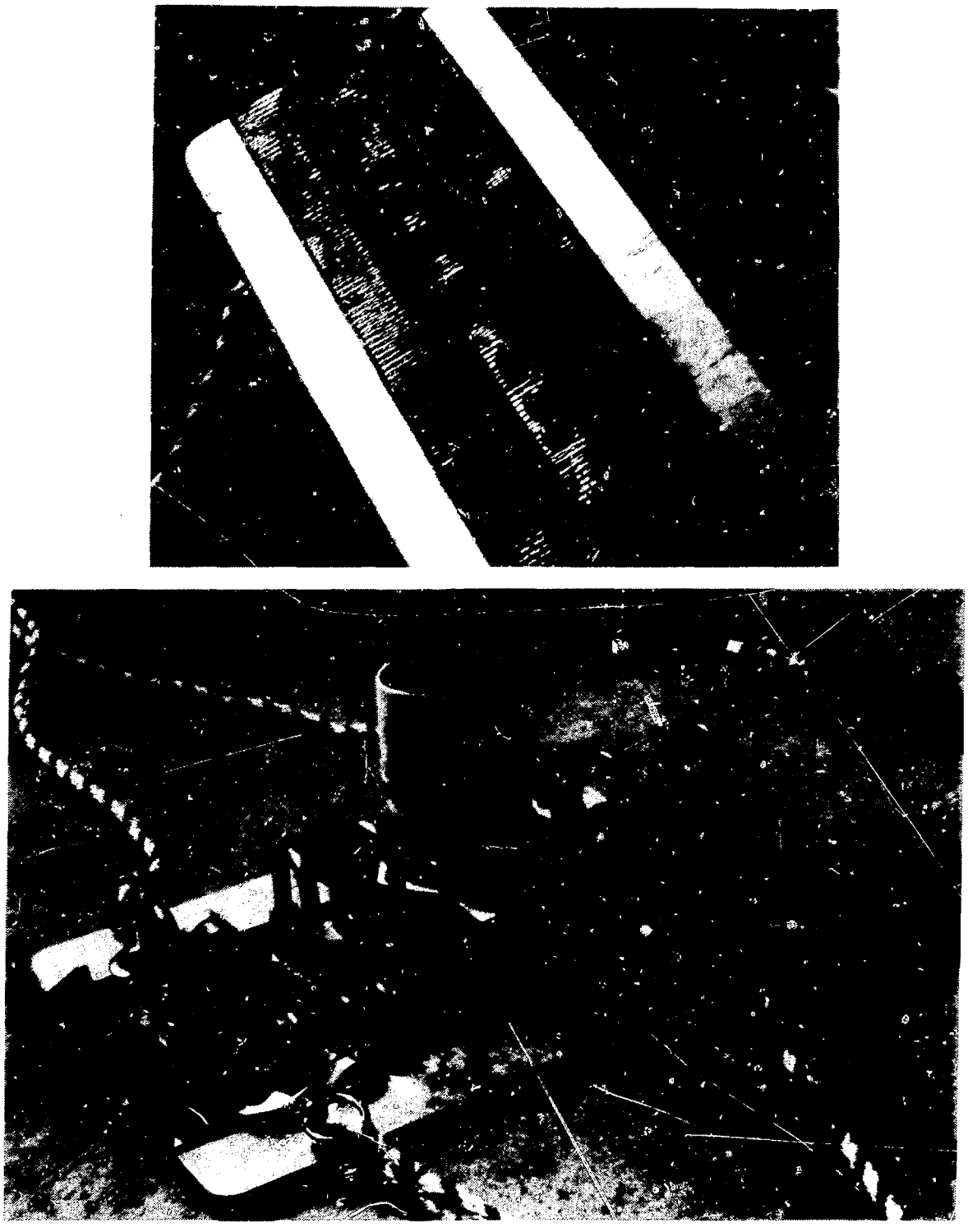

Figure 4. The Vacuum Head as Modified for Use in the Fuel Storage Basin.

After considering personnel radiation exposure and waste packaging requirements, a decision was made to move the sediments to the transfer pits for interim storage. Following cleanup of the transfer pits to ensure that no fuel elements remained prior to sediment storage in the transfer pits, dams were fabricated for the transfer pit areas at the basins to add depth and prevent solid items from entering the pits. Portable dams were devised to help direct the sediment to the pits and to provide sufficient settling time for the sediment. See Figure 5 for the conceptual drawing of the sediment transfer. In addition, dewatering manifolds consisting of a matrix of CUNO filters (Figure 5) were placed at the bottom of the transfer pits along with 18 in. of sand and gravel to dewater the sediment after transfer was complete.

UNC processed approximately 80,000-gal of water from C basin and discharged it to the 105-C discharge pond. All water processed was well within the release limits.

UNC moved approximately 600 ft^3 of sediment to the C basin transfer pits. Approximately 50% of the sediment was transferred utilizing the vacuum head described above. Because of the large amounts of debris (plastic sheeting, paper, clothing, etc.) present in the sediment, it was necessary to transfer the remaining material using fire hoses. It had proved too difficult to manipulate the vacuum head through the debris. The water was supplied to the fire hoses by recirculating water out of the transfer pits, through the filter press, and back to the basin. As a result, approximately 30 ft^3 of sediment were removed by the filter press and subsequently packaged for disposal as low-level waste based on sediment sample analysis. Final sediment transfer was performed manually using shovels. Screens were devised to make sure that no fuel fragments or other debris were accidentally allowed into the pits.

While fuel elements (real or suspect) were stored at the 105-C basin facility it was necessary to provide additional security measures to protect the material.

A full-time security patrolman was assigned to the basin area. The security procedure called for a patrolman to check the building and to all entrances to the storage basin area.

UNC completed the sediment transfer and water removal from the 105-C basin in August 1985. Containment covers fabricated to place over the sediment in the pits were put in place August 16.

UNC resumed water processing at 105-B basin in September 1985 with the improved system developed at 105-C basin. Approximately 650,000 gal of water were processed to release limits and discharged to the 105-B discharge pond. UNC's experience at the 105-C basin with the equipment and the situation led to two major decisions about handling the sediment at the 105-B basin: 1) Because of personnel exposure and waste packaging requirements, a decision was made to move the sediments into the transfer pits for interim storage; 2) Because of the operational difficulties experienced with the vacuum head at the C basin, the decision was made to utilize the washdown approach for transferring the sediments into the transfer pits except for high dose-rate areas, where the vacuum system was required to reduce dose rates prior to washdown. The vacuum system removed enough sediment from the high dose-rate areas so that the use of the dams that had been built as a contingency to maintain water levels for shielding were not required.

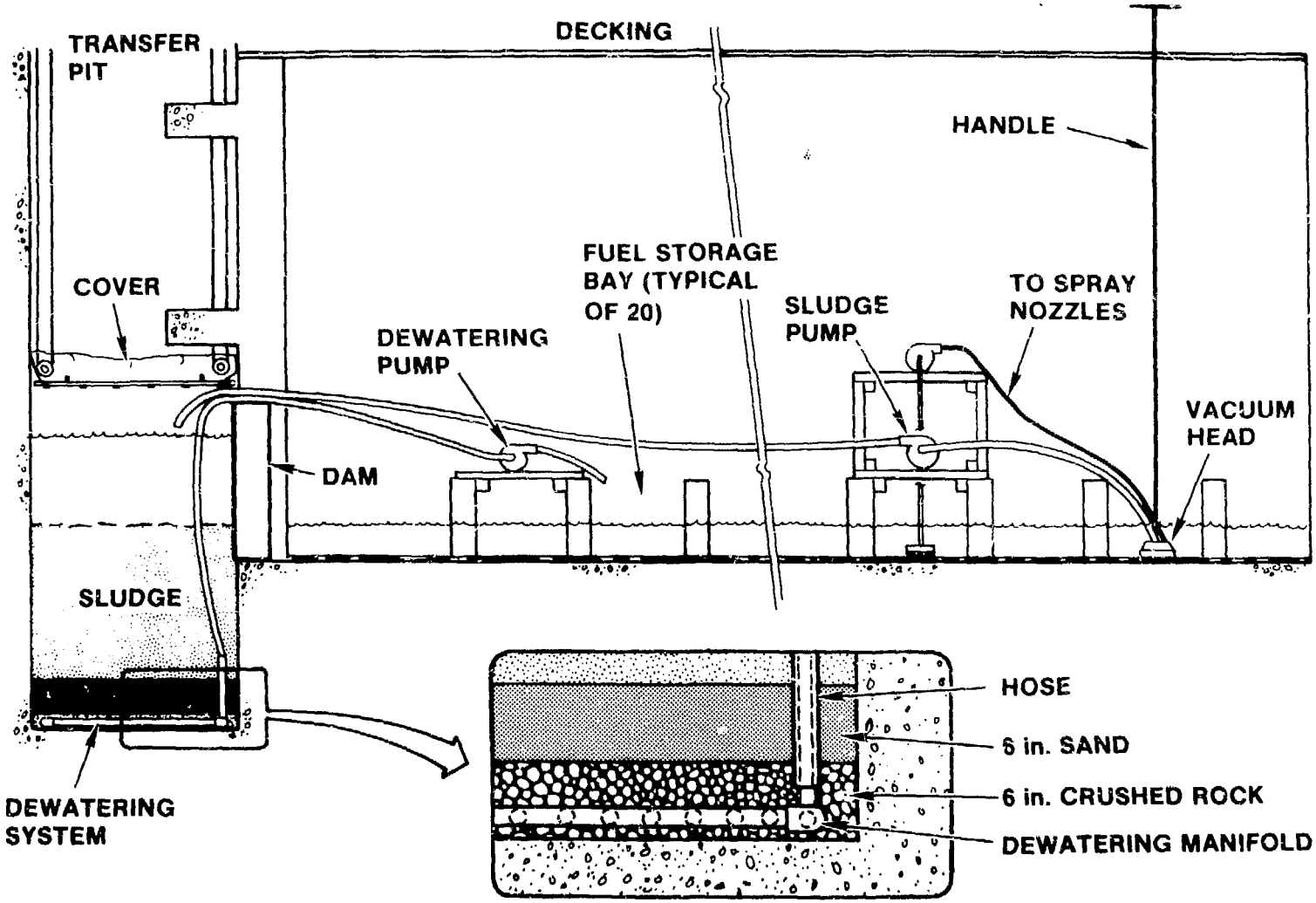

Figure 5. Transfer Pit Containment Design

Approximately 600 ft^3 of sediment was moved to the transfer pits. Approximately 30 ft^3 of sediment was removed from the filter press after the recirculation of washdown water from the transfer pits.

UNC completed removal of the water and transfer of the sediment in the 105-B storage basin in December 1985. Containment covers were placed over the sediment in the pits January 20, 1986. Figure 6 shows a cover in place. Final dose rates over the sediment were 15 mR/hr at the deck level at each pit except the south pit at 105-C, where the rate was 5 mR/hr. Dose rates on contact (the cover was not shielded) were 1 R/hr at both 105-B pits and 1.5 R/hr (south pit) and 350 mR/hr (north pit) at 105-C.

A total of 14 suspect fuel elements were found during the 105-B basin cleanout. These elements were ultimately transferred to 100-N for final examination and identification.

Figure 6. 105-C Transfer Pit after Installation of Cover

While the final cleanout of the 105-B and 105-C basins was still underway, preparation for the final fixing of contamination on the concrete walls and floors of the basins was initiated. Several sealers were tested and an asphalt emulsion (ATCO 1840) was selected as the stabilizing material.

Due to the deterioration of the concrete in many areas, the walls were swept and vacuumed to remove dust and loose pieces. To control airborne contamination during this task, the basin deck was covered with plastic and two HEPA ventilation units were installed. Before the asphalt was applied, concrete samples were taken for final characterization[2]. The average working dose rate before and after fixing the contamination was less than 5 mrem/hr.

The ATCO 1840 was rolled on and stabilization of the 105-D basin was completed on September 3, 1985. Similarly, stabilization was completed at the 105-DR basin on September 23, 1985.

Sweeping of the 105-C basin was completed on October 17, 1985. A new Graco Bulldog airless sprayer was used to apply the ATCO 1840. The use of the sprayer reduced the time required to coat the basin from 14 days to 4 days. Similarly, sweeping and coating of the 105-B basin was completed on January 24, 1986.

The present condition of the four fuel storage basins is considered to be sufficiently stable to require minimal surveillance and maintenance until final disposition. The final disposition of the basin facilities will be determined by the NEPA process currently underway for decommissioning the surplus Hanford production reactors.

Over the course of the project, there was one lost workday accident, ten minor first aid injuries, and eleven skin contamination cases. Five of those individuals received skin contamination when a hose clogged, became disengaged, and sprayed the workers. They were all successfully decontaminated. One Radiation Occurrence Investigation was held. During packaging of radioactive waste, radiation dose rates exceeded those permitted by the Radiation Work Permit. One individual received 160 mrem which could have been avoided if proper work methods had been followed. Disciplinary action was taken against a Supervisor and a Manager as part of the corrective action. A total of 41 man-rem of exposure was used over the course of the project.

Of the final inventory of suspect or identified fuel elements, 45 pieces were identified as fuel, 55 pieces were identified as non-fuel (spacers, test material holders), and 33 pieces required further testing.

Waste Management. A total of 24,300 ft^3 of solid wastes were removed from the basins and packaged and transported to the Hanford 200 Area low level waste burial facility. The total volume of liquids that were treated and released to the environment was about 1,210,000 gallons.

Lessons Learned. The complexity and duration of the basin cleanout project provided several areas where the original approach to the task did not provide the most efficient or cost-effective solution. As the project progressed, there were a number of design and operation changes that were made to the equipment systems and a number of evaluations and decisions made that can benefit future decommissioning projects. Of special note are the following items:

- The division of project tasks between UNC and the subcontractor caused some delays in the project schedule when high-dose-rate items were found in the hard packed sediment.

- A Project Engineer with the necessary responsibility and authority must be assigned to major projects; this in lieu of an engineer assigned only as a technical advisor.

- Major projects where the exact complexity of the work is not entirely understood require detailed up front planning and engineering rather than attempting to "fast track" the project.

- Subcontractor equipment design should be reviewed in depth to provide assurance that the design is adequate for the work.

- A realistic assessment of the complexity and magnitude of a project must be made before commitments to milestones, schedules, and costs are made.

- Adequate contingency must be included in decommissioning project cost and schedule estimates to allow for the unexpected.

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SECTION IV

DECONTAMINATION TECHNIQUES

DECONTAMINATION TECHNOLOGY - A U.S. PERSPECTIVE

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ABSTRACT

Substantial progress has been made in the U.S. since the 1982 International Decommissioning Symposium in developing, demonstrating and applying chemical and selected nonchemical decontamination techniques. Although much of this progress has been motivated by the decontamination requirements of operating facilities, the resulting technology is directly applicable to decommissioning-related decontamination needs. This paper provides an overview of these advances and applications to provide a U.S. perspective for the decontamination technology and applications presented at the 1987 Symposium.

INTRODUCTION

Decontamination is an integral and very important part of almost all decommissioning operations. Representative applications include decontaminating major systems and components as a precursor to other decommissioning operations to reduce radiation exposure to decommissioning personnel; the selective removal of contaminated concrete and other surface layers to permit the handling and disposal of the underlying material as nonradioactive waste; the decontamination and reuse of tools, equipment and protective clothing used in the decommissioning operation; the decontamination of removed components and materials for volume reduction or recycle and reuse; and the decontamination of structures and sites to prescribed residual radioactivity limits for reuse or release.

The objective of this paper is to provide an overview of major U.S. decontamination technology advances and applications in the five years since the 1982 International Decommissioning Symposium. Although many of these decontamination techniques were developed or applied primarily for major maintenance/repair operations, to support post-accident recovery activities, or even for non-nuclear applications, the resulting decontamination technology and experience base is directly applicable to decommissioning needs.

The status of decontamination technology in 1982 is well represented by the various techniques and applications reported in the Symposium papers[1] and other conference proceedings and reports[2-4] of that period. For example, loose or smearable contamination on structural and component surfaces was removed by vacuuming, hand or mechanical scrubbing with detergents and chemical agents, the application of foam containing decontamination agents, high-pressure water, strippable coatings, and the use of organic solvents and high-pressure Freon.®

Techniques used or developed for stripping paint and tile and the removal of contaminated concrete surface layers included solubilizing, grinding, chipping, mechanical and flame spalling, sandblasting and water blasting. Metallic surfaces with fixed contamination were decontaminated using acids, immersion and in-situ electropolishing, and electropickling. Abrasive cleaning and vibratory finishing were used to remove fixed contamination from both metallic and nonmetallic surfaces.

Other decontamination techniques in use in 1982 included ultrasonic/chemical decontamination of removed components, steam/hot water cleaning, and the application of decontamination gels and pastes. The chemical decontamination of utility-owned nuclear plant primary coolant circuits was just beginning. Site decontamination was accomplished using conventional earthmoving equipment, such as backhoes, front-end loaders and bulldozers. Some studies had been conducted exploring the use of polyurethane foam to strip contamination from soil surfaces, and the removal of soil contaminants by wet screening and attrition scrubbing.

The following sections provide an overview of the further refinement and application of these decontamination techniques and new developments since the 1982 International Decommissioning Symposium.

CHEMICAL DECONTAMINATION TECHNIQUES

Substantial progress has been made during the last five years in further developing chemical decontamination techniques for application to major nuclear power reactor systems and components.[5-7] More than 30 successful decontamination operations have been conducted during this period on BWR recirculation and reactor water cleanup systems and on PWR channel heads. Although the primary motivation for the development of this chemical decontamination technology is exposure reduction

associated with special maintenance operations such as the replacement of recirculation system piping, the resulting technology is directly applicable as a precursor to plant decommissioning operations.[8]

The majority of these decontaminations have been performed using a dilute chemical process in which a low concentration of reagent is circulated through the component or system. Either feed-and-bleed or fill-and-drain methods are employed, and the radioactive waste is usually removed with ion exchange resins. The two types of dilute chemical decontamination systems currently available employ either combinations of mildly reducing organic acids and chelating agents, or a stronger reducing agent based on a low oxidation state metal ion (LOMI). Both of these processes are effective and produce adequate decontamination factors with proper application. However, both may require an oxidizing pretreatment for corrosion films with a high chromium content of the type commonly encountered in PWR systems.

Concentrated chemical decontamination processes also have been further developed and applied. The decontamination of the retired Dresden-1 reactor using this alternative approach[9] produced decontamination factors ranging from 2 to 25 in the various reactor systems and subsystems.

As a result of this recent development activity, chemical decontamination of reactor systems and major components is a rapidly maturing technology with several vendors offering a variety of proven decontamination processes and services. It should be noted, however, that the major process selection considerations for an operating plant, such as minimum impact on outage time and corrosion concerns, are not important for decommissioning-related decontamination operations. Considerations such as cost, decontamination effectiveness for maximum exposure reduction, and radioactive waste volumes and the associated handling, processing, and disposal requirements[8,10] may dictate selection of a different chemical decontamination process for a decommissioning operation than those preferred for operating plants.

NONCHEMICAL DECONTAMINATION TECHNIQUES

The nonchemical decontamination techniques discussed in this section refer to the variety of surface cleaning and removal methods developed or adapted for component and facility-type decontamination applications. Since these techniques and their applications have been described in a recent review paper,[11] only significant recent developments or decommissioning-related applications of the nonchemical decontamination techniques will be highlighted in this paper.

Manual Techniques

Although there have been continuing improvements in equipment and procedures[12] for facility decontamination, the emerging interest in using robotics and other remote technologies to perform manual decontamination tasks such as washing, scrubbing, vacuuming and steam cleaning has the most potential for significant savings in cost and exposure. Recent examples include the Nine Mile Point Unit 1 testing of an underwater robotics device for scrubbing and vacuuming[13] and the extensive use of robotic systems at the West Valley Demonstration Project for decontamination and other decommissioning-related tasks.[14]

Abrasive Cleaning

The recent development and application of abrasive cleaning technology has focused on the air slurry technique,[15] in which a water-abrasive slurry is propelled against the contaminated surface by a jet of compressed air. This three-component approach permits better control of the surface removal rate than with conventional air or water abrasive blasting methods. The amount of material removed can be matched to the specific decontamination requirements to permit decontamination of a variety of metallic and nonmetallic materials with minimal secondary waste generation.

Ultrahigh-Pressure Water

The initial development work on an ultrahigh-pressure water scarifier for contaminated concrete surfaces that was reviewed at the 1982 Decommissioning Symposium[16] has been extended to include the decontamination of metals[17,18] and the development and demonstration of a remotely operated cleaner/scarifier.[19] The metal decontamination studies demonstrated that ultrahigh-pressure water (>70 MPa) can remove corrosion layers and even erode stainless steel surfaces for the proper combination of nozzle stand-off distance and traverse rate.

The effectiveness of these higher pressures for decontamination applications was confirmed using TMI-2 leadscrew sections with cesium contamination incorporated in a tightly adherent surface layer. This fixed contamination was readily removed at a pressure of 240 MPa and a stand-off distance of 15 cm to give essentially bare metal and a decontamination factor of more than 140. It should be noted that the severe surface roughening produced by this process, which represents a serious disadvantage for operating systems subject to recontamination, is not a concern for decommissioning-related applications.

The further development of ultrahigh-pressure water technology for metal and concrete decontamination has resulted in a remotely operated cleaner/scarifier that has a swath width of 35 cm and can remove concrete to a depth of 0.6 cm at 1 cm/s. The device is equipped with a shroud and vacuum system that contains and collects the water and cutting debris to leave a clean and virtually dry surface.

Electrochemical Techniques

Electropolishing is developing into a mature decontamination technology that is used for a variety of exposure reduction and waste management applications.[20] Although phosphoric acid remains the primary electrolyte, some work has been done to explore alternative electrolytes that are more amenable to purification and recycle.[21] Sodium nitrate electrolytes of the type used for electrochemical machining will decontaminate metal surfaces, with most of the removed metal precipitating as a hydroxide. However, electrical efficiencies are quite low unless very concentrated, acidified solutions are employed.

The recent focus for electropolishing has been on the development of procedures and equipment for the in-situ decontamination or prepolishing of large components and surface areas. Examples include the in-situ decontamination of steam generator channel heads to reduce radiation exposure during tube inspection and repair operations, the prepolishing of hot cell liners and the surfaces of reactor refueling cavities to facilitate subsequent decontamination, and the prepolishing of replacement reactor recirculation piping to retard radiation buildup. The inside surfaces of the 0.3 to 0.7-m diameter replacement piping for the Monticello Plant, for example, were electropolished at the site using special movable internal cathodes that could accommodate elbows, reducers, crosses and other complex pipe geometries.

Although the development of in-situ electropolishing methods for large components and surface areas has been primarily for prepolishing applications, the same technology is directly applicable to decommissioning needs. Specific consideration also has been given to remote applications of electropolishing and cerium redox decontamination for decommissioning operations.[22]

Strippable Coatings

The major advance in fixatives technology since 1982 has been the development of a self-stripping coating material. This polymer penetrates and bonds to the corrosion layer and, upon curing, develops sufficient internal stress to spell the corrosion layer from the substrate. The spalled coating and incorporated contamination can be collected by vacuuming. As with other strippable coatings, application thickness must be closely controlled for ready removal or spalling. However, unlike the other coatings, the cured self-stripping coating is water soluble and can be removed by scrubbing for areas that fail to spall.

Although the self-stripping coating was originally developed for rust removal from carbon steel,[23] tests show that it also is effective in removing smearable and even some fixed contamination from stainless steel.[24] Moreover, no essential difference in decontamination effectiveness was noted for coatings removed by self-stripping as compared with thinner coatings that did not spall and were removed by scrubbing.

Mechanical Techniques

Several industrial mechanical descaling techniques have been evaluated for decontaminating the inside of piping.[25] These include various types of rotating brushes, cutters, scrapers and hones; and cylindrical scrapers and devices with abrasive surfaces that are forced through the piping by water pressure. Both the rotating wire brush and the rotating brush hone (a brush with nylon bristles tipped with silicon carbide pellets) appeared promising for in-situ pipe decontamination applications. Both techniques removed surface corrosion layers to base metal and were capable of negotiating and cleaning pipe elbows. Radioactive tests on 30-cm diameter BWR pipe indicated ready removal of accessible surface contamination at a traverse rate of about 1 mm/s, but very slow removal of contamination in pits and other subsurface features.

The mechanical decontamination work also encompassed the development and demonstration of unique delivery systems for high pressure water used to flush loose contamination from drain piping systems and other comparatively inaccessible areas.[26] This included evaluation of a 50 MPa, 1600 cm^3/s rotating cavitating jet system for decontaminating drains; an eight-jet, self-feeding mole nozzel for decontaminating small-diameter piping; and a unique segmented, cable-manipulated

nozzel positioning mechanism that can be inserted through a tube or other existing access point and remotely operated to deliver high pressure water to specified offset locations within a reactor head or other constricted area.

High-Pressure Freon Cleaning

Freon cleaning has become an accepted and important decontamination technique[27] for a number of component and facility applications. Although higher pressures (>30 MPa) have been explored, most of the development work has been oriented toward new or more demanding applications such as the decontamination of large cranes or hot cell equipment.

Application Examples

In addition to the preceding discussion of specific nonchemical decontamination techniques, there are a number of recent examples of the application of one or more decontamination methods that provide valuable information on planning, costs, exposure, waste volumes and other considerations that are important for decommissioning-related decontamination operations. These examples include:

- Use of grit blasting, a high-temperature steam cleaning/vacuum cleaner, an ultrasonic Freon degreaser, a bead blasting glove box, and manual techniques to decontaminate the steam generator channel heads, the reactor refueling cavity, a variety of tools and equipment, and facility surfaces as part of a year-long outage to replace the steam generators at the H. B. Robinson Nuclear Plant.[28]

- Use of various facility decontamination techniques including the removal of contaminated concrete surfaces by grinding as part of the West Valley Demonstration Project.[29,30]

- Use of scabbling techniques to decontaminate concrete shielding slabs from the TMI-2 plant to substantially radwaste disposal volumes.[31]

- Use of scabbling, scrubbing and vacuuming techniques to remove contaminated concrete and coatings as part of the decontamination operations at TMI-2.[32]

SOIL DECONTAMINATION

Although various soil decontamination or segregation processes have been explored,[33] in-situ vitrification, presently under active development, is the only alternative to soil removal . The extensive experience gained through recent site cleanup activities has, however, provided a substantial data base and methodology for planning and conducting soil removal operations.[34,35] In addition, potentially applicable new technology for pavement cutting and soil excavation is being developed by the Gas Research Institute[36] to reduce costs associated with the installation and replacement of underground piping. This includes improved pavement-and concrete-cutting techniques; robotic digging systems; a hand-held rotary excavation tool; a supersonic air nozzle for excavating soil; and a multipurpose field support vehicle that provides a soil stabilizer mixing system, a vacuum soil removal system, and compressed air and hydraulic power supplies for the new cutting and excavation equipment.

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DECONTAMINATION STUDIES RELATED TO GARIGLIANO BWR DECOMMISSIONING

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ABSTRACT

Garigliano BWR is a 160 MWe power plant which operated from 1964 to 1978. In 1982 the ENEL Board of Directors declared it definitively out-of-service.

Specific studies were started to select the best decommissioning alternatives. A feasibility study on dismantling and decontamination of all the building components was performed in preparation for decommissioning of the turbine building in a few years.

Specific experimental activities were also started specifically in the decontamination field. A programme which deals with the dismantling and decontamination of the tube bundle of a feed-water preheater is the most important and significant task.

The aim of this research is to decontaminate the tubes as much as possible in order to reach the unrestricted material release limits.

Preheater No. 4, which was operated at the highest temperature and pressure, was selected for the experiments. The bundle has tubes with 16 mm OD in AISI Type 304; the full length of tubes in the preheater is about 12000 m. The tubes are covered with an adherent contaminated oxide layer both inside and outside; the contamination was measured to be about 65 Bq/cm^2 of 60-Co mainly.

Other decontamination studies such as decontamination tests on small valves with aggressive chemical, spent decontaminant treatment and decontamination tests with more recent chemical processes are also reviewed.

Part of the work presented in this paper is being carried out as a jointly funded project with the Commission of the European Communities, as part of the Community Research Programme on the Decommissioning of Nuclear Installations (1984-1988).

INTRODUCTION

Garigliano power plant is a 160 MWe Boiling Water Reactor with dual cycle. It operated from 1964 to 1978 generating 12.5 10^9kWh with a capacity factor of 68%. In August 1978 it was stopped due to damage to one of the two Secondary Steam Generators (SSG).

In March 1982 the Italian Electricity (ENEL) Board of Directors declared it definitively out-of-service. In December 1982 ENEL approved an

"Action Plan" to put the Garigliano plant in safe storage conditions for at least 30 years. This decision was supported by the fact that more than 30 man-Sv are needed for immediate dismantling; moreover there is no storage area or final waste disposal at present in Italy.

At present the irradiated fuel elements are being taken out of the plant and transported to the Avogadro storage pool; high and medium level radioactivity wastes generated during the plant life are also being treated and conditioned[1].

In this context several R & D activities were started in the decontamination field in particular. These activities are being performed in collaboration between the Special Unit for Decommissioning (USD of ENEL G & T Division) and the Thermal and Nuclear Research Centre (CRTN of ENEL R & D Division), with CISE lab support.

Some research was and is sponsored by the Commission of the European Communities as part of the Community Research Programmes on the Decommisioning of Nuclear Plants and Installations (1979-1988).

DISMANTLING AND DECONTAMINATION OF THE TURBINE BUILDING: FEASIBILITY STUDY

In the safe storage of the plant the turbine building is considered a boundary part and it could be decommissioned in a few years. Therefore a feasibility study on dismantling and decontamination (up to unrestricted release limit) of all the building components was performed as a possible alternative to decommissioning[2]. Direct melting of scraps for both release and reuse in the nuclear field, was also considered as another alternative.

About 1930 tons of materials are installed in the turbine building with about 33000 m^2 of contaminated surface. The base materials are carbon and low alloyed steels. The radioactivity inventory is about 38 GBq (1021 mCi) mainly due to 60-Co (80%) and 137-Cs (15%) with traces of 134-Cs, 54-Mn and 90-Sr. The highest contamination levels were measured inside the primary steam pipe (3.7 kBq/cm^2), inside the off-gas line (1.1 kBq/cm^2) and in the condenser hot-well (0.7 kBq/cm^2).

The average dose rate levels inside the turbine building are about 10 μSv/h with a maximum of about 100 μSv/h near the condensate water treatment system. All inventary data are detailed in Table I.

The purpose of the feasibility study was to select techniques which can decontaminate components as much as possible, in order to allow the unrestricted release of the materials (a limit of 1 Bq/cm^2 or 1 Bq/g for beta and gamma emitters, was considered).

The dismantling and cutting of all building components before decontamination was considered as the basic goal, in order to minimize the secondary wastes. Several decontamination techniques were considered and compared in terms of effectiveness, secondary waste generation, plant and safety impacts, costs and so on.

The results show that there is no single decontamination technique which can be used for all components or parts and that the use of more

decontamination techniques is needed.

Four decontamiantion techniques were selected for decontaminating all components: electropolishing, for pieces with simple geometries (flat and concave shapes); water jet with abrasives, for quite complex pieces (mainly convex shape); ultrasounds in aggressive chemicals, for heat-exchangers and condenser tube bundles; freon jet, for engines and all parts with loose contamination.

The relative quantities for each technique in terms of weight, contaminated surface and radioactivity are summarized in Figures 1-3. A small part of the materials (3.4% wt.) was considered for direct storage without decontamination, for economic reasons in particular.

To perform the decommissioning of the turbine building components, a requirement of more than 200000 man-hour (of which 70000 man-hour in surveilled areas) was estimated; 20 people for about 5 years were considered. A total exposure of 0.5 man-Sv (50.5 man-rem) was also estimated as detailed in Figure 4.

The decontamination activities will generate about 50 tons of secondary wastes and all the dismantling works will generate more than 200 tons of other wastes such as insulation, clothes and so on.

Among the selected decontamination techniques, the use of ultrasounds in aggressive chemicals did not appear to be fully developed. In order to acquire more knowledge in this field, a specific experimental research programme was started. This programme deals with the dismantling and decontamination of the tube bundle of a feed-water preheater[3].

Moreover, in order to reduce as low as possible the quantity of materials not to be decontaminated, some experiments were carried out on chemical decontamination of small valves[3].

DEVELOPMENT OF THE AGGRESSIVE CHEMICAL WITH ULTRASOUND DECONTAMINATION TECHNIQUE

Foreword

The tube-bundles of feedwater preheaters are characterized by very high contaminated surfaces (both ID and OD) and relatively low weights. As a consequence direct melting could be considered only for disposal and not for release.

Among decontamination techniques the use of aggressive chemicals could meet the objective of unrestricted release. Nevertheless direct (dynamic) chemical decontamination needed too large solution volumes. Therefore chemical decontamination in a tank after dismantling the tube bundle was considered.

Taking into account the geometric features of the bundle pin-tubes and previous results from aggressive chemical decontamination experiments, the use of ultrasounds in connection with chemicals was considered in order to enhance the effectiveness of the process.

In this context the experimental research programme was outlined as

follow:
- lab tests on pin-tube specimens in order to select operative conditions;
- in scale testing on small tube assemblies;
- selection of procedures for residual radioactivity measurements;
- total dismantling and decontamination.

Actually the lab tests are concluded, meanwhile the scale tests and the selection of radioactivity measurement procedure are in progress.

For the experimental activity the preheater No. 4, which was operated at the highest temperature and pressure, was selected.

The bundle has tubes with 16 mm outside diameter in AISI Type 304; the full length of tube in the preheater is about 12000 m. The tubes are covered with an adherent contaminated oxide layer both inside and outside. The contamination was measured to be about 60 Bq/cm^2 of 60-Co on the inside surface and about 5 Bq/cm^2 (90% 60-Co and 10% 137-Cs on the outside surface). The contaminated oxide layers are mainly composed of magnetite and hematite with traces of Ni, Cr and Cu[4]. Some SEM images are given in Figure 5.

Laboratory testing

The lab tests were performed to optimize the process parameters such as the kind and concentrations of aggressive chemicals, temperature, time, ultrasonic frequency and power.

Concerning aggressive chemicals the use of strong acids such as HCl, HF, HNO_3 and their mixtures was considered on the basis of previous experimental works on hard chemical decontamination[5,6,7,8]. The concentration of chemical ranges from 2 to 10% vol.

The laboratory tests were carried out at the ENEL/DECO laboratory using 3 cm long specimens cut from the tubes; some more tests were performed using 40 cm long tubes.

To perform the experiments, the tube specimens were immersed in a thin beaker (or in a proper tank) with the aggressive chemical solution. The beaker (or the tank) was then put in the ultrasonic tank which was filled with demineralized water.

The decontamination effectiveness was commonly evaluated by measuring the 60-Co radioactivity before, during (as steps), and after the tests by Na-I detector.

Before testing real specimens of tube-bundle, several preliminary tests were performed with aluminium and steel specimens in order to check the best position in the ultrasonic tank, and the power and frequency of the ultrasonic transducer. As a result, the following experiments were carried out with the specimens in the centre of the tank and with 20 W/l of ultrasonic power at frequency 20/40 kHz.

The series of tests on 3 cm long tube specimens were carried out in the following conditions:
- 2 temperatures: 40 (low) and 70-80°C (high);
- 2 aggressive solutions: HCl and HF/HNO_3;

- presence or absence of ultrasounds.

Moreover some reference tests were made in demineralized water with ultrasounds.

Some more tests were performed on tube specimens after the total decontamination of the outside surface by electropolishing, in order to check the process effectiveness on the inside surface.

The tests at low temperature were started at room temperature. Nevertheless, after about 1 h, the solution reached 40°C because of the action of ultrasounds. In the tests at 70°C the solution was previously heated by a common electric immersion heater which allowed control of the test temperature thoughout the test.

Some results are presented in Figures 6-7. They show the presence of a synergic effect between aggressive chemicals and ultrasound. In particular the weight losses (ΔP) and the decontamination effectiveness (DF) can be expressed by:

$$\Delta P = \Delta P_u + \Delta P_c + \Delta P_s$$
$$DF = DF_u \; DF_c \; DF_s$$

where u is the ultrasonic effect in water, c is the aggressive chemical effect without ultrasounds and s is the synergic effect between chemicals and ultrasounds. ΔP_s and DF_s are function of many parameters such as kind and nature of oxide layers, temperature, time and so on.

The synergic effect on tube specimens is more evident in the HF/HNO_3 solution.

During these tests, information on the contaminated oxide was also recorded. It appeared to be composed of two separate layers: the former, red in appearance, is dishomogeneous and inconsistent, it is easy to remove (ultrasound in water can take it off) and it retains about 80-90% of the radioactivity; the latter, black in appearance, is more compact, tenacious and difficult to remove. Figure 8 shows the 60-Co and 137-Cs radioactivities as a function of the thickness removed.

Three more tests were performed on 40 cm long tube specimens using one tube specimens in each test.

In order to ensure renewal of the solution the chemical solution was circulated slowly by a small pump.

The results show that the HF/HNO_3 solution is most effective in removing the contamination: at the end of the test the tube treated with HF/HNO_3 solution appears to be clean and shiny, while the tube treated with HCl solution appears to be still covered with the oxide.

A single test was also carried out on several specimens, about 10 cm long, loaded in bulk using a HF/HNO_3 solution at room temperature for 240 min; the solution was not recirculated.

The results show tube specimens with different behaviours. The removed radioactivity ranges from 96 to 100%. Looking at the specimens, some are clean and shiny, others are still black and oxidized on the entire surface and others are partly covered with oxide.

At the end of these lab tests, it was decided to use the solution HF/HNO_3 with ultrasound at low temperature for in scale tests (even if other conditions were shown to be effective). On making this choice it is always possible to improve the decontamination effectiveness by increasing the temperature (from low to high) and/or the concentration of chemicals.

In scale test

After the laboratory testing a preliminary test on an assembly of 20 tubes about 40 cm long was performed in the hot chemical laboratory of the BWR Garigliano power station using a special rig.

The test was performed in the following conditions: volume of solution: 27.5 litres; total surface: 72.8 dm^2; chemical solution : 3% vol. HF+10% vol. HNO_3 for 5.5 h and: 5% vol. HF+10% vol. HNO_3 for 1 more h. The test started at room temperature; nevertheless because of the effects of ultrasound and pump, the temperature increased to 40°C in 3 h.

The decision to add some more 2% vol. HF was taken during the test when a visual check of the assembly showed that some tubes were not clean after 5.5 h of test time.

Looking at the tube assembly after the test, some tubes appeared to be still partially dark, covered with a black oxide, on the external surface. The weight losses and the average arithmetic values of the removed radioactivity for each tube, are given in Figures 9-10.

A preliminary analysis of this results shows that:

- the tube assembly is partially contaminated; the gross DF is about 120 as arithmetic average and about 30 as harmonic average;
- only some tubes of the assembly appear to be totally cleaned: the position in the tank seems too have some effect;
- the increase of the solution aggressiveness by adding more HF did not appear to have any appreviable effect.

At the end of the test, the spent decontaminant solution was discharged and neutralized up to pH 8 by adding about 4.5 kg of NaOH. The residual sludges recovered on the conical bottom of the tank were discharged after separation from the floating solution. The volume of the sludges was about 3 l with a radioactivity of about 40 kBq/l (about 100% 60-Co). The volume of the floating solution was 35 l with a radioactivity of 0.4 kBq/l (85% 60-Co and 15% 137-Cs).

Residual radioactivity measurements

Direct gamma spectrometry was selected as base techniques for measuring the residual radioactivity on the decontaminated tubes. Gamma counts were performed on each test specimen at the end of the test and compared with the measurement before testing. Usually the measurements were carried out by Na-I detectors and referred to 60-Co.

In order to validate these measurements specific gamma spectrometries were performed. The measurements on 40 cm long tubes were performed using a

large cylindrical (5"x4") Na-I detector connected to a multichannel analyzer. It was located in a shielded box commonly used for Whole Body Counter measurements and it has a very low background level (2.5 nSv/h).

To optimize the system, several measurement geometries were checked. The selected geometry considers the tubes located vertically around the detector at a distance of about 5 cm. This geometry has the highest sensitivity and it is axysymmetric allowing us to measure from 1 to 24 tubes with the same calibration.

The system was calibrated using a standard 60-Co source to mark the inside surface of tubes similar to the tubes to be measured both for dimension and material.

As a preliminary conclusion, this system configuration allows us to measure the residual radioactivity on a number of 40 cm long tubes from 1 to 24 in a few minutes (5-10 min) with a detection limit much lesser than 1 Bq/cm^2; in the worst case of a single tube the detection limit is about 0.1 Bq/cm^2 and 0.1 Bq/g (Figure 11).

Some views of the radioactivity measurement system are given in Figures 12-13.

Concerning the more general problem of radioactivity measurements a specific program to determine low beta and X emitters such as 55-Fe, 59-Ni and 63-Ni in the oxide contamination layer, before and after decontamination, was started. Preliminary results show, on the inside surface of the tubes before decontamination, the presence of about 20 Bq/cm^2 of 55-Fe and 10 Bq/cm^2 of 63-Ni (60-Co is 36 Bq/cm^2); on the outside surface of the tubes, before decontamination 0.6 Bq/cm^2 of 63-Ni were measured. All measurements after decontamination were less than 0.1 Bq/cm^2.

OTHER DECONTAMINATION STUDIES

Aggressive chemical decontamination of small valves

In order to check the effectiveness of direct chemical decontamination on small and complex components, actually considered for storage without decontamination (3.4% wt. in Fig. 1), some tests were performed on the DECO experimental loop. Four small stainless steel valves from the primary system of Garigliano BWR were decontaminated using mainly aggressive chemicals such as HCl, HF, HNO_3 and their mixtures. In two valves before aggressive chemicals a step with soft chemical (oxalic and citric acid mixture) was performed in order to see whether a softening action can help the following aggressive decontamination.

The results are summarized in Table II. They allow the following considerations:

- before to take the decision whether to decontaminate or not, for decommissioning needed, a small and complicate component or parts, is absolutely needed to establish clearly how to measure the residual radioactivity after treatment;
- using aggressive chemicals, the final residual radioactivity is quite

independent of the initial dose rate level;
- whatever the decontamination process may b some hote spots remain in the valve (or components); these hot spot will be surely greater than 1 Bq/g or 1 Bq/cm^2;
- when the decontamination is performed, it is very difficult to establish "a priori" if the valve (or component) can be released or not; valves (or components) which are more simpler or covered with loose contamination will more probably be totally decontaminated than valves (or components) which are more complex or covered with thick and tenacious contamination.

Treatment of spent waste solution arising from aggressive chemical decontamination

Many batch tests were performed in order to establish the experimental ranges of precipitation of Co and Fe in varying the nature of the spent solution ($HF+HNO_3$, $HCl+HNO_3$ and HCl), the initial concentration of iron and/or cobalt, the kind of neutralization agent, presence of flocculant and so on.

As a conclusion, spent radwaste solutions in aggressive mineral acids coming from surface decontamination can easily be treated for volume reduction by means of the simple chemical process of neutralization with NaOH and/or CaO. Improvement of the purification was noticed by adding chitosan to the early radwaste solution[9]. The radioactivity of the treated spent rad solution was less than or around 0.1 Bq/cm^3 [10].

The residual sludges arising from neutralization and precipitation were also characterized. The main results are:
- the residual sludges retain more than 90% of water;
- three phases are usually present in the dry residual: hydroxides such as $Fe(OH)_3$ and so on, salts and particulate oxide matters;
- the radioactivity of course is retained only in hydroxide and particulate phases.

Chemical decontamination tests with more recent processes

A specific qualification program using chemical decontamination in operating BWR plants to save man-rem, was started some years ago with reference to Caorso BWR power plant. Samples arising from the primary system of Garigliano BWR were used for the experiments.

Althought the main purpose of this program is the evaluation of material compatibility with reference to stress corrosion cracking, the decontamination effectiveness measured on Garigliano samples with different processes is reviewed.

The decontamination tests were performed in DECO loop with the following peocesses: Can-Decon, LOMI, Diluite Citrox and CORD/OZOX [11,12]. The main test conditions and results are summarized in Table III.

Looking at these data a large range of DF_s can be seen. Apart from the intrinsic differences on each process, some variations of DF_s are related to the differences in the nature of contaminated oxide layer.

A correlation between decontamination and corrosion appears quite obvious in the sense that as effective is the process as the corrosion can occur.

These chemical processes do not allow surely to reach the unrestricted release of materials, nevertheless they could be same very interesting in decommissioning activities mainly for reducing radioactivity levels before dismantling works.

CONCLUSION

Studies related to Garigliano BWR decommissioning show that releasing the turbine building components decontamination is an interesting possibility; four decontamination techniques would need: electropolishing, water jet with abrasives, ultrasounds in aggressive chemical and freon jet.

Direct melting for release or reuse in the nuclear field is another alternative.

The preheater tube bundles are a particular case and for their release decontamination is the only possible way. The experimental tests on the aggressive chemical with ultrasound decontamination process on tube bundle samples show that this technique is very promising.although more experiments are needed.

The experimental tests performed on small and complex components (valves), by aggressive chemical decontamination, show that some hot spots always remain on the decontaminated parts. Therefore it is very difficult to establish "a priori" if the component or part can be release or not.

Spent waste solutions from aggressive chemical decontamination can easily be treated by a chemical neutralization and precipitation. Purification can be improved by adding chitosan powder.

The lab testing performed on the soft chemical decontamination processes such as Can-Decon, LOMI, Diluite-Citrox and CORD/OZOX, show that they do not allow us to reach unrestricted release although they could nevertheless be very interesting in decommissioning activities.

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Table I - Inventory of components in the Garigliano BWR turbine building.

Components	Weight (ton)	Contaminated surface (m^2)	Radioactivity (GBq)
Pipes and valves	260	2500	19.2
Pumps	80		0.4
Heat-exchange	180	12500	5.2
Tanks	40	500	1.1
Condenser	460	16500	10.0
Turbine	910	1000	1.8
Total	1930	33000	37.7

Kind of material	Weight (tons)
Carbon and low alloyed steels	1510
IP 304 stainless steel	150
13 % Cr steel	180
Copper alloy	115
stellite	5

Table II - Aggressive chemical decontamination tests on valves from Garigliano BWR.

Valve No.	Type	DN (*)	Weight (kg)	Surface inside (dm^2)	Surface total (dm^2)	Initial dose (mSv/h)	Decontamination process (*)	Residual Radioactivity (kBq) Co-60	Residual Radioactivity (kBq) Cs-137	Specific Residual Radioactivity Weight (Bq/g)	Specific Residual Radioactivity Total surface (Bq/cm^2)
1	GLOBE	1	10.5	12.5	21	0.5	US + AC1 + US + BR	4.5	0.2	0.44	2.2
2	Y-TYPE	1 ½	25.5	7.0	19	2.0	US + AC2 + US + EL	85.9	0.9	3.41	45.8
3	GLOBE	1	10.2	12.5	21	10.0	US + WL + SC + AC2	2.8	1.5	0.41	2.0
4	Y-TYPE	1 ½	24.5	7.0	19	10.0	US + SC + AC2	50.5	1.9	2.08	26.8

(*) Decontamination steps:

- US = ultrasounds in demineralized water: 20-25 kHz; 15 w/l; 10-70 min; 30°C.
- AC1 = aggressive chemicals: HCl 4.1% vol.; 6h; 40°C.
- AC2 = aggressive chemicals: HF (0.75-1.5 % vol.) + HNO_3 (2.5-5 % vol.); 8-10 h; 40-80°C.
- SC = soft chemicals: OX 0.125 % wt + CIT 0.125 % wt; 24 h; 80-110°C.
- WL = water pressure: 4 kg/cm^2; 10 min.
- BR = brushing (outside surface); with detergents.
- EL = electropolishing: 75% H_3PO_4, 0.1 mA/cm^2; 9 h.

Table III - Chemical decontamination tests with more recent processes on contaminated AISI 304 type samples from Garigliano BWR.

Decont. process	Test conditions	Decontamination Factors (*) A	B	C	Decontamination Factors after 5 min ultrasounds in demineralized water A	B	C
CAN-DECON	0.1%, 95°C, 0.2-3 m/s, 25 h	1.6-1.8	-	1.9	1.9-2.0	-	4.1
	0.2%, 115°C, 0.2-3 m/s , 24 h	7.9	-	8-53	8.4	-	27- >100
DILUITE CITROX	0.125% CIT + 0.125% OX, 80°C, 1 m/s, 24 h	2-2.9	9.7	-	-	-	-
LOMI	0.006 M/l V II + 0.018 M/l Formate + Form. acid	1.4	1.1	1.3	1.8	1.1	1.7
	+ 0.036 M/l Pic. acid, 90°C, 0.3-4 m/s, 3h	1.1-1.4	2-2.9	-	1.3-1.5	2.5-7	-
CORD/OZOX	$HMnO_4$ 200 ppm, OXALIC 2000 ppm, 90°C, 3 steps, 24 h, 0.2-4 m/s	4.1-5.7	13.3- >100	>100	35- >100	38- >100	>100

(*) Decontamination Factor as ratio between initial and final counters with Na-I or Ge detector.

A : samples from 24" primary recirculation pipe.
B : samples from separation baffle of channel head of SSG.
C : samples from auxiliary lines of primary system.

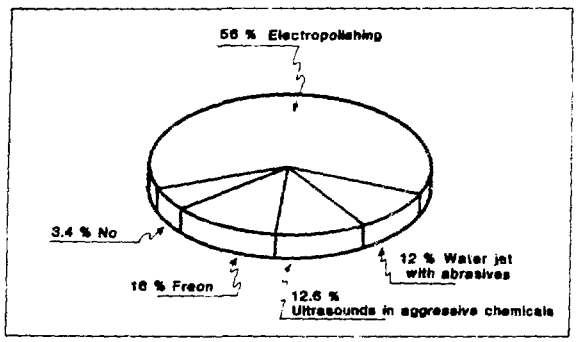

FIG. 1 - FEASIBILITY STUDY ON DISMANTLING AND DECONTAMINATION OF TURBINE BUILDING COMPONENTS OF GARIGLIANO BWR : % OF DECONTAMINATION TECHNIQUES IN TERM OF WEIGHT (TOTAL WEIGHT 1932 TONS)

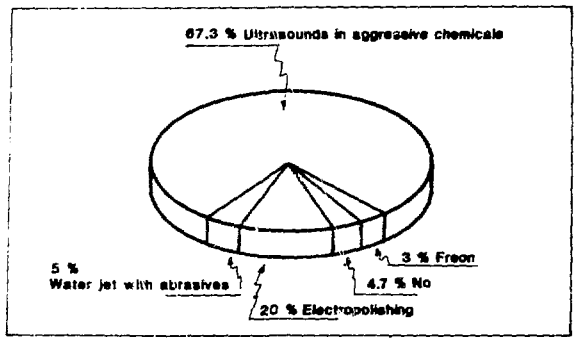

FIG. 2 - FEASIBILITY STUDY ON DISMANTLING AND DECONTAMINATION OF TURBINE BUILDING COMPONENTS OF GARIGLIANO BWR : % OF DECONTAMINATION TECHNIQUES IN TERM OF CONTAMINATED SURFACE (TOTAL SURFACE 33072 m^2)

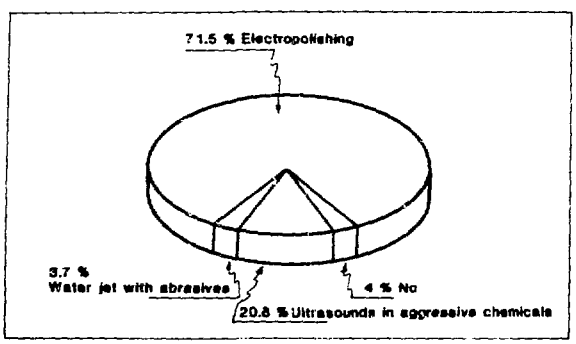

FIG. 3 - FEASIBILITY STUDY ON DISMANTLING AND DECONTAMINATION OF TURBINE BUILDING COMPONENTS OF GARIGLIANO BWR : % OF DECONTAMINATION TECHNIQUES IN TERM OF RADIOACTIVITY (TOTAL RADIOACTIVITY 1021 mCi, FREON<0.2 %)

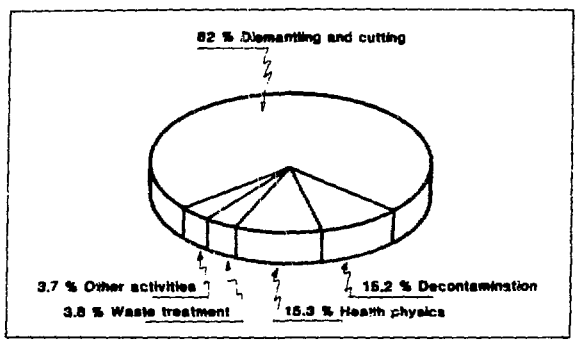

FIG. 4 - FEASIBILITY STUDY ON DISMANTLING AND DECONTAMINATION OF TURBINE BUILDING COMPONENTS OF GARIGLIANO BWR : MAN-REM DISTRIBUTION IN EACH WORK ACTIVITY % OF A TOTAL OCCUPATIONAL DOSE OF 50.5 MAN-REM

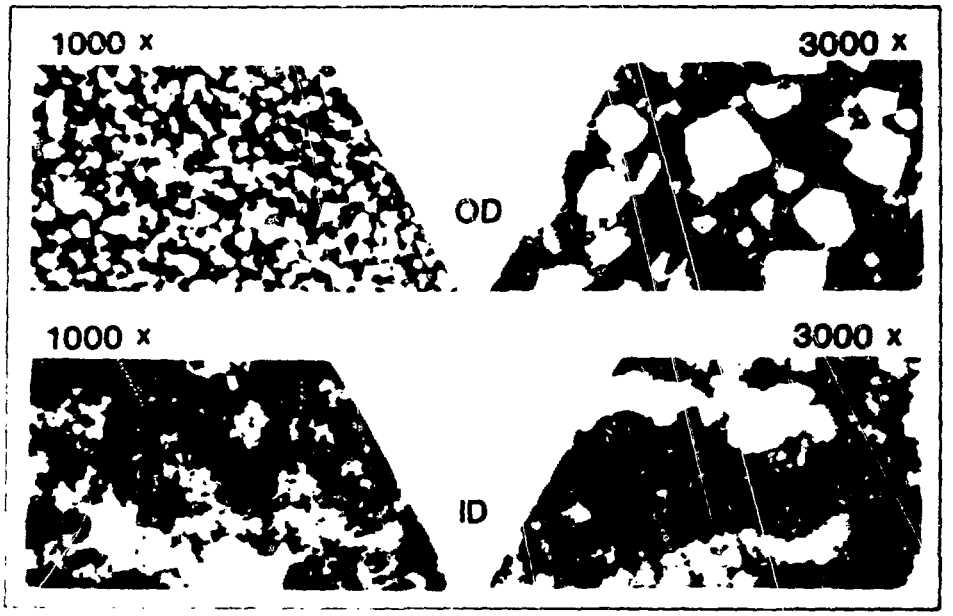

FIG. 5 - SEM VIEWS OF THE CONTAMINATED SURFACE OF A TUBE OF THE PREHEATER No. 4 OF GARIGLIANO BWR

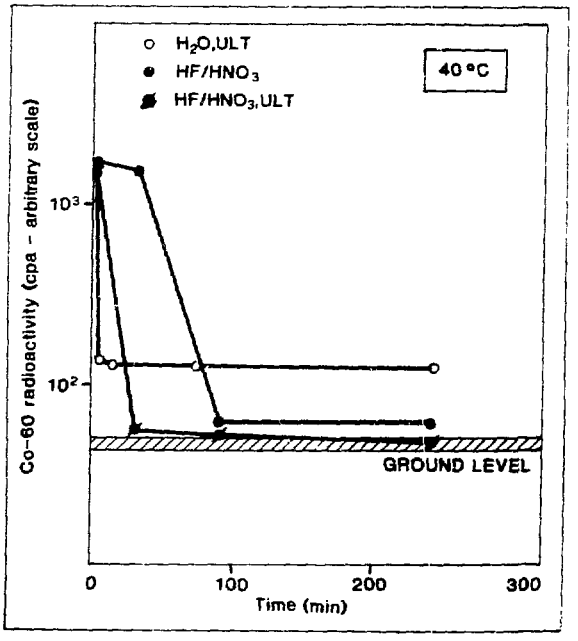

FIG. 6 - 60-Co RADIOACTIVITY VERSUS TIME IN TESTS WITH ULTRASOUNDS AND HF/HNO_3 SOLUTION AT 40 °C

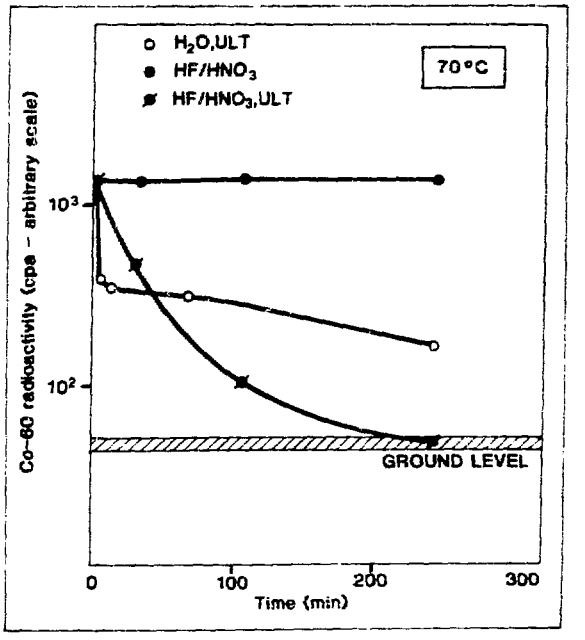

FIG. 7 - 60-Co RADIOACTIVITY VERSUS TIME IN TESTS WITH ULTRASOUNDS AND HF/HNO_3 SOLUTION AT 70° C

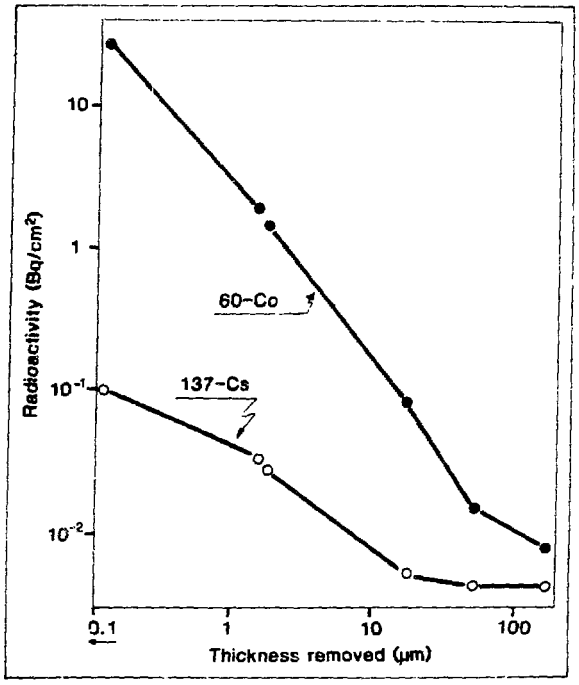

FIG. 8 - 60-Co AND 137-Cs RESIDUAL RADIOACTIVITY AS A FUNCTION OF THE THICKNESS REMOVED ON THE INSIDE SURFACE

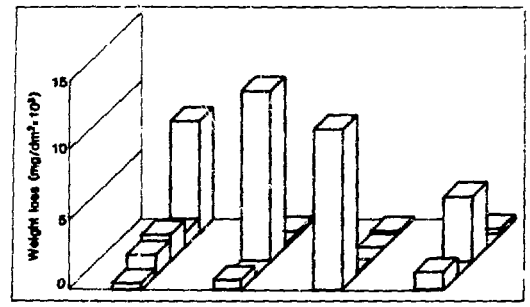

FIG. 9 - WEIGHT LOSSES OF TUBES USED IN THE PRELIMINARY IN-SCALE TEST

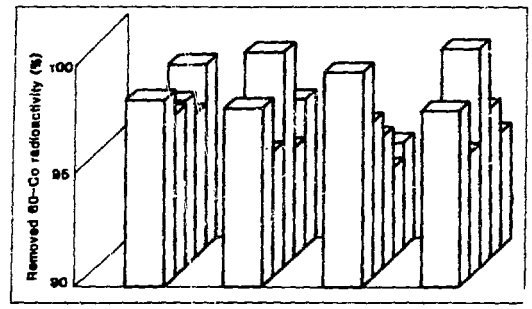

FIG. 10 - REMOVED 60-Co RADIOACTIVITY ON TUBES USED IN THE PRELIMINARY IN-SCALE TEST

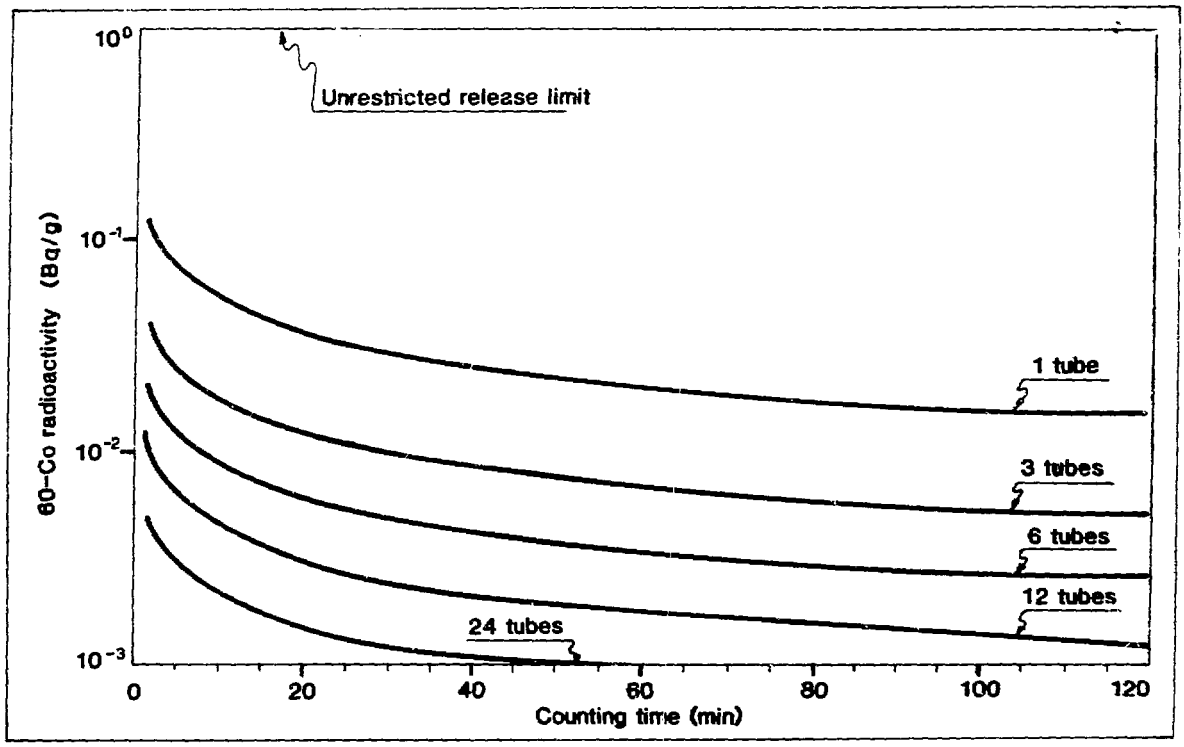

FIG. 11 - MINIMUM DETECTABLE RADIOACTIVITY VALUES AS A FUNCTION OF COUNTING TIME (t) AND NUMBER OF TUBES (N)

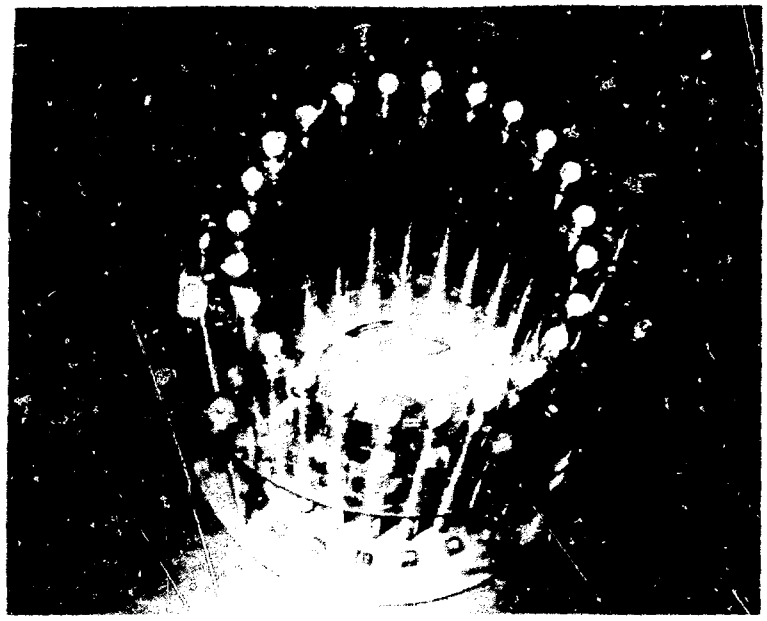

FIG. 12 – VIEW OF THE SUPPORT OF THE TUBES FOR THE RADIOMETRIC MEASUREMENT SYSTEM

FIG. 13 – VIEW OF THE RADIOMETRIC MEASUREMENT SYSTEM

TMI-2 CLEANUP TECHNIQUES APPLICABLE TO DECOMMISSIONING

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ABSTRACT

This paper discusses techniques and equipment used for cleanup at Three Mile Island Unit 2 (TMI-2) that could be applied directly for decommissioning any chemical or nuclear facility. The technology discussed concentrates on those approaches that either substantially reduce exposure to the work force engaged in decommissioning or those actions that result in volume reduction of contaminated materials. The key issues presented are surface decontamination, removal of contaminated surface material, internal pipe cleaning, remote tools, and the workhorse robot.

Acknowledgement

GPU Nuclear Corporation (GPUNC), the Electric Power Research Institute (EPRI), the U.S. Nuclear Regulatory Commission (NRC), and the U.S. Department of Energy (DOE) entered into a joint research agreement to transfer technological information from the Three Mile Island Unit 2 (TMI-2) cleanup to the utility industry. This paper reflects the collective efforts of the above, with assistance from Bechtel, EG&G Idaho, and other onsite contractors. In addition, Carnegie-Mellon University and the Ben Franklin Partnership made significant contributions to the development of the remote equipment discussed herein. Finally, at TMI-2 Paul R. Bengel, Michael D. Pavelek, and David L. Giefer have been significantly responsible for developing and implementing the technologies discussed in this paper.

INTRODUCTION

The March 28, 1979 accident at TMI-2 contaminated the reactor building with fission products to such an extent that no entries were possible until 1 1/2 years after the accident. Even today, the basement level is only accessible to remote operated equipment.

As the accident was brought under control, and the plant was returned to cold shutdown, experts convened at Three Mile Island to determine the cause of the accident and to assist in the accident cleanup. Now, eight years after the accident, the reactor building

is readily accessible in the upper levels, and the damaged core is being removed. While it was predicted in 1983 that it would take between 13,000 and 46,000 person rem to accomplish this task, the actual exposure is currently expected to be less than 8000 person rem. Total exposure to date is approximately 3800 person rem with over half the expected exposure having been received.

The cleanup at TMI-2 has been accomplished with a joint agreement between GPU Nuclear, the Department of Energy, and the Electric Power Research Institute to assist in the research and development areas. The accident itself and ensuing damage were carefully studied, and the most effective tools and techniques were developed for cleanup. While the effort at TMI-2 is not aimed at decommissioning, many of these cleanup techniques are directly applicable to activities associated with cleaning a plant suitably for its decommissioning.

SURFACE DECONTAMINATION

The accident had released 200,000 gallons of contaminated water from the reactor coolant system, which carried fission products to all areas of the reactor building. The initial effort in cleanup was aimed at washing down the resultant residue from manned areas to the basement. A combination low- and high-pressure spray was used for maximum effectiveness. Squeegee brooms were used to continually move the water/debris from cleared areas to minimize recontamination. The water source was accident-generated water which had been removed from the basement and filtered.

The spraying was a manual operation using standard, commercially available equipment. The filter process for the water supply was developed at TMI-2. This system, called the submerged demineralizer system, used zeolite resins which underwent testing to determine effectiveness and disposability.

The initial use of water spray achieved a dose rate reduction of approximately 50%. The reason for this limited effectiveness was because much of the contamination had penetrated into the paint and concrete and was bonded to the surface. The water spray was effective as the first step in reducing the manrem even though it did not achieve desired cleanup. Temperature of the water spray was not a factor.

Flushing the walls required less water flow rate and pressure than flushing the floors, and the overall approach had to be planned to minimize recontamination. Final disposition of the dirt and contamination that was flushed to low points was to collect it with a wet, vacuum-type system and separate out the solids and contaminants for disposal.

With the introduction of the teleoperated mobile robot, ROVER, water spraying is now conducted as a remote, unmanned operation. The use of a water spray continues as part of surface decontamination just as it was introduced in 1980.

High-Pressure Water Flush. Water at a flow rate of 18 gal/min and nozzle pressure of 6,000 psi was found to be effective in covering large areas and in moving heavy contamination to pickup points. Directing the water was difficult at higher flow rates, and over spray tended to redistribute contaminants to clean areas.

High-Pressure Water Jet. Water at a flow rate of 7-12 gal/min and a nozzle pressure of 3,000-6,000 psi was found to be effective at dislodging surface contamination on complex surfaces. This tool had to be used cautiously because the water jet could remove surface material at close distances.

Ultra-high pressure sprays (to 35,000 psig) were effective in removing contaminated paint from surfaces in most cases, including epoxy-based paints. However, it should be noted that the enamel-based paint on the manway cover of a steam generator could not be removed in this way.

Mechanical Scrubbing. Industrial-type scrubbers removed a small amount of the surface when operated normally in conjunction with a water flush. This method was very effective, especially on painted surfaces.

Steam Jet and Vacuuming. Water at 330°F was sprayed onto the surface under a local vacuum of approximately 16.7-in. hg. using a Vacu-mac^{tm}. This was especially effective in removing layered contamination bound by films of oil, etc. and was particularly useful on cable trays, deck grating, and other complex surfaces where acceptable alternatives did not exist. An important benefit of this system was that it used a minimum of water and collected it in a single vessel.

Water Scrubber. A 15° rotating water jet floor scrubber at a pressure of 6,000 psi and at flow rates of 15 to 18 gpm was used on horizontal surfaces. Recontamination was a serious problem when using this method.

Strippable Coatings. Strippable coatings (Imperial ALARA DECON 1140) were used on floors and equipment. These were effective in removing surface contamination and in temporarily protecting clean surfaces from being recontaminated. This was one of the fastest and least labor intensive decontaminaton methods, although it contributes to the waste bulk. It has an added advantage over water

in that no secondary processing is required prior to disposal. Best results were obtained by spraying the coatings onto the surfaces.

A self-stripping, non-toxic copolymer was also tested on heavily corroded reactor vessel guide studs and found to be effective in removing oxides and loose surface contamination. Two applications reduced surface contamination from 260,000 dpm to less than 500 dpm essentially releaseable as no longer contaminated. The copolymer flakes off the surface as it drys trapping the rust and contamination within the material's matrix.

SURFACE REMOVAL

A significant amount of the contamination from the accident penetrated porous, unpainted surfaces or bonded to the paint or metallic surfaces. This required removal of some portions of the surface, varying from the paint layer or the metallic oxide film, to fractions of an inch of concrete.

Many different techniques were evaluated, including liquid abrasive blasting, chemical treatment, scabbling, ultra-high pressure water, and vacuum blasting. Important criteria in evaluating the surface removal techniques were the minimization of airborne contamination or added waste and the removal efficiency.

Abrasive Blasting and Grinding. These conventional methods were discounted early in the evaluation process because they tended to create airborne contamination and/or added to the waste disposal problem.

Mechanical Scabbling. Controlled impact breakdown of concrete surface using reciprocating pistons is an industrial technique for preparing concrete for resurfacing. A standard commercial, manually operated system was evaluated for removing contaminated surface material from floors early in the cleanup with very good results. Later, EPRI developed a remote operated 7-piston scabbler with a vacuum system mounted on a remote operated vehicle, identified as the MOOSE to eliminate operator exposure and fatigue. The MOOSE can scabble up to 400 square feet of concrete surface per hour, removing 1/16 inch of the surface. Although not completed in time to contribute directly to TMI-2 cleanup, this has been used successfully at other nuclear plants. A 3-piston head scabbler was built on the LOUIE-2 remote operated vehicle and has been successfully used in limited access areas of TMI-2. Mechanical scabbling has proven to be very effective in removing the necessary amount of surface to achieve the desired decontamination, minimizing the waste disposal problem. The reacting forces created by the mechanical operation make it difficult to use on vertical surfaces

and the inherent size of the vehicle restricts its use to fairly open areas. Hand tools are employed for smaller areas and trim work.

Hydro Scabbling (Scarification). Ultra-high pressure water jets were used to remove contaminated surface from vertical walls. Pressures up to 36,000 psig were used with both a single jet for detail scarifying or rotating jets for wide area coverage. The actual gun weighed only 13 pounds and could be operated manually, with a remote manipulator or with a robot.

The water jet erodes the surface away and its effectiveness is dependent on surface hardness, pressure, flow rate, stand-off distance, and rate of movement.

The hydro scarifier offers the advantages of low reaction force, small light-weight tool, and relatively high production rate. The disadvantages are the spread of contamination, the need to use another process to pick up and dispose of the waste, and the difficulty in controlling the actual desired amount of surface removal, which is affected by the variation in hardness of the different materials and aggregate of concrete.

The residue from the water scabbling was removed by water flush and vacuuming and subsequently separated from the water and disposed as waste.

INTERNAL PIPE CLEANING

Mechanical and hydro non-chemical devices were evaluated for cleaning reactor coolant system piping. The evaluation phase consisted of off-site laboratory type testing by EPRI of a wide range of devices and the on-site evaluations of two devices found most suitable for TMI-2.

The basic cleaning problem defined for the laboratory was to remove corrosion layers and loose fuel debris in pipes with 90° elbows varying from 0.75-in ID tubing to 3-in ID stainless steel pipe. The corrosion was simulated with paint and high-temperature oxides; the debris was simulated with tungsten powder. The devices tested varied from water jets to rotating tools to pipes. The criteria for evaluation was cleaning effectiveness, ease of operation, and adverse effect on pipe finish.

As a result of the laboratory evaluation, three devices were selected for evaluation at TMI-2: a rotating brush, FLEX HONE tool, and two rotating water jets, the MOLE NOZZLE and the Cavijet nozzle.

The FLEX HONE used a motor-driven rotating brush assembly of nylon bristles with silicon carbide tips. The tips were available in 2-, 4-, and 6-in diameters. The MOLE NOZZLE used an 8-jet nozzle at 10,000 psig. The cavijet nozzle used a special nozzle operating at 7,000 psig to produce bubbles for cleaning action.

The FLEX HONE and MOLE NOZZLE were initially tested at TMI-2 for effectiveness in cleaning drains which were believed to be substantial radiation sources. The success of the FLEX HONE and MOLE NOZZLE on the drains was inconclusive, principally because the test conditions were considerably different from the original requirements for the reactor coolant piping. The FLEX HONE and MOLE NOZZLE have not been tested on the reactor coolant piping to date.

The ideal use of these tools was to use the MOLE NOZZLE to remove loose debris and then the FLEX HONE to remove corrosion products, followed by the MOLE NOZZLE to flush the pipe again. Soft materials such as paint or PVC tended to clog the FLEX HONE silicon tips.

The cavijet, while not yet tested at TMI, was found to be very effective in removing surface coatings and is still planned for use in the reactor vessel.

REMOTE TOOLS FOR DEFUELING

Removal of the damaged core from the reactor vessel nessitated design, fabrication and testing of a family of tools. These tools consisted of a special purpose end effector attached to a long-handled pole. The end effectors varied from a simple hook to hydraulicly operated shears; a catalog of over one hundred different end effectors was created. The long-handled poles were segmented into 22-ft and 7-ft sections with the longest reach being 36 ft. High-pressure lines were run through the center of the tools.

The tools operated through a slotted, rotating work platform installed above the reactor vessel. This work platform includes power and controls for both the work platform and the tools.

Upon removal of all the loose material from the reactor vessel, ACES, an Automated Cutting Equipment System will be employed to dismantle the lower core support structure and access fuel that has flowed from the core area. ACES is an industrial robot that can manipulate a plasma arc torch in five degrees of freedom to cut stainless steel structures varying from 2.5-cm to 35-cm thick and under 35.4-ft of borated water.

MANFRED, MANipulator For REmote Defueling, is a double-armed remote manipulator that will be used in conjunction with ACES to dismantle the structures in the reactor vessel. A 40-in arm will be used to stabilize the MANFRED, while a 72-in arm will be used for actual dismantling. Position information from both ACES and MANFRED will be fed to a display system for use by the operators for collision avoidance between the two systems.

WORKHORSE

The Workhorse is a remote work vehicle developed by Carnegie-Mellon University specifically for the TMI-2 basement environment. It consists basically of a telescoping boom mounted on a wheeled mobile platform with outriggers for stability. The boom can reach to a height of 23 ft above the floor and includes a family of end effectors for performing work, such as grinding, cutting, etc. Electric motors supply the basis power including a 20-horsepower hydraulic power plant. A manipulator arm, attached to the boom, can be taught to perform tasks, making the Workhorse a true robot.

CONCLUSION

The cleanup techniques developed at TMI-2 are useful for surface decontamination during decommissioning. The benefits derived at TMI-2 have been to:

- Reduce worker exposure
- Reduce the amount of material to be disposed of as contaminated waste
- Reduce the quantity of high-level waste
- Increase the production rates.

The key factors in achieving these benefits are more effective tooling and extending the workers' reach via remotely operated tooling.

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MICROWAVE DECONTAMINATOR FOR CONCRETE SURFACE DECONTAMINATION IN JPDR

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ABSTRACT

Radioactive contamination of nuclear facilities must be evaluated to determine appropriate decommissioning procedures which address both worker-exposure and waste management. The Japan Atomic Energy Research Institute (JAERI) measured concrete surface contamination prior to the dismantlement of the Japan Power Demonstration Reactor (JPDR). Measured penetration-depths of the contamination were within 2cm of the surface.

JAERI has developed a microwave decontaminator for concrete surface removal. It can remove the surface of concrete to a depth of more than 1cm with a single-pass, and be used for walls as well as floors.

Concrete surface contamination of the JPDR will be removed by both the microwave decontaminator and conventional machines to evaluate the usefulness of the microwave decontaminator.

INTRODUCTION

Concrete surface contamination occurs at nuclear facilities from spills and leaks of radioactive liquids during operation.

The decommissioning of a facility produces radioactive wastes which must be treated and disposed of in a controlled manner. The removal of the contaminated concrete surface layer before the demolition of buildings reduces the volume of radioactive waste.

Concrete surface contamination was measured in the JPDR to make a decommissioning plan prior to the dismantlement of the facility. Penetration-depths of contamination were within 1-2cm of the surface. Contamination was found not only on floors but also on walls.

Scabblers and plainers used for construction work are planned to be used for concrete surface decontamination. However, they can not remove a 1cm of thickness of concrete surface with a single-pass. Thus, much time is needed to remove contamination which has penetrated to the depth of 2cm.

Starting in 1983, JAERI has been developing a microwave decontaminator to remove concrete surface contamination.* It can remove more than 1cm of concrete surface in a single-pass. Its usefulness for concrete surface decontamination was confirmed by performance tests.

This presentation will describe the concrete surface contamination of the JPDR, performance of the microwave decontaminator and its applicability.

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* This work was performed by the Japan Atomic Energy Research Institute under contact from the Science and Technology Agency of Japan.

CONCRETE SURFACE CONTAMINATION OF THE JPDR

Concrete surface contamination was surveyed as part of the plan for the dismantlement procedure of the JPDR.

Loose contamination was measured by the smear method, and penetrating contamination was measured by analyzing drilling samples.

Contamination was found in the reactor containment, liquid waste treatment building, turbine building and spent fuel storage building. Radioactivities in other buildings were lower than the background level. The contaminating nuclide was mainly ^{60}Co, but ^{137}Cs was found in limited areas of the liquid waste treatment building, turbine building and spent fuel storage building. Figure 1 shows the distribution of contamination on the first basement of the turbine building. Contamination was widely distributed on the floor, and some was partly found on the walls. Penetration-depths were within 2cm of the surface of the concrete.

Figure 2 shows an example of the contamination penetration depth of the floor of the liquid waste treatment building. Contamination was not found within 5cm of the surface, but more than $10^{-4}\mu Ci/g$ was found at a depth of 6-7cm. A review of the JPDR operation record disclosed that a large spill of radioactive liquid occured, and the contamination had been covered by mortar in order to prevent it from spreading. This indicates that documentation of contamination during operation is very useful for planning reactor decommissioning.

Figure 3 shows contaminated area as a function of penetration-depth. Eighty-three percent of contamination was within 2cm of the surface. Where scratches or cracks in the concrete surface had occurred, the radionuclides penetrated more than 2cm. The deepest penetration depth was 11cm.

MICROWAVE DECONTAMINATOR

Device

Contamination limited to surface coatings can be removed by scabblers and plainers. However, they do not have sufficient performance to remove penetrating contamination. A microwave decontaminator has been developed for removing this type of contamination. It is able to be used on walls and ceilings as well as floors.

Microwave energy irradiated to concrete is absorbed within a few centimeters of the surface of the concrete. The absorbed energy heats water of hydration in concrete. This causes both high steam pressure and subsequent concrete surface removal by a spalling process as shown in Fig.4.

An isometric view of the prototype decontaminator is shown in Fig.5. It consists of three 5kw magnetron units in parallel (① in Fig.5), wave guides (②) and irradiating heads (③). 2450MHz microwaves generated by magnetrons propagate through the wave guide to the head and onto the concrete surface. The head moves at a constant speed for continuous surface removal, and the head can be oriented to walls, ceilings and corners. The standoff (distance between the head and concrete surface) is adjustable by raising or lowering of the head. Concrete debris is collected by the vacuum system attached to the microwave irradiating head.

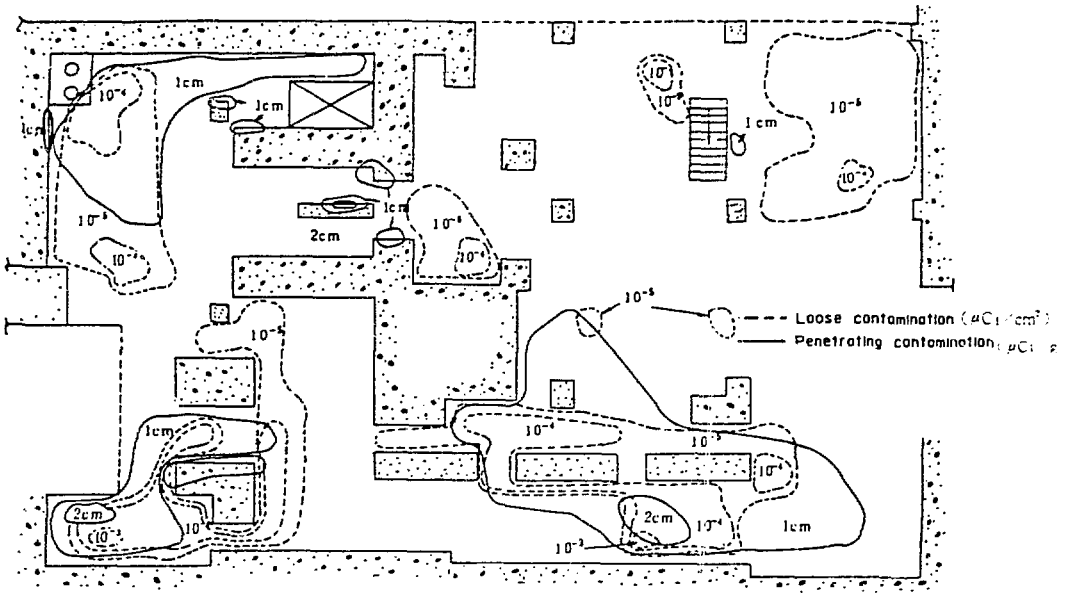

Fig.1 Contour map of radioactive contamination of JPDR turbine building B1F

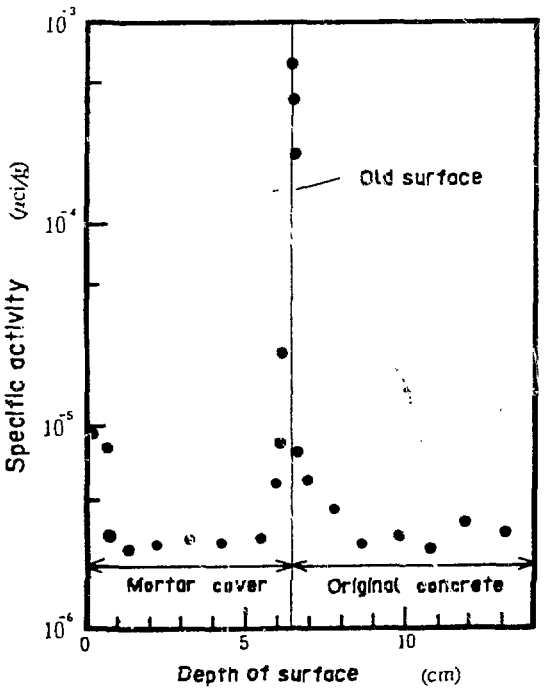

Fig.2 Contamination penetrating depth of JPDR liquid waste treatment building

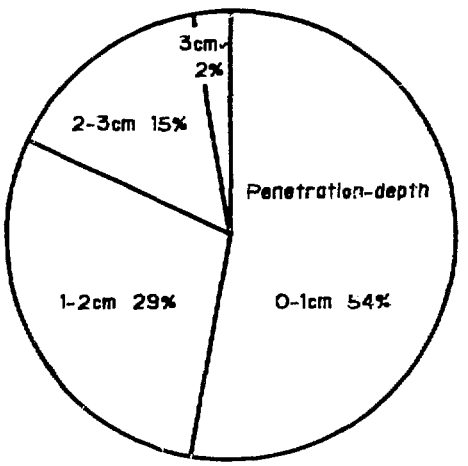

Fig.3 Contaminated area by penetration depth

Performance Test and Discussion

Performance tests were conducted to obtain operational data for the microwave decontaminator. The results were:

Shape of irradiating head. Figure 6 shows shapes of three types of irradiating heads, a spread type, a straight type and a tapered type. It also shows a cross section of the concrete removed by each. The spread type removes the concrete surface deeply and smoothly, compared with the other heads. Therefore, the spread type was chosen as the irradiating head of the microwave decontaminator.

Effect of parallel irradiation. The material volume removed was compared for one, two and three magnetrons. Three units in parallel removed up to 14 times the volume of one unit, and 6 times the volume of two units. This demonstrates the effectiveness of parallel irradiation.

Standoff. The removal depth increases as the standoff decreases as shown in Fig.7. A standoff of 20mm was selected because of good performance and because a concrete surface is usually a little irregular.

Head moving rate. The effect of the head moving rate on removal depth is shown in Fig.8. The removal depth varies inversely as the head moving rate. Contamination partly remains at more than 4mm/sec. Then the head moving rate was determined to be 3mm/sec. Furthermore, overlap of successive scanning passes is necessary as shown in Fig.9(a). Contamination remains without overlap as shown in Fig.9(b).

Dust concentration and size of concrete debris. Controlling dust concentration is important to minimize worker exposure. Dust concentration was $2mg/m^3$ using the vacuum system, and $200mg/m^3$ without the vacuum system at about 2m from the concrete surface. Figure 10 shows a weight distribution of concrete debris by size. About 70% of concrete debris was larger than 5mm in diameter.

Collecting of concrete debris by the vacuum system. At first, 50% of the concrete debris was collected by the vacuum system. But, through the improvement of the shape of the attachment, this was increased to 70%. An increase of the vacuum system capacity should provide more improvement.

Influence of water content in concrete and cracks on concrete surface. Two concrete blocks with different water content were tested to compare the removal depths. Table I shows the composition of the blocks. The removal depth was 22mm for A, and 16mm for B. The water content was 4.3% for A, and 3.5% for B. The removal depth increases in accordance with water content. Cracks also effect removal depth, occasionally having an infavorable effect on spalling because of the escape of steam.

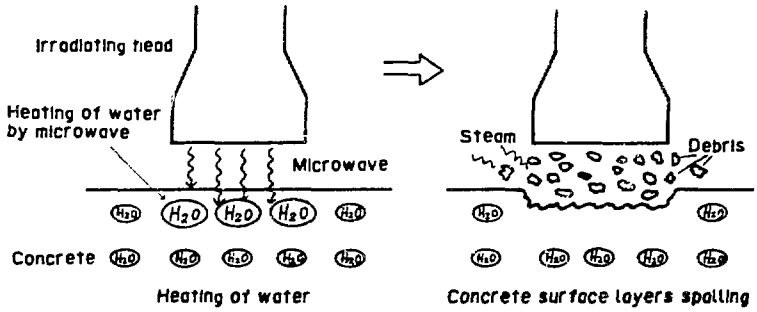

Fig.4 Concrete surface removal mechanism by microwave decontaminator and cross section of irradiated paths in concrete blocks

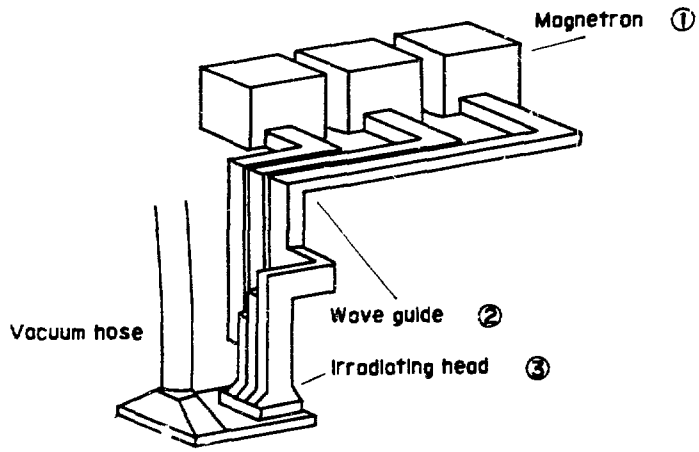

Fig.5 Isometric view of microwave decontaminator

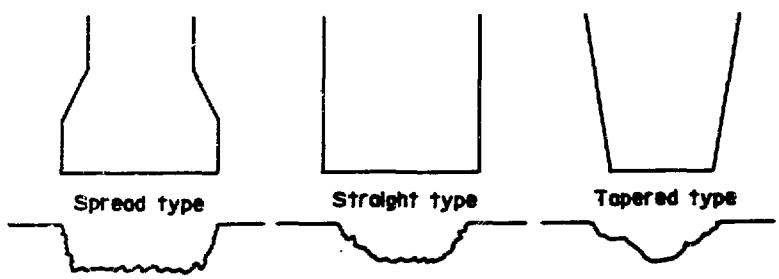

Fig.6 Shapes of microwave irradiating head tested

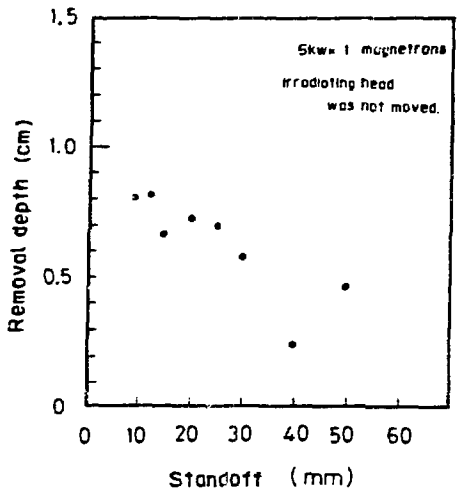

Fig.7 Effect of standoff on removal depth

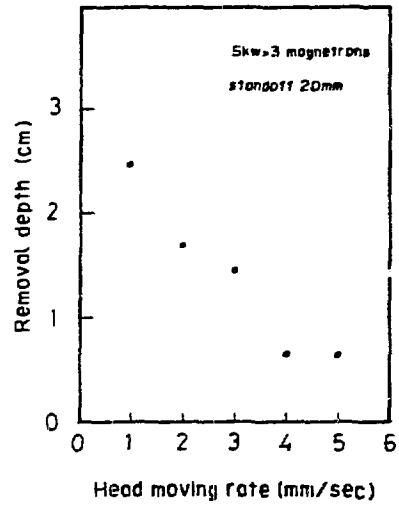

Fig.8 Effect of head moving rate on removal depth

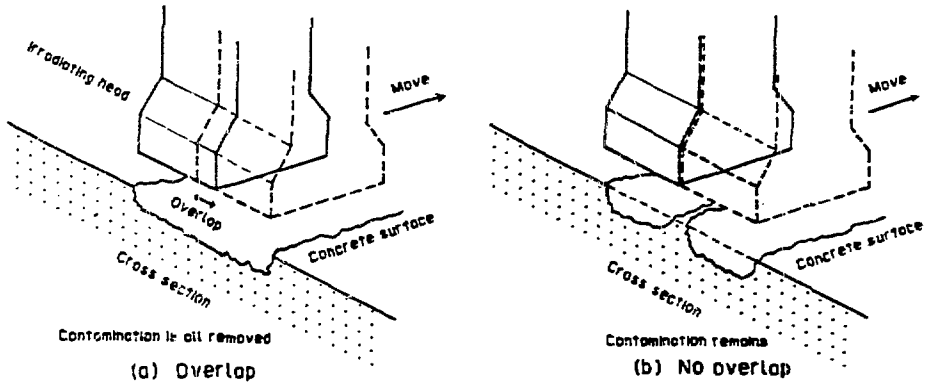

Fig.9 Concrete surface removal with overlap or no overlap

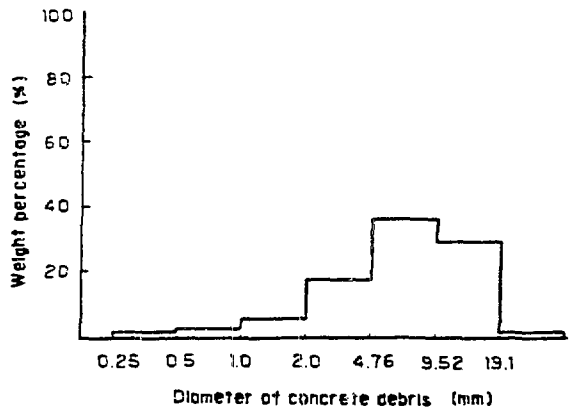

Fig.10 Distribution of concrete debris generated by microwave decontaminator

Effect of coating on concrete surface. The effect of paint coatings was investigated using epoxy and urethane resins. The removal depth was reduced to 80% of concrete without coating. Gases produced were analyzed. Toxic gases were to be undetectable.

Performance in comparison with conventional techniques. The performance of the microwave decontaminator was compared with those of various existing decontamination machines. Their features and results are surmarized in Table II. The removal depth and volume of the microwave decontaminator were larger than those of other machines. The microwave decontaminator was effective for wall decontamination. However it was difficult to use a scabbler and a plainer for a wall, because they were too heavy to be supported. The microwave decontaminator presently does not remove sufficient material from inside corners. A special head will have to be developed. A hand-breaker was effective for walls and corners, though it needs much time to decontaminate large areas. A steel grit blast generated a large quantity of secondary wastes, and work efficiency was small. It is not suitable for decontamination of a large area.

As the prototype microwave decontaminator is larger than other machines, it has to be compact in order to decontaminate in a confined place of a nuclear facility.

Applicability for nuclear facility decontamination. The microwave decontaminator can remove penetrated contamination within 10-20mm of the concrete surface by a single-pass operation, and 20-40mm by a double-pass operation. This will remove most contamination on uncoated concrete surfaces in nuclear reactor facilities. It is also applicable for decontamination of coated concrete floors and walls.

CONCLUSION

A microwave decontaminator is being improved further to make it compact and to equip it with a self driving system and a flexible wave guide for easy operation.

In the JPDR, concrete surface decontamination will start in 1989. The microwave decontaminator and conventional decontamination machines such as scabblers and plainers will be used. Operational data about these decontaminators will be obtained to demonstrate the usefulness of microwave decontaminator for concrete surface decontamination. These data will be useful for decommissioning of nuclear power reactors in future.

ACKNOWLEDGMENTS

The authors sincerely appreciate Mr. M. Hatakeyama and Mr. F. Satoh for evaluating concrete surface contamination of the JPDR. The authors also wish to thank Dr. E. Tachikawa for his helpful suggestions in the development of decontamination technology and to thank Mr. Y. Iwasaki for conducting performance tests of the microwave decontaminator.

Table.] Composition of Concrete Blocks Tested

Mix proportion	A	B
Water	170 kg/m³	139 kg/m³
Portland cement	325 kg/m³	290 kg/m³
Sand	650 kg/m³	784 kg/m³
Aggregate	1200 kg/m³	1205 kg/m³
Water-cement ratio	0.52	0.48
Slump *	18.0 cm	7.5 cm

* Slump is an index of consistency of concrete

Table.] Comparison of Concrete Removal Characteristics among Decontaminators

Tool	Power source	Concrete removal mechanism	Removal depth by single-pass	Removal volume per hour
Microwave decontaminator	Electricity	Spalling caused by steam pressure	15--30mm	$40\times10^{-3}m^3/hr$
Scabbler	Compressed air	Impact of steel head	3 - 7mm	$40\times10^{-3}m^3/hr$
Plainer	Electricity	Removal by cutter blades	2 - 4mm	$30\times10^{-3}m^3/hr$
Hand-breaker	Compressed air	Impact of steel chisel	5 - 10mm *	$20\times10^{-3}m^3/hr$
Steel grit blast	Compressed air	Steel grit blasting	1 - 5mm	$1.0\times10^{-3}m^3/hr$

* Considered to be only a single-pass operation.

DEVELOPMENT OF A PROCESS FOR THE REMOVAL OF RADIOACTIVELY CONTAMINATED COATINGS FROM CONCRETE STRUCTURES, WHEN REPAIRING OR SHUTTING DOWN NUCLEAR PLANT *

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1. ABSTRACT

In nuclear installations, to prevent radioactive contamination of structural and plant components, their surfaces are given protective coatings. The aim of the project was the development and trial of a process for the removal of contaminated coatings.
An experimental plant was built. Coatings, as used in practice, where applied to concrete, cement screed, cement-lime mortar and steel plates.
By application of liquid nitrogen, removal of the coatings was achieved under certain circumstances.

2. INTRODUCTION

Component surfaces, so-called building structures, must be decontaminable within the control area, i.e. **unavoidable radioactive** surface contamination must be at least partly removable through washing processes. The contamination must not result in an irreversible coating of the concrete and steel construction materials. For this reason, the pertinent component surfaces are given special coatings. Typical structures are:

* The research project was sponsored by the Ministery of Research and Technology of the Governement of the Federal Republic of Germany

On concrete

Dispersion improved cement filler, thickness about 500 µm.
Epoxy cover coating, thickness about 100 µm.

On screed surfaces

Epoxy intake primer (non-coating).
Epoxy cover coating, thickness about 1000 µm.

On steel surfaces

Epoxy zinc dust primer, thickness about 50 µm.
Epoxy cover coating, thickness about 50 µm.

On cement-lime plaster surfaces

Epoxy filler, thickness about 500 µm.
Epoxy cover coating, thickness about 100 µm.

If repairs are carried out on these structures, or if a shutdown of the complete plant is under consideration, a partial decontamination of the surfaces will initially take place, where required using chemical-physical cleaning processes. Afterwards, the residually contaminated coating will be removed. After the removal of these radioactive surface coatings, a repair or demolition of the no longer contaminated components can be carried out. Up to now, the following conventional processes are available for coating removal:

- The sand blasting process
- The high-pressure water jet process
- Flame cleaning
- Pickling
- Chiselling off the coated concrete zone close to the surface.

Especially the blasting processes have been known for some considerable time, and are extremely effective removal processes. However, these unavoidably produce dust, aerosols and smoke. In

addition, secondary refuse is produced, such as shotblasting residues, water, etc.. If the processes are used in this case, both primary as well as secondary waste would be radioactively charged, which would make their disposal increasingly difficult. Spreading of these radioactively contaminated particles in the substrate materials concrete, screed, steel and lime-cement plaster, as well as in the air, is unacceptable. The most serious disadvantages of this process have thus been specified.

This was the reason for developing a new removal process, which is not subject to these disadvantages. The requirement specification for such a removal process is:

- The coating must be removable, insofar as it is contaminated radioactively.
- Spreading of radioactively contaminated components, both in building structures as well as in the air must not occur.
- Critical secondary waste, must not occur.
- Disposal of the removed coating must be possible in a simple and safe manner.

3. ACTION PRINCIPLE OF THE REMOVAL PROCESS

If materials are strongly cooled, they contract and become brittle. This rule can be utilized under certain conditions for the removal of coatings. Fig. 1 shows a concrete beam coated on one side with a uniform polymer coating. If this compound structure is cooled, due to the material specific expansion behaviour, different

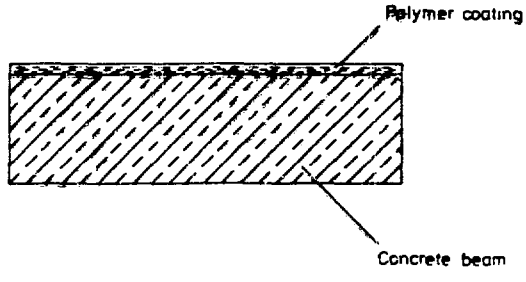


Fig 1 Coated concrete beam

negative expansions in both layers are produced thus generating constrained stresses. These stresses can be accurately stated by means of Hooke's law:

$$\sigma = E \cdot \varepsilon \qquad (1)$$

With the compound structure under consideration, the E module of the polymer coating is larger or smaller than the concrete substrate. Normally, it will be slightly smaller. However, it does not have a dominant effect in our deliberations regarding the intended stress development. By comparison, the expansion dimension, is of major importance in relation to the target, which in the event of a temperature change can be specified as follows:

$$\varepsilon = \frac{\Delta l}{l_0} = \alpha_\vartheta \cdot \Delta\vartheta \qquad (2)$$

The thermal length change coefficient α_ϑ is that of the material specific dimension, which characterises the elongation behaviour of the material under changing temperatures, i.e. with a temperature interval $\Delta\vartheta$. The coatings bound by organic high polymers, always have a higher thermal length change coefficient than that of the substrate concrete. This produces the finding that the intended action mechanism of the tension structure is especially determined by this thermal length change coefficient in connection with a trial induced operated temperature interval.

In principle, two alternatives lead to the intended forced tensions:

A) The total compound system is cooled, in that it is subjected on all sides to a "refrigeration medium". The compound loaded forced stresses are then produced from the resultant difference in the compound body contraction stresses between concrete (a) and coating (b):

$$\Delta\sigma_I = E_b \cdot \alpha_{\vartheta,b} \cdot \Delta\vartheta - E_a \cdot \alpha_{\vartheta,a} \cdot \Delta\vartheta \qquad (3)$$

B) The compound system is cooled, as rapidly as possible (cold shock), by the application of a "refrigeration medium" solely from the coated side. The temperature interval $\Delta\vartheta$ of the concrete is thus practically zero, which causes contraction to occur solely in the polymer coating.

$$\Delta\sigma_{II} = E_b \cdot \alpha_{\vartheta,b} \cdot \Delta\vartheta - 0 \qquad (4)$$

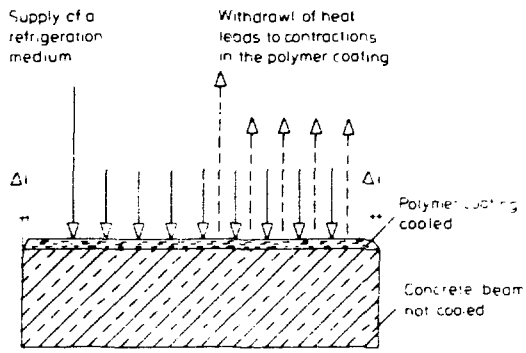


Fig 2 One-sided application of a refrigeration medium

It is easy to see that the forced stresses produced in Alternative B are greater in size than the forces arising from the considered difference from the total cooling of the compound structure. In the latter case, the complete contraction efforts of the coating is faced with the virtually uncontracted concrete beam. Fig. 2 clearly shows these details.

The second alternative is preferred below, especially as it is simpler to implement on building structures of the considered installation. Through one-sided, rapid supply of a "refrigeration medium" and the consequently initiated contraction efforts within the coating plane, forced stresses are produced in the threshold

between the concrete beam and the coating, which under certain circumstances result in breakage between both these compounded substances. In the specified target analysis, the radioactively contaminated coating would separate in splinters from the non-radioactively contaminated concrete, without decomposing into fine particles, thus permitting relatively easy disposal.

4. SELECTION OF AVAILABLE "REFRIGERATION MEDIA"

The specification made on potential "refrigeration media", is in principle:

It must,

- cause extremely strong cooling,
- be applicable easily and safely,
- be easy to dispose of, and
- have an acceptable price.

From several possible materials, after a rough selection, there remained the following four "refrigeration media":

- Air
- Oxygen
- Nitrogen
- Carbon dioxide

When liquefied, air and oxygen are easy to apply, and cause very strong cooling. Disposal directly into the atmosphere is not problematic. However, due to the oxygen, neither of the substances is inert, and in the liquefied state form dangerous explosive compounds. For safety reasons, both these media must therefore be excluded. There thus remain nitrogen and carbon dioxide, which require further fine selection:

Nitrogen is liquid under normal pressure, when cooled to - 196°C. It can be applied in the liquefied state, as well as in the gaseous state at practically the same low temperature, producing very considerable cooling in both aggregate conditions. Nitrogen, applied in the liquefied state, evaporates under normal ambient conditions. The required heat energy is obtained by the liquefied nitrogen from its environment and thus the coating to which it is applied. This effect, which can also be described as the production of "evaporation coldness", leads to an additional, quite considerable cooling effect, which does not arise where cold, gaseous nitrogen is applied.

Where liquefied nitrogen is applied, a detrimental effect is the so-called Leidenfrost phenomenon. The gas envelopes produced around the liquefied nitrogen droplets, produce a heat transfer resistance between the "refrigeration medium" nitrogen and air resp. the coating to be cooled. Rapid cooling of the item to which the medium is applied, is thus hindered.

Initial touch trials have shown that the removal of a coating by the application of liquid nitrogen is successful, and that maximum heat removal is present, when the liquid nitrogen is applied finely atomized.

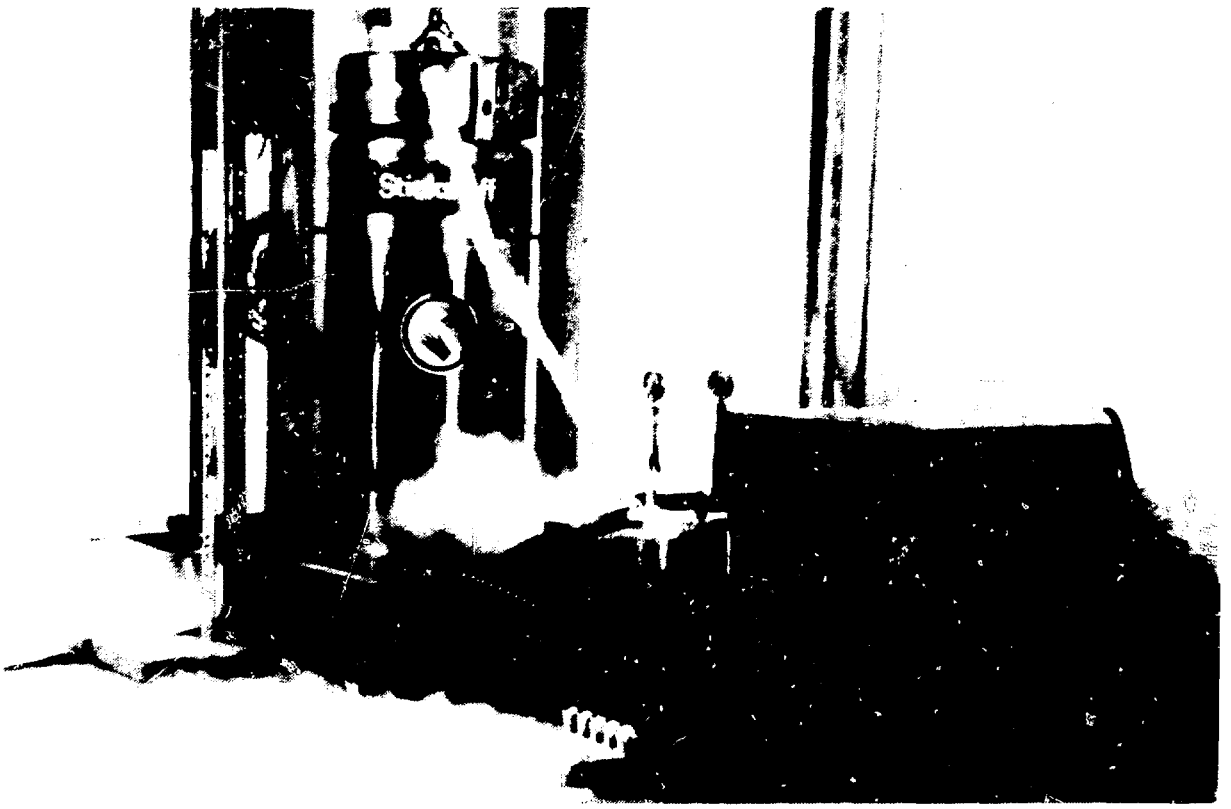
Carbon dioxide is available in the liquefied state in cylinders, at room temperature subject to a pressure of 50 bar. During rapid escape of the liquid carbon dioxide partial evaporation occurs, the unevaporated residue initially solidifies to so-called dry ice. During evaporation and subsequent sublimation of the dry ice, a temperature of - 78°C is produced. Compared with the application of nitrogen, considerably reduced cooling occurs. Additionally detrimental is the foam structure of the dry ice, since the heat energy removing medium cannot be exchanged rapidly enough due to its inertness from the rheological aspect. Variations in the

manner of application of the dry ice did not produce any improvements. The removal of coatings from the substrate was not possible in this way by any of the methods used.

5. DEVELOPMENT AND OPTIMISATION OF AN EXPERIMENTAL INSTALLATION

Based on the conclusions from the touch trials, it appeared that coating removal through the application of finely atomized liquid nitrogen was the most promising. A nitrogen jet system was developed and built, with the following major components:

A 160 l capacity storage tank for liquid nitrogen, mounted on a mobile support structure. Between the storage tank and support structure, a pressure gauge is fitted, which indicates the consumption of liquid nitrogen during application. Via a dip tube, liquid nitrogen is taken from this storage tank by means of a pressurising device.



Picture 1: Nitrogen jet system

A nitrogen jet carriage, consisting of a vehicle propelled by electric motor, which can horizontally move at variable speeds of between o.5 and 1.5 m/min. A spraying device is fitted on this vehicle, which is supplied with liquid nitrogen from the storage tank, via a special, thermally insulated supply line.

Picture 1 shows a photograph of the trial plant.

With regard to the intended coating removal success, considerable importance was given to the design of the nozzle head. Two details had to be given precedence:

The nitrogen must be supplied to the spray head by a line system, in which evaporation is prevented. If this is incomplete, pulsation occurs to a greater or lesser extent, depending on the gaseous proportion produced in the liquid nitrogen jet, which results in an inhomogeneous spray pattern. This can be counteracted by a phase separator upstream of the spray head. Fig. 3 shows the design of this type of spray head with phase separator. The special characteristic of this component is that the liquid nitrogen is supplied separately to each nozzle.

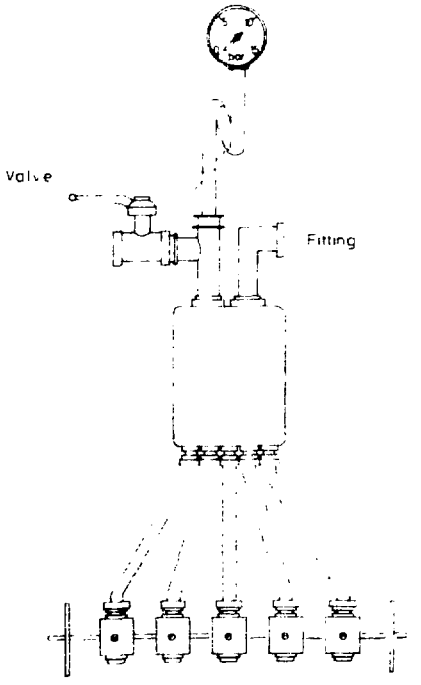


Fig 3 Nozzle head with phase separator

An alternative spray head design is shown in Fig. 4. The liquid nitrogen is supplied here to the side of a nozzle module system. At the other end of the nozzle row, via a valve, gaseous nitro-

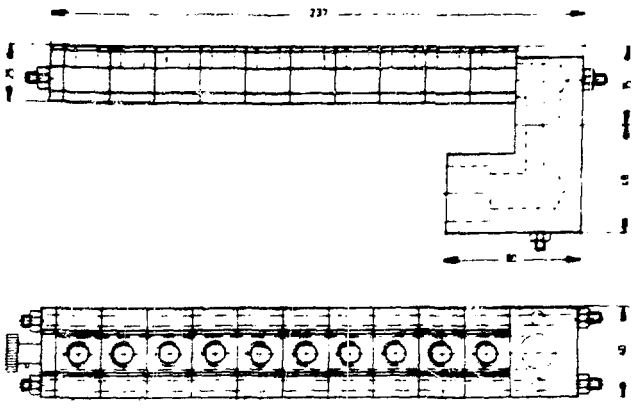


Fig 4 Nozzle head modulesystem

gen can be released during the initial phase of nitrogen application (in this phase, evaporation of the liquid nitrogen is preferred).

Optimisation regarding the maximum supply of liquid nitrogen at the spray head is generally achieved by an adequate flow speed in the line system, combined with appropriately produced pressure.

The nozzle head design (cross-section and outlet opening) as well as the setting angle of the spray head and the distance of the surface to be applied, must be selected so that a uniformly covering spray pattern is produced. Fig. 5 shows the spray pattern that occurs, as a function of the distances from a nozzle brush equipped with 5 nozzles (dia. 1.5 mm, setting angle 60°), which is passed across the surface at an angle of 45°.

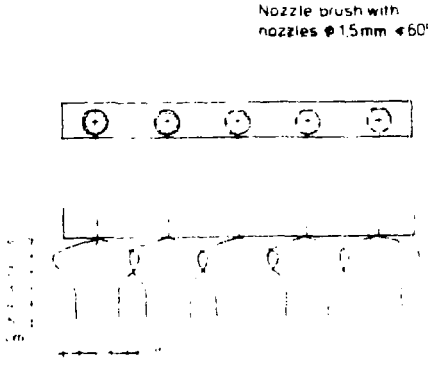


Fig 5 Spray pattern

As originally stated, the aim is to subject the compound system to forced stresses, through strong refrigeration, which will result in separation of the coating. The temperatures which occur during nitrogen application, were determined during a separate series of trials. Temperature sensors were fitted at differing depths under the

surface of coated and uncoated concrete slabs. Using the specified jet system, liquid nitrogen was applied to the concrete slabs prepared in this manner. A nozzle head was used, located transverse to the direction of travel, which was equipped with nozzles of 1.5 mm diameter and outlet openings of 60° each. The speed of travel was set at 0.8 m/min. The angle of the nozzles to the direction of travel was 45°.

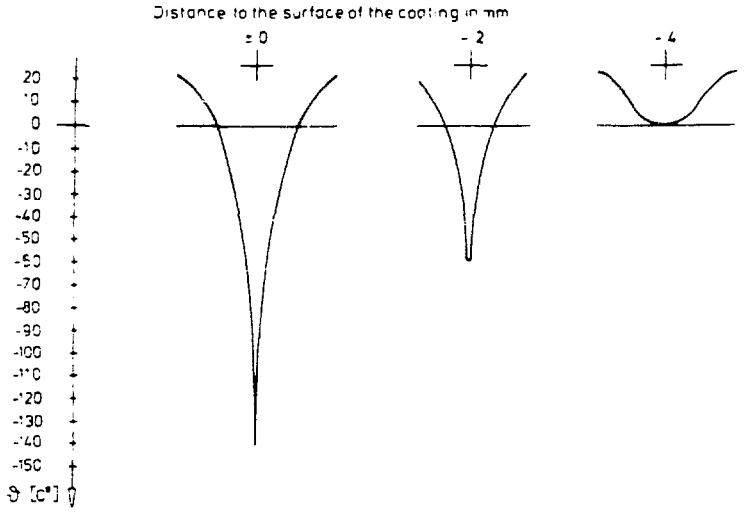


Fig 6 Temperature profiles

Fig. 6 shows a diagram of the measured temperature profiles:

- In direct proximity to the surface: about - 140°C
- About 2 mm under the surface, i.e. in the threshold area between coating and concrete slab: - 60°C to - 70°C
- About 4 mm under the surface: about 0°C

6. TRIAL RESULTS

Using the experimental facility, liquid nitrogen was applied to numerous trial slabs of concrete, cement screed, steel and lime-cement plaster, which had been given so-called decontamination coatings. The result of one series is given here, since this is representative of several trials. Using this series, the laws, the possibilities as well as the removal limits of coatings by

heat removal, based on the present state of the art, can be representatively shown. The concrete slabs used as the carrier of the coating, are within quality category B 35. So-called Agepan slabs were used as moulding for the concrete, such as are used during the concreting of nuclear power plant. The concrete slabs with a fine-coarse, and low-shrinkage surface structure were primed with a non-pigmented, epoxy impregnation and cover coated with various polurethane coatings. The coating structures and the ageing conditions were varied as follows:

- The pigment volume concentration (pvc), i.e. the volume related ratio between binder and pigment/filler material amount of the cover coatings, was used in the trial series in the following stages:

Variant 1	0%
Variant 2	12%
Variant 3	24%
Variant 4	36%

- The particular cover coating was executed in three different dry layer thicknesses:

Variant 1	1.0 mm thick
Variant 2	0.5 mm thick
Variant 3	0.25 mm thick

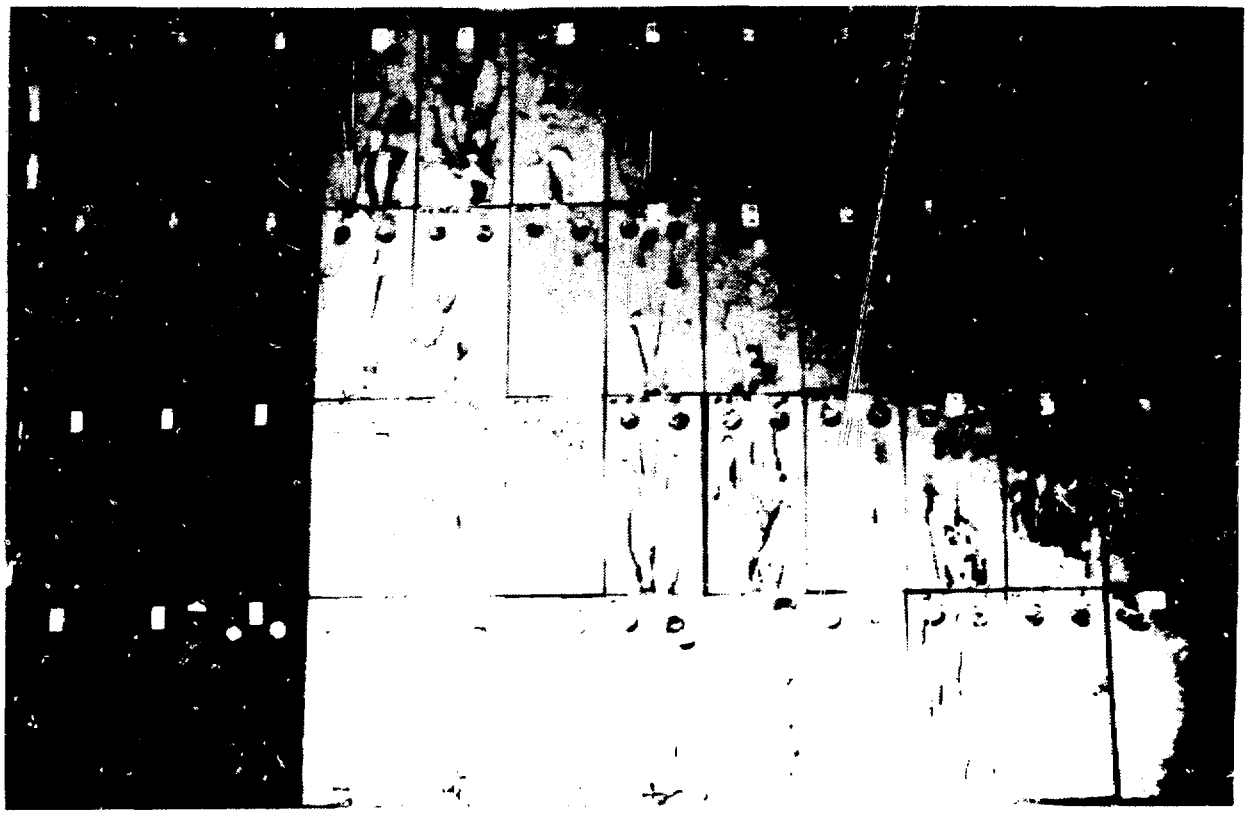
- The nitrogen application was made parallel, after the previous air, water and furnace storage of the coated concrete slabs:

Variant 1	Storage at normal climate 20/65
Variant 2	Water storage for 6 weeks at 20°C
Variant 3	Furnace storage for 6 weeks at 60°C
Variant 4	Combined furnace and water storage for 6 weeks at 60°C in the furnace and 20°C in the water

All trial slabs were placed horizontal under the above specified conditions. The jet applicator vehicle was traversed across the surface at a speed of 0.8 m/min. The liquid nitrogen was applied across the direction of travel by a fitted nozzle brush system, inclined at an angle of 45° from the vertical. 6 spot nozzles were fitted on the nozzle brush. Each nozzle had a diameter of 1.35 mm and an opening angle of 60°.

Depending on the variants of the used samples, virtual or partial separation of the coating occurred, with a 1 ... 2 mm thick layer of the concrete sticking to the back of the separated coating. The separation process can be characterised as follows:

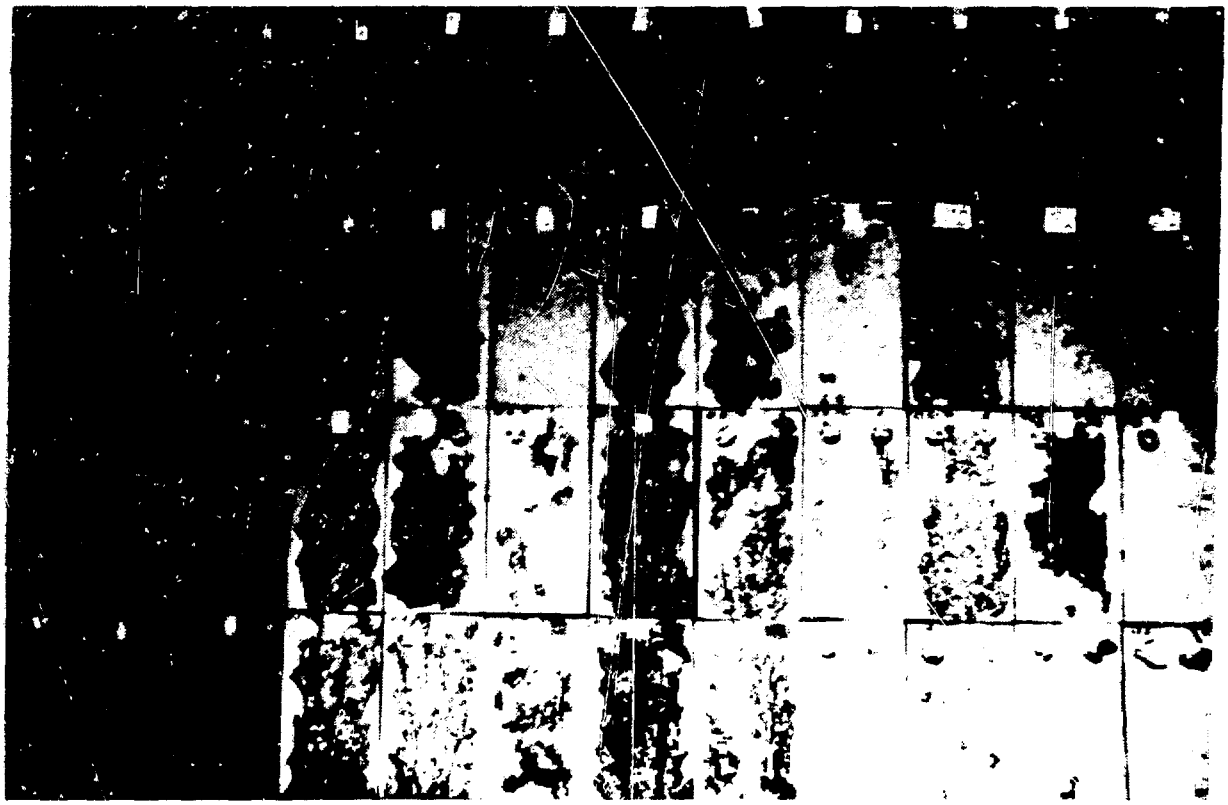
Initially, cracks occurred in the coating parallel to the direction of travel. Generally, these were between the spot nozzles and covered the complete coating thickness. After tearing longitudinally, the coating initially curved tranversely, then



Picture 2: Concrete slabs after nitrogen application

along the direction of travel and then came away, so to speak, in single strips from the concrete. Where successful, the raising was always limited to those zones of the specimens that were away from the edge. In the edge zones, the coating did not separate due to the disturbing effects that arose there.

Pictures 2 and 3 show photographs of the surfaces of the blasted slabs immediately after the application of the nitrogen (Picture 2) and after additional removal of the loose coating strip (Picture 3). Vertically, 3 neighbouring slabs each are grouped in a column with the same pvc, with the triple group on the left being pvc = 0% (unpigmented), whilst the next triple group being pvc = 12%, then pvc = 24% and the group on the outside right having a coating with a pvc = 36%. The three individual columns of the pertinent pvc groups differ in the thickness of the coating, with the slab of the first individual row being 1 mm thick, the second individual row 0.5 mm thick and those of the third individual row having a coating with a thickness of 0.25 mm.



Picture 3: Concrete slabs after nitrogen application and after additional removal of strips

The horizontal rows are assignable through appropriate conditioning to the preset age conditions of the coating systems. The uppermost row is that of the normal climate - stored slabs, the row underneath contain in sequence the water stored, the furnace stored and the furnace/water stored slabs.

The mechanism of separation and the separation patterns can be explained as follows:

Over the nozzle brush, fitted across the direction of travel, considerable cooling of the coating also occurs across the direction of travel. Consequently, contractions occur in the cover coatings, again across the direction of travel. During these contraction efforts, the coating is locally subjected to excess tensile strength, so that a crack is produced in the longitudinal direction.

That this tearing generally occurs between two nozzles, is evidently due to the overlapping of the spray cone of two neighbouring nozzles, as shown in Fig. 5. The forced heat removal generated there by the higher supply of "refrigerating medium" generates between every two nozzles a maximum stress, within which the coating tears. The continued contraction effort then leads in the neighbouring sector to a lifting of the coating.

On both sides of the crack, the fracture surfaces are characterised by a different type of structure to that in the subsenquently curving sector of the coating. This indicates that in the tearing area only shearing forces occur which can separate in a zone of the concrete that is close to the surface, whilst in the neighbouring sector vectoral added stresses from shearing and curving processes lead to a separation of the coating.

In Picture 3, if one views all slabs in their variants, it is evident that the best removal success is achieved with a coating with a pvc of 12%. By comparison, with reducing and increasing pvc, the efficiency drops. With reducing pvc, the determined behaviour can be explained by the fact that the polymer binder can avoid the forces produced by the temperature drop by creeping, due to its high proportion of plastic deformers, which then counteract the removability. With rising pvc, the increasing proportion of fillers and pigments produces a reduced creep capacity as well as a smaller thermal length change coefficient α_T. Whilst the creep capacity is adequately reduced, the thermal length change coefficient is not sufficiently large so that adequate force stresses occur which actuate the separation. However, the smaller the thermal length change coefficient of the coating becomes, the smaller are the contraction efforts and the associated forced stresses in the neighbouring area.

It was additionally noted that the coating removal success rate was at its maximum when the thickness of the applied coating was about 1 mm. With a thickness of 0.5 mm, partial removal is still possible, although the success rate decreases noticeably when the thickness is only 0.25 mm. This reaction can be explained by the fact that the forces which lead to forced stresses between coating and substrate, increase with increasing coating thickness. In addition, the coating is subjected to increased cooling, since the heat flows out of the concrete base at a slower rate due to the increased thickness. By comparison, the binding forces between coating and concrete are in principle the same with thicker and thinner coatings.

It has also been found that the coating removal success rate is increasingly improved, the smaller the (close to the surface) stability of the coated concrete. This is easy to explain, if one realises that a low-grade layer of concrete also has a lower shear and tensile strength.

7. SUMMARY AND FORECAST

Based on the deliberation, that coatings on construction materials are subject to contracting stresses under strong cooling, which can lead to separation, a trial installation was built, with which liquid nitrogen (- 196°C) can be applied to a coating as a "refrigerating medium".

Trials have shown that the separation of coatings is successful under certain conditions. Favourable factors for the separating process are stability that is not too high in the close-to-the-surface coating substrate, a high thermal length change coefficient α_{ϑ} of the coating, a minimum creep capacity and a specific layer thickness of the coating. In addition, the success rate can be improved by rapid, shock cooling, which can be obtained especially through installation and application technical optimisation of the heat transfer during liquid nitrogen application.

The tests have also indicated the possibility of the specific use of coating systems, which are formulated from the outset for easy separation, e.g. through incorporated nominal fracture levels, which will allow at least partial removal. There is also the possibility of removing thin layered coatings when combined with thick layered auxiliary coatings.

DECONTAMINATION OF URANIUM-CONTAMINATED EQUIPMENTS AND PARTS

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ABSTRACT

A Uranium enrichment pilot plant was decommissioned in 1985-1986. This paper concentrated on decontamination concept, methods and results for the equipments and parts decontaminated. The kinds of metals involved in the decontamination action were Copper, Nickel, Aluminium alloy, mild steel, stailess steel and so on. Decontamination results showed the surface contamination levels of most parts decontaminated achieved the required level. The Uranium contents in Aluminium ingots after melt refining were from 33 to 232ppm. The decontamination liquid wastes were treated with multiprecipitation method. The contents of Uranium, Nickel, fluoride in supernatant were 0.02-0.1mg/l, 0.02mg/l and 0.13mg/l respectively.

INTRODUCTION

A Uranium enrichment pilot plant was decommissioned in 1985-1986. After decommissioning the buildings would be used for other purpose. The equipments were not expected to be reused. It was decided to dismantle and decontaminate these equipments for metal recycling. The equipments were contaminated by Uranium compounds, mainly UO_2F_2, with nearly the same isotope composition of natural Uranium. According to the operation history, the residual UO_2F_2 was estimated to amount to 15-20Kg. The contamination levels of inner surface of dismantled components were measured and ranged from 0.002 to 4 Bq/cm^2 for alpha contamination, from 3 to 140 Bq/cm^2 for beta contamination.

Considering the nature of contamination chemical methods were selected as the principal decontamination methods, mechanical methods and melt refining were also used when necessary.

The decontamination methods used in routine maintenance were not suitable for the purpose. To develop decontamination methods of high efficiency, some chemical decotamination and melt refining experiments were carried out. On the basis of experiment results some decontamination procedures were recommended.

EXPERIMENTS OF CHEMICAL DECONTAMINATION

Selection of Decontamination Agents

In order to select the suitable decontamination agents

for various metals, more than 20 kinds of agents with different compositions were tested with representative samples. The results were as follows

Decontamination of Copper parts Using 5%$(NH_4)_2CO_3$+2%H_2O_2, 5%$(NH_4)_2CO_3$ (pH9), 5%$Al(NO_3)_3$+1.5%$Fe(NO_3)_3$+0.2%H_2SO_4 (pH1.5), 2M HCl, or 1.5M HNO_3 as decontamination agent with soaking time of 10 min., the decontamination efficiencies (to be simplified as DE) were higher than 97%. The best decontamination effects were obtained when 5% $(NH_4)_2CO_3$ + 2% H_2O_2 was used. During decontamination the reactions proceeded violently. So the operation must be performed at a site with good ventilation and necessary safety measures. If 5% $(NH_4)_2CO_3$ solution was used, reaction was more gentle, but it needed much longer time to achieve the required effect.

Decontamination of Nickel parts The results showed when 5% $(NH_4)_2CO_3$ + 2% H_2O_2, 1.5M HNO_3, 1M HCl, 5% Na_2CO_3 + 1% EDTA-Na2, 5% $Al(NO_3)_3$ +1.5% $Fe(NO_3)_3$ + 0.2% H_2SO_4 was used as decontamination agent separately with soaking time of 1 hr., the DE were all higher than 87%. For the first three agents, the DE were higher than 98%. The Uranium contents in decontaminated Nickel plates with 0.2mm of thickness were below 5ppm.

Decontamination of Aluminium alloys parts Some decontamination agents were given out in following sequence according to their decontamination effectiveness: 1.2M HCl ≐ 1.5M HNO_3 ≐ 5% NaOH > 5% $Al(NO_3)_3$ + 1.5% $Fe(NO_3)_3$ + 0.2% H_2SO_4 > 5% $(NH_4)_2CO_3$ + 2% H_2O_2 ≐ 5% $Al(NO_3)_3$ ≐ 1% $Fe(NO_3)_3$ > 5% $(NH_4)_2CO_3$ > 5% Na_2CO_3. For 5% Na_2CO_3, 5% $(NH_4)_2CO_3$, 1% $Fe(NO_3)_3$ or 5% $Al(NO_3)_3$, when soaking for 2 hr., the obtained DE were 80%, 90%, 95% and 92% respectively. The results were close to their maximal DE. So there was no need to extend the soaking time further. For 5% $(NH_4)_2CO_3$ + 2% H_2O_2, 5% $AL(NO_3)_3$ + 1.5% $Fe(NO_3)_3$ + 0.2% H_2SO_4, 1.2M HCl or 1.6M HNO_3, with soaking time of 1 hr., the Uranium contents in decontaminated Aluminium plates with 4mm of thickness were below 7ppm.

All of decontamination results for various agents showed that alpha emitter (U) was easier to be removed from the surface than beta emitters (UX_1,UZ,UX_2)(See Table I), and the beta emitters were easier to be sorbed on the surface. It was found that after decontamination the beta surface residual activity reduced with time. After 300 days the beta surface contamination levels would be near to the alpha residual contamination levels measured immediately after decontamination. It showed the residual beta activity was mainly contributed by shortlife daughter of Th-234.

The Uranium contents in decontaminating solutions effected the DE obviously, especially for beta DE. Fig.1 and 2 also showed that Uranium was easily complexed or disoluted into solution and it s daughters had greater tendency to remain on the surface. For complexing agents, for example 5% $(NH_4)_2CO_3$, it was found when Uranium contents (added in the form of UO_2F_2) in decontamination solution were higher than 5 gU/L, DE (alpha, beta) would be improved. For corrosive agents, for instance HNO_3, Uranium and its daughters came into solution all together. More recontamination happened when the concentrations of Uranium and its daughters in solution increased,

Table I Alpha, Beta Activities Residual On the Surface of Aluminium Parts After Decontamination

sample	chemical decontamination				laying aside after decon.	
	composition of decon. solution	time (hr)	alpha (%)	beta (%)	time (d)	beta (%)
wing plate (cast)	5%$NaHCO_3$(pH8)	22	4	30	300	less 2
	5%Na_2CO_3+2%H_2O_2	1	0.6	73	300	less 2
	5%$(NH_4)CO_3$+2%H_2O_2	1	0.7	6	300	BDL*
	5%$(NH_4)CO_3$(pH9)	22	less 1	8	301	BDL
	5%$Al(NO_3)_3$+1.5% $Fe(NO_3)_3$+0.2%H_2SO_4	1	1	3	302	BDL
	5%Na_2CO_3(pH13)	22	2	7	301	BDL
	5% NaOH	0.5	1	BDL	301	BDL
	1.5M HNO_3	1	0.6	less 1	300	BDL
	1.2M HCl	0.5	0.3	less 1	300	BDL
valve barrel	5% $NaHCO_3$(pH8)	24	BDL	22	301	BDL
	5% Na_2CO_3(pH13)	4.5	10	24	302	9
	5%$(NH_4)_2CO_3$+2%H_2O_2	1	1	1	302	BDL
	5%$(NH_4)_2CO_3$(pH9)	4.5	5	8	302	1
	5%$Al(NO_3)_3$+1.5% $Fe(NO_3)_3$+0.1%H_2SO_4	1	0.6	3	302	BDL
	5% NaOH	1	10	2	301	BDL
	1.5M HNO_3	1	0.5	2	300	BDL
	pH 1 HCl	4.5	0.1	1	300	BDL
compressor plate	1% $Fe(NO_3)_3$(pH2.5)	2	0.2	3.6	300	BDL
	5% $Al(NO_3)_3$(pH4)	2	0.3	7.5	300	BDL

* BDL- Below the Detection Limit

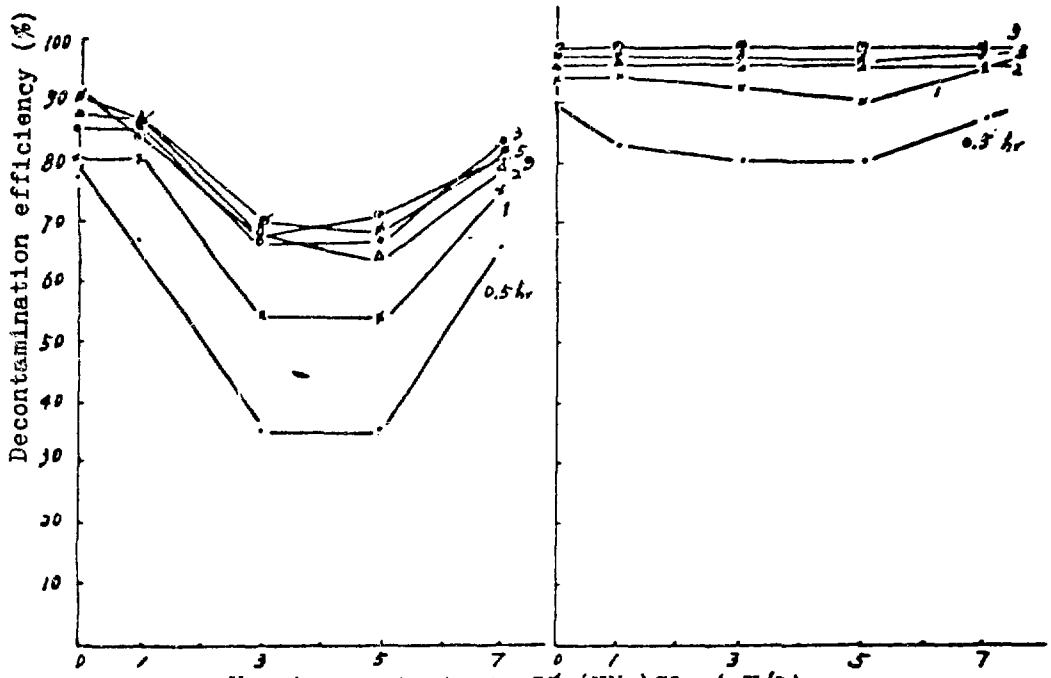


Fig. 1 Uranium contents in decontamination solution versus decontamination efficiency.

Left- for beta DE, Right- for alpha DE

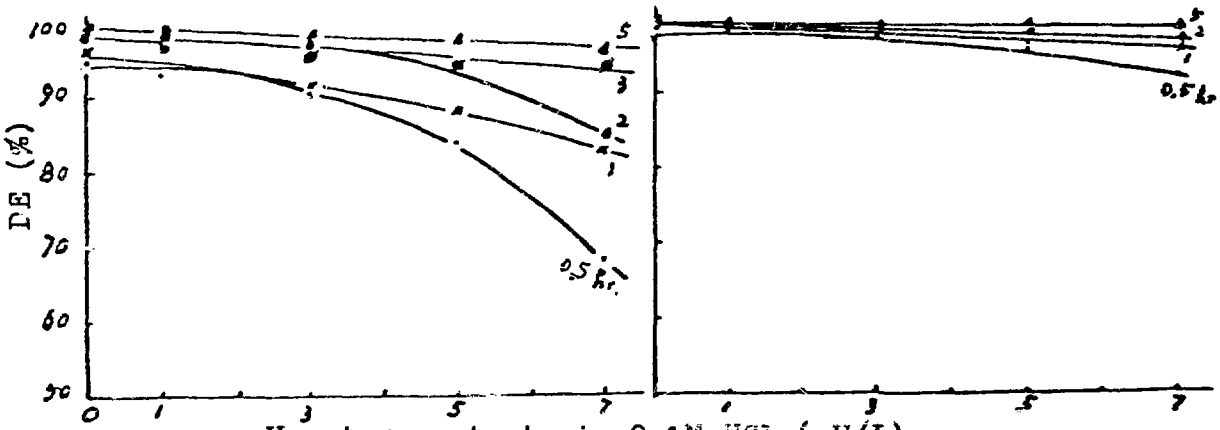


Fig.2 Uranium contents in decontamination solution versus decontamination efficiency.

Left- for beta DE, Right- for alpha DE

especially for the daughters.

Decontamination of mild steel parts Most parts of the equipments were made of mild steel. Among them the steel pipes were the most difficult to be decontaminated, because there were rusty and crud on the inner surface in varying degrees. This condition also caused quite different decontamination effects for different agents.

The beta contamination levels on the inner surfaces of steel pipe with slight corrosion were 30-200 Bq/cm^2. For these pipes only 2M HCl or of higher concentration and 15% H_3PO_4 + 8.5% CrO_3 had satisfactory effects (decontamination factor more than 300). The Uranium contents in the pipe with 3mm of thickness were below 8ppm after decontamination with soaking for 4hr..

When the ratio of the volume of hydrochloric acid solution to the surface of decontaminating parts equal to 10-15, the concentration of hydrochloric acid solution should be higher than 1M to obtain effective decontamination (Fig.3). During practical operation the concentration of hydrochloric acid solution should be kept higher than 2M. In decontamination some fraction of UO_2^{2+} was reduced into U^{4+} and then deposited on the surface of pipe. Porous rust could obsorb much more Uranium compounds, so that the surface contamination level would increase rapidly until rust on the surface disappeared (Fig.4). The Uranium contents in hydrochloric acid solution must be kept below 3 gU/L and the concentration of hydrochloric acid solution higher than 2M during decontaminating steel pipes.

Hand scrubbing of unwieldy parts There were a lot of unwieldy parts that could not be decontaminated with soaking, so hand scrubbing decontamination technique was employed. Experiment results showed after scrubbing with 5% $(NH_4)_2CO_3$ or 1-2M HNO_3, the surface contamination levels could be decreased to required level (Table II).

The Decontamination Procedures Recommended

Based on the experiment results and considering some other factors, following decontamination procedures were recommended:

Copper-

(1) 5% $(NH_4)_2CO_3$ + 2% H_2O_2, soaking for 10-20 min., rinsing. (2) 2M HNO_3, soaking 10 min., rinsing.

Nickel-

(1) 5% $(NH_4)_2CO_3$ + 2% H_2O_2, soaking for 10-20 min. rinsing. (2) 1M HNO_3, soaking for 10-20 min., rinsing.

Aluminium alloy-

(1) 5% $(NH_4)_2CO_3$, soaking for 1-2 hr., rinsing. (for slightly contaminated parts)

(2) 2-4M HNO_3, soaking for 10-20 min., rinsing. (for heavily contaminated parts)

(3) 10% NaOH, soaking for 10-20 min., washing. (for shallow layer contaminated parts)

When the required level could not be achieved using above mentioned procedures, melt refining would be employed.

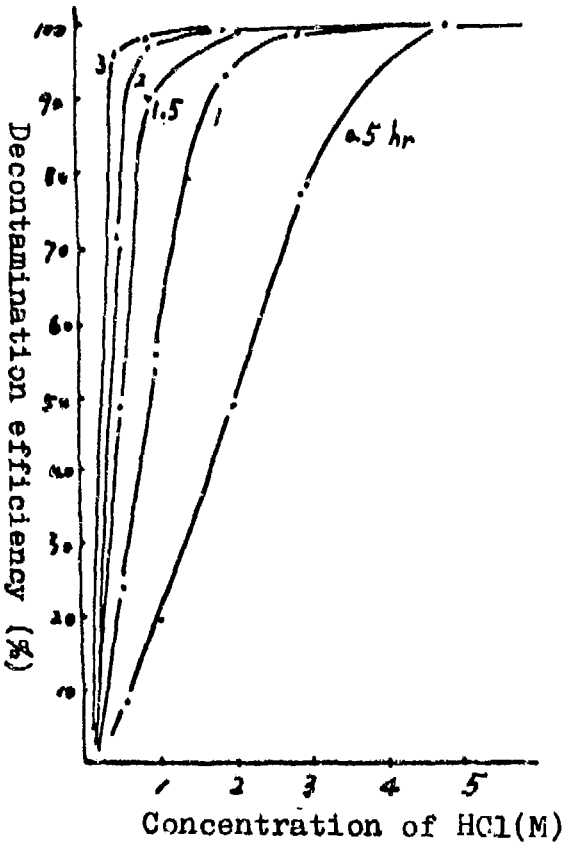


Fig.3 Effects of concentration of HCl on decontamination efficiency (mild steel)

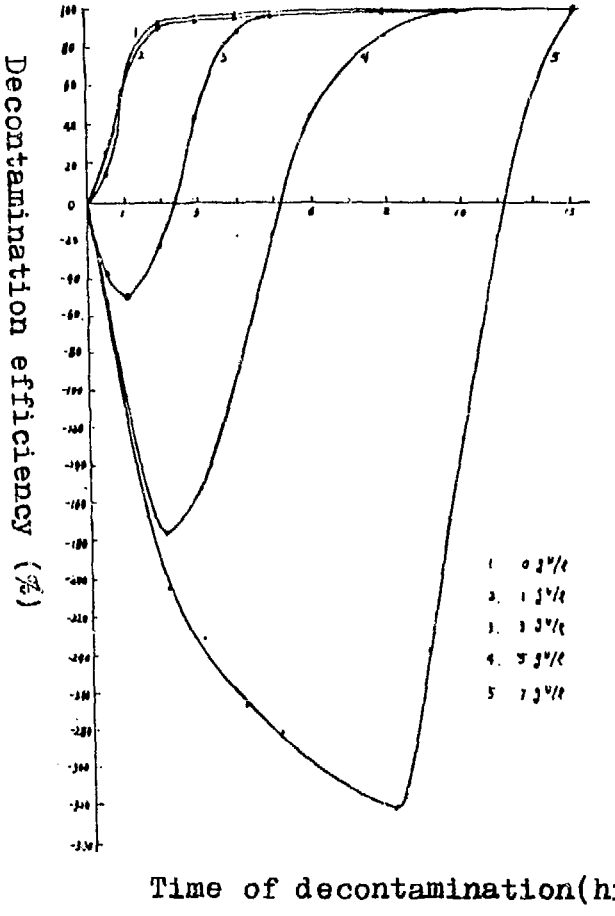


Fig.4 Uranium contents in 1M HCl versus decontamination efficiency (mild steel pipe)

Table II Results of Hand Scrubbing

Parts	Metal	Charact. of surface	Decont. solution	Beta DF				Alpha DF			
				origi.	1 st	2 nd	3 rd	origi.	1st	2nd	3rd
OK 7/8	steel	light yellow	5% Na_2CO_3	147 Bq/cm^2	2.1	3.2	3.2	43	1.7	1.9	7.2
		item	5%$(NH_4)CO_3$	160	6.4	10.7	14.5	30	33.3	8.2	22.9
		item	0.8M HNO_3	87	8.3	11.7		26	28.5	312	
		green-yellow	5%Na_2CO_3+2%H_2O_2	280	7.1	16	46.9	40	8.5	8.9	
		item	5%$(NH_4)_2CO_3$+ 2%H_2O_2	400	23.5	29.3	67.1	37	9.1	36	
		item	0.8M HNO_3	310	40.4	55		33	152	597	
appara-tus tank	Steel	black	5%$(NH_4)_2CO_3$	103	7.4	42	4.10^4	27	3.4	16.6	
		green-yellow	1M HNO_3	86	6.2	24.6	32.7	26	10	36.4	262
		yellow	5% Na_2CO_3	23	1.1	1	82.3	40	4.2	4.6	
		dark grey	5%$(NH_4)_2CO_3$	15	2.7	3.9	9.3	7	1.5	2.8	
		grey	item	8		20.8		43		18	
		grey	5% Na_2CO_3	6	5	12	21.8	2	1.5	2.3	
compre-ssor cast	steel	grey-green	1.5M HNO_3	666	43			21	3.7		
		yellow	0.8M HNO_3	283	13.1	17.8					
		yellow	5% Na_2CO_3	170	6.2						
oxle sleeve	s.s.	green	1.5M HNO_3	30	12						
spring box	steel	grey	1.5M HNO_3	12	1.9			1.2	12		

Mild steel-

Hand scrubbing with 5% $(NH_4)_2CO_3$ or 1-2M HNO_3; or 2-4M HCl soaking for 1-8 hr., rinsing.

Decontamination Results of Equipments and Parts

1300Tons of various metals were decontaminated with recommended procedures. The measurement results showed that the beta surface contamination ranged from 0.2 to 17 Bq/cm^2 on 7207 parts and the alpha surface contamination levels from 0.003 to 1.51 Bq/cm^2 on 330 parts. The contamination levels of most of the parts achieved the required level after decontamination.

MELT REFINING EXPERIMENTS

During long period of operation, Uranium contaminants could entry into the inside of metal, especially for cast Aluminium parts. Measurements showed the Uranium contents in different depth of cast aluminium parts were 0.5-1 mm, 5300ppm; 3-4 mm, 3350ppm; 9-10 mm, 13ppm. The beta contamination levels in different depth were given in Table III.

Table III The beta contamination levels in different depth of cast aluminium parts (Bq/cm^2)

No.	surface	1 mm	2 mm	3 mm	4 mm	5 mm	6 mm	7 mm
62	36.3			18.2		3.6		3.5
63	2.5		3.2	5.6		1.6	0.04	
64	14.8		12.1	BDL*				
71	15.0	18.0		0.47		BDL		
72	18.1	10.4		0.5				
73	104		77.6	6.1		BDL		

* BDL- Below the detection limit

For the sake of metal recycling or volume reduction of solid waste, the melt refining experiments were carried out with the cast aluminium parts mentioned above.

Tatsuhika, Uda.(1) and Heshmatpour, B. (2,3) had reported their experiment results of melt refining. They pointed out fluoride-containing fluxes were effective. The goal of our experiment was to select suitable fluxes for large scale melt refining. At high temperature whether the Uranium can be removed from the basic metals, it depended on the molecular free energy of their oxides. When melted, the shallow layer of the aluminium parts could be removed as Al_2O_3 from the melted aluminium and UO_2F_2 decomposed to UO_2, came into slag with Al_2O_3. Because of the very active chemical properties of aluminium at high temperature, it could reduce UO_2 into U ($4Al + 3UO_2 = Al_2O_3 + 3U$) and form (Al-U) alloy. Suitable fluxes could delay or inhibit this reaction. During the experiments

more than one hundred kinds of fluxes with various compositions were tested. Effects of melt temperature, melt period, composition of aluminiumalloy and times of melt on the DE were studied. Total gamma activity and residual uranium in aluminium ingots were used to assess the effects.

Results of Melt Refining Experiments

The results showed all of the fluxes that contained fluoride had better effects. Other useful fluxes were also given in Table IV

The total gamma activity of aluminium ingots decreased with time, as presented in Fig. 5 and 6. It indicated that Uranium was easier to be removed from aluminium by melt refining than its daughters. Th-234 was the main contributor of gamma activity. Those phenomena were similar to that in chemical decontamination.

Melt temperature appeared to have evident influence on DE of Uranium, except when the mixtures of fluoride and chloride were used as fluxes (Fig.7). That was because high temperature was favourable for Uranium to exist in form of oxide (Fig.8). Short melting time was advantageous to melt refining of aluminium (Fig.9). Within 10 min. after melt down of aluminium, Uranium oxide could be concentrated into slag. It was found that the composition of aluminium alloys had great influence to Uranium residual in ingots (Table V).

Table IV Decontamination efficiency of various fluxes on cast aluminium alloy*

No.	flux composition (Wt%)	sample (g)	total gamma residual(%) min.	laying (d)	U residual(%) min.	avg
1	14NaF76KCl10BaCl$_2$	100	29.4	9.5 (173d)	8.6	17±11
2	14NaF1Na$_2$CO$_3$76KCl 10BaCl$_2$	100	43.5	10 (167)	8.9	26±15
3	14NaF76NaCl10CaCl$_2$	100	35.2	9.2 (128)	8.6	28±17
4	14LiCl14NaF62KCl 10BaCl$_2$	100	33.3	9.1 (151)	8.3	19±12
5	15LiCl15NaF50BaCl$_2$ 20KCl	100	45.5	16.1 (151)	14.9	18±7
6	KCl	100	23.8	16.9 (149)	15.8	26±14
7	BaCl$_2$	100	16.1	12.3 (195)	11.2	27±17
8	50KCl40BaCl$_2$10CaCl$_2$	100	20.1	10.4 (167)	10.2	20±12
9	50KCl50BaCl$_2$	100	34.5	23.8 (167)	19.7	39±20
10	90NaCl10Na$_2$CO$_3$	100	32.3	27.8 (167)	26.6	47±18
11	50BaCl$_2$50Na$_2$SO$_4$	50	22.7		22.6	45±16
12	26NaCl74Na$_2$SO$_4$	50	29.1		23.5	29±4

* Amount of fluxes: 10% Of sample wight
Melting temperature: 800 °C
Melting time: 1hr.

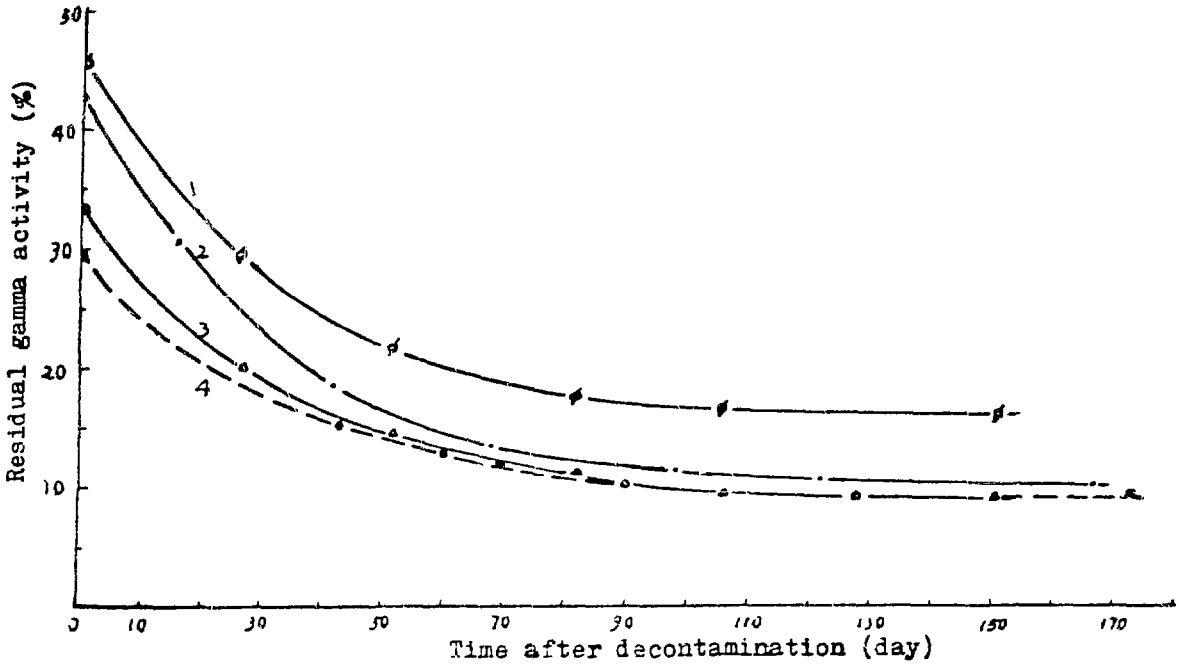


Fig.5 Time after decontamination versus residual gamma activity in ingot

1 - 15LiCl 15NaF 50$BaCl_2$20KCl 2 - 13NaF 1Na_2CO_3 76KCl 10$BaCl_2$

3 - 14LiCl 14NaF 62KCl 10$BaCl_2$ 4 - 14NaF 76KCl 10$BaCl_2$

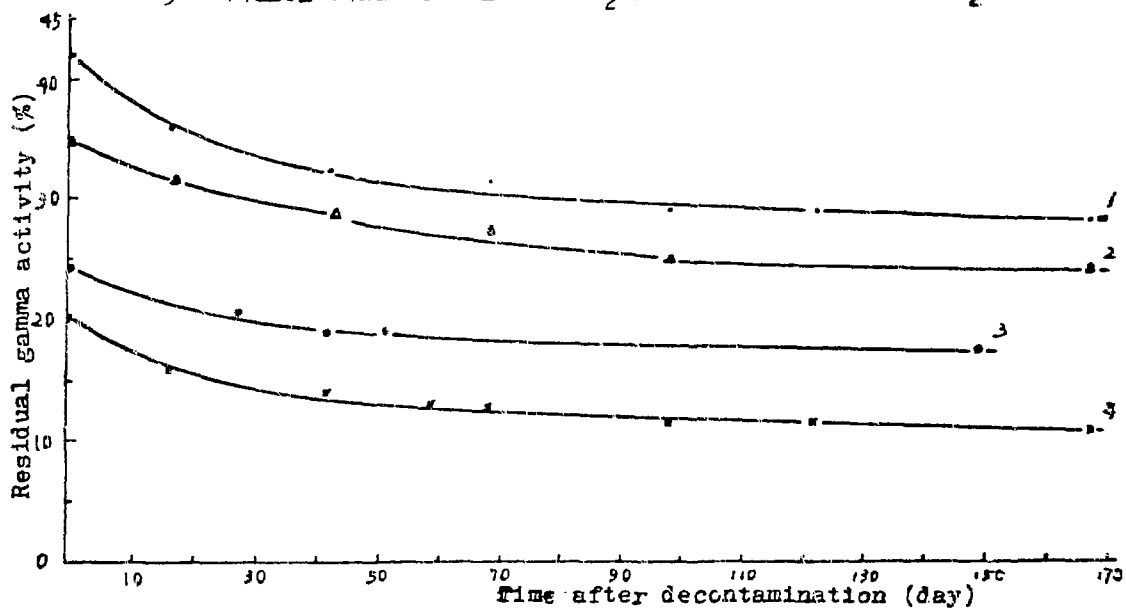


Fig.6 Time after decontamination versus residual gamma activity in ingot

1 - 90NaF 10Na_2CO_3 2 - 50KCl 50$BaCl_2$

3 - KCl 4 - 50KCl 40$BaCl_2$ 10$CaCl_2$

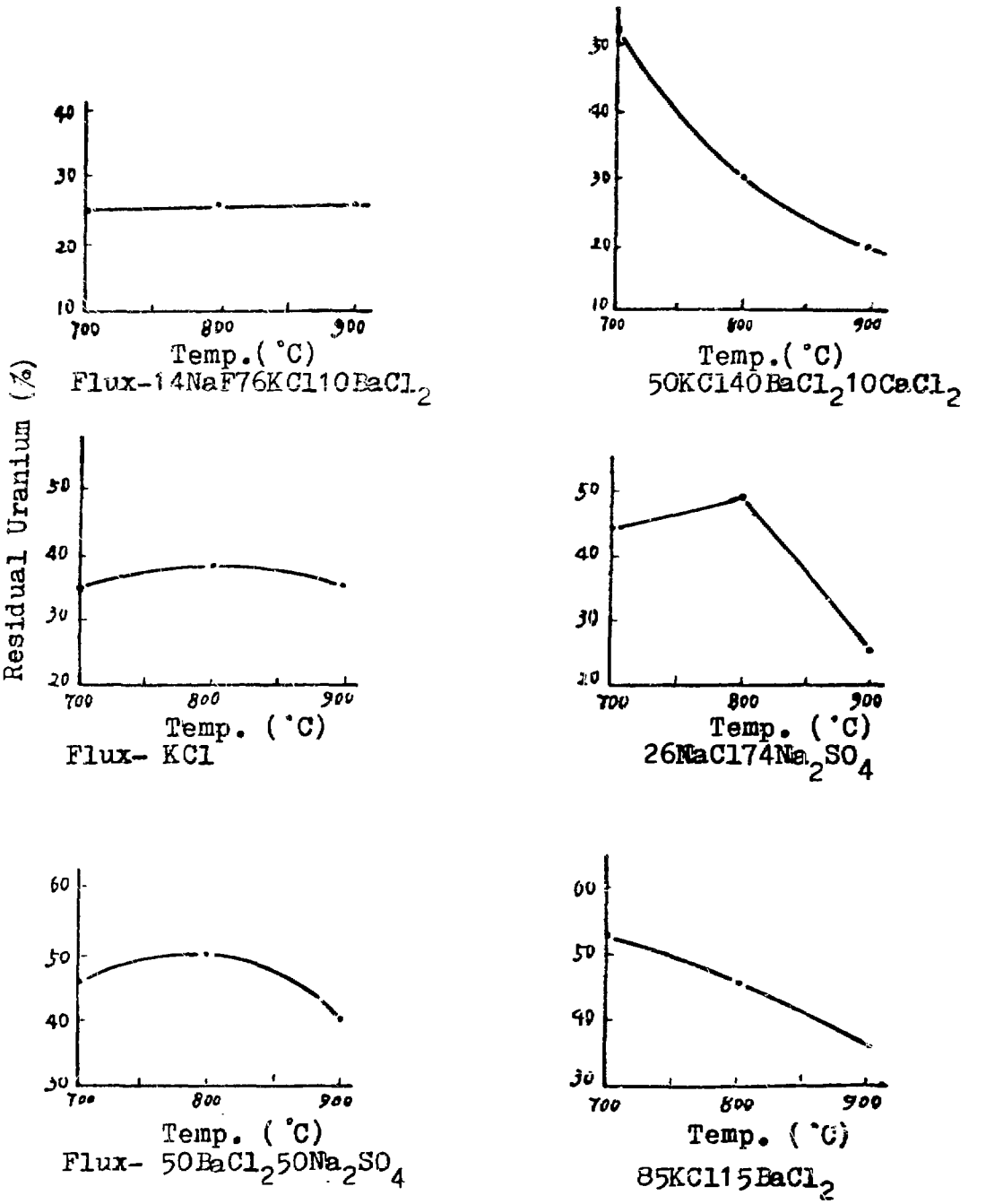


Fig.7 Effects of melt temperature on residual Uranium in the Al ingots

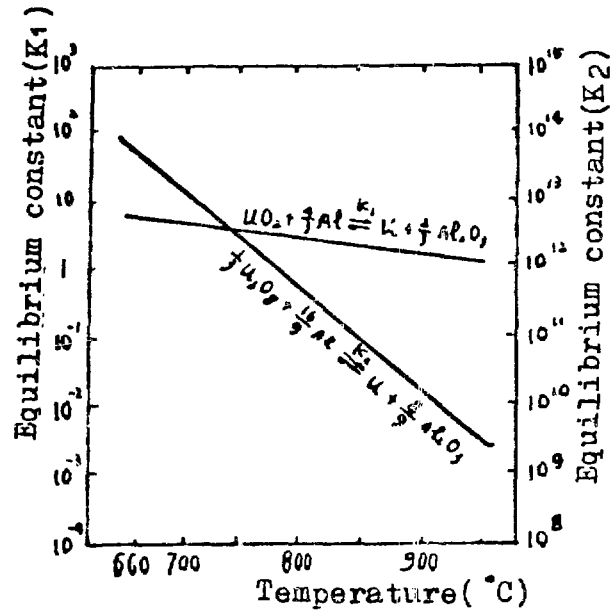


Fig.8 Equilibrium constants versus melt temperature

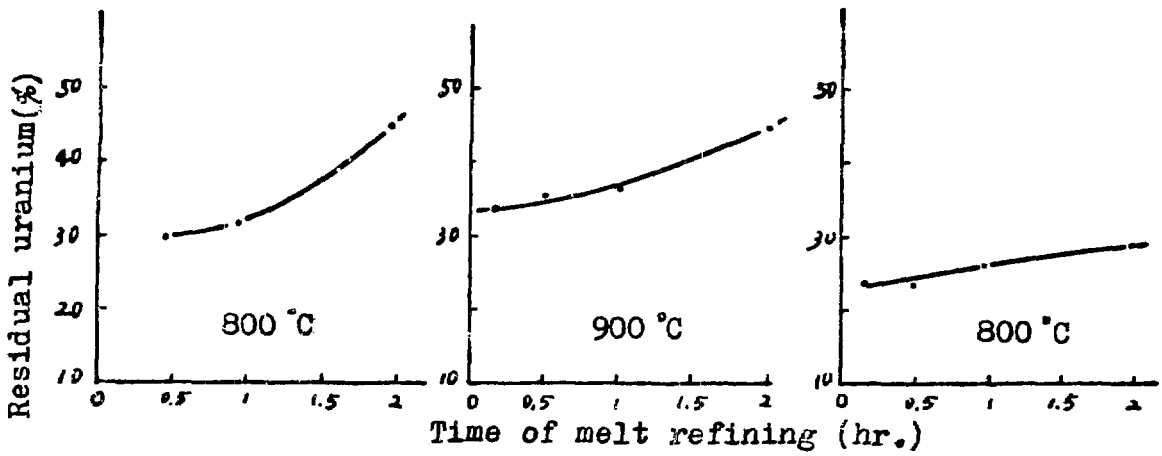


Fig.9 Effects of melting time on residual uranium

Flux: Left- 50KCl40$BaCl_2$10$CaCl_2$ Right- 14NaF76KCl10$BaCl_2$

Medium- 26NaCL74Na_2SO_4

Table V Effects of composition of aluminium alloys to the uranium residual (%)

flux composition (Wt%)	composition of aluminium alloys(%)		
	88.5Al 9.19Si 0.21Mg 0.5Zn 0.03Mn 0.005Fe 0.002Cu	89.97Al 7.5Si 0.29Mg 0.3Zn 0.003Fe0.001Cu	91.76Al0.04Si 1.66Mg0.03Mn 0.01Cu0.001Fe
14NaF76KCl10$BaCl_2$	13.4±0.5	20.0±2.0	18.0±1.0
26NaCl74Na_2SO_4	13.7±0.7	29.3±0.1	28.0±3.0
KCl	5.0±0.4	26.9±0.1	4.0±2.0
50KCl40$BaCl_2$10$CaCl_2$	3.2±0.6	40.0±3.0	10.0±2.0

The first and third aluminium alloy appeared to have higher DE. The reason perhaps was that there was higher contents of Si (9.19%) in the first alloy and higher contents of Mg (1.66%) in thethird one. When increase of Si content, Uranium had more chance to form uranium silicides. It would reduce the probability of forming (Al-U)alloy. For the third one the formation of (Al-Mg) alloy resulted in the decrease of free energy(Fig.10). The equilibrium constant of (Al-Mg) alloy with UO_2 would also decrease with increasing Mg contents (Fig.11). So uranium would be removed more easily. Duplicating melt refining was not helpful to decontamination.

In the experiment 3 tons of uranium-contaminated aluminium alloy parts were refined. Original beta contamination levels were between 0.5-24 Bq/cm^2, and uranium contents between 38 - 2100ppm. Melting temperatures were about 750 - 800 C. Melt period was 20 - 30 min. The maximum uranium content in the aluminium ingot after melt refining was 232ppm (beta surface contamination level less than 2 Bq/cm^2) and its average content was 102ppm. Those value (102ppm) responded to 7.2 10^{-8}Ci/Kg. Partition radio (defined as the amount of contaminant in ingot divided by the amount of contaminant in slag) was 0.013 - 0.005.

Table VI and VII presented the results of melt refining tests. All of the results were better than those in laboratory experiment. The reason was different melt technique was employed.

Melt Refining Procedure Recommended

Based on the results obtained following fluxes and other parameters were recommended:

Fluxes - (1) 14% NaF + 76% KCl + 10% $BaCl_2$
(2) KCl
(3) 50% KCl + 40% $BaCl_2$ + 10% $CaCl_2$
(4) 26% NaCl + 74% Na_2SO_4

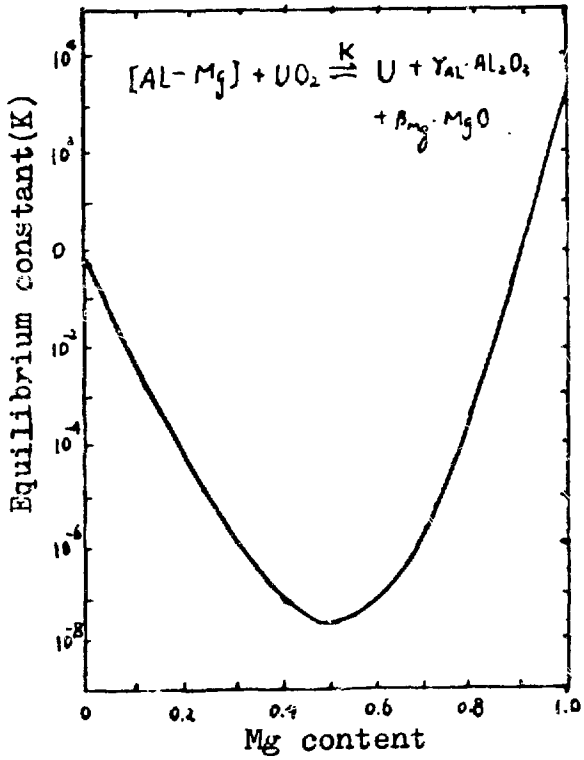


Fig.11 Equilibrium constants versus Mg contents

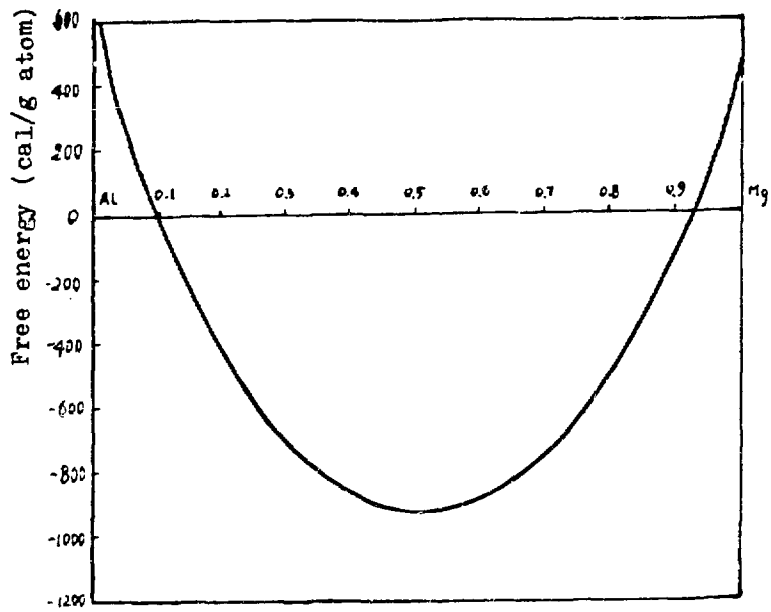


Fig.10 Free energy of (Al-Mg) versus Mg content

Table V! Results of Melt Refining Test*

No.	sample (Kg)	flux composition (Wt%)	flux (%)	origin U (ppm)	ingot U (ppm)	slag (ppm)	partition ratio,**
2	150	$BaCl_2$	10	780	187	4370	$4.3.10^{-2}$
3	150	$BaCl_2$	10	780	182		
4	150	50KCl40$BaCl_2$10$CaCl_2$	5	1570	150		
5	150	item	5	1600	118		
7	150	item	5	1300	135		
8	150	item	10	1300	160	7320	$2.2\ 10^{-2}$
9	110	item	10	1600	232		
10	140	item	7	1600	63		
11	120	item	7	1600	134		
12	140	item	7	1600	125	9990	$1.3\ 10^{-2}$
13	150	item	7	1800	156		
14	150	90NaCl 10Na_2CO_3	10	1800	139		
15	170	70NaCl 30Na_2CO_3	9	2100	71		
16	150	50KCl40$BaCl_2$10$CaCl_2$	5	1100	34	2470	$1.4\ 10^{-2}$
17	145	10NaCl80$BaCl_2$10$CaCl_2$	10	1600	33		
18	120	70NaCl 30Na_2CO_3	10	500	34	6840	$5.0\ 10^{-3}$
19	190	90NaCl 10Na_2CO_3	5	600	68		
20	150	70NaCl20Na_2CO_3 10$BaCl_2$	5	200	29	1350	$2.1\ 10^{-2}$
21	150	70NaCl 30Na_2CO_3	5	100	71	4690	$1.5\ 10^{-2}$

* Melting temperature 750-800 °C
Melting time 20-30min.

** Partition ratio defined as the amount of Uranium in the aluminium ingot divided by the amount of Uranium in the slag.

Table VII Surface contamination levels of aluminium ingots after melt refining

No.	sample (Kg)	flux composition (Wt%)	origin* (Bq/cm^2)	ingot(Beta)** (Bq/cm^2)	ingot(Alpha) (Bq/cm^2)***
2	150	$BaCL_2$	8	0.7	0.009
3	150	$BaCl_2$	8	0.6	0.007
4	150	50KCl40$BaCl_2$10$CaCl_2$	16	0.7	0.007
5	150	item	17	0.6	0.002
7	150	item	13	0.7	
8	150	item	15	1.8	
9	110	item	16	2.0	0.011
10	140	item	16	1.4	
11	120	item	17	0.4	
12	140	item	17	0.9	0.005
13	150	item	18	1.1	0.009
14	150	90NaCl 10Na_2CO_3	18	0.7	0.001
15	170	70NaCl 30Na_2CO_3	24	0.3	0.008
16	150	50KCl40$BaCl_2$10$CaCl_2$	15	0.7	0.002
17	145	10NaCl80$BaCl_2$10$CaCl_2$	16	0.8	0.003
18	120	70NaCl 30Na_2CO_3	12	0.8	0.002
19	190	90NaCl 10Na_2CO_3	14	0.2	0.002
20	150	70NaCl20$Na_2CO_3$10$BaCl_2$	0.7	0.005	BDL****
21	150	70NaCl 30Na_2CO_3	0.5	0.11	BDL

* That was an estimated value.
**It was measured after 60 days of decontamination.
*** It was measured after 120 days of decontamination.
****BDL means below the detection limit.

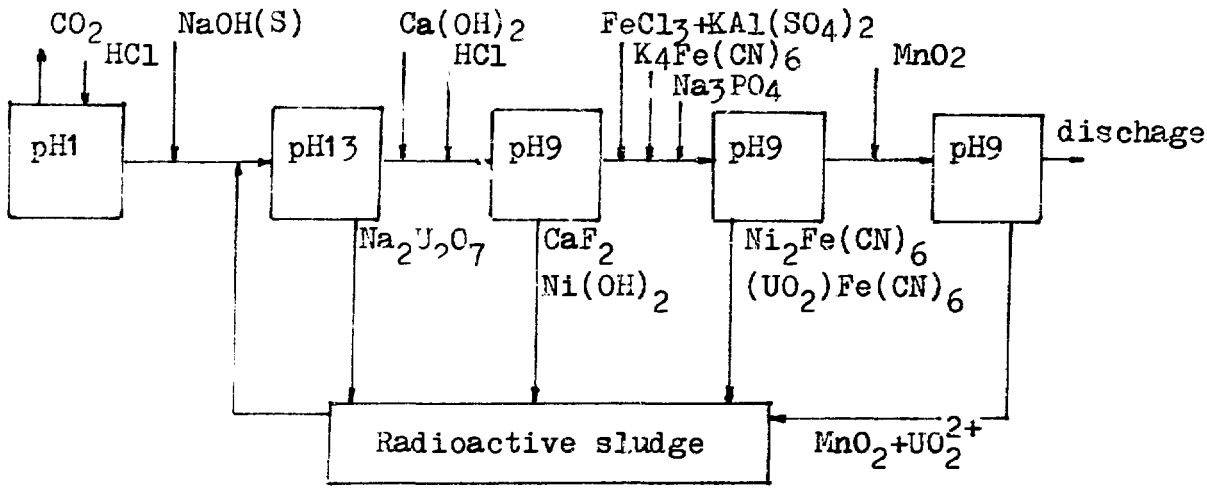
Melt temperature: 900 °C
Melt period: 10 - 20 min.

RADIOACTIVE WASTE MANAGEMENT

Liquid Waste Management

In decontamination three kinds of liquid wastes were generated. Alkaline liquid waste mainly came from decontamination of separation membrane . Contents of U, Ni, F in this waste were 4.8 g/L, 5.4 g/L, 2.8 g/L respectively. Acidic liquid waste came from decontamination of aluminium, copper and mild steel parts. Acidity was about 1M and uranium content was about 2 mg/L. Liquid waste with $Al(OH)_3$ was generated in decontamination of aluminium parts with shallow layer contamination by 10% NaOH, uranium content was near to 0.7 mg/L.

Precipitation and sorption methods were used to treat the first two kinds of waste. The flow sheet was asfollows;



Contents of U, Ni, F in the supernatant after treatment were 0.02 - 0.1 mg/L, 0.02 - 0.5 mg/L and 0.02 - 0.13 mg/L respectively.

The liqiud waste containing $Al(OH)_3$ was solidified with cement.

Solid Waste Management

Radioactive slug sent to plant to recover uranium. Combustible wastes, such as cotton, cloth, masks, gloves, wood and so on, would be burnt to reduce the volume. 21 Tons of metal parts that were difficult to decontaminate were stored as solid waste.

CONCLUSION

In the decontamination experiment and practice valuable results and experiences were obtained for the decontamination of nuclear facilities contaminated by uranium. The experiences indicated that following chemical decontamination procedures can be used:

Copper- (1) 5% $(NH_4)_2CO_3$ + 2% H_2O_2, soaking for 10-20min.
(2) 1M HNO_3, soaking for 10-20min.

Nickel- (1) 5% $(NH_4)_2CO_3$ + 2% H_2O_2, soaking for 10-20min.
(2) 1M HNO_3, soaking for 10-20min.

Aluminium alloy-
(1) 5% $(NH_4)CO_3$, soaking for 1-2hr. (for slightly contaminated parts)
(2) 2-4M HNO_3, soaking for 10-20min. (for heavily contaminated parts)
(3) 10% NaOH, soaking for 10-20min. (for shallow layer contaminated parts)

Mild steel-
(1) hand scrubbing with 5% $(NH_4)_2CO_3$ or 1-2M HNO_3
(2) 2-4M HCl, soaking for 1-8hr.

Melt refining for aluminium alloy with internal contamination:

Fluxes- (1) 14% NaF + 76% KCl +10% $BaCl_2$
(2) KCl
(3) 50% KCl + 40% $BaCl_2$ + 10% $CaCl_2$
(4) 26% NaCl + 74% Na_2SO_4

Melting temperature- 900 C. Melt time- 10-20min.

Precipitation and sorption procedure can be used to treat the radioactive liquid waste generated from decontamination. The contents of U, Ni, F in supernatant were 0.02-0.1 mg/L, 0.02-0.5 mg/L and 0.02-0.13 mg/L respectively.

ACKNOWLEDGMENTS

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DECONTAMINATION TECHNIQUES

POSTER SESSION

DECONTAMINATION FOR DECOMMISSIONING OF NUCLEAR POWER PLANT A-1

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ABSTRACT

The paper presents results of phase analysis of corrosion product layers formed on a low-alloy steel of the secondary circuit and stainless steel of explosive mixture combustion system in the NPP A-1 and results of experimental verification of decontamination efficiency of different procedures for chemical and electrochemical decontamination of materials mentioned above. A methodology is described for determination of acceptable residual contamination of metallic materials from decommissioning of nuclear power plant for their introduction to the environment after remelting and reprocessing.

INTRODUCTION

The Nuclear Power Plant A-1 (NPP A-1) with CO_2 cooled and heavy-water moderated reactor with 150 MWe output was definitely shutdown in Febr. 1977, after 5 years of operation. Considering existing technological base, availability of waste management and disposal facilities in Czechoslovakia the NPP A-1 is currently under decommissioning to the first upgraded stage (according to the IAEA classification), with partial dismantling of the secondary and selected auxiliary circuits. Among operative decommissioning problems, the issues concerning decontamination of materials from the circuits and metallic LLW management have to be studied. The objectives of the research were:

a) to characterize nature of contaminated corrosion layer on both secondary and auxiliary circuit surfaces,

b) to develop chemical and electrochemical decontamination techniques enabling reduction of personnel exposure during dismantling of equipments or material decontamination on the level permitting its unrestricted reuse,

c) to elaborate fundamental approach to assessment of residual contamination level of metallic materials for their unrestricted reuse.

CHARACTERISTICS OF CONTAMINATED CORROSION LAYER

For decontamination experiments and also for characterization of corrosion products layer by optic microscopy, SEM, X-ray diffraction and Moessbauer spectroscopy samples were taken from different parts of secondary circuit (carbon steel) and from the inlet pipe of the explosive mixture combustion system (stainless steel). The samples were cut to the size of 30 x 30 mm.

Results of phase analysis have proved [1] that corrosion layer on carbon steel surface can be divided into two specific parts:

a) upper layer is thin, compact. It consists of substituted magnetite with higher content of Cu, Mn and Zn.

b) lower layer, adhering to the base material represents a bigger part of the entire corrosion layer. This part consists of compact, mechanically firm magnetite (thickness from 0.1 to 0.3 mm) with different degree of nonstoichiometry and smaller portion (up to 30 %) of hematite, goethite and small particles of oxides. The whole layer is unhomogeneous.

A part of the material has contaminated layer loosely bound to the base material. The layer is formed by remarkably

nonstoichiometrical magnetite and by comparable amount of iron oxides in the higher oxidation stage and oxides in the form of small particles. Contamination is represented by at least 94 % of ^{60}Co and the rest is ^{137}Cs.

The contaminated layer of stainless steel contains a small portion of iron oxides (cca 4 %) and from 40 % to 50 % of another compounds of Fe^{2+} and Fe^{3+}. A presence of $FeC_2O_4 . 2H_2O$ was proved by X-ray analysis. On a basis of analyses results and determination of metals in decontamination solutions we can presume that contamination layer is formed by metallic compounds (Fe, Ni) with radiation-chemical products of D_2O and entraining CO_2 radiolysis.

Mean specific activity of samples $a_s = 700 \pm 270$ Bq.cm^{-2} with a main part of ^{60}Co and ^{137}Cs.

CHEMICAL DECONTAMINATION

For experimental verification of efficiency we have chosen decontamination agents referred to in the literature [2-4] as successful for low-alloy and stainless steels and solutions, whose composition was based on better familiarity with processes of corrosion layers dissolving. An influence of ultrasonic (US) impact on decontamination efficiency was experimentally verified at simultaneous treatment with ultrasound and the agent and also the impact of ultrasound after chemical decontamination of samples.

Experimental Procedure

In experiments, 21 cm^3 of decontamination solution per 1 cm^2 of sample was used. The decontamination process was carried out in a glass beaker placed into thermostat. The samples were rinsed with distilled water after extracting from the decontamination solution and dried by hot air or ultra-

sonically treated finally in glass beaker, placed into a laboratory ultrasonic tank.

Decontamination efficiencies were evaluated by decontamination factor (DF) and by residual count rates of the samples (n_{res}) after the decontamination.

Test Results and Discussion

Decontamination of Low-Alloy Steels. The decontamination efficiency by oxalic acid of 10 - 50 $g.dm^{-3}$ concentration at T = 90 °C and exposure up to 3 hours, was low and DF < 3. The efficiency was increased by addition of H_2O_2 or H_2SO_4 respectively, but in spite of this, it was still low.

The DF $\leq$ 2.8 was obtained with the mixture of acids, such as: HNO_3, H_2SO_4, H_3PO_4, concentrations within 0.2 - 1.0 $mol.dm^{-3}$, at T = 50 °C and 3 hours exposure. The samples, mainly the bulk material, were considerably damaged after the decontamination and local deep corrosion of the base material was visible.

The results of performed decontamination tests in the reducing mixture of CH_2O-HCOOH are reported in Tab. I and Tab. II.

Table I Decontamination tests in CH_2O-HCOOH mixture at 50 °C with subsequent ultrasonic treatment

Solution conc. /mol.dm^{-3}/	t = 3 hours		t = 5 hours		+ 0.5 hour US in H_2O	
	Δm /%/	DF	Δm /%/	DF	Δm /%/	DF
CH_2O /0.48/ HCOOH /0.48/	1	2.97	1.8	25.8	1.81	60
CH_2O /0.48/ HCOOH /0.24/	0.48	2.6	1.85	42.5	1.94	112
CH_2O /0.24/ HCOOH /0,48/	1.08	2.86	1.9	45.7	1.94	86.8

Δm - mass loss of a sample after decontamination as percentage of the initial mass

Table II Decontamination efficiency in the CH_2O [0.48 mol.dm^{-3}] - HCOOH [0.48 mol.dm^{-3}] solution at 20 °C

Time [hour]	Δm [%]	DF
5	0.03	1.24
48	1.9	74
72	2.2	142
300	3.75	27.2

In the process of chemical decontamination not only dissolving of corrosion layer occurred but also its cracking and peeling off so thin scales were deposited on the bottom of beakers.

The decontamination efficiency of CH_2O-HCOOH solution can be enhanced 2 - 3 times by subsequent ultrasonic treatment of samples in pure water after their chemical decontamination (Tab. I). Ultrasonic treatment increased DF on already damaged corrosion layer after chemical treatment but had not increased the corrosion effect.

The decontamination efficiency of this solution was also very good at the laboratory temperature but the long exposure of samples was necessary. We have observed that after a certain exposure time a recontamination occurred . The decontamination results obtained after 3 days showed a possibility to use this solution even at the decontamination of NPP components and surfaces of circuits prior to dismantling.

On a basis of favourable preliminary results with the solution of H_2SO_4 - $Na_2S_2O_3$, where reducing agent had positive effect on dissolution of the corrosion layer and it was the only possible source of e^- for reduction of Fe^{3+} in the oxide

layer, we have used for this purpose elemental sulphur. The results of planned experiment with H_2SO_4-S-$\dot{E}$DTANa_2 carried out in order to obtain dependence of n_{res} and DF on composition of the solution are in Tab. III.

Table III Dependence of DF on composition of H_2SO_4-S-EDTANa_2 solutions
T = 50 °C, t = 3 hour and 0.5 hour ultrasonic treatment in clean water (50 °C)

Sample No	Solution concentration /⁻mol.dm^{-3}_7			Δm /⁻%_7	DF
	H_2SO_4	S	EDTANa_2		
1	0.5	0.062	0.008	2.01	8.1
2	0.5	0.062	0.018	1.985	7.46
3	0.5	0.25	0.008	2.25	21.05
4	0.5	0.25	0.018	2.12	43.6
5	1.0	0.062	0.008	2.48	41.15
6	1.0	0.625	0.018	2.49	36.08
7	1.0	0.25	0.008	2.97	29.2
8	1.0	0.25	0.018	2.48	69.6

Obtained regression equations after conversion from dimensionless data x_i to real concentration data /⁻mol.dm^{-3}_7 in coordinates Z_i have the following form:

$$y(n_{res}) = 1\,278.3 - 1\,028.7\,Z_1 - 3\,968.73\,Z_2 + 9\,661.2\,Z_3 + 3\,813.7\,Z_1Z_2 - 13\,712\,Z_1Z_3 + 2\,986.7\,Z_2Z_3 \quad (1)$$

The values of coefficients at parameters Z_1, Z_2, Z_3 and at interaction member Z_1Z_2, Z_2Z_3 in the equation (1) prove that all 3 components of the solution take part on the total decontamination effect. The given regressive equation can be used for the decontamination process optimization.

From the DF values obtained after further decontamination in the given solutions we can further state that, in general, it is possible to achieve DF = 100 - 1000 (even more) at various concentrations in the given concentrations range after 3 - 5 hours decontamination.

The decontamination of samples in different agents was not effective enough to reach a residual level of contamination $n_{res} \leq 0.37$ Bq.cm^{-2} even if the ultrasonic treatment has been applied afterwards. That is why an influence of simultaneous action of solution and ultrasound, alternative impact of chemical agents and ultrasound in water and an influence of decontamination solutions specific activity on DF and n_{res} was investigated.

The results have shown that by alternative action of the agent and ultrasound $n_{res} \leq 0.37$ Bq.cm^{-2} can be achieved on that material in H_2SO_4 (1 mol.dm^{-3}). An efficiency of H_3PO_4 and solutions based on HCOOH + $EDTANa_2$ was not sufficient. The process of recontamination has been observed after longer exposure to the agents. At the decontamination with solutions already used, especially with lower concentrations. the lower DF were obtained. Utilization of fresh solutions at repeated decontamination of the samples enabled to reach $n_{res} \leq 0.37$ Bq.cm^{-2}.

A comparison of decontamination effects of solutions H_2SO_4 and H_3PO_4 without and with simultaneous ultrasonic treatment showed that ultrasound accelerates the decontamination process in average 2 - 3 times and this speeding up depends on an intensity of ultrasonic field.

Decontamination of Stainless Steel. With regard to the characteristics of contaminated corrosion layer on stainless steel from the explosive mixture combustion system, the oxidizing and oxidizing-reductive agents such as: HNO_3 (8 g.dm^{-3}),

CH_2O (15 $g.dm^{-3}$) + CHOOH (10.34 $g.dm^{-3}$) + $EDTANa_2$ (3.36 $g.dm^{-3}$) or NP (nitric acid - permanganate) + Citrox respective were successful for decontamination prior to dismantling.

ELECTROCHEMICAL DECONTAMINATION

Experimental Procedure

Experiments were carried out in standard arrangement, where the potential of the working electrode was measured against SCE, and conductance with temperature were followed-up continually during the electrolysis [1]. The electromagnetic stirrer with heating agitated and heated the solution to the required temperature. After electrolysis the electrodes were pulled out under voltage from the solution, rinsed with water, dried and activity was measured.

Test Results and Discussion

Decontamination of Low-Alloy Steels. During anodic oxidation and cathodic reduction in the electrolytes on the basis of HNO_3, H_2SO_4, H_3PO_4 and oxalic acid, decontamination efficiency was observed depending on the process main parameters, as e. g. electrolyte concentration and temperature, current density and duration of electrolysis.

Cathodic reduction in the 10 % H_2SO_4 and in the mixture of 10 % H_2SO_4 with 10 % H_3PO_4 was successful in the aspect of decontamination. Regarding the properties, in the further experiments H_3PO_4 was used as an electrolyte.

H_3PO_4 increase was demonstrated in increased rate of corrosion layer dissolution and decontamination efficiency. DF = 258 was reached in 40 % (vol) H_3PO_4 after 20 min. of electrolysis.

Sample mass losses after the decontamination were up to 6 %, depending on the electrolyte concentration and duration of the electrolysis.

From the effect of electrolyte (20 % H_3PO_4) temperature, at current density i = 0.5 $A.cm^{-2}$, on the time dependence of the sample residual activity (n_t) it follows that the most suitable electrolyte temperature was about 80 °C. Non-linear regression analysis of different curves $n_t = f(t)$ showed that at given flow density and material, at 80 °C, independently on acid electrolyte type and its concentration, the following polynomial best fitted the curves

$$\log n_t = a - ct^2 + et^4 \qquad (2)$$

The shape of time dependence of residual activity was changing at lower temperature, so its course was most suitably described by following polynomial

$$\log n_t = a - bt + ct^2 \qquad (3)$$

Solution of polynomials for real cases (known coefficients a, b, c, e) enables to determine duration of decontamination necessary for obtaining required residual contamination. Necessary condition for solution existence is, that for given polynomial, following condition is satisfied for big t; $n_t = f(t) = 0$.

Results from the performed experiments showed [1] that it was possible by use of electrochemical decontamination to reduce the residual contamination level up to the level of specific activity, allowing unrestricted use of the material off the controlled zone. Necessary time of the electrochemical decontamination for given electrolyte depends mainly on the size of current density and thickness of the corrosion layer that is to be dissolved.

For the given material and current density of 150 $A.dm^{-2}$ this time was 30 - 60 min. At the given current density the DF value was increasing with time and for the time given it was increasing with the current density, but the dependence was not linear.

Decontamination of Stainless Steels. Effect of electrode polarity, flow density, type, concentration and temperature of electrolyte on decontamination efficiency was studied for contaminated stainless steels from the explosive mixture combustion system. Experiments were carried out with samples not decontaminated as well as with chemically decontaminated ones, in order to verify effect of previous chemical decontamination on electrochemical decontamination efficiency.

From the results obtained at anodic and cathodic polarization of samples as well as in regime with polarity reversation in one minute intervals was obvious that anodic oxidation and regime with polarity reversation are for decontamination rate more effective than regime of cathodic reduction. For samples previously not decontaminated, background level at anodic oxidation or polarity reversation was reached after 5 - 6 min, for samples chemically decontaminated it was reached later, after minimum 14 min.

Regardless chemical composition of decontamination solution, previous chemical decontamination negatively effected DF time dependence at following electrochemical decontamination.

Effect of flow density observed in the range of 0.1 - 0.5 $A.cm^{-2}$ has influenced decontamination rate. For samples previously chemically decontaminated, the background level has been reached in 10 % H_3PO_4, at T = 80 ^{o}C and i = 0.5 $A.cm^{-2}$ in minimum 14 min.; at i = 0.1 $A.cm^{-2}$ after the same time about 4 % of original activity as n_{res} was obtained.

The dependence of decontamination efficiency on electrolyte temperature is significant. With increasing temperature (up to 80 °C) time needed for obtaining given DF is decreasing.

RELEASE OF MATERIALS FROM NUCLEAR INSTALLATION DECOMMISSIONING TO THE ENVIRONMENT

Decision on equipment or material reuse can be based on assessment of acceptable limits for radionuclides activity in material given either per weight or surface units.

There are the following basic aspects for assessment of the acceptable limits of radionuclide activity in material released for reuse (recirculation):

- assessment of limit for contribution to the total dose to public (annual effective dose equivalent) caused by this performance,
- preparation of "scenarios" describing the most probable pathways for material release to the environment (recirculation procedure),
- calculation of maximum permitted activity of individual radionuclides in the released material, by which any limits of annual dose equivalent will not be exceeded at any stage of its use according to considered scenarios.

Regarding the potential increasing amount of released material from the decommissioning of nuclear installations it seems useful to regulate even the total amount of released material (total amount of radionuclides). Assessment of limit collective dose equivalent from material release seems to be very suitable for this purpose.

Detailed analyses of present and anticipated radiation sources to public in Czechoslovakia as well as evaluation of approaches to this issue in other countries led to recommendation of following criterion for material release:

- annual effective dose equivalent to an individual from the

critical population group will not exceed limit value of 10 μSv.

Scenario of Material Release to the Environment

In the first stage we considered the following scenario: transportation of material from NPP – interim storage at municipal site – melting in the furnace – unlimited use of remelted material. The main attention was focused on evaluation of the risks to the public from the material on the site and remelting in furnance [1, 5].

Three potential exposure pathways to the public were considered during the interim storage of materials:
- external irradiation from material,
- radionuclide release from material via soil to water sources,
- release to the atmosphere.

Effect of material melting as well as effect of resulting remelted material depends on behaviour of radionuclides present in melting process. Three boundary cases were considered:
- all radionuclides are during melting released to the atmosphere,
- the whole radionuclides inventory stays in the resulting material (metal) after remelting,
- the whole radionuclides inventory stays in slag.

Apart from this a case with theoretical values of homogeneous radionuclide distribution among all three components (i. e. metal, slag, atmosphere) was considered.

Immediate impact of material remelting on environment is connected with radionuclide release to the atmosphere during melting. In the further stage, radiological impact of other by-product of remelting – slag – was assessed.

Up to now, wide possibilities of unlimited use of material by population in the form of objects for everyday use (cars, furniture, dishes) were not assessed. In order to

get basic orientation of the range of radiological burden to the public caused by material application we narrowed an application mode to one case: Steel from material remelting will be used for building construction. We assumed its build into concrete foundation of service rooms with annual stay of 2000 hours. Regarding building practice the considered dilution factor of iron in concrete was 0.25.

Conclusion on the Base of Scenario Evaluation. Results of verifying calculations showed principial applicability of our scenarios for assessment of effect of uncontrolled release of material from NPP decommissioning to the environment or at its reuse. In spite of interim character of calculations, following conclusions can be formulated from the results obtained (on the relation to the annual limit of the effective dose equivalent of 10 (uSv):

a) In order to lower risk annual dose equivalent limit it seems suitable to eliminate interim off-site storage of material prior its processing. Material should be transported directly from NPP to metallurgical plant and it should be immediately processed there.
b) Important risk is represented by slag material. Possibilities of its use will be limited by level of radionuclide concentration in slag as well as by its physical and chemical properties. It would be useful to consider slag storage at shielded controlled sites.
c) Regarding the very conservative condition of calculations, the material can be released without any risk at radionuclide activity of 1 - 10 $Bq.g^{-1}$ and at total value of surface contamination of 4 $Bq.cm^{-2}$.
d) Value of collective dose equivalent will be decisive for assessment of the total amount of material that can be released to the environment. It appears necessary to include the other activities connected with material release into

the calculation of collective dose equivalent (preparation to transportation, loading and unloading of material, material handling at furnace etc.).

Data on material melting characterization contain considerable conservativism. So in order to obtain reliable data on radionuclide behaviour by remelting, experimental material melting with very low radionuclide contents is under preparation.

CONCLUSION

This report presents results of phase analysis of corrosion products on carbon steel from the secondary circuit and stainless steel from the explosive mixture combustion system in NPP A-1, together with results of research in the field of decontamination efficiency dependence of various procedures of chemical and electrochemical decontamination of given materials. It also gives methodology of acceptable residual contamination assessment for metallic materials from NPP A-1 decommissioning and their introduction to the environment after remelting.

The layer of corrosion products formed on carbon steel of the secondary circuit consists of a layer of compact, mechanically stable magnetite (thickness 0.1 - 0.3 mm) with different degree of non-stoichiometry and substitution of Fe-ions (Zn, Cu, Mn) with portion (up to 30 %) of maghemite, hematite, goethite and small particles of oxides. Corrosion layer is unhomogenous, a part of material has less adhering corrosion layer to the base material.

The contaminated layer on stainless steel from the explosive mixture combustion system contains iron oxides (relatively 4 %) and other compounds with Fe^{2+} and Fe^{3+}. RTG analysis determined presence of $FeC_2O_4.2H_2O$ in the layer.

Decontamination of samples from secondary circuit was very

successful with the "reducing" solutions of CH_2-HCOOH at 50 °C and H_2SO_4 + S + $EDTANa_2$.

The effect of ultrasound on removal of chemically undamaged layer is relatively low. US caused 2 - 3 times increase of DF on damaged corrosion layer after decontamination, without an increase of corrosion effect. Effect of US decreases with increasing sample-US transducer distance.

Combined treatment by agent (H_2SO_4) and US enables to obtain residual contamination level lower than 0.37 $Bq.cm^{-2}$.

Material from the explosive mixture combustion system can be decontaminated in HNO_3 solution (7.5 $g.dm^{-3}$); it can be followed by subsequent decontamination with simultaneous US treatment in H_3PO_4 solution.

The electrochemical cathodic reduction is effective for decontamination of materials from carbon (low-alloy) steel, whose surface is covered with cca 150 μm thick corrosion layer, consisting mainly of magnetite.

Regarding the obtained efficiency of these materials, as suitable electrolytes appeard solutions of H_3PO_4 and H_2SO_4 or their mixture.

Electrolyte temperature significantly effects rate and efficiency of electrochemical decontamination. The most suitable temperature is about 80 °C. Rate efficiency of the process decreases at lower temperature.

Anodic oxidation and regime with polarity reversation, as for decontamination rate, is more effective than cathodic reduction regime for stainless steel from the explosive mixture combustion system.

In the methodology for assessment of acceptable residual activity on material the preliminary exposure of the public was assessed for the given material and its surface contamination.

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CHEMICAL DECONTAMINATION OF THE IN PILE "CART-TC" LOOP FOR DECOMMISSIONING PURPOSES, SOME EVALUATION ON THE MANAGING ASPECTS OF THE AGGRESSIVE SOLUTIONS IN RADIOACTIVE AREAS

by

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Abstract

The paper briefly describes the experience made and the lesson learned in testing both soft and hard chemical solutions in the decontamination of the in pile CART-TC loop, located in ESSOR reactor at the European Communities JRC of Ispra ITALY.

The main objective of the work is to extend laboratory tests to a real small scale loop in order to get operating experience on chemical decontamination of significant nuclear reactor systems and to evaluate the real advantages and limits of this technique in decommissioning activities.

The main investigations were conducted in order to get knowledge on:

- testing both commercial and experimental chemical solutions
- operating problems in aggressive environment
- waste reduction and management

and, as a conclusion, to make an evalutation on the extension of this technique to the power plant. The main results are presented.

The research programme was set up by ENEA and CISE, and is carried put by CISE assisted by ESSOR reactor's staff.

1. Introduction

A common feature of both experimental and power boiling-water reactors is the formation of oxides containing activated particles or fission products on all surfaces affected by the coolant flow. These oxides may either be in the form of a compact mass or in powder or slag form, depending on the coolant temperature and speed. This behaviour of the oxides gives rise to a uniform exposure rate in simple geometry zones (especially if operated at high temperatures). In complex geometry zones or in zones affected by a low-speed coolant flow (especially if low temperatures are involved), a considerable exposure rate nonuniformity may by encountered.

A nuclear plant dismantling must primarily take into account the exposure rate in the different parts of the plant so that the job may be organized in such a way as to minimize the integrated exposure rate. Due consideration should also be given to the fact than once these loops are open active particulates with highly volatile particles of less than 1 /um may be released. This circumstance will further complicate dismantling operations both as regards the safety of the personnel and the possible contamination of the area involved.

These reasons prompted a number of researchers to study and test chemical decontamination procedures for inner nuclear plant surfaces.

The following points were investigated during the course of the experiment described in this paper.

- the Decontamination Factor (D.F.) obtained by using different chemical

solutions;

- the expenditure in terms of dose equivalent adsorbed during the course of the chemical decontamination,
- the dose equivalent adsorbed on the assumption of a dismantling with and without preliminary decontamination,
- an evaluation of the secondary waste management.

2. Plant description

The plant on which the test is carried out is called CART T.C., which stands for Cirene Assembly Reactor Test Twin Channel, an experimental nuclear plant located in the Ispra CCR Euratom Essor reactor. This plant, commissioned in 1968, has worked for about 4 years with a single channel set in the reactor core. Subsequently, after a modification which doubled its radiation capacity and consisted in the installation of a second channel, the plant operated from 1976 until the end of 1980.

Technical particulars (see Fig. 1):

- Totale thermal power: 3 MW
- Maximum flow rate for channel: 12 t/h
- Design pressure: 100 kg/cm2
- Maximum operating pressure: 60 kg/cm2
- Coolant conditions at the reactor core output: 254 °C, two- phase at a max. 15% steam
- Coolant circulation: open loop with reciprocating pumps and pressure control valves
- Expansion tank with stripping functions
- Low-pressure (10 kg/cm2) and low-temperature (50°C) chemical section.

3. Decontamination tests

With the radiological situation described in Fig. (1) (the numbers appearing in Fig. (1) and in the following figures represent the dose equivalent rate in /uSv/h), six tests were carried out using both soft and hard acids. The plant was divided into different zones by means of acid and pressure-resistant flexible joints. This division was due to the need of investigating the behaviour of oxides with a differtent morphology due to their having grown at different coolant temperatures:

- between 50 and 60 °C
- between 130 and 150 °C
- between 250 and 260 °C.

For the purpose of a "real time" monitoring of the behaviour od the oxide involved, a corrosion test section based on the linear polarization technique Fig. (1) was set up, in which different pipe specimens were munted for each new test.

The functions of preparing, pumping, heating, purification through filters and resins and final collection of the test solution were carried our by the plant's own equipment.

Systems that did not fully meet test requirements were modified accordingly.

The first configuration to be carried out affected a large part of the plant, a part common to all the other configurations which includes the pumping assembly, heater, expansion and stripping tank and the chemical section. Whith the nuclear plant in operation, most pipes and components of this zone worked at temperatures rangin from 50 to 60 °C whilst the heater temperature ranged from 240 to 254 °C. Soft acid was used for decontamination purposes and the temperature did not exceed 90 °C Fig. (1).

The configuration involved and the dose equivalent intensity values at the end of the test are shown in Fig. (2).

The second configuration concerned a piping section operates at 130 - 150 °C which was treated with the same soft acid but at a temperature of 120 °C.

Fig. 3 shows the configuration and the results in terms of dose equivalent rate values.

The third test was carried out on a configuration concerning a zone operated at 250 °C. The temperature of the solution used was 120 °C.

Fig. 4 shows the configuration and the test results in terms of exposure rate.

The fourth and fifth tests were carried out on the fourth configuration. Low-concentration hard acids were used. The results in terms of dose equivalent rate in the different locations of the plant are shown in Fig. 5. Two values are recorded for each measuring point: the left-hand value refers to test N° 4 and the right-hand value to test N° 5.

Test N° 6 concluded the first test run and was carried out on a plant configuration which included almost the whole primary loop. A high-concentration ($\sim$ 4%) hard-acid solution was used. The results of this test are shown in Fig. 6.

The figures show that in simple geometry zones there is a significant decrease in dose equivalent rate values whereas an increase in activity is noted in complex geometry zones and in zones with very low coolant speeds.

In order to verify the reliability of the activity values measured in the corrosion test section, a pipe specimen of about 100 mm was witndrawn during the course of a decontamination test.

The specific activity measurements made on this pipe specimen have

confirmed the validity of the corrosion test section measurements.
The above pipe specimen interior was examined by means of the smear-test method. The results thus obtained showed that the transferable contamination and especially the airborn migratory part of it is almost nil.

4. Modification of the plant for test purposes

4.1 Plant function restoration

In 1980, upon completion of the irradiation programme, the plant was left in a stand-by condition and full of demineralized water with a good conductivity value. After about five years of inactivity however, many plant functions, especially the control functions, had to be restored in the original operating conditions by replacing or repairing the damaged parts.

4.2 Accommodating the plant to meet test requirements.

The work carried out in order to make the plant suitable for decontamination tests included following items:

- the heater designed to operate at three power levels (100 - 200 - 300 kW) without regulation was provided with regulation facilities for each power level,
- The demineralized water feed tanks were insulated and provided with a 25 kW electric heater for the purpose of heating the soft acid solution,
- a corrosion test section was installed behind the heater for the purpose of monitoring corrosion parameters. The corrosion test section

was provided with two gamma activity sensors enclosed in a demountable 2-ton lead shield for whose installation the floor has to be suitably reinforced. Arrangements were also made for cooling the gamma sensors located inside the lead shield,

- the chemical section had to be converted in order to make it suitable for the in-line purification of the soft acid solution,
- in addition to the restoration of existing gamma chains, further gamma chains were installed in strategic points of the plant for the purpose of monitoring the gamma activity course directly on the pipes and components involved,
- many sealing joints between pipes and components had to be modified in order to mount the flexible joints used for setting up the different configurations.

4.3 Test configuration set-up

Two kinds of armoured 2 to 7-metre long flexible joints were used for setting up the various test configurations:

- teflon joints
- ethylene-propylene joints.

The former, which are both acid and heat-resisting, feature a certain rigidity after their first usage so that once used they cannot be reused for a different configuration. The latter, which maintain their flexibility after many hours of continuous duty, show their limitation at around 110 °C even if used only for soft acids.

It has consequently been necessary to optimize the performance of both types in order to keep the secondary waste production within reasonable limits.

4.4 Test procedure

Tests were preceded by a preparatory stage and followed by a conclusive stage. The following are the typical test stages common to all decontamination tests.

4.4.1 Pipe specimen withdrawal and subsequent preparation for their insertion into the corrosion test section.

4.4.2 Gamma count be means of the (Ge-Li) spectrometer and weighing of the pipe samples.

4.4.3 Mounting the pipe specimens in the corrosion test section.

4.4.4 Filling the unit with demineralized water.

4.4.5 Forced circulation for seal testing.

4.4.6 Forced circulation of demineralized water at operating temperature for temperature regulation facility setup.

4.4.7 Lead shield erection around the corrosion test section and gamma sensor cooling system startup.

4.4.8 Flowrate adjustment in the chemical section.

4.4.9 Chemical and radiological instrument setup.

4.4.10 Stopping plant and emptying the water into the effluent tanks.

4.4.11 Filling the unit with an acid solution at 90°C (prepared about 12 hours before the beginning of the test).

4.4.12 Acid solution circulation and degassing.

4.4.13 Heater starting and beginning of test.

4.4.14 Whilst the test is in progress, data from the corrosion test section is monitored, solution samples are withdrawn (from 10 to 30 sample withdrawals per test) and gamma sensors are switched on for the purpose of monitoring the activity progress in a significant piping zone.

4.4.15 Heater turnoff at the end of the test; emptying the solution

into the effluent tanks.

4.4.16 Corrosion test section isolation and rinsing.

4.4.17 Rinsing the unit interior with demineralized water (almost four m3).

4.4.18 Filling the unit with demineralized water and circulation via filters and resins.

4.4.19 Lead shield and corrosion test section dismounting.

4.4.20 Gamma count and weighing of pipe specimens in the corrosion test section.

4.4.21 Gamma mapping of the most significant plant zones involved in the test.

4.4.22 Neutralization with sodium carbonate of the solution in the effluent tanks until a pH ranging from 6 to 7 is obtained.

4.4.23 Treatment of the acid solution in the plant storage tank for onward conveyance to the Ispra Treatment Station.

All these operations, excluding the test itself, employ four persons and take on the average about 10 working days.

5. Trouble caused by the acid solution management

Only troubles of a certain significance are listed.

- Chemical section

  Clogging and subsequent breakage of the sinterized porous filter of one of the two resin beds, with resin pouring into the plant.

- Reciprocating pumps

  Damage of the main pump seals and chromium-plates and ground pistons.

Major repairs were required to put the pumps back into operation.

- During the preparation of a soft acid test, whilst the seals were tested with demineralized water only, and ethylene-propylene joint slipped off. This brought about the contamination of a considerable part of the casemate where the loop is located.

- During the course of a hard acid test, acid solution was spilled on the lower level of the casemate floor, because of a damaged secondary pump seal. As a consequence of this accident, neutralization, washing and final decontamination operations had to be carried out.

- Acid vapours escaping from above damaged pump seals have put one of the pump control cubicles out of action, following the damage suffered by its electronic components.

6. Active waste

Active waste produced during these tests was chiefly of the liquid type and consisted of a fair amount of water containing active nuclides both in solution and in suspension. The suspended particulate was separated by decantation and the remaining liquid was neutralized making sure that specific activity and pH do not exceed the Ispra CCR Treatment Station acceptability levels. The particulate was packed in special containers and sent to the Ispra Treatment Station for subsequent cementation or bituminization. During the course of the next test run, several liquid waste pretreatment procedures will be tried out with the view of minimizing the work of the Ipsra Treatment Station.

7. Dose equivalent assessment from the beginning of 1985 to the middle of 1987

During the period under review, four persons worked for about 1300 man-hours in a casemate zone where the average dose equivalent rate amounted to about 10 /uSv/h (conservative estimate).
During the same period of time the same four persons worked for about 150 man-hours in another casemate zone where the dose equivalent rate amounted to about 100 /uSv/h (conservative estimate).
To date, the total test expenditure in terms of dose equivalent rate is $2.8.10^{-2}$ man-Sv (2.8 man-rem).

8. Conclusions

From the dose equivalent rate values appearing in the figures it follows that the chemical decontamination of a plant inner surfaces is effective in simple geometry zones, i.e. pipes and such-like. On complex components and in zones where the coolant speed is comparatively low, concentrations of particles that loosened themselves from the simple geometry zones may be found.
For this reason, a second test run will commence before long, with the view of carrying out the chemical decontamination of complex plant components with the aid of the same plant pumping and heating functions and of the same solutions used during the first test run.
The dose equivalent of the second test run is estimated to be in the same order as that of the first test run, i.e. about 3.10^{-2} man-Sv.
Several considerations may be made starting from two opposite assumptions, namely:

- dismantling without preliminary decontamination,

- dismantling with preliminary decontamination.

In the first case dismantling will have to be carried out by using special protective suits and masks. Personnel will have to work in the "hottest" casemate zone (see Fig. 1 for dose equivalent intensity values) for about 1000 man-hours and the dose equivalent adsorbed in that particular zone alone would amount to about 3.10^{-1} man-Sv (30 man-rem).

After the preliminary decontamination, personnel will work in the same zone without special protective suits for about 800 man-hours (dismantling time is obviously shorter than in the first case) at an average exposure intensity of about 30 /uSv/h whilst the integrated dose equivalent will amount to about $2.4.10^{-2}$ man-Sv (2.4 man-rem).

To this latter value must be added the dose equivalent required to complete the chemical decontamination of the plant which is estimated to amount by the end of the test cycle to about 6.10^{-2} man-Sv (6 man-rem).

Consequently, the total expenditure in terms of adsorbed dose equivalent for a dismantling with preliminary chemical decontamination would amount to $8.4.10^{-2}$ man-Sv (8.4 man-rem).

This value could be further reduced if proven rather than experimental techniques were to be used. On the other hand, a careful evaluation will have to be made of the secondary waste invclved and of the cost of the modifications which will have to be carried out in order to make the plant suitable for this kind od treatment. As regards the latter figure the type and quality of these accommodations will have to be taken into account:

- a simple accommodation from both a constructive and procedural viewpoint is comparatively inexpensive but may be a source of possible accidents whose cost can obviously not be estimated;
- an accommodation based on normal nuclear constructive criteria may

practically reduce the probability of accidents to zero but its cost would be so high as to make any type of chemical decontamination prohibitive.

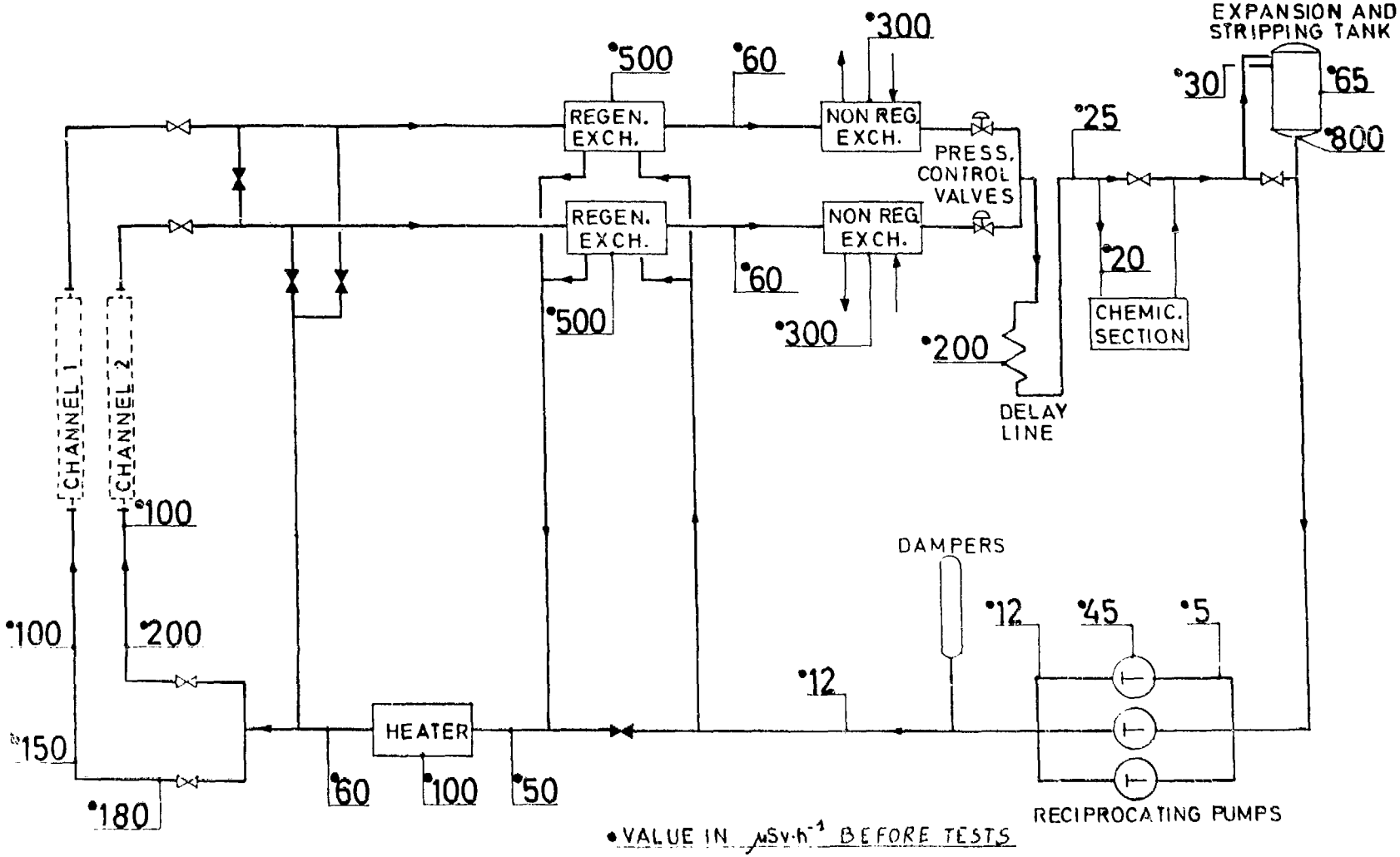

EXPANSION AND STRIPPING TANK
•30
•65
•800
•300
•500
•60
REGEN. EXCH.
NON REG. EXCH.
PRESS. CONTROL VALVES
•25
REGEN. EXCH.
NON REG. EXCH.
•60
•20
•500
•300
•200
CHEMIC. SECTION
DELAY LINE
CHANNEL 1
CHANNEL 2
•100
DAMPERS
•12
•45
•5
•100
•200
•12
HEATER
•150
•60
•100
•50
•180
RECIPROCATING PUMPS
• VALUE IN μSv·h⁻¹ BEFORE TESTS

FIG. 1

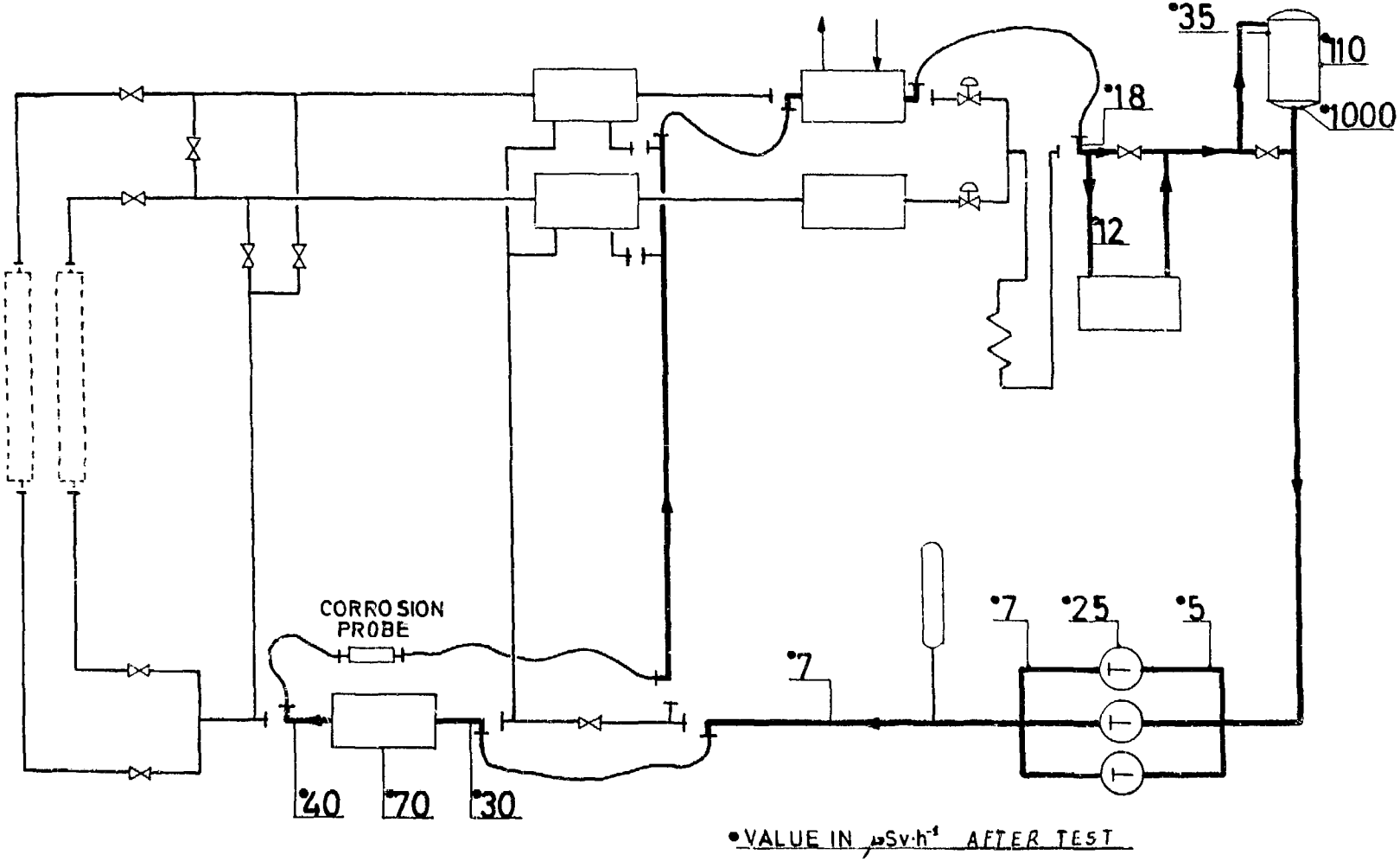
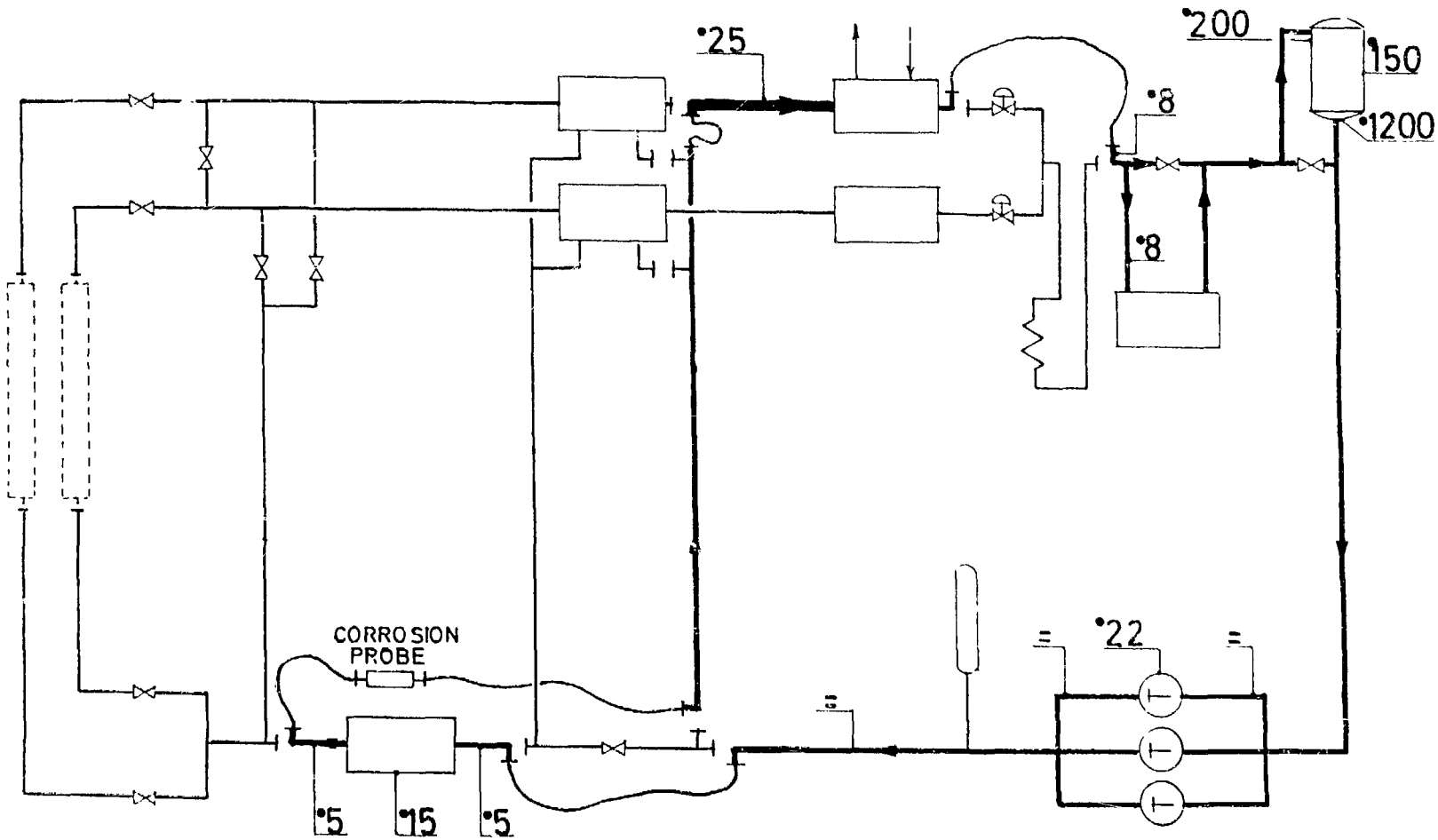

•35
•10
•1000
•18
•12
CORROSION PROBE
•7
•25
•5
•7
•40
•70
•30
•VALUE IN μSv·h⁻¹ AFTER TEST

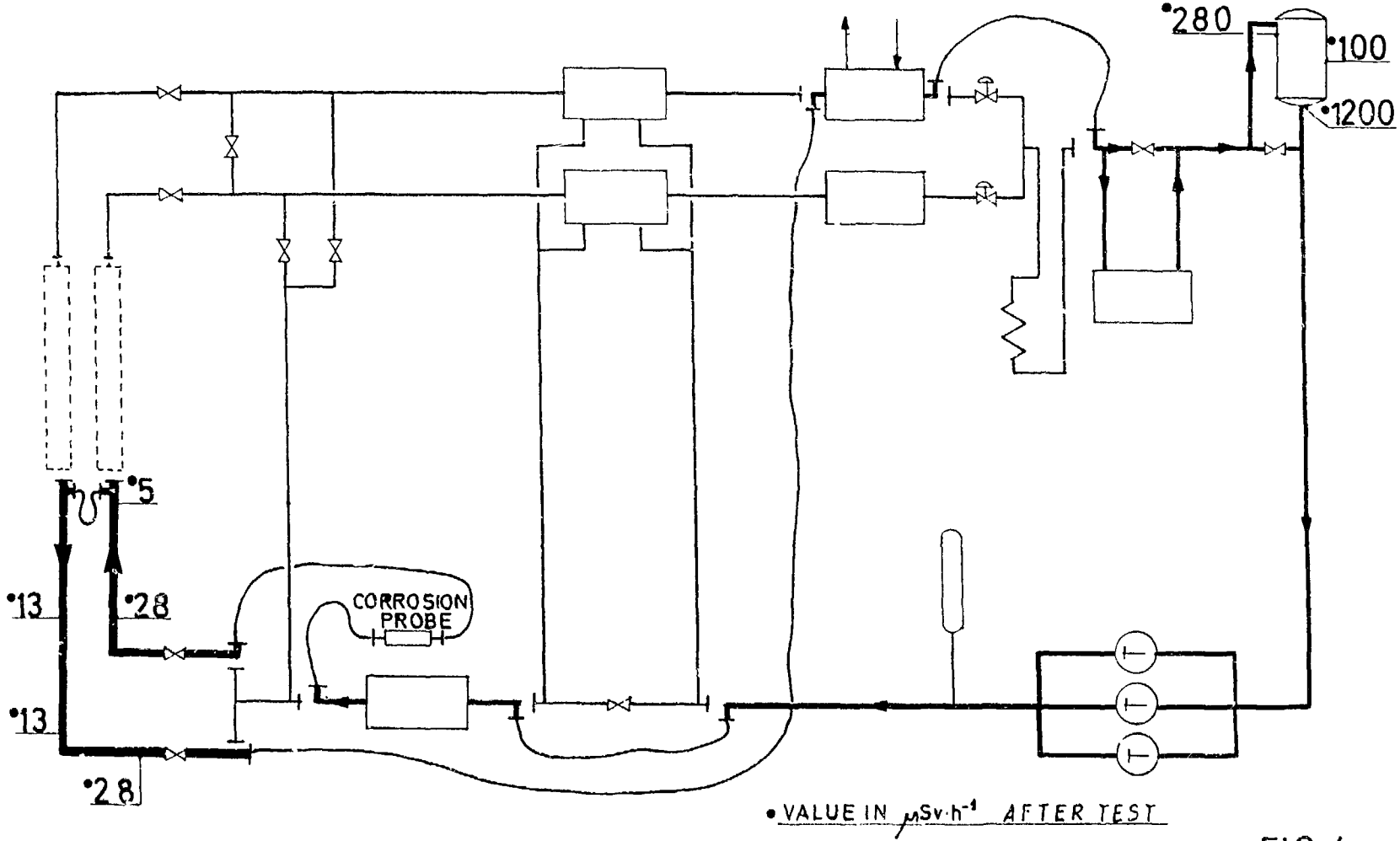
FIG. 2



•VALUE IN $\mu Sv \cdot h^{-1}$ AFTER TEST

= GROUND LEVEL 1-2 $\mu Sv \cdot h^{-1}$

FIG. 3


•280
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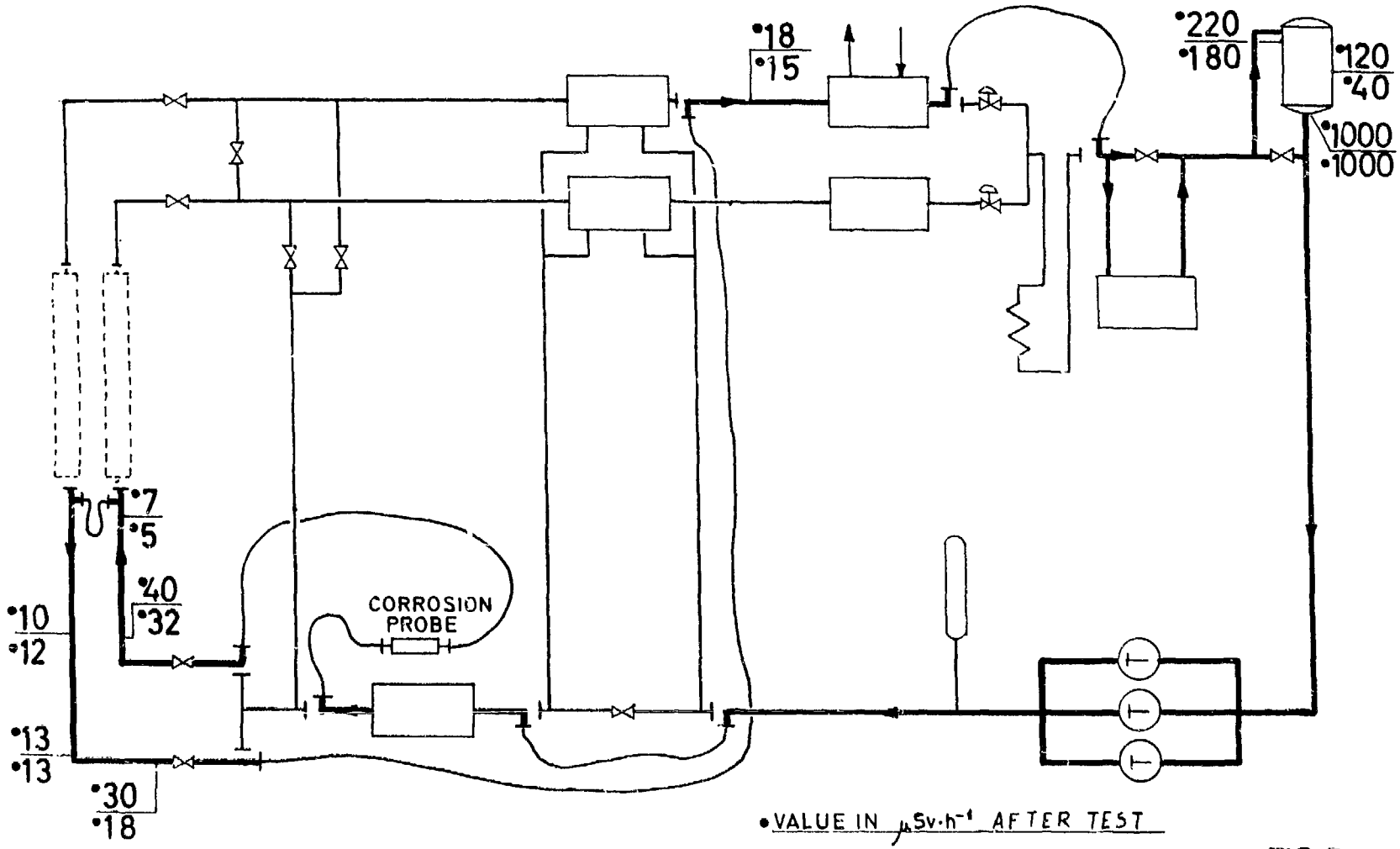
•1200
•5
•13
•28
•13
•28
CORROSION PROBE
• VALUE IN μSv·h⁻¹ AFTER TEST

FIG. 4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•1000
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•7
•5
•40
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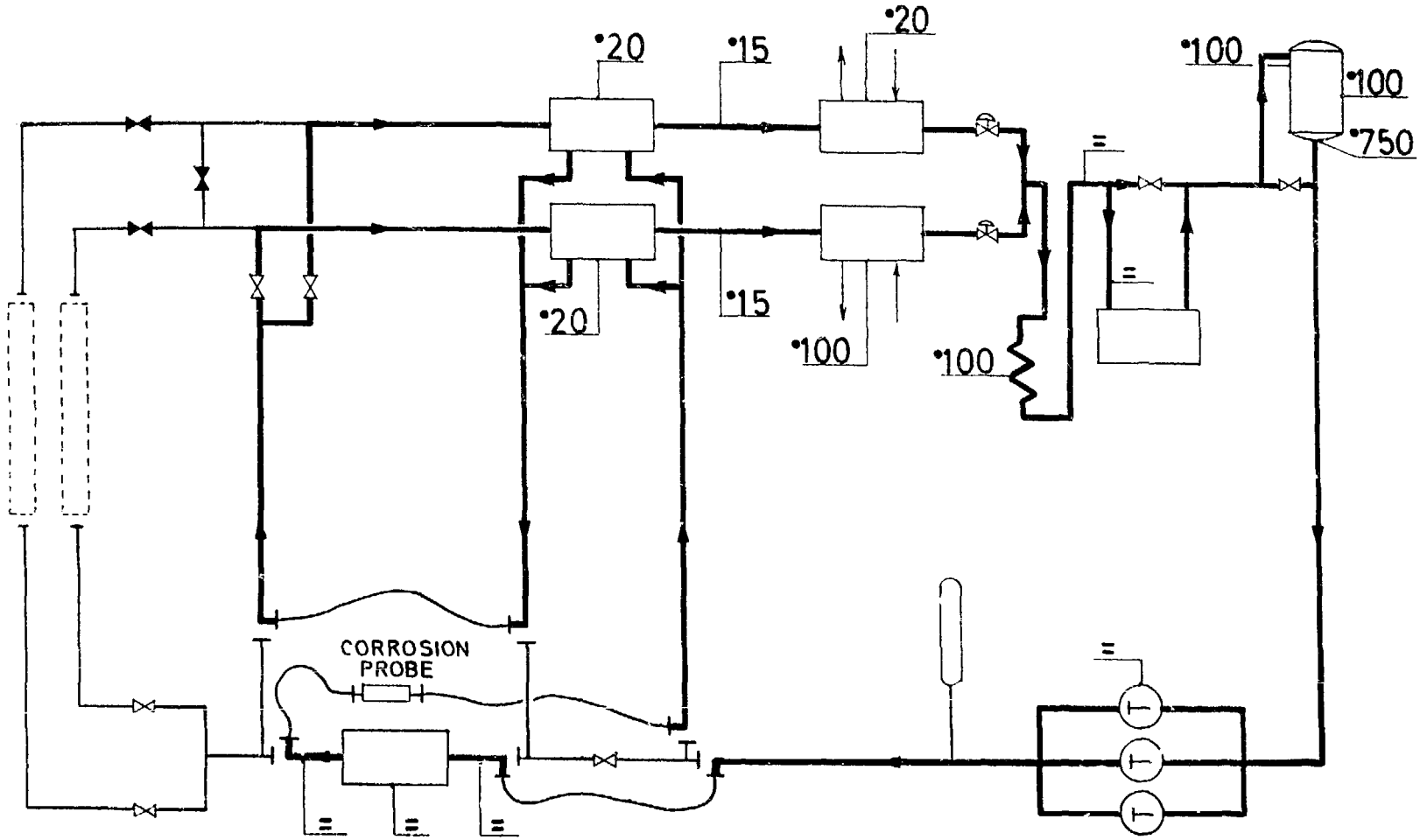
•12
•13
•13
•30
•18
CORROSION PROBE
• VALUE IN μSv·h⁻¹ AFTER TEST

FIG. 5



• VALUE IN $\mu Sv \cdot h^{-1}$ AFTER TEST

= GROUND LEVEL 1-2 $\mu Sv \cdot h^{-1}$

FIG. 6

DECONTAMINATION OF NUCLEAR FACILITIES BY ELECTROCHEMICAL METHODS

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ABSTRACT

Three electrochemical, remotely operated decontamination equipments were developed for the Paks NPP. The first equipment was constructed for the decontamination of the main circulating pump case, the second equipment for the collectors of the steam generator and the third equipment for the main gate valve case of the primary circuit. All the equipments use movable cathods. The main circulating pump case was decontaminated eight times. The electrolyte contained sulfuric, oxalic and phosphoric acid. The decontamination factor of plant decontamination was 10-500. The duration of the process was 8-12 hours. The collective dose for the staff (3-4 workers) was about 2-3 mSv. Only 100-200 l volume of radioactive effluents was produced.

INTRODUCTION

In Hungary the Paks Nuclear Power Plant is operating since the end of the year 1982. It has pressurized water reactors of VVER-440 type. Three units of four hundred forty MWs are in operation, one unit is under construction and will be put into operation this year. The long-term plans include the installation of two one-thousand MW units.
Systematic and efficient maintenance and inspection are essential conditions for the safe operation of nuclear power plants, however, maintenance and supervision are dangerous because of the personnel exposure hazard. Radiation doses can be reduced by the decontamination of the primary circuit. The exposure of the decontamination staff can be reduced using remotely operated devices. The primary circuit of the VVER-440 type reactor is made of stainless steel. Therefore agressive physico-chemical processes can be used thus increasing the efficiency of decontamination.
In the last three years the development of electrochemical decontamination was carried out in the institute as a part of a coordinated research program of the International Atomic Energy

Agency on the development of decommissioning and decontamination of nuclear facilities,

ELECTROCHEMICAL DECONTAMINATION PROCESS

Scope

Electrochemical decontamination is an application of electropolishing of metal surfaces.
Thin /0.01-0.05 mm/ oxid-layers are removed from the contaminated metals using direct electric current between the workpiece and a cathode in an acidic electrolyte. The anodic oxidation, taking place on the surface of the workpiece, occurs more slowly in crevices and more quickly at peaks, and results in a polishing effect without severe damage of the surface. When electropolishing a tank containing acidic electrolyte, low direct voltage and high current density are usually used. Tools, fittings and accessories or mobile equipment can be decontaminated in such a way.

Figure 1. shows the current density versus potential.

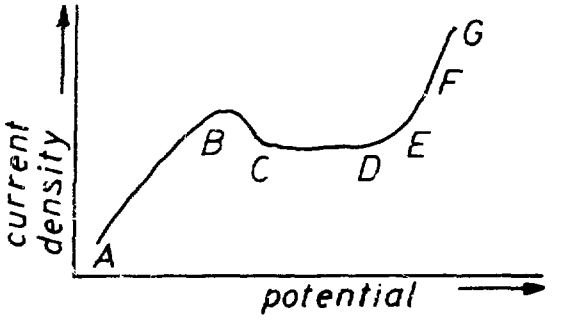

Fig. 1. Current density versus potential.

The curve between A-B follows Ohm's-law, the anode dissolves, but polishing effect can't be observed.

$$Me \longrightarrow Me^{++} + 2e$$

The oxidation of metal surface begins at point B.

$$Me + 2OH^{-} \longrightarrow MeO + H_2O + 2e$$

The electric resistance increases because of the forming oxid-layer while the current density decreases (B-C). However the acidic electrolyte dissolves the metal-oxids and thus equilibrium sets in between the formation and dissolution of the oxid-

layer (C-D). Polishing effect can be detected in this section. The O_2 formation begins at point C:

$$4\ OH^- \longrightarrow O_2 + 2H_2O + 4e$$

First gas-bubbles separate slowly from the anode, causing striped corrosion. The gas formation increases in the E-F section, the current yield decreases. The metal passivation begins at point G. Section C-D proves to be the best for decontamination and section D-F for polishing.
The process produces very smooth (0.02-0.03 μm), non-reactive and non-adsorbing surface with is resistant to recontamination during the following operation.

Electrochemical Decontamination By Movable Cathod

When contaminated large-sized workpieces are sunk into a tank for electropolishing a big volume of radioactive waste water is produced.
An in-situ process was developed to decontaminate large equipments that cannot be immersed in an electropolishing tank. Electropolishing using movable cathode is one of the decontamination methods which does not use big volume of liquid and does not produce big quantitites of effluents. The method can be seen in Fig. 2.

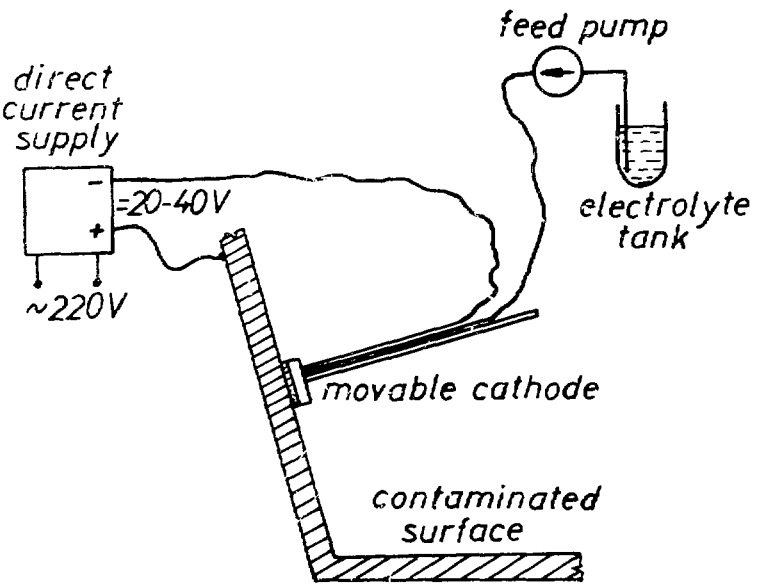

Fig. 2. Decontamination my movable cathode

The metal object to be decontaminated serves as an anode. Acidic electrolytes /H_3PO_4, H_2SO_4, oxalic-acid, citric-acid/ low direct voltage and high current density are usually used.

The electrolyte has to dissolve the removed metal-oxides, to advance the forming of a protective oxid layer, to conduct effectively the current and must not cause corrosion. The cathode is made of stainless steel.

To avoid short-circuit the cathode should be isolated from the surface of the workpiece /anode/ by some water-absorbing material /sponge, felt/. This material is continuously wetted by the help of an electrolyte pump, and a power supply provides electric current through the felt to the decontaminated surface. The cathode is slowly moved by the operator or by remote handling.

The electrochemical process has the following advantages:
- High decontamination factor (20-500)
- Short application time
- Produces smooth surface
- Can remotely be handled
- Low volume of liquid radioactive waste

High personnel exposure can be taken into account when manhandling and difficulties can arise with complicated geometry or profile.

REMOTELY OPERATED EQUIPMENTS FOR DECONTAMINATION

A remotely operated equipment was constructed for the decontamination of the main circulating pump case. Figure 3. shows the principle of operation. The equipment consists of the support and the traversing mechanism of the movable decontamination head and the supply unit. (Fig. 4. and Fig. 5.) The supply unit is carried by a small truck and can be craned. There is a 40 l volume electrolyte tank on the truck. A feed pump provides the accurate dosage of the electrolyte for the movable decontamination head. The used electrolyte accumulates on the bottom of the treated pump case and it can be discharged by a plunger pump into the radioactive sewage. The traversing mechanism is equipped with a multi-jointed arm moved by pneumatic cilinders on the inner surface of the pump case. The central bearing axle can be rotated by a rotative gear right and left 360 degs. The multi-jointed arm pulls away the movable decontamination head from the decontaminated surface after one turn-around, moves it 45 mm down and presses on the surface again. The compressive force can be regulated between 40-400 N. The position of the movable decontamination head always follows the curving of the pump case and the felt fits close to the surface. The uniform contact between the felt and the surface is ensured by pressure springs. The movable decontamination head is equipped with mechanical sensing device to detect if the head reaches the pipe connection of the primary circuit.

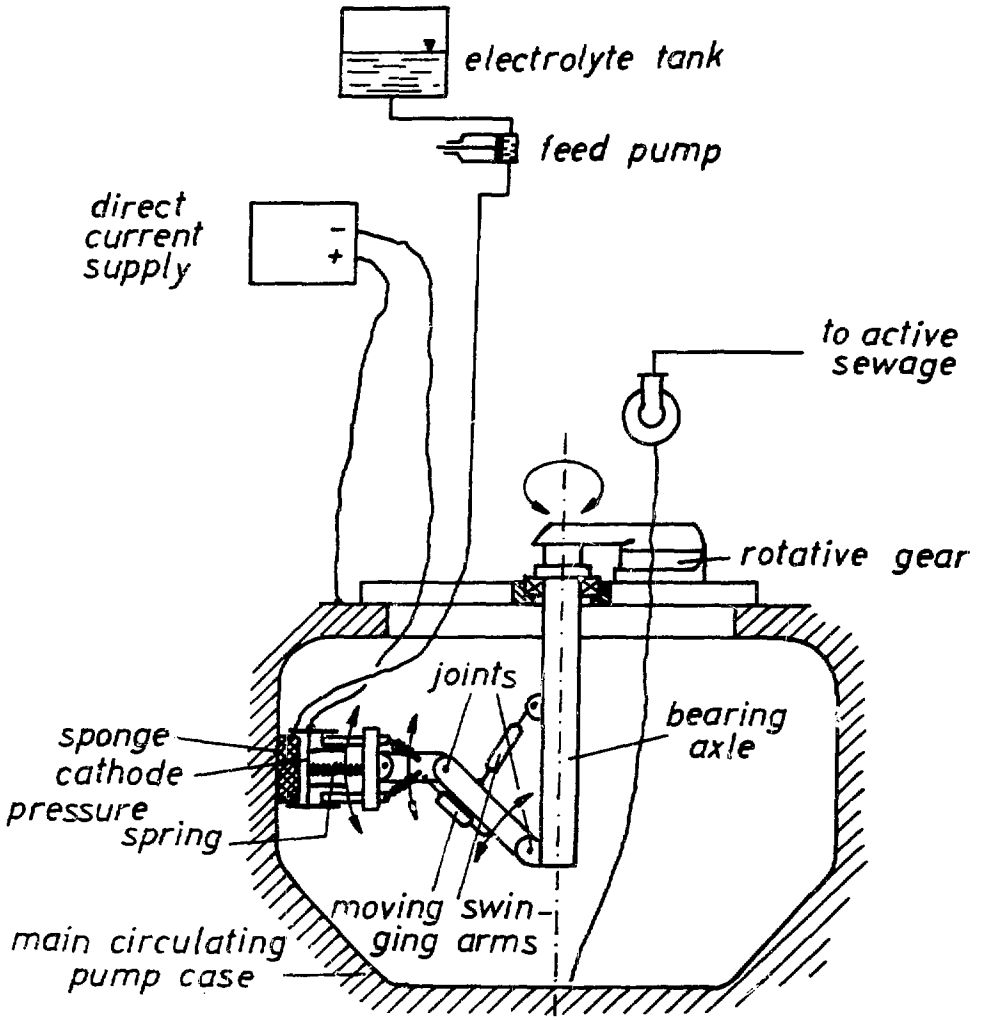

Fig. 3. Electrochemical decontamination equipment for the main circulating pump case

Results Obtained From Plant Decontamination

Effective decontamination procedures were conducted at the Paks Nuclear Power Station during the shutdown periods by the remote controlled electrochemical decontamination equipment for the main circulating pump case. The main circulating pump cases (MCP) No.4. and 5. of the first unit and No. 3. and 6. of the second unit were decontaminated in 1985, No 1. and 3. of the first unit and No. 1. and 4. of the second unit in 1986. The electrolyte contained phosphoric, sulfuric and oxalic acids. Before decontamination the inlet nozzle for reactor coolant was plugged in such a way that the electrolyte could not enter the connecting pipe. The spent electrolyte accumulated on the bottom of the treated pump case and was drained into the radioactive sewage. After electrochemical decontamination the inner

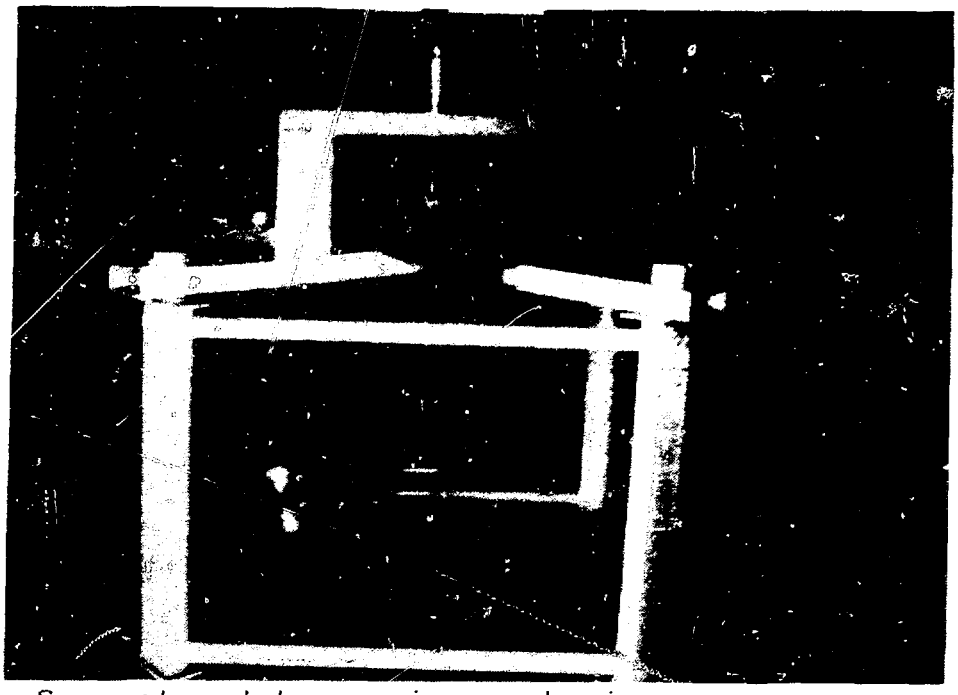

Fig. 4. Support and traversing mechanism

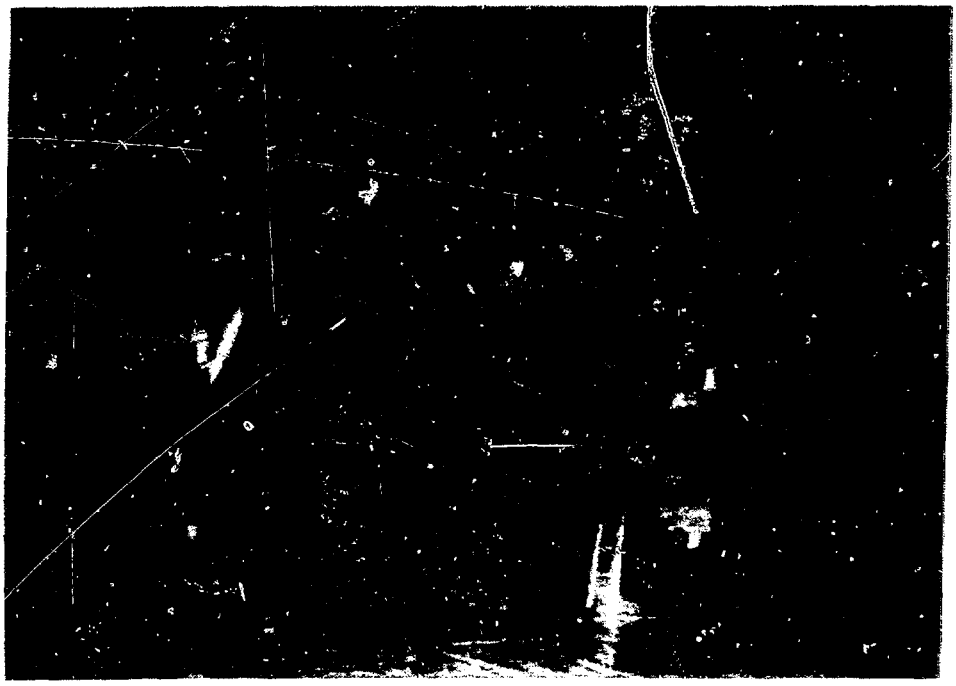

Fig.5. Movable cathode

surface of the pump case was rinsed with a 12 g/l boric acid solution.
The concentration of the radioactive isotopes and the corrosion products was measured in the spent electrolyte and in the rinsing solution too. The activity of the used felts was measured after the first decontamination because only max. 1-2 % of the total activity could be found in the felts. About 150 l liquid radioactive waste (together with rinsing water) was produced during the decontamination procedure.
The duration of the decontamination was 6-8 hours. The procedure was carried out by 3-4 workers, their collective dose was 2-3 mSv. The decontamination was followed by chemical and radiometric analyses.
Table I. shows the removed activity, the removed corrosion product and the decontamination factor (DF).

Table I.

Date and location	DF	removed activity (MBq)	removed corrosion products (g metal)
1985. I. unit			
4. MCP	27	1500	-
5. MCP	1800	-	-
2. unit			
6. MCP	490	1850	52
3. MCP	11	4200	42
1986. 1. unit			
1. MCP	-	3800	76
3. MCP	110	2200	47
2. unit			
1. MCP	46	2600	54
4. MCP	52	2200	70

Table II. and III. show the nuclide and metal composition of the removed activity and metal as percent of total in 1986.

Table II.

Location	Activity %							
	^{51}Cr	^{54}Mn	^{58}Co	^{59}Fe	^{60}Co	^{95}Nb	^{110}Ag	^{124}Sb
1. unit								
1. MCP	1.0	11.5	58.6	2.8	21.3	-	4.0	0.5
3. MPC	2.0	8.6	47.7	2.8	25.3	4.4	8.5	0.4
2. unit								
1. MCP	2.0	7.2	27.0	1.4	9.1	-	51	2.3
4. MCP	9.5	12.3	45.7	4.0	16.1	-	10.6	1.8

Table III. Composition of the removed metal as percent of total

Location	Fe	Co	Ni	Mn	Cr
			%		
1.unit					
1. MCP	74.1	0.1	10.0	0.8	14.8
3. MCP	78.1	0.1	6.1	1.3	14.2
2. unit					
1. MCP	74.6	0.4	11.3	1.5	12.2
4. MCP	76.7	0.3	10.0	1.4	11.6

Figure 6. shows the main circulating pump case after the decontamination. It can be seen that the dark oxid-layer was effectively removed from the inner surface, i.e. the pump case was cleaned to metal.

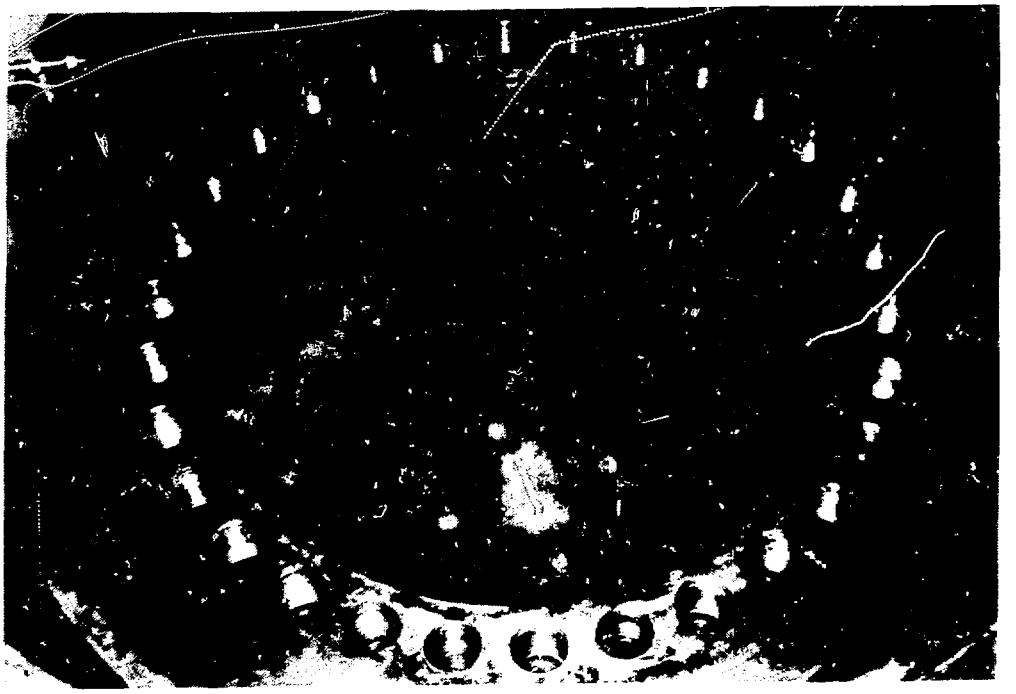

Fig. 6. The main circulating pump case after the decontamination

Decontamination Equipment For The Collectors Of The Steam Generator

A remote operated decontamination equipment, showed in Fig.7. was constructed for the collectors of the steam generator. The equipment consists of the support, the traversing mechanism of the decontamination head and the control unit including electrolyte and voltage supply.

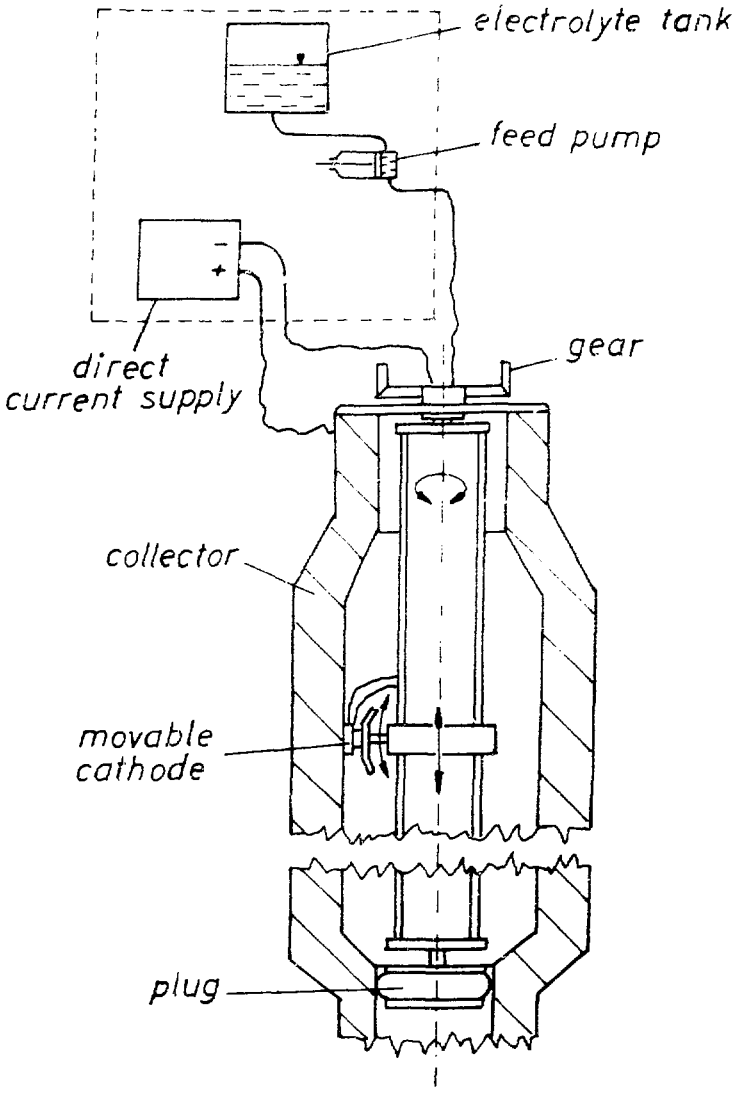

Fig. 7. Decontamination equipment for the collectors of the steam generator.

The whole mechanism can be craned into the collector and the control unit inflates the inner tube of the plug. The tube is fixed by friction and seals between the pipewall and the plug. The decontamination head is fitted swinging on the axle of the pneumatic cylinder which presses the head on the surface to be decontaminated. The head moves down at a speed of 20 mm/min. The whole surface of the collector can be treated in twelve hours. In the bottom position the control unit stops rotating, moving and feeding the electrolyte too.

The control unit is carried by a small car and can be craned. A piston type pump feeds the electrolyte from a tank to the movable decontamination head. The spent and contaminated electro-

lyte accumulates on the bottom of the treated collector and is removed by a plunger pump. The bottom of the collector is closed by a plug.

The guide rail of the support and traversing mechanism of the decontamination head is fitted with bearings and is free-wheeling. The lower bearing socket is built in the plug at the bottom of the collector. The upper bearing socket is built in a bridge placed on the upper plane of the collector. The turning is performed by a gear fixed on the bridge.
The decontamination head returns after one turning to avoid any damage to the vertical rails. The speed of the turning is 0.4 turn/min during continuous moving. The cathode can also be moved step by step to reduce the damaging of the felt.

Decontamination Equipment For The Main Gate Valve Case

An electrochemical decontamination equipment was designed and is under construction to treat the inner surface of the main gate valve case. /Fig. 8./

The decontamination of the gate valve case can be conducted only after removing the cut-off slide-valve. First the two pipe-ends must be plugged to prevent from the electrolyte or other impurities getting in. The feeding and removing of the electrolyte is performed as in the case of the two electrochemical decontamination processes mentioned before.
The support and the traversing mechanism is craned on the flange of the gate valve case. The decontamination equipment has two movable decontamination heads. One of them /head II./ moving around the vertical symmetry axis of the case treats the upper part of the case. The central bearing axle is only rotated right and left 180 degs to avoid damaging the wires and conduits.
The lower spherical part of the case is treated by head II. moving round the horizontal axis of the centre of the pipe ends. The spherical surface of the case and the outer surface of the pipe ends is simultaneously treated by cathode I.
The moving of both movable heads is complicated because they have to go round the two guide rails of the cut-off slide valve as well. About 80 percent of the contaminated surface can be treated in such a way. The guide rails, the inner and the sealing surface of the pipe-ends can only be decontaminated by manually moved cathods.

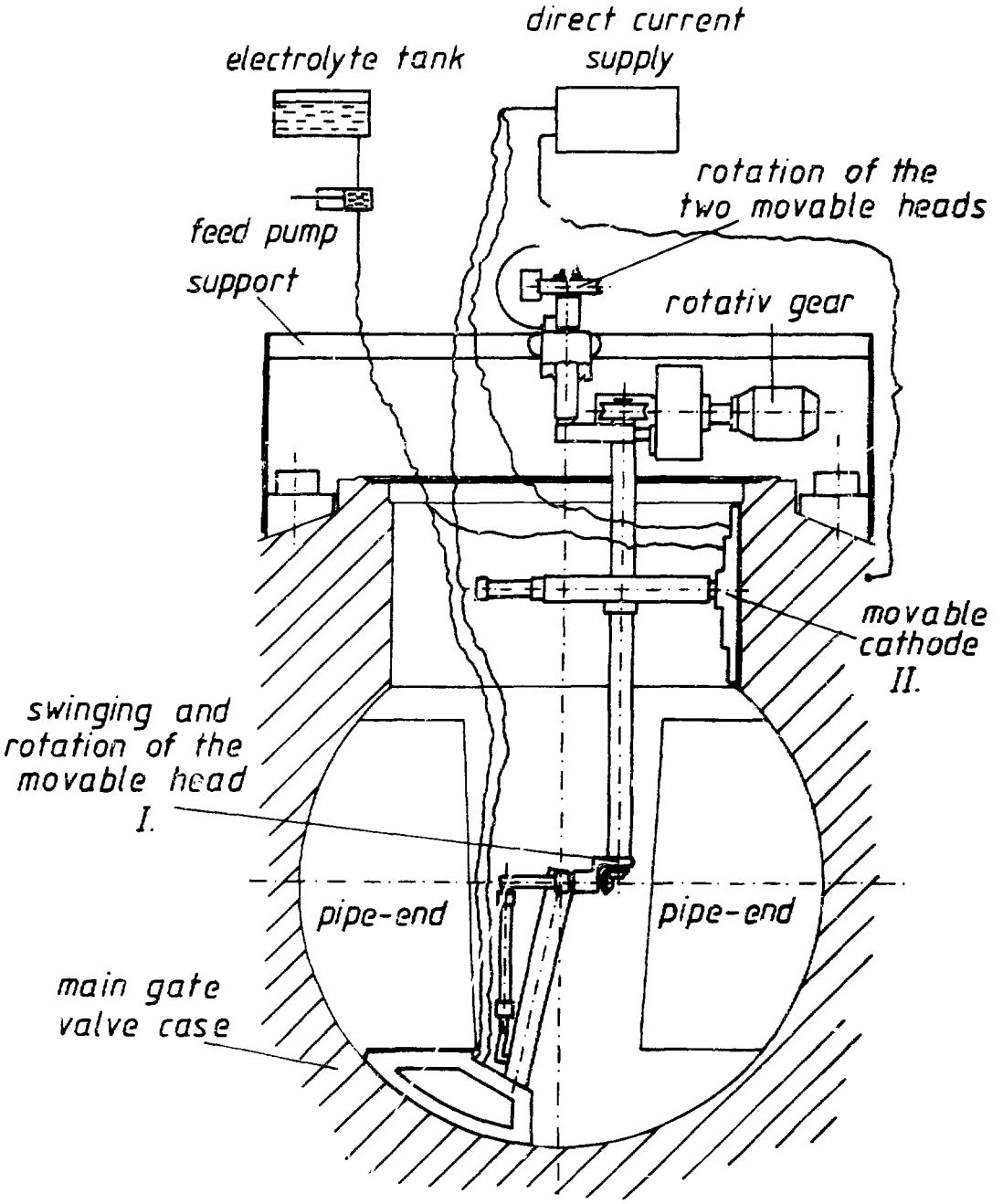

Fig.8. Decontamination equipment for the main gate valve case

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CHARACTERISATION OF THE RADIOACTIVE DEPOSITS ON PWR PRIMARY CIRCUIT SURFACES

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ABSTRACT

Results from detailed examination of the oxides on a range of Inconel 600 and stainless steel specimens from a number of PWRs are reported. A variety of techniques have been employed. These include gamma spectrometry, alpha spectrometry, scanning electron microscopy and wet chemical analysis. In addition, surface analytical techniques including secondary ion mass spectrometry (SIMS) have been used to characterise the oxides. The sources of the major radionuclides present on circuit surfaces are considered. Procedures for decontaminating PWR surfaces are described. In addition to the PWR specimens, results from a limited number of specimens from BWRs and CANDU reactors are presented.

INTRODUCTION

In assessing the decontamination and decommissioning options for LWRs, characterisation of the radioactive deposits is required to provide information on the radioactive inventory in the system. Knowledge of the quantities of activation products, fission products and actinides deposited on reactor pipework during operation is essential in formulating the overall decommissioning strategy. Similarly, in the assessment of whether or not to decontaminate and what process to use, knowledge of the physical and chemical form of the radioactive deposits is required to assess the potential efficacy of any decontamination options. In spite of this importance there is only limited information available on the nature of PWR corrosion products on out-of-core surfaces. This paper describes detailed examinations performed on a number of Inconel 600 and stainless steel specimens from PWRs.

CORROSION AND ACTIVITY TRANSPORT PROCESSES IN PWR CIRCUITS

The major constructional materials used in PWR primary circuits are austenitic stainless steels (AISI 304, 308, 309, 316 and 321) and high nickel alloys (Inconel 600, 690, 718 and Incoloy 800). For the fuel cladding Zircaloy is used on all modern PWRs. In addition, there are small areas of various other alloys in the circuit; the most important of these is the high cobalt material, Stellite, which is used as a hard-facing in pumps, valves and various other components.

The majority of the PWRs of Western design use AISI 304 stainless steel for the primary circuit pipework and Inconel 600 for the steam generator (SG) tubes. The major exception is the West German KWU plants which use Incoloy 800 for the SG tubes. In addition, AISI 308 and 309 is used as weld overlay material in the main pressure vessel and areas such as the SG channel heads. On Russian designed VVER plants the major material of construction is an AISI 321-type stainless steel which is used for both the primary circuit pipework and SG tubes.

In pressurised water at 300^{0}C, stainless steel and nickel alloys corrode at a slow but finite rate, forming surface oxides and also releasing soluble species to the circulating coolant. Particles of oxide formed by either precipitation or break-up of surface oxides also circulate in the coolant. The circulating soluble and particulate material can deposit on the Zircaloy fuel cladding where it is neutron activated; re-release and deposition and sorption of activated material onto out-of-core surfaces can then take place. In addition, release of activated material from stainless steel and Inconel components in the core can also occur. The Zircaloy fuel cladding also corrodes but releases very little material to the circulating coolant. In the majority of PWRs, activated corrosion products are the major contributors to radiation fields and doses. Typically, about 60% of station dose arises from ^{60}Co (from an n,γ reaction on ^{59}Co) with a further 15% from ^{58}Co (n,p reaction on ^{58}Ni) and 10% from other activation products. Fission products generally contribute < 10% of station dose on PWRs. In considering decommissioning scenarios, of course, cognisance has to be taken of the pure beta-emitting and electron-capture radionuclides, and also actinides. These do not contribute to radiation fields but may come to dominate radioactivity arisings. In this paper, results from gamma and alpha spectrometry of reactor specimens are presented. Work to determine the levels of beta-emitting and electron-capture radionuclides is in progress.

EXAMINATION OF SPECIMENS

The specimens examined include Inconel 600 SG tube and stainless steel from a number of Westinghouse type PWRs, specimens from a number of other PWRs, including a Russian type VVER reactor, CANDU SG tube and specimens from two BWRs. The specimens were sectioned into small coupons (approximately 1 cm^{2}) for subsequent examination. On the PWR specimens examined there is a fairly clear distinction between the loose (particulate) oxide on the surface and the fixed (grown-on) oxide. The former was removed by a few minutes treatment in an ultrasonic bath and collected on a 0.1 μm Nuclepore filter. With some of the other specimens examined the distinction between fixed and loose oxide was less clear, although in general the oxide was separated into these components for analysis using the ultrasonication technique.

Gamma Spectrometry

Typical results from gamma spectrometry of the Inconel 600 SG tube specimens are given in Table I. The predominant isotope on all the

TABLE I

Radioactivity Levels on PWR Inconel 600 SG Tube Specimens

Reactor (EFPY)	Radio-nuclide	Total Activity (kBq cm^{-2})
P-21 (1.6)	^{60}Co	41.4
	^{54}Mn	4.9
	^{125}Sb	0.4
B-13 (2.1)	^{60}Co	41.1
	^{54}Mn	5.2
	^{125}Sb	0.5
L-25 (1.4)	^{60}Co	17.8
	^{54}Mn	3.6
	^{125}Sb	0.4
I-4 (2.2)	^{60}Co	66.2
	^{54}Mn	1.5
	^{125}Sb	0.07
C (2.2)	^{60}Co	74.0
	^{54}Mn	4.1
	^{125}Sb	0.4
Indian Point 2 (2.2)	^{60}Co	144
	^{54}Mn	10.7
	^{125}Sb	0.4
Ringhals 2 (5.4)	^{60}Co	253
	^{54}Mn	4.4
	^{57}Co	0.5
	^{58}Co	170
	^{65}Zn	2.2
	^{94}Nb	0.012
	^{106}Ru	0.15
	^{110m}Ag	0.015
	^{125}Sb	0.46
	^{137}Cs	0.0037
	^{144}Ce	0.24
	^{152}Eu	0.0078
	^{154}Eu	0.011
	^{155}Eu	0.011
Doel 2 (6.0)	^{60}Co	194
	^{54}Mn	12.6
	^{57}Co	2.7
	^{58}Co	992
	^{65}Zn	2.3
	^{94}Nb	0.007
	^{124}Sb	2.2
	^{125}Sb	1.3
	^{154}Eu	0.004

TABLE II

Radioactivity Levels on PWR Stainless Steel Specimens

Reactor (EFPY)	Radio-nuclide	Total Activity (kBq cm^{-2})
Surry 2 Manway Insert (< 1)	^{60}Co	133
Ringhals 2 CVCS Pipe (2.2)	^{60}Co	248
	^{58}Co	522
	^{54}Mn	33.7
	^{65}Zn	5.6
	^{144}Ce	0.56
Ringhals 2 Manway Insert (5.8)	^{60}Co	958
	^{54}Mn	26.3
	^{57}Co	1.7
	^{58}Co	433
	^{65}Zn	5.2
	^{94}Nb	1.7
	^{106}Ru	0.3
	^{125}Sb	0.2
	^{144}Ce	0.8
	^{154}Eu	0.06
	^{155}Eu	0.04
Loviisa 2 CVCS Pipe	^{60}Co	4.1
	^{54}Mn	0.3
Loviisa 2 SG Tube (1.5)	^{60}Co	0.26
	^{54}Mn	0.13
	^{57}Co	0.011
	^{65}Zn	0.018
	^{108m}Ag	0.27
	^{110m}Ag	0.34
	^{152}Eu	0.0015

specimens was ^{60}Co, apart from Doel 2 where it was ^{58}Co. The majority of the ^{60}Co, 72-96%, was associated with the fixed oxide. The other isotopes detected on all the specimens were ^{54}Mn (from n,p on ^{54}Fe) and ^{125}Sb (from n,γ on ^{124}Sb and n,p on ^{125}Sn). In the case of the Ringhals 2 and Doel 2 specimens, the solutions of oxide stripped from the specimens as described under chemical analysis were examined after removing the cobalt isotopes from the solution. This drastically reduced the Compton continuum background and enabled a number of radionuclides present in very small amounts to be detected. The additional radionuclides detected on the Ringhals 2 specimen included the fission products ^{106}Ru, ^{137}Cs and ^{144}Ce and the activation products ^{94}Nb (from n,γ on ^{93}Nb), and the europium isotopes ^{152}Eu (from n,γ on ^{151}Eu), ^{154}Eu (from n,γ on ^{153}Eu) and ^{155}Eu which is an activation product of ^{154}Eu.

The ^{60}Co radioactivities are plotted in Fig. 1 versus EFPY, together with earlier Doel 2 and Ringhals 2 data obtained at BNL and EPRI data[1]. The data from the BNL studies fall into the same spread as the data from the EPRI report. A reasonable fit of the data points beyond 1.5 EFPY is provided by a simple ^{60}Co build-up/decay equation $\frac{R}{\lambda}(1-e^{-\lambda t})$, with a ^{60}Co deposition rate of 48 kBq cm^{-2} per EFPY. If this rate continued throughout reactor life it would lead to a saturation level of ^{60}Co on SG tube surfaces of about 350 kBq cm^{-2} after 25 years.

Results from gamma spectrometry of various stainless steel PWR specimens are presented in Table II. The major isotopes were again ^{60}Co, ^{58}Co and ^{54}Mn. For the Ringhals 2 manway insert specimen the ^{60}Co level was about a factor of 4 greater than on the SG tube (Table I) which had experienced a similar EFPY. As with the SG tube specimens the majority of the ^{60}Co radioactivity (75-95%) was present in the fixed oxide. The ^{60}Co levels on the Loviisa PWR specimens are about 2 orders of magnitude lower than on the other PWR specimens. Low levels of ^{60}Co were also found on SG tube from CANDU reactors (Table III). The Bruce 4 specimen, in fact, had substantially more ^{65}Zn than ^{60}Co. The reasons for the much lower levels of ^{60}Co on the Loviisa and CANDU reactor specimens are discussed later.

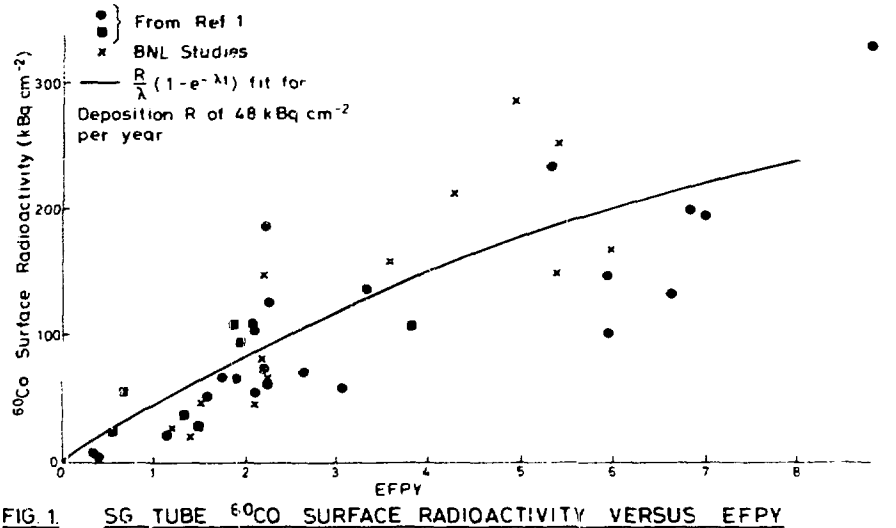


FIG. 1 SG TUBE ^{60}Co SURFACE RADIOACTIVITY VERSUS EFPY

TABLE III

Radicactivity Levels on CANDU SG Tube Specimens

Reactor (EFPY)	Radio-nuclide	Total Activity (kBq cm^{-2})
Bruce 3 (5.9)	^{60}Co	0.22
	^{54}Mn	0.05
	^{65}Zn	0.15
	^{106}Ru	2.0
	^{124}Sb	0.70
	^{125}Sb	0.16
	^{144}Ce	3.9
Bruce 4 (3.5)	^{60}Co	1.7
	^{54}Mn	0.81
	^{65}Zn	18.9
	^{106}Ru	0.73
	^{125}Sb	0.027
	^{137}Cs	0.005
	^{144}Ce	0.77

TABLE IV

Radioactivity Levels on BWR Stainless Steel Specimens

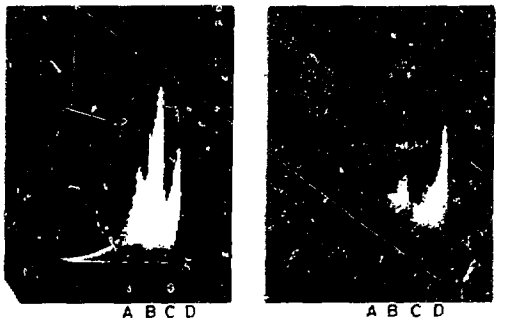
Reactor (EFPY)	Radio-nuclide	Total Activity (kBq cm^{-2})
T (4.6)	^{60}Co	133
	^{54}Mn	4.4
	^{58}Co	1.2
	^{65}Zn	64.6
	^{103}Ru	0.63
	^{106}Ru	13.9
	^{125}Sb	0.47
	^{144}Ce	2.8
	^{155}Eu	0.019
M (8.7)	^{60}Co	107
	^{54}Mn	1.1
	^{65}Zn	2.4
	^{106}Ru	1.4
	^{125}Sb	0.25
	^{137}Cs	0.023
	^{154}Eu	0.070
	^{241}Am	0.032

The two BWR specimens examined also showed ^{60}Co as the predominant isotope (Table IV). However, on the reactor T specimen a number of other radionuclides were present in appreciable amounts, in particular ^{65}Zn (65 kBq cm^{-2}) and the fission product ^{106}Ru (14 kBq cm^{-2}). The large amount of ^{65}Zn present is consistent with the high level of zinc in the oxide on the reactor T specimens. This zinc originates from the brass condensers and other components in the feed train.

Alpha Spectrometry

The alpha radioactivity on the specimens was determined directly using an ORTEC alpha spectrometer equipped with 600 mm^2 surface barrier detectors. This was coupled to a Canberra multi-channel analyser. The system counting efficiency in 4π geometry was 25%. Typical examples of alpha spectra obtained from the specimens are shown in Fig. 2 for Doel 2 and Ringhals 2 SG tube. The spectra generally consisted of three or four main peaks corresponding to the following energies and major actinides:-

A 5.05-5.10 MeV ^{239}Pu + ^{240}Pu
B 5.40-5.45 MeV ^{238}Pu + ^{241}Am
C 5.70-5.75 MeV ^{244}Cm
D 6.00-6.05 MeV ^{242}Cm [$T^{1}/_{2}$ 163 d]



(a) Doel 2 (1983) SG Tube (b) Ringhals 2 (1984) SG Tube

FIG 2. TYPICAL ALPHA SPECTRA OBTAINED FROM REACTOR SPECIMENS

The spectra were analysed using fitting routines and results from the PWR specimens are summarised in Table V. The data are expressed in terms of Bq cm^{-2}; the curium isotopes are decay corrected to the time when the specimens were removed from the reactor. Two of the reactors, L-25 and Ringhals 2 (1977) showed < 0.05 Bq cm^{-2} of alpha radioactivity. It is possible that this level may represent the quantity expected from uranium impurities in the circuit, e.g. tramp uranium on fuel. On the other specimens, the total alpha radioactivity varied from 0.9-26 Bq cm^{-2}. However, if the ^{242}Cm data are neglected the range is reduced to 0.9-4.9 Bq cm^{-2}. The quantity of short-lived ^{242}Cm on surfaces at shutdown will be very dependent upon when the actinides were deposited. For instance, if the actinides were deposited as a result of fuel failures a number of years previously, very little ^{242}Cm would remain.

The reactors examined have experienced up to 6.8 EFPY of operation. The actinide arisings (excluding ^{242}Cm) are plotted versus EFPY in Fig. 3. It is clear that there is no systematic trend with EFPY. In fact, it is probably unreasonable to expect any progressive relationship between the concentrations observed and operating time since fuel failures (which are probably the major source of the actinides) are not systematic events. This contrasts with the data for ^{60}Co where a $\frac{R}{\lambda}(1-e^{-\lambda t})$ relationship can be fitted to the reactor data. Clearly, it is not possible, therefore, to perform a simple extrapolation to predict the actinide levels on out-of-core surfaces at the end of reactor life after a possible 40 calendar years of operation. Calculation of the actinide inventory at the end of life also requires an assessment of the impact of the growth of daughter products. The major change will result from decay of the beta-emitter ^{241}Pu ($T^{1}/_{2}$, 15 y) to ^{241}Am ($T^{1}/_{2}$, 433 y). Therefore, measurement of ^{241}Pu through chemical separation and liquid scintillation counting is required.

The actinide concentrations on three BWR reactor specimens were also measured. The levels recorded (Table VI) were in the range 81-174 Bq cm^{-2}, i.e. somewhat higher than on the PWR specimens.

TABLE V

Actinide Radioactivity Measured on PWR Specimens

Reactor (1)	EFPY	Total Alpha Activity (Bq cm^{-2})	Actinide Activity (Bq cm^{-2})			
			$^{239}Pu+^{240}Pu$	$^{238}Pu+^{241}Am$	$[^{244}Cm]$	$[^{242}Cm]$
B-13	2.1	0.89	0.48	0.37	0.0052	n.d.
P-21	1.6	1.96	0.70	1.11	0.13	n.d.
L-25	1.4	0.022	0.0078	0.013	0.0015	n.m.
I-4	2.2	4.85	1.37	2.85	0.63	n.m.
Indian Pt 2	2.2	4.92	1.74	2.70	0.48	n.m.
Doel 2 (1984)	6.8	1.67	0.14	0.89	0.56	0.093
Ringhals 2 (1977)	1.2	0.044	0.022	0.020	0.0026	n.m.
Ringhals 2 (1984)	5.4	25.9	0.78	1.41	0.48	23.3
Ringhals 2 (1979) 304 SS	2.2	3.07	1.41	1.30	0.34	n.m.
Ringhals 2 (1984) 304 SS	5.8	7.7	0.41	0.78	0.32	6.15

[ ] Decay Corrected ^{244}Cm and ^{242}Cm data. n.d. = not detected

(1) = Inconel 600 SG tube except where stated otherwise. n.m. = not measured

TABLE VI

Actinides Measured on BWR Specimens

Reactor	EFPY	Total Alpha Activity (Bq cm^{-2})	Actinide Activity (Bq cm^{-2})			
			$^{239}Pu+^{240}Pu$	$^{238}Pu+^{241}Am$	$[^{244}Cm]$	$[^{242}Cm]$
M	8.7	111	16.7	70.3	23.7	0.4
T	4.6	174	5.9	11.5	8.1	148
L	4.7	81.0	21.1	54.0	5.6	-

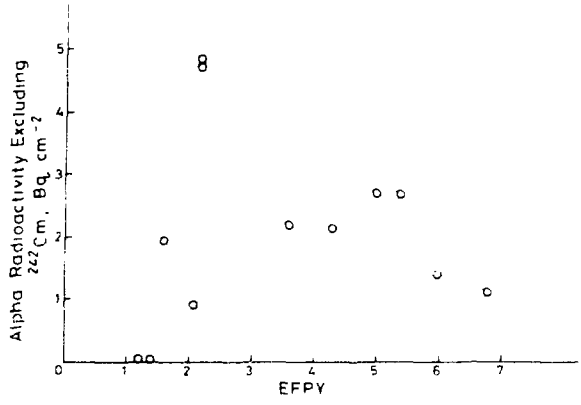


FIG 3 SG TUBE SURFACE ACTINIDE CONCENTRATIONS VERSUS EFPY

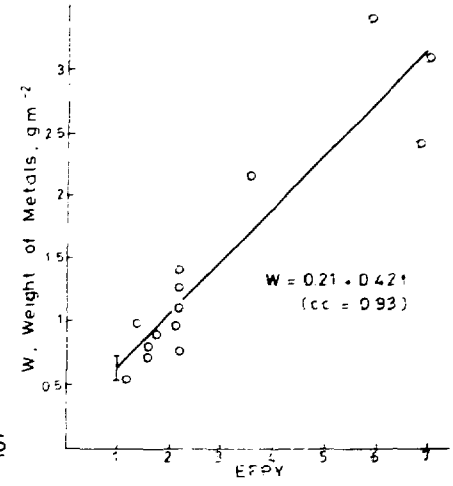


FIG 4 TIME DEPENDENCE OF TOTAL METALS (Fe + Cr + Ni) WEIGHT IN OXIDE ON INCONEL SG TUBES

Chemical Analysis

After removal of particulate oxide by ultrasonic bath treatment the fixed oxide on the specimens was stripped using a 20% bromine/methanol solution. For Inconel SG tube this was a rapid procedure (< 10 mins required). However, for stainless steel specimens it was very slow and it was easiest to cut a thin slice from the specimen at the surface and dissolve away the base metal. The stripped oxide released as flakes was collected on a 3 μm Nuclepore filter and rinsed with methanol. A few individual flakes of oxide were retained for scanning electron microscope (SEM) examination and the rest was fused in a platinum crucible using a mixture of sodium carbonate and sodium tetraborate. The fused mixture was then dissolved in nitric acid for atomic absorption analysis. Cobalt was determined by electrothermal atomic absorption after extraction into chloroform using 1-nitroso-2-naphthol[2].

Results from analysis of Inconel SG tube for iron, nickel, chromium and cobalt are shown in Table VII. With respect to the base metal the fixed oxides are all enriched in chromium which varied from 30-55%. Iron concentrations were in the range 21-38% and nickel 23-48%. The loose particulate oxide was lower in chromium than the fixed oxide and higher in iron and nickel. The range of compositions observed were chromium 16-30%, iron 22-48% and nickel 28-53%. Cobalt concentrations in the fixed oxide range from 0.46-1.7% and show a considerable enrichment (7-30) over the cobalt in the base metal. The oxide metal weight on the SG tubes is shown in Fig. 4 versus EFPY of operation. A straight line fit with a correlation coefficient of 0.93 was obtained. This showed that the total weight of oxide was increasing by about 60 μg cm^{-2} per EFPY.

TABLE VII

Inconel SG Tube Oxide Composition (Normalised to 100% (Fe+Ni+Cr) and Oxide Weight

Plant	(EFPY)	Oxide	Percentage Composition				% Co in Base Metal	Total Metals in Oxide ($g\ m^{-2}$)
			Fe	Ni	Cr	Co		
Reactor C	2.2	Fixed	26	36	38	0.48	0.057	1.25
		Loose	27	52	21			n.d.
Indian Point 2	2.2	Fixed	21	24	55	1.7	0.073	1.10
		Loose	47	37	16			n.d.
Ringhals 2	1.2	Fixed	38	23	39	1.5	0.042	0.55
		Loose	22	53	25			n.d.
Ringhals 2	3.6	Fixed	27	29	44	0.95	0.043	2.15
		Loose	42	28	30			n.d.
P-21	1.58	Fixed	21	48	30	0.62	0.055	0.46
		Loose	38	37	25	0.23		0.26
B-13	2.12	Fixed	26	25	48	0.66	0.059	0.77
		Loose	48	36	16	0.28		0.20
L-25	1.37	Fixed	31	26	42	0.46	0.057	0.71
		Loose	44	41	16	0.30		0.27
I-4	2.21	Fixed	24	32	44	0.50	0.058	0.56
		Loose	44	36	20	0.33		0.20

n.d. = not determined

Results from the analysis of various stainless steel oxides are given in Table VIII. For the fixed oxide chromium concentrations ranged from 28-44%, iron 32-58% and nickel 12-34%. In general, compared with the Inconel SG tube, the stainless steel oxides contained similar amounts of chromium but less nickel and more iron. Cobalt concentrations, with the notable exception of the Loviisa 2 specimens, were similar (0.44-1.15%) to those in the SG tube oxide and also showed an enrichment over the base metal composition. The total weight of metals in the oxide on the stainless steel specimens is somewhat greater than on the SG tube specimens of a similar EFPY.

TABLE VIII

PWR Stainless Steel Oxide Composition
(Normalised to 100% Fe+Ni+Cr) and Oxide Weight

Reactor	EFPY	Oxide	Percentage Composition				% Co in Base Metal	Total Metals in Oxide ($g\ m^{-2}$)
			Fe	Ni	Cr	Co		
B	n.k.	Fixed Loose	47 81	15 5	36 14	0.85 n.d.	0.068	2.32
D	n.k.	Fixed Loose	46 84	12 7	40 9	n.d. n.d.	n.d.	5.0
Surry 2 Manway Insert	1	Fixed Loose	46 38	16 46	38 17	1.15 0.40	0.13	11.6 0.27
Ringhals 2 CVCS Pipe	2.2	Fixed Loose	47 60	16 22	37 18	0.55 0.22	0.13	2.25 0.83
Ringhals 2 Manway Insert	5.8	Fixed	39	19	43	0.44	0.13	10.4
Beaver Valley Manway Insert	0.77	Fixed Loose	32 51	34 45	33 4	0.30 0.11	n.d.	3.9 1.1
	1.02	Fixed Loose	38 72	17 23	44 5	0.49 0.17		7.4 7.8
Loviisa 2 CVCS Pipe	1.5	Fixed	44	12	41	0.059	0.078	6.4
SG Tube	1.5	Fixed	58	13	28	0.084	0.061	1.0

n.d. = not determined
n.k. = not known

The cobalt concentrations observed in the oxide on the Loviisa 2 specimens are of particular interest in relation to data from the other reactors. The cobalt concentrations in the fixed oxide from Loviisa 2 show no enrichment over the concentration in the base metal. In contrast, the cobalt concentrations in the fixed oxide on the other reactor specimens, mainly from Westinghouse plants, show enrichment factors over the base metal cobalt concentration ranging from 4-30. The Loviisa reactors are reported to contain no Stellite. This suggests strongly that the high cobalt concentrations observed in the oxides from Westinghouse type PWRs arise from Stellite wear and corrosion. This conclusion is also supported

by a mass balance study for Westinghouse PWR circuits[3]. It is unlikely that the fact that Loviisa 2 is an all stainless steel plant is significant. Reactor B (Table VIII) is also an all stainless steel plant which is known to contain Stellite, and the oxide analyses show a cobalt concentration similar to those observed on the Westinghouse type plants. On the specimens from BRUCE-CANDU reactors (Table III) very low ^{60}Co arisings were also noted. On these reactors, steps have been taken to replace the majority of the Stellite by other hard-facing materials and steam generator tubing containing a low concentration of cobalt (< 0.015%) has also been used.

For comparison with the PWR specimens, results obtained from two BWR specimens are shown in Table IX. On the reactor T specimen there was enrichment of chromium in the inner fixed oxide and substantial quantities of zinc were present in the oxide. The loose oxide contained less chromium and more iron. Hence, apart from the zinc, the fixed oxide on this specimen was similar in composition to the oxide on stainless steel PWR specimens. With the reactor M specimen both the fixed and loose oxide were predominantly iron based.

TABLE IX

BWR Stainless Steel Oxide Composition and Oxide Weight

Reactor	EFPY	Oxide	Percentage Composition						Total Metals in Oxide ($g\ m^{-2}$)
			Fe	Ni	Cr	Zn	Mn	Co	
T	4.6	Fixed	41	7	31	12	8	0.30	6.3
		Loose	65	6	12	16	1	0.45	3.8
M	8.7	Fixed	60	15	13	7	4	0.28	3.9
		Loose	86	5	2	4	1.9	0.35	2.4

Metallographic Examination

A section through the Indian Point 2 SG tube oxide is shown in Fig. 5a. The oxide is less than 1 μm thick. The surface of the tube was very smooth with no evidence of pits or oxide penetration down grain boundaries. However, some of the SG tube specimens showed penetration of oxide into the metal, e.g. the Ringhals 2 (1981) specimen shown in Fig. 5b. Typically, the oxide extended up to 10 μm into the metal.

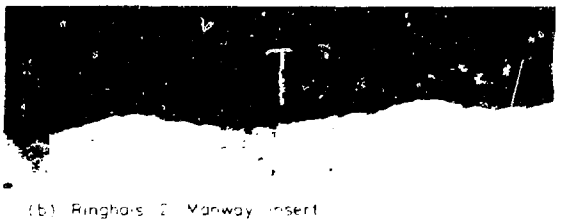
A section through the Ringhals 2 CVCS pipe is shown in Fig. 6a. The pipe surface is quite rough with a peak to valley distance of 20 μm and the oxide appears to follow the surface contours. The manway insert which had an oxide layer up to 5 μm thick in places is shown in Fig. 6b. In general, the stainless steel specimens examined tended to have rougher surfaces than the SG tubes which were invariably almost completely smooth.

(a) Indian Point 2 20 μm

(b) Ringhals 2 (1981)

FIG. 5.
SECTIONS THROUGH PRIMARY SIDE OXIDE ON SG TUBE

(a) Ringhals 2 CVCS Pipe 20 μm



(b) Ringhals 2 Manway Insert

FIG 6. SECTIONS THROUGH PRIMARY SIDE OXIDE ON STAINLESS STEEL SPECIMENS

Scanning Electron Microscopy

An electron-micrograph of the oxide on Ringhals 2 (1983) SG tube is shown in Fig. 7a. There are a number of small particles on the surface but they only occupy a small proportion of the surface area. A more interesting electron-micrograph is that shown in Fig. 7b which is of the underside (i.e. oxide-metal interface) of the oxide stripped from the tube using bromine/methanol solution. The pattern shown corresponds to where oxide has penetrated down grain boundaries as shown on the metallographic sections. The Ringhals 2 and Doel 2 specimens all showed varying degrees of oxide penetration down grain boundaries to a maximum depth of about 10 μm. On the other SG tube specimens examined there was very little evidence of any oxide penetration into the metal.

Electron-micrographs of the stainless steel specimens examined tended to show a rather rougher surface than the SG tubes, as expected from the metallographic sections. The CVCS pipe shown in Fig. 8a appeared to consist of exposed metal grains up to 50 μm across and areas where grains may have been torn from the surface, possibly during manufacture. The manway insert surface shown in Fig. 8b appeared to have been polished in two directions with 50 μm grit.

Analyses of the stripped oxide flakes using X-ray analysis on the SEM tended to confirm the results of the atomic absorption analyses for iron, nickel and chromium. A number of other elements were detected in the oxide including manganese, titanium, silicon, copper and zinc; however, on the PWR specimens these were generally present at levels of less than one or two percent. Analyses of discrete 1 μm spots showed little variation in

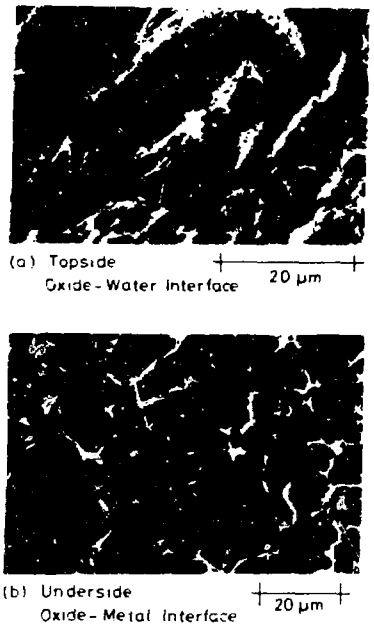


FIG 7 ELECTRON - MICROGRAPHS OF PRIMARY SIDE OXIDE ON RINGHALS 2 (1983) SG TUBE

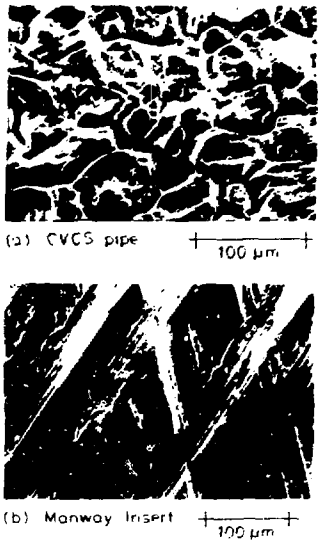


FIG 8 ELECTRON - MICROGRAPHS OF PRIMARY SIDE OXIDE ON RINGHALS 2 STAINLESS STEEL SPECIMENS

composition; although where oxide had penetrated into the Inconel base metal it tended to be slightly more chromium rich than the surface oxide.

Surface Analytical Techniques

Secondary Ion Mass Spectrometry (SIMS). Specimens were examined using a VG SIMS system with an argon ion beam. The beam energy was 4.5 kV, and the current used was generally in the range 20-30 nA. Specimens were exposed for times up to 4 hours, depending on the changes observed in the mass spectrum.

A spectrum from a Ringhals 2 specimen after 5 mins sputtering is shown in Fig. 9. Chromium-52 and iron-56 give strong signals but the nickel-58 signal is very weak. There are strong signals from potassium-39, aluminium-27 and sodium-23. The SIMS sensitivity factors for different elements vary markedly, for instance by three orders of magnitude between alkali metals and transition metals. The raw SIMS data must be processed using relative sensitivity factors to produce quantitative data. In the present work these factors were derived experimentally for iron, nickel and chromium; for other elements literature data[4] were used. The other main elements which were detected in the spectra included magnesium -24 , silicon-28, titanium-48, manganese-55, zinc-64 and zirconium-90. The signals at 68 amu (atomic mass units) and 72 amu are probably due to $^{52}CrO^+$ and $^{56}FeO^+$, respectively. In the region above 100 amu signals from Me_2^+ species, e.g. Cr_2^+ and $CrFe^+$, appear.

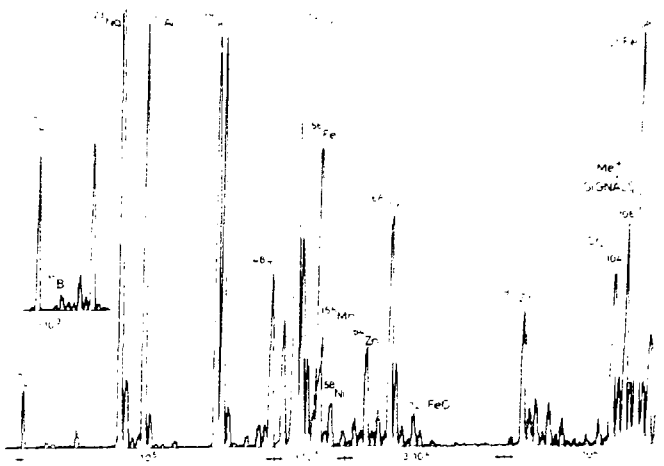


FIG.9 SIMS SPECTRUM OF RINGHALS 2 AFTER 5 MINS SPUTTERING

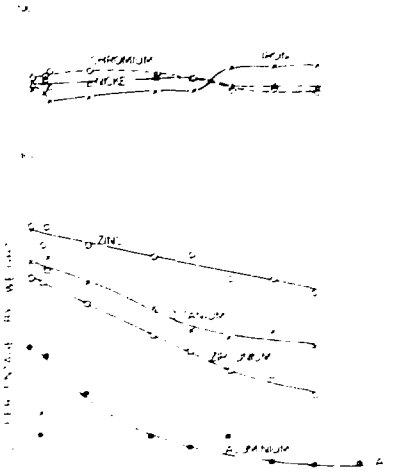


FIG.10.
SIMS SPUTTERING PROFILE THROUGH RINGHALS 2 OXIDE

The results obtained from sputtering through the Ringhals 2 oxide are plotted in Fig. 10. The profile shows a slight decrease in chromium and a slight increase in iron through the oxide. The nickel concentration is fairly constant. Aluminium, zirconium, titanium and zinc are present at levels from 0.5-3.0% at the oxide surface but their concentrations fall as sputtering proceeds through the oxide. The Inconel SG tube specimen examined showed similar profiles to the stainless steel specimens.

One of the main points to emerge from the SIMS study was that there is no clear evidence of a layer-type structure for the PWR fixed oxides; for instance, an inner chromium rich oxide layer and an outer layer consisting mainly of iron and nickel oxides. Instead, the concentration profiles for the major oxide constituents, i.e. iron, nickel and chromium were relatively smooth demonstrating that the oxide is fairly homogeneous in composition.

X-Ray Diffraction and Laser Raman Spectroscopy. Results from examination of the PWR reactor specimens with low angle X-ray diffractometry on a Philips vertical diffractometer showed that the oxide in all cases corresponded to a magnetite type spinel structure. This was also revealed by examination using Laser Raman Spectroscopy at Harwell Laboratories. To date, it has not proved possible to determine whether the oxide is a normal or inverse spinel.

OVERALL SUMMARY

Inconel SG and stainless steel specimens from a number of different PWRs have been examined. The chemical composition of the oxides is very similar, they are all enriched in chromium by about a factor of two

over the base metal chromium concentration of 16-18%. The other major elements in the oxide are iron and nickel. In addition, manganese, titanium, silicon, cobalt, copper and zinc are present at levels of a few percent or less. The oxide thickness on stainless steel specimens is higher than on the Inconel SG tube after a similar EFPY. For instance, the thickness on the Ringhals 2 stainless steel manway insert after 5.8 EFPY is up to 5 μm, compared with about 1.5 μm on SG tube. On the SG tube the oxide appears to be growing at a linear rate. There are insufficient data available to determine the rate of growth of stainless steel oxides. The stainless steel specimens are much rougher than the SG tube specimens, this may partially account for the higher oxide burden on the former.

The predominant gamma-emitting radionuclide on virtually all the specimens examined, including some from BWRs, is ^{60}Co, formed from ^{59}Co. It is considered that the major source of the ^{60}Co is the high cobalt alloy Stellite. This conclusion is supported by data from the Russian designed Loviisa 2 VVER reactor. The ^{60}Co levels on specimens from this reactor which contains no Stellite are about two orders of magnitude lower than on Westinghouse reactors. Low levels of ^{60}Co are also found on Inconel SG tube specimens from CANDU reactors where efforts have been made to remove the majority of the Stellite and to lower cobalt impurities in the Inconel SG tubes.

Most of the other gamma-emitting radionuclides measured in significant quantities on the specimens, e.g. ^{54}Mn, ^{58}Co, ^{65}Zn, ^{106}Ru, ^{125}Sb are shorter-lived than ^{60}Co. Therefore, the ^{60}Co deposited on out-of-core surfaces will continue to dominate radiation fields for tens of years after shutdown. The long-lived gamma-emitting radionuclide, ^{94}Nb, which is likely to be the dominant contributor to radiation fields after long decay periods, was detected on one of the specimens examined.

On nearly all the specimens examined alpha-emitting actinides have been detected; typical levels range up to 5 Bq cm^{-2} of longer lived actinides (^{238}Pu, ^{239}Pu, ^{240}Pu and ^{241}Am) on PWR specimens and 100 Bq cm^{-2} on BWR specimens. These actinides are thought to arise as a result of fuel failures during operation. Clearly, their presence on out-of-core surfaces must be taken into account in assessing decommissioning scenarios. At the time of writing the levels of pure beta-emitting and electron capture radionuclides on the specimens have not been determined. However, these radionuclides, e.g. ^{63}Ni and ^{59}Ni, may well dominate residual radioactivity, although not dose-rates, over long decay periods.

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NEW METHOD FOR DECONTAMINATION OF CONCRETE WITH MILLING CUTTER

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ABSTRACT

The techniques up to this time on removing the contaminated concrete surface have some problems, such as irregular depth of removal, difficulty of collecting the scraped debris and so on. As a solution of these problems, new method with original milling cutter and vacuum collecting system had been developed.

Milling cutter can scrape the concrete surface to a few millimeters depth accurately by one pass. The scraped debris is shaped uniform powder and is collected almost 100 % by vacuum collecting system. This method has so many advantages, such as radioactive waste reduction, prevention from internal exposure of workers, recontamination prevention and easy measurement of residual radioactivity after decontamination.

This report describes the development, demonstration, experience and outline of new method completed as "Clean Cut Removal System".

INTRODUCTION

When radioactively contaminated nuclear facilities are to be decommissioned, the decontamination of concrete surface of floors and walls in the building is necessary. Purposes of the decontamination are as follows:

- to minimize the radiation exposure of personnel
- to minimize the activity of the surface contamination to a level which permits unrestricted use of previous controlled area, i.e. to permit materials from a demolished plant to be treated as normal building waste.

The depth to which radioactivity penetrates into concrete is generally a few millimeters depending upon the nature of concrete (its porosity), the duration of exposure, the composition of contamination and the condition of protective coating. The peneration depth is reported that the KRB-A Gundremmingen Power Plant, West Germany, approximately 1 mm for protective coating by epoxy paint and approximately 5 mm for no coating. The Gentilly-1, Canada, shows the same results.

Required efficiency for concrete decontamination methods is as follows:

(1) Prevention from internal exposure of workers: a non-dust method or a method which can completely collect the produced dust.

(2) Minimization of radioactive waste: accurate removal of contaminated concrete surface.

(3) Easy collection of radioactive waste.

(4) Easy measurement of residual radioactivity after contaminated surface removal (smoothly removed surface).

(5) Non-recontamination.

Contamination which penetrates into concrete cannot be removed by using hand scrubbing method or chemical decontamination methods, but can be removed using mechanical methods such as chipping hammers, grinders, scabblers, drills & spallers and sand blasters. All these methods, however, have such defects that they entail the internal exposure of operating personnel as well as the spread of contamination due to the aerosol activity, increase radioactive waste due to the removal of non-contaminated concrete, or involve extra labor for debris recovery.

So, it was necessary to develop a new method which can safely and easily remove contaminated concrete surface. After the basic research, the concrete cutting method was selected by the following reasons:

(1) In order to remove concrete surface with a depth of several millimeters uniformly and smoothly, scraping (thin cutting off) is a more effective method rather than hitting and scarifying.

(2) Fine-grained particles are effectively obtained by cutting and so it is easy to collect by vacuum system.

(3) There are no secondary radioactive wastes because cutting method does not use extra materials.

DECONTAMINATION OF CONCRETE WITH MILLING CUTTER

Concrete milling technique and the produced debris collecting technique had been researched & developed since 1983. Many subjects had been solved by experiments, fabrication of experimental machines and demonstrations, and prototype machines for practical use was completed as "Clean Cut Removal System".

Development of Concrete Surface Milling Technique

It is very difficult to scrape the high strength concrete surface, because concrete includes hard aggregate and its material characteristics are not uniform. There were many subjects to solve, such as change of reaction force, impact force, excessive power load and tear and wear of cutter tips. Several equipments were made to get basic data for accurate concrete surface milling technique, and various experiments were pursued.

The milling cutter has many tips arranged at regular intervals on the outer circumference of the cylindrical body.

Experiments were practised by equipments which include the milling cutter rotated with electric motor, the situation between cutter and concrete sample was kept constantly and the speed of cutter movement was changeable.

Test samples were made as plates using the concrete of compressive strength, Fc = 300 kg/cm^2 and maximum coarse aggregate size, 25 mm.

Full view of experiment is shown in Fig. 1. Experiment on concrete cutting is shown in Fig. 2.

Experiment for cutting device. The experiments for cutting device had been pursued to gain the data of the following factors. (1) width of cutter (2) rotating speed of cutter (3) moving speed of cutter (4) cutting depth. These data had been examined from the following points of view. (1) pressing force of cutter to concrete (2) power for cutter movement (3) power for cutter rotation.

The result indicated that the moving speed of cutter had most influence on the pressing force of cutter to concrete and the cutting depth had next weight on it. The pressing force was influenced a little by rotating speed of cutter. As the rotating speed of the cutter became faster, the pressing force tended to be less. The pressing force was in proportion to the width of cutter. The maximum pressing force of cutter to concrete was 1000 kg under the condition of cutter width 20 cm, cutting depth 5 mm and moving speed 100 cm/m.

The power of cutter movement was not influenced by above mentioned factors and it indicated constant value. Provided that the cutter width was constant, the power for cutter rotation was influenced mainly by rotating speed, secondary by moving speed and a little by cutting depth.

Measurement of vibration. To gain the design factors for experimental machine, vibration was measured during the experiment of milling cutter. The vibration acceleration showed the peak value when the cutter scraped aggregates in concrete and it indicated the necessity of shock absorbing mechanism. The biggest value was measured in cutting device, the next was experimental equipment, and the third was concrete samples. The acceleration of vibration showed different value depending on the frequency, and three frequency zones, low (50 - 150Hz), middle (300 - 500Hz), high (800 - 900Hz), indicated high value. Maximum acceleration was measured more than 30 G.

Improvement for cutting material. Experiments had repeated to select most suitable material as bits for cutter. The important factors for bits are hardness and tenacity. After trials and errors, some kind of cemented carbide tip was found as most suitable material. And for the practical use, the cutter had designed with exchangeable tips.

During the experiments, it sometimes happened that the tips were torn and worn by the big impact force. To solve this defect, the hydraulic shock absorber was adopted, then the tear of tips lost and the wear of tips diminished. The tips wore 1 mm by scraping the concrete surface about 80 m long. The cemented carbide tip has sharp-angled edges on its four faces so that one tip can be used four times. Relationship between wearing length and cutting volume is shown in Fig. 3.

By using these experimental data, many subjects like the shape and size of cutter, rotation number, motor power, control system for smooth and accurate cutting and shock absorber, were examined and new type machines were designed. Fabrication of experimental machines and test cuttings were repeated several times and the original milling cutter system was completed.

Development of Concrete Debris Collecting Technique

It is very important subjects to collect the debris which was arisen by scraping the concrete surface, because scattering of debris causes the internal exposure of workers and the recontamination of scraped surface. Experiments was pursued to create the suitable collecting device.

Measurement of scraped debris size. Scraped debris which is smaller than 10 μm becomes aerosol particles, and bring on internal exposure. It is rather difficult to transport the debris bigger than 1 mm. So the distribution of dust was measured in weight. Measurement was practised by standard sieve-analysis method and liquid precipitate method.

The distribution of scraped debris size is shown in Fig. 4.

The two straight lines were lead from the data which was measured by standard sieve-analysis test and liquid precipitate method.

Rosin-Rammler logarithmic function which was guided from two straight lines is as follows :

$$R = 10^{-0.00081 \times D^{1.158}} \times 100$$

R : weight of debris left on sieve (%)

D : debris size (μm)

The scraped debris by milling cutter was distributed 90 % to the range of 10 μm - 1 mm. Bigger than 1 mm was 7 % and smaller than 10 μm was about 2.2 - 2.8 %. The dust was such uniformly distributed and almost every part was smaller than 1 mm, so it was very suitable for transportation.

Measurement of extracted air volume and pressure loss. In actual concrete decontamination works, it is required to transport the scraped debris through at least about 30 m long flexible hose. Air velocity that does not leave the dust in the flexible hose, is over 15 m/s which is twice of terminal settling velocity 7.5 m/s for 1 mm size particles. To get the required air velocity, relationship between extracted air volume and pressure loss were measured by changing the size and shape of flexible hose. After above mentioned examination, the capacity of blower was decided and the design of prototype machine was practised.

Examination of mechanism required for dust collection. Several methods that could collect dust safely and easily were examined. For example, clear away method of dust which is stuck on bag filter, method of filter and drum exchange, volume detection of debris in the drum, minimizing debris volume in the drum and security assurance against breaking filter or flexible hose, etc.

Clean Cut Removal System

Clean Cut Removal System is a method which can accurately scrape the contaminated part of concrete by rotating the cylindrical milling cutter with cemented carbide tips and collect the concrete debris and transport it simultaneously into a drum. This system can cut off concrete surface with aggregate & coating smoothly to a depth of several millimeters.

Prototype machines. The original milling cutter is the feature of Clean Cut Removal System which can be applied to all types of concrete surface by changing the supporting and moving mechanism, according to the shape and space of objective facilities. Presently, large type for spacious floors and high walls and small type for narrow places and corner were completed as prototype machines for practical use.

Large type cutting device is shown in Fig. 5. Large type collection device is shown in Fig. 6. Small type cutting device and collection device are shown in Fig. 7.

- Clean Cut Removal System
  - Large type
    - Large type cutting device
    - Large type collection device
  - Small type
    - Small type cutting device
    - Small type collection device

Advantages. Clean Cut Removal System has the following advantages.

- This system, which had various control system, is able to scrape concrete to a depth with aggregate or coating accurately and smoothly. So it can minimize the volume of radioactive waste.

- The scraped dust are simultaneously collected and accumulated directly into a drum, so there are no troublesome works for recovering, no fear of the operator's internal exposure and no recontamination of the surroundings.

- The surface of concrete after cutting is very smooth and free from dust, making it easy to measure the residual radioactivity and to carry out the repair work.

- Secondary radioactive waste is not produced because abrasives and cooling water are not used.

DEMONSTRATION BY NON RADIOACTIVE CONCRETE STRUCTURE (COLD TEST)

Various kinds of cold test were carried out in a concrete structures to confirm the effectiveness of each device. One of the structures was 6 m wide, 8 m long and 3.7 m high, and one part of its wall and floor was finished up epoxy coating. Concrete strength was 240 kg/cm^2 and the maximum size of coarse aggregate was 25 mm.

Cutting Range of Demonstration

The large type cutting device was able to cut the surface of wall automaticaly from 30 cm to 210 cm at the height of floor level, and cut the surface of floor except the range of 50 cm adjacent to wall side. As it can be estimated usually that the contamination zone is spread to about 150 cm high from floor level, the height of demonstrated cutting is enough.

The small cutting device was able to cut the surface of floor perfectly, and cut the surface of wall to 85 cm high from floor level, except the range of 10 cm near the corner. The cutting range of demonstration is shown in Fig. 8

Milling Test of Steel Plate

Many steel materials are laid on the surface in the nuclear facilities, and are sometimes contaminated. Therefore, the milling test of the steel plate on the wall and floor surface was made to confirm whether the cutter and/or the tips were broken or not.

As a result, steel plates were able to be scraped with adjusting the cutting speed & depth, but the conditions were not same to the concrete. The cutter and tips did not receive any damage by steel milling.

Surface After Removal

The surface after removal by the small type cutting device is shown in Fig. 9 and surface after removal of epoxy coated concrete is shown in Fig. 10. The epoxy coated concrete surface, which was removed 3 to 5mm deep at one pass, was very smooth, and the cutter scraped aggregates evenly. Because the surface temperature of cutter tips weren't hot just after cutting, and can be touched by finger without any problem. This was caused by air cooling effect of collection device, and epoxy coating could be scraped easily without melting.

In order to confirm whether the scraped dust is left on the cut concrete surface or not, fluorescent material applied concrete surfaces were scraped. As the result, there were rarely the fluorescent materials left on the smoothly removed concrete surface, and it can be said that no recontamination after removal.

Efficiency of Collection Device

Measured efficiencies of collection device are the followings. 99.9 % of the concrete dust was collected in the bag filter and directly put into a drum. Fine concrete dust having passed through the bag filter was perfectly collected by the pre-filter and HEPA filter.

The number of dust particles was counted with the particle counter in order to check the purity of exhausted air. As a result, the number of dust particles above 0.3 μm was 0 - 1/0.01 ft^3 at the outlet of the HEPA filter unit, though it was 10,000 - 200,000/0.01 ft^3 at the inlet of the HEPA filter unit.

Thus the air exhausted from the blower is very pure, and contaminated concrete can be removed safely. Concrete dust collected in drum is shown in Fig. 11.

Dust Leak Test

In order to confirm whether the concrete dust was scattered from the cutter hood and upper part of drum under exchange, the leak test was carried out by the following way.

The green house was 2 m wide, 2 m long and 3 m high, covered with non-charged film. The clean air, which passed through the HEPA filter, was sent to the green house and made aerosol particles minute and steady. After such preparations, the number of particles were countered four cases, before, during the cutting and before, during drum exchange, then the difference of data were examined.

The samples were collected from the side & bottom of cutter hood and the upper part & the 30 cm outside of drum under exchange. View of dust leak test is shown in Fig. 12.

EXPERIENCE ON CONTAMINATED FACILITY (HOT TEST)

Decontamination Work Condition

The decontamination of the concrete surface was carried out on the floor of the radioactive waste disposal facility for about a week in November, 1986.

This facility was intended to reform after decontamination and partial dismantlement.

Although activity was low, it was spread to the entire surface of the floor. The maximum activity was 2.5 kBq/100cm^2.

The small type cutting device and the large type collection device were used.

The operators were wearing a half-covering mask for safety. No temporary particular fences and ventilation equipment for the operation were used. The removal operation was carried out by three persons: an operator for the small cutting device, an assistant for measuring radioactivity after the removal, and a worker required for shift. Experience of contaminated facility is shown in Fig. 13.

Decontamination Effect

After one-pass cutting of 3 mm removal depth, the surface contamination density in the most part of surfaces was reduced to the background level. Particular areas, where the contamination had penetrated deeply, was spottily found in several places, and the contamination was reduced to the background level by several extra removals (15 mm). The removed surface was smooth, so the residual contamination was measured very easily.

Radiological Safety

In order to examine the influences of the removed dust to the surroundings, radioactive concentrations in the air were measured at the following locations: the working room, around the milling cutter hood, outlet of the HEPA filter and upper part of drum under exchanging. The results indicated the same value to the background level and the radiological safety was confirmed against internal exposure of workers and recontamination of the surroundings.

Cleaning of Devices After Operation

The cutting device and the waste collection device, which have been used for the concrete decontamination, were carried away from the facilities after cleaning with water and radiation survey.

Short hoses of the cutting device to which radioactive dust adhered, all filters, and parts of the flexible hose were disposed as the radioactive waste.

CONCLUSION

The most important subject while decommissioning works is safety. The essentials for maintaining radiological safety are reduction of radiation exposure to workers and prevention of the contamination spread. Safety is assured by easy collection of radioactive waste and simple operation.

Clean Cut Removal System can remove concrete surface accurately and collect the scraped debris into a drum systematically, so this system is more safe than others. The present system will be of great help to maintenance, repair and decommissioning of nuclear facilities.

Sato Kogyo will continue its endeavours to improve Clean Cut Removal System to meet the user's expectations.

ACKNOWLEDGEMENTS

The authors wish to express the greatful appriciation to the support and cooperation of many individuals of the Japan Atomic Energy Research Institute. This development program was pursued by receiving a subsidy from the Science and Technology Agency, Japan.

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Fig. 1 Full view of experiment

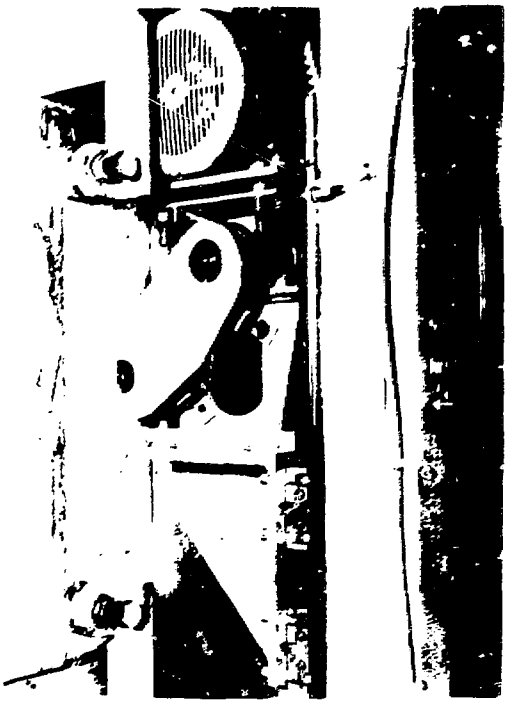

Fig. 2 Experiment on concrete cutting

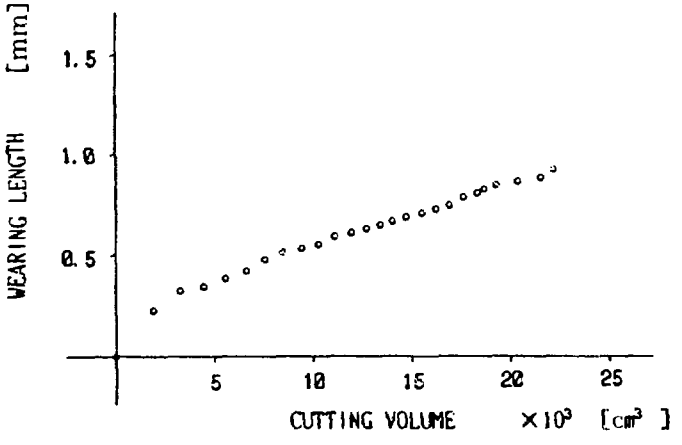

Fig. 3 Relationship between wearing length and cutting volume

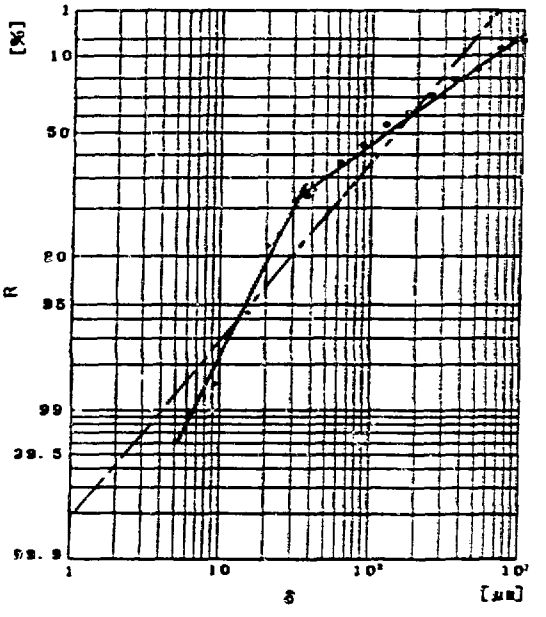

Fig. 4 Distribution of debris size

Fig. 5 Large type cutting device

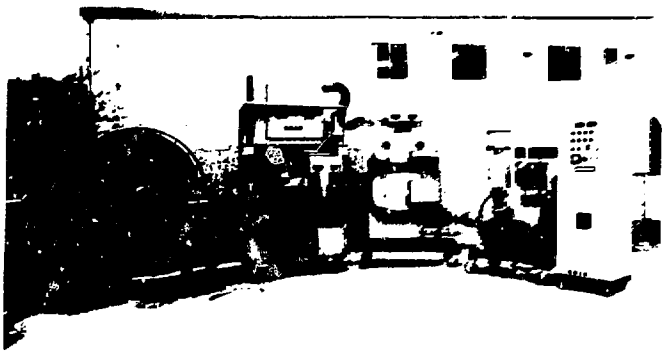

Fig. 6 Large type collection device

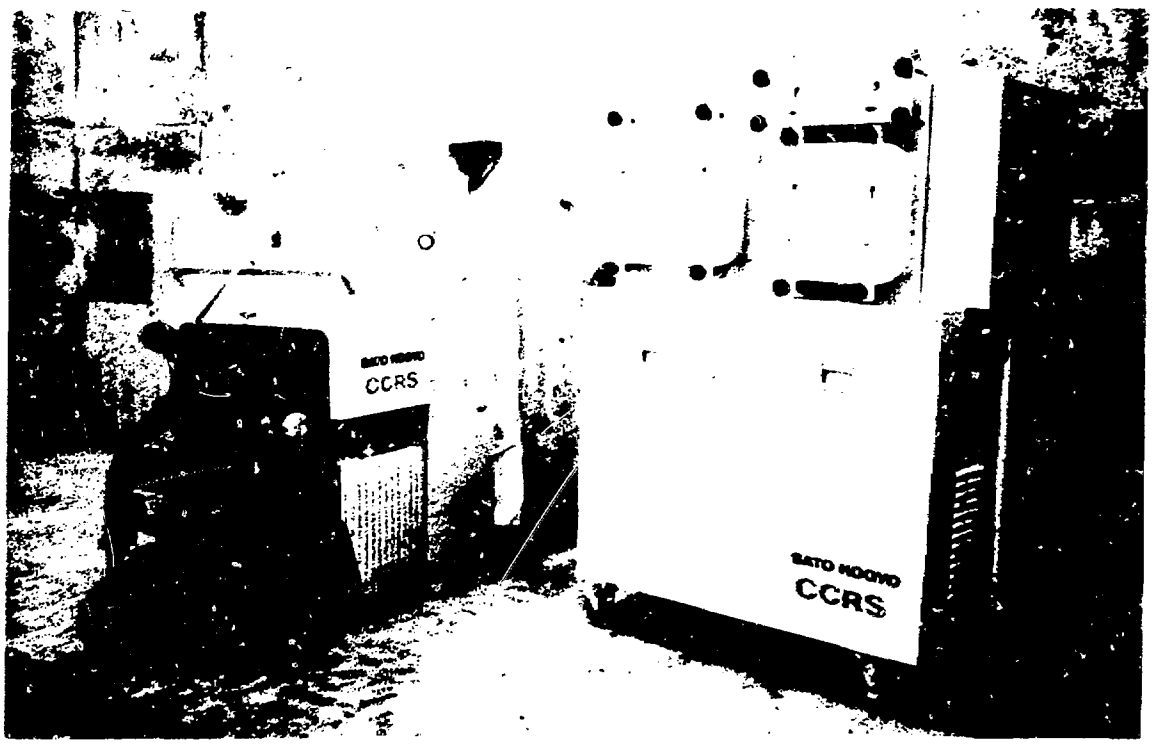

Fig. 7 Small type cutting device and collection device

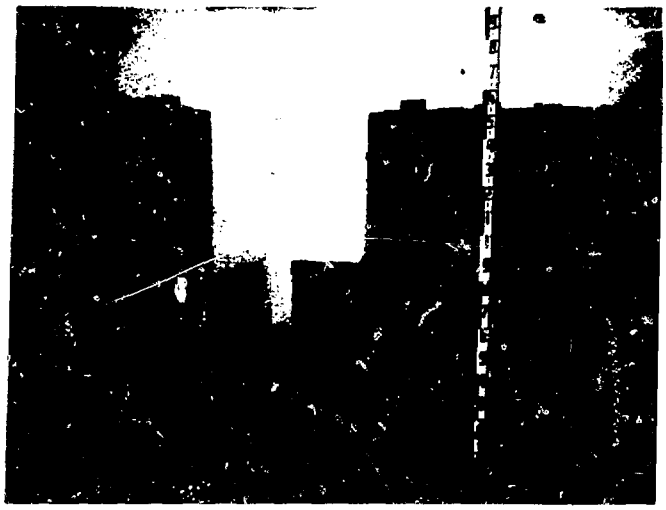

Fig. 8 Cutting range of demonstration

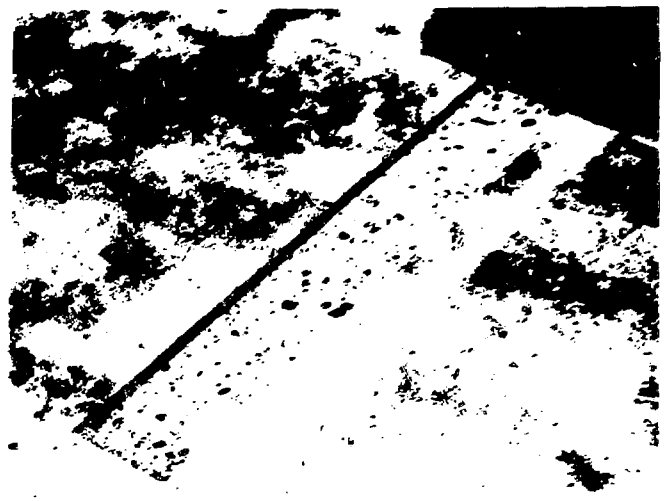

Fig. 9 Surface after removal by small type cutting device

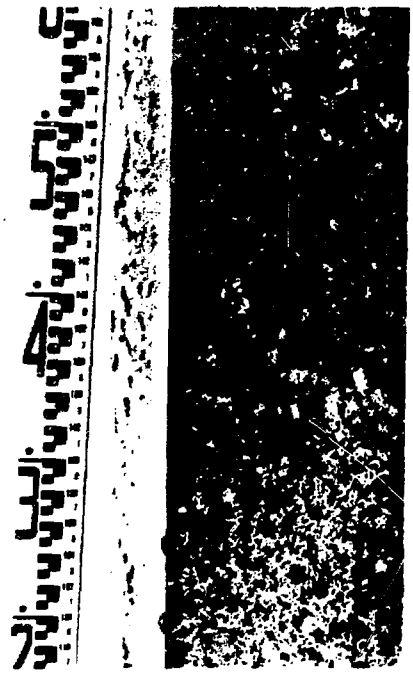

Fig. 10 Surface after removal of epoxy coated concrete

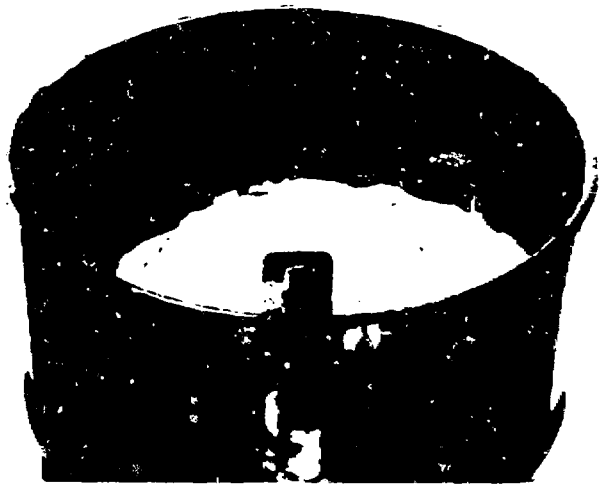

Fig. 11 Concrete dust collected in drum

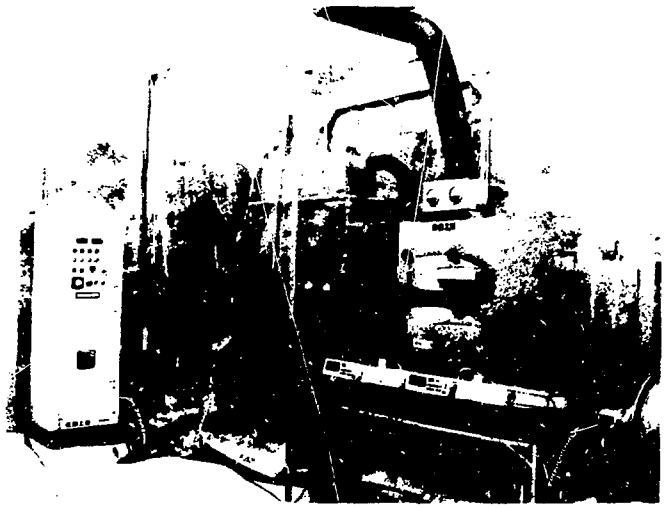

Fig. 12 View of dust leak test

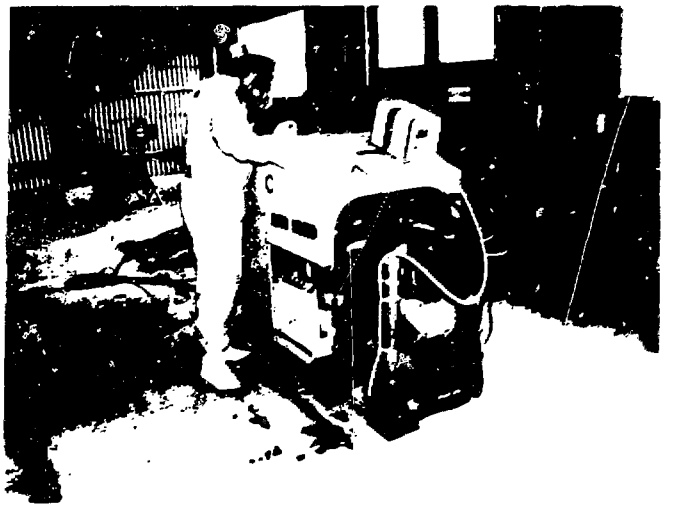

Fig. 13 Experience of contaminated facility

POSTER SESSION

DECOMMISSIONING TECHNIQUES AND TOOLING

METAL SEGMENTING USING ABRASIVE AND RECIPROCATING SAWS

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ABSTRACT

This paper presents the results of a study conducted at the Pacific Northwest Laboratory (PNL) under U.S. Department of Energy sponsorship to evaluate a light-weight, high-power abrasive saw for segmenting radioactively contaminated metal components. A unique application of a reciprocating mechanical saw for the remote disassembly of equipment in a hot cell also is described. The results of this work suggest that use of these techniques for selected remote sectioning applications could minimize operational and access problems and be very cost-effective in comparison with other inherently faster sectioning methods.

INTRODUCTION

Electrically powered, hand-held abrasive saws are used extensively for industrial metal working operations. Although these hand-held units can achieve reasonable cutting rates, the units are heavy and physically demanding on the operators. The studies reported in this paper were conducted to evaluate a lightweight, high-power hydraulic cut off saw as a possible alternative to plasma torch technology for the remote sectioning of contaminated equipment.

These studies included a detailed investigation of cutting rate, blade life and waste generation under constant cutting force or constant cutting rate conditions. Special waterproof blades were obtained and the saw was modified to permit testing both under dry cutting conditions and with a water spray incorporated in the blade guard to attenuate the cutting smoke and sparks. Remote tests of the high-power hydraulic saw also were conducted in the PNL size reduction facility[1] by sectioning various contaminated components with the saw mounted on the end of a master-slave manipulator.

This paper also describes the adaptation and application of a standard industrial pneumatically powered reciprocating saw for the remote disassembly of equipment in a hot cell. Information presented includes an illustration of the saw modifications and examples of the advantages of this cost-effective alternative sectioning technique.

ABRASIVE SAW

The abrasive saw used for these studies is shown in Figure 1. The unit was 28-cm wide by 50-cm long and had a 3.2-mm thick by 25-cm diameter blade. It weighed only 10.5 kg, yet was powered by a 7500 W hydraulic motor. The light weight and high power-to-weight ratio made this saw comparatively easy to use for manual cutting operations, with good potential for adaptation for remote applications using power manipulators or robotics.

FIGURE 1. High-Power Hydraulic Cut-Off Saw

A series of initial cutting tests demonstrated the versatility of the high-power hydraulic saw. It could rapidly section a variety of metals (carbon steel, stainless steel, aluminum) plus nonmetallics such as plastic and wood without changing the blade or operating conditions. Unlike the plasma arc torch, the saw could section components with double-wall construction and it proved relatively insensitive to changes in material thickness or part geometry. Moreover, the hydraulic saw did not require the component rigidity needed for reciprocating saws or other mechanical cutting devices when sectioning thinner materials.

A special pivoted test fixture (Figure 2) was constructed that permitted evaluation of cutting rate, blade life and waste generation under constant cutting force or constant cutting rate conditions. Sections of 6.4-mm thick stainless steel were cut into strips at a series of constant force levels ranging from 20 to 160 N, and at constant feed rates ranging from 2 to 17 mm/s. Tests also were conducted to compare cutting rates for 1.3-cm, 2.5-cm and 5.1-cm thick stainless steel plate.

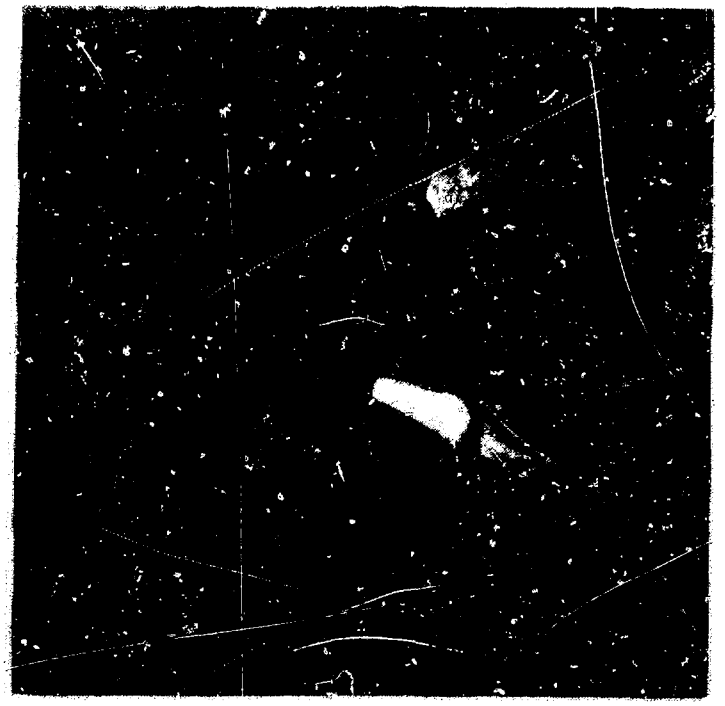

FIGURE 2. Test Fixture Used to Evaluate Metal Cutting Parameters

The diameter of the abrasive blade was measured before and after each run to determine blade wear. As part of these tests, special waterproof blades were obtained and the saw was modified to permit testing both under dry cutting conditions and with a water spray incorporated in the blade guard to attenuate the cutting smoke and sparks.

The results of the cutting rate tests are shown in Figure 3 for dry cutting and for wet cutting using a 35 cm^3/s water spray. For dry cutting of 6.4-mm thick stainless steel, the cutting rate increased approximately linearly with the cutting force, with a threshold value of about 25 N for initiation of cutting. At the lower cutting rates (<8 mm/s), there were sparks but little dust or smoke. The kerf was 5.8-mm to 6.4-mm wide. At the highest cutting rates the kerf was narrower (~4.0 mm), but there was substantial dust generation.

The dry cutting rate decreased for thicker material, even when normalized for the cross-sectional area of the cut. At 100 N cutting force, the respective cutting rates for 0.6-cm, 1.3-cm and 2.5-cm thick stainless steel were 6.8 mm/s, 2.2 mm/s and 0.8 mm/s.

The wet cutting rate (Figure 3) was nonlinear with cutting force. The cutting rate remained low (<3 mm/s) and almost constant for cutting forces less than about 125 N, and then increased sharply to values equivalent to those for dry cutting. The rate for both dry and wet cutting changed slightly with blade wear and decreasing blade diameter. At low cutting rates, the rate increased with blade wear, but decreased with wear at high cutting rates.

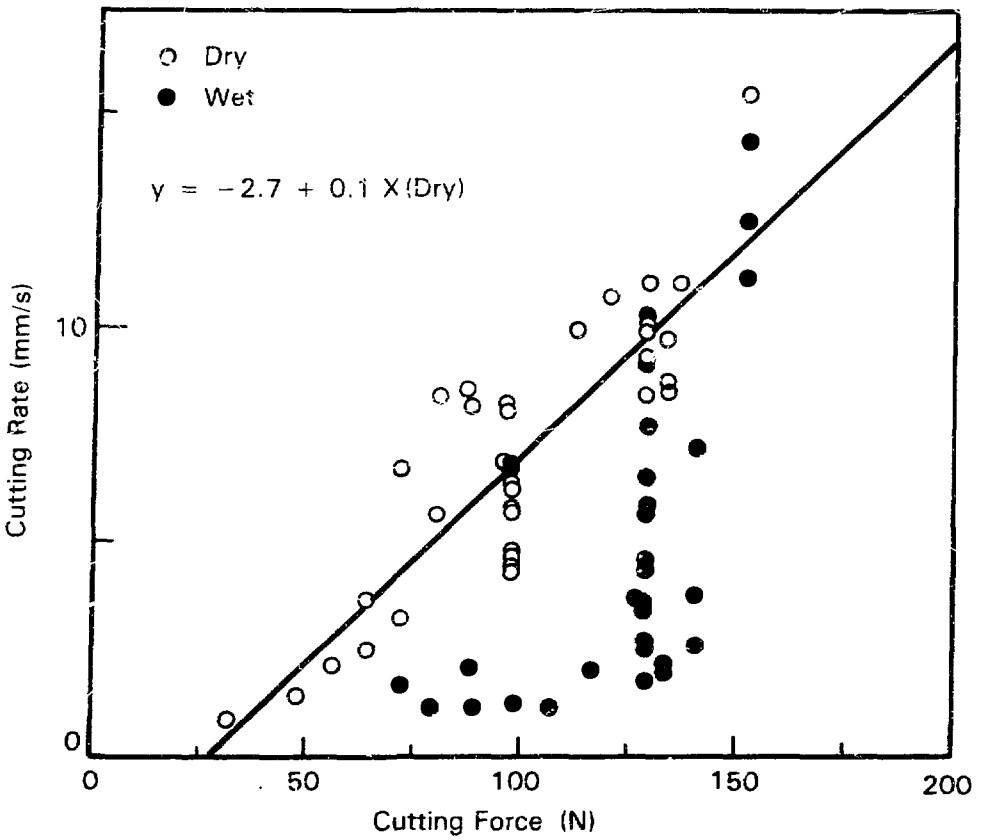

FIGURE 3. Cutting Rate as a Function of Cutting Force for 6.4-mm Thick Stainless Steel

The blade wear data for dry and wet cutting are shown in Figure 4 as a function of cutting rate. Blade wear is normalized with respect to the amount of metal cut, and reported as cm^2 of blade wear per cm^2 of metal removed. Blade life, or the area of metal that can be cut before blade changeout at ~15-cm diameter, can be estimated by dividing the blade wear/metal removal ratio into 325. The cubic centimeters of equivalent waste solid volume also can be estimated from this ratio and the previously noted kerf width values. The waste from abrasive cutting is a coarse, nonadherent, granular material that is easily collected for disposal.

The most important observation with respect to the wear data summarized in Figure 4 is that blade wear increases rapidly and linearly with cutting rate for both dry and wet cutting. This means that blade life can be extended significantly and waste volumes can be reduced by operating at moderate cutting rates. For example, approximately 8.2 m of 6.4-mm thick stainless steel could be cut per blade at 1.7 mm/s as compared with less than 24 m per blade at a 10.2 mm/s cutting rate.

Constant cutting rate tests were conducted using a motor drive on the test system to move the blade through the test samples at rates ranging from 2 to 17 mm/s. The normalized blade wear/metal removal ratios were similar for dry cutting either 0.6-cm, 2.5-cm or 5.1-cm thick stainless steel plate, and

tended to be slightly lower than the constant cutting force ratios for similar cutting rates. However, the wear ratios for wet cutting at a constant rate were almost 50% lower than the ratios for the constant cutting force tests over the entire range of cutting rates.

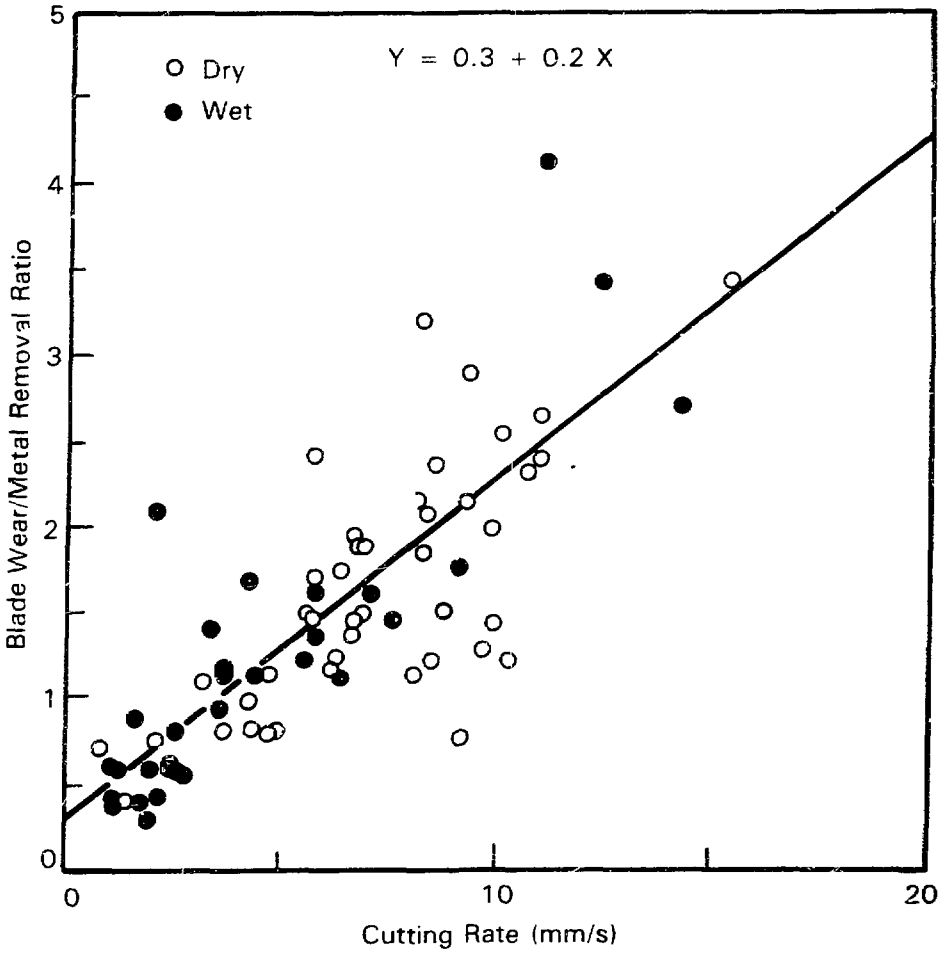

FIGURE 4. Ratio of Blade Wear to Metal Removed as a Function of Cutting Rate for 6.4-mm Thick Stainless Steel

Remote tests of the high-power hydraulic saw were conducted in the PNL size reduction facility by sectioning various plutoniumcontaminated components with the saw mounted on the end of a masterslave manipulator as illustrated in Figure 5. The saw could be guided with the manipulator and adequate force could be exerted to produce acceptable cuts. The hoses connecting the saw to the hydraulic power unit somewhat limited maneuverability and control, but this would not be a problem for a power manipulator or robotics application. The most encouraging finding was that the saw could tolerate substantial misalignment and twisting during cutting without blade breakage.

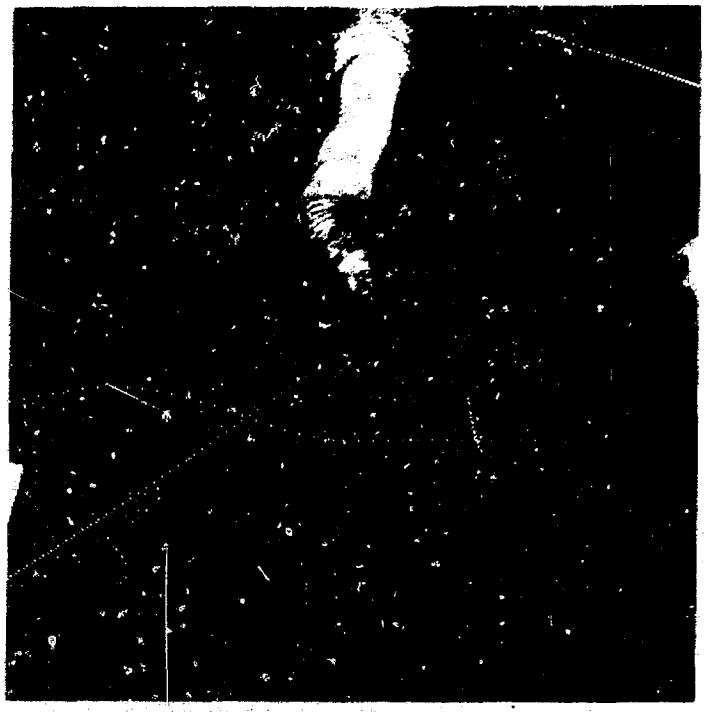

FIGURE 5. Remote Testing Arrangement for the High-Power Hydraulic Cut-Off Saw

RECIPROCATING SAW

A standard industrial pneumatically powered reciprocating saw was adapted for remote operation and used to section large components inside a PNL hot cell[2] at a significant savings in time and cost over more rapid cutting techniques such as the plasma arc torch. This remote sectioning work included the disassembly of a large furnace by using a 41-cm long saw blade to cut 10- and 15-cm diameter pipes and 5.1 x 13 cm stainless steel rails, and the dicing of 70-cm diameter stainless steel vessels and contents using a 94-cm long blade.

The saw was adapted for remote hot cell service as shown in Figure 6 by equipping it with a special pneumatic clamp and a pneumatic feed system to hold the saw against the work piece. Using the overhead crane in the cell, the saw could be remotely attached to pipes, beams and other equipment protrusions as illustrated in Figure 7. Because of the pneumatic feed system, the saw would operate equally well in any position including upside down.

This versatility in placement and operating position saved substantial time and cost compared to the use of a plasma arc torch and other faster cutting devices, since these latter techniques required extensive repositioning of the equipment within the cell to provide adequate operator access with the manipulators. Labor costs also were minimal for the reciprocating saw despite its low cutting rate because it could operate with almost no operator attention. The pneumatic drive was immune to damage from stalling and the system was designed to automatically shut off at the completion of each cut. The only required maintenance was periodic blade replacement, approximately once per shift.

FIGURE 6. Reciprocating Saw with Pneumatic Clamp and Blade Feed System for Remote Sectioning Applications

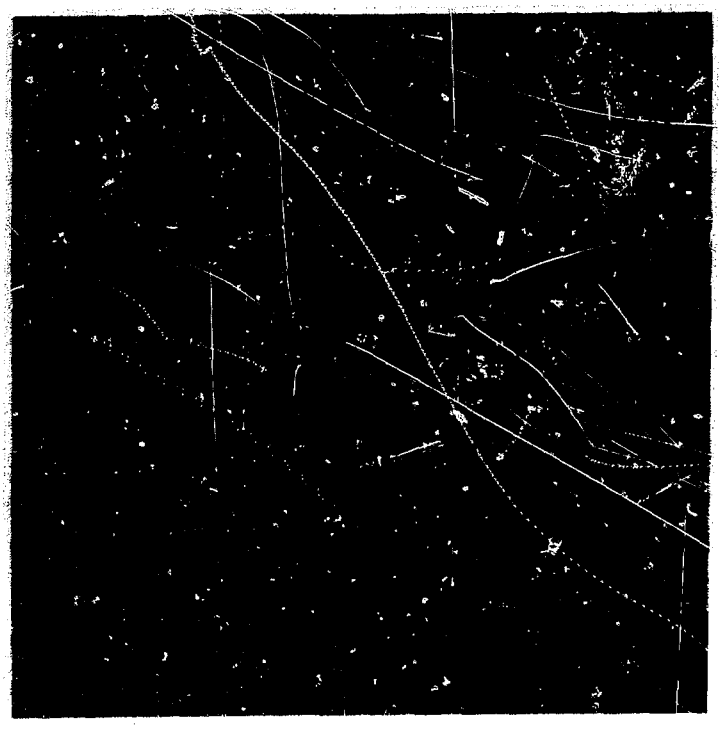

FIGURE 7. Remote Sectioning of Hot Cell Equipment Using the Reciprocating Saw

SUMMARY AND CONCLUSIONS

This paper describes the evaluation and application of two different metal segmenting devices: a light-weight, high-power abrasive saw and a reciprocating saw adapted for remote operations. Based on these studies, it is concluded that the hydraulically powered abrasive saw, because of its high power-to-weight ratio, versatility and comparative insensitivity to adverse cutting conditions, represents a practical alternative to the plasma arc torch for many remote sectioning applications using power manipulators or robotics. Blade wear, waste generation and smoke problems can be minimized by operating at moderate cutting rates and by incorporation of a water spray.

Similarly, the modified reciprocating saw, although inherently slow, can be used very effectively for remote disassembly operations where access is limited for other types of sectioning equipment. The studies and experience with both of these techniques suggest that their use for selected remote sectioning applications could minimize operational and access problems and be very cost-effective in comparison with inherently faster sectioning methods.

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2. SCHARNHORST, N. L., G. H. BRYAN, and W. J. BJORKLUND, "Remote Removal of Contaminated Equipment from a Radiochemical Engineering Cell," Proceedings, 32nd Conference on Remote Systems Technology, Vol. 2, pp. 87-93, 1984.

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A High Efficiency Suction System for Use in Controlled Areas

Since more than six jears our company effectualle employs special equipments of the following specification in the course of back-fitting, revision and decommissioning operations in nuclear-units.
The main components of the system are

- the suction and conveyor hose (inner diameter up to 100 mm)
- the blower with electric drive
- the rubble and dust separator including a precleaner (cyclon) and high efficiency particulate filters.

Precleaner, filters, and blower are mounted in a mobile rack. the conveyed materials are directly discharged into drums or container for subsequent further conditioning.

The system is capable of suctioning and conveying pneumatically

- dust of various consistency
- rubble and rubbish of oncrete and brickwork
- ironexchangerand filter materials e. g. charcoal, resins, diatomite, sand
- insulating materials e. g. asbestos, stone and glass wool
- sludges and residual liquids

over a distances up to 180 metres and level differences up to 25 metres.

The efficiency of the dust separation and filtersystem is such that the exhaust air can be discharged directly to the working area atmosphere. The respective limits of radioactivity concentration and of hazardous materials in the exhaust air applicable to controlled areas are permanently met. For instance, the particulate concentration during the conveying of asbestos materials was lower than 0,25 fibres/m³.

The suction system can be applied directly or in combination with size reduction operations (shredding etc.) or mechanical decontamination tools such as brushing, according to the needs of the various jobs.

The suction system has mainly been used so far for

- removal of insulating material from the interspace between the pressure vessel and the bioshield,and from turbine casings, with and without size reduction prior to suction
- removal of dust layers from surfaces in the containment vessel and the reactor hall.

The experience gained so far has shown definitely the adavantages

- avoidance of spread of contamination
- minimum staff requirement
- minimum working time

the advantages of the application from economic as well as health-physical.

Aerosol-activity concentration of conveying-air returned into the controll-area atmosphere meets the limits concerned.

The particle-concentration during shredding of asbestos-mats was measured to be 0,25 partcl./m³.

- special brushing devices (protected by patent) in combination with the exhauster-system.

The various devices suit the geometry of parts and structures to be decontaminated. So far they have been employed

effectively to decontaminate parts of the following specification by means of a dry, mechanical procedure for instance in the course of decomissioning operations at the Gundremmingen Nuclear Power Plant:

- shell plates of tanks, vessels and heat-exchangers
- inner and outer surfaces of pipes and tubes with a minimum diameter of 300 mm
- plain or large-area structures, for instance turbine and condensr casings
- concret floors and walls.

Furthermore contamination protective-coating can be removed, conveyd and racked without contamination of working - area and immediate environment. The advantages of employing dry, mechanical decontamination methods are:

- no production of additional secondary-waste
- conditioning for disposal of the brushed-off contaminated layer-material can comparativly easily be achieved
- preliminary treatment by brushing followed by chemical or electrolytic decontamination procedures reduces processing time and amount of secondary waste-material.

Power Clean Industry Reiniguns- und HandelsgmbH is authorized tc employ its staff in controll-areas as ruled by § 20 a Strahlenschutzverordnung.

Specifications of the unit

The measurement:

Aggregat

2,70 m lenght, 1,60 m width, 2,20 m hight, 2 100 k weight
30 kw power, 2 950 rotations/minute
700 mbar = 7 000 mm head of water

Prefractionator

1,50 m square, 1 800 kg weight, 1 vibrator

F i l t e r

1 150 m square, 1 850 m hight of the supports,
1 vibrator, 25 m² filters contents,
3 53o m total hight, 1 300 kg weight

Special Safety Container

is employed in suction of high-contaminated sludge and water for safety overflow device.
1,20 m square, 4 supports, 2,80 m hight, 1 vibrator

S h r e d d e r

electrical, 1,50 m square, 2,80 m hight with 4 supports,
2 500 kg weight

Pipes/Tubes

The connections between the components are hoses or pipes with quick lockings. we arrange the several components in rooms/ places as there are difficulties with the working-place.
Our tubes are flexible plastic-tubes 100 mm diameter.
High contaminated materials will be sucked by double tubes.
The lenght of tubes/pipes can reach 200 - 220 m. In special cases we need steel or zinc-plate tubes.

Transport of Equipment

All operations of transport will be carried out like a "nuclear transport".
The equipment ist shipped in a special container 20 ft. and the transport has to make special-carriers with all the necessary licensees.
Between the devices we stock the equipment in the container in a special hall with licence for nuclear employment. So we save the very high costs for cleaning and decontamination.

Examples of our employment in controll-areas

- Suck-off the reactor sphere, 2 - 3mm contaminated dust lenght of pipes from sphere to the aggregat 220 m.
- Removal of insulating materials out of the slot between pressure vessel and biological shield, high contaminated insulation material. It is a very difficult job, because the distance-holders are irregular situated and the pipes reach till to the bottom. Workers has to wear masks.
- Suck-off asbestos insulation of turbines. Lenght of the pipes 180 m.
- Suck-off wheel-swart/rust below the turbine. Workers has to wear complete safety-masks.
- High contminated filter-sludge out of sett-off-basins.
- Bears, supports, walls, plains concrete, after the shut down.

Summary

We are able to carry out all operations with the high-capacity -exhausters and our special equipment by suctioning.
In the same way we carry out transport and packing in all nuclear areas. We have a very interesting time-reduction.
The contamination of workers will be reduced to a minimal burden.
The cleaning-succes ist very high. We produce no secundary waste.
Big plains and walls, high contaminated dust, concrete-pieces till 95 mm diamter, all goods with a diameter till 95 mm can be sucked off.
We work without dust and that is a important succes for the health of people.

Victor Beyer

Paper for presentation at the 1987 International Decommissioning Symposium, 4-8 October 1987, D L Lawrence Convention Center, Pittsburgh, Pennsylvania, USA

REMOVING THE WAGR CO-AXIAL DUCTS - AN EXAMPLE OF 'HANDS-ON' DISMANTLING

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ABSTRACT

This paper describes the methods used for dismantling two of the WAGR co-axial gas ducts which connected the reactor pressure vessel to the heat exchangers. Each duct system is 4½ metres long, 860 mm diameter and weighs 1½ tonnes.

After assessing several cutting techniques the choice of a standard oxy-acetylene gas torch was made. (A market survey yielded a semi-remote profile cutter which was used on the second duct). The containment boundaries and provision of ventilation system are detailed.

Radiation dose figures included in this paper show that materials with surface radiation levels in the 1-10 mSv hr^{-1} range can be safely removed and disposed of at modest cost and with acceptable dose uptake. In particular the lessons are those of good planning, adequate training and a high level of team involvement. The successful completion of this work allows the planned manual removal of the reactor pressure vessel top dome to proceed with confidence.

INTRODUCTION

The dismantling of nuclear reactors involves the removal and disposal of materials of a wide range of specific activities. For the higher activity materials remote cutting and handling techniques are essential. As the bulk of the plant is not radioactive then for those parts manual demolition methods are appropriate. Deciding where to draw the line between remote and 'hands-on' methods is of vital importance in planning the decommissioning task. Keeping absorbed radiation doses as low as reasonably achievable is the guiding principle for these decisions. What is reasonable? The answer can only be given properly if all the facts are available. However, ignoring the opportunity to

use quick, simple, manual methods in the proper circumstances is likely to lead to unacceptably high costs. One of the objectives of the decommissioning of the Windscale AGR is to provide information of this type. This paper describes the removal and disposal of contaminated parts of the plant within an acceptable radiation dose uptake.

The Windscale AGR was constructed with a built-in neutron shield immediately above the graphite core. As a consequence the top part of the steel pressure vessel and its refuelling branches have not been subject to neutron irradiation in any quantity. The decommissioning plan includes the manual dismantling of the pressure vessel top dome region, confirmed by comprehensive radiation surveys. Part of this manual cutting work is concerned with the removal of the four large co-axial gas ducts which transferred the CO_2 coolant between the reactor pressure vessel and the four heat exchangers (boilers). The hot gas from the hot box of the reactor passed along the inner duct with the cooled gas returning via the annulus formed by the inner and outer ducts (Fig 1). Each duct system is $4\frac{1}{2}$ metres long, 860 mm diameter and weighs $1\frac{1}{2}$ tonnes. The inside of the inner duct is lined with stainless steel insulation material in a sandwich construction.

Cutting and removal of the ducts connected to two of the heat exchangers ('B' and 'D') was necessary early on in the project programme to allow for construction of the intermediate level waste handling route. This route uses part of two of the heat exchanger bioshields which meant that the heat exchangers had to be moved. This was achieved by lifting them up by some 12.4 metres to free the necessary space.

DUCT CUTTING STRATEGY

Experiments carried out in WAGR during its 18 year operating life included the study of cladding failures. Some fuel pins were retained in the reactor to see how their failures would develop. Consequently the gas circuit was contaminated with volatile fission products to a greater degree than occurs in a civil nuclear power station. The contamination is largely Cs-137 and Cs-134 with a smaller proportion of activation products, dominated by Co-60. Caesium tends to plate-out on the first of the cooler surfaces it encounters, so the inside of the inner duct has the highest contamination levels. The contamination was not evenly spread, levels in 'D' duct being about twice those in 'B'. It was clear that 'B' duct should be cut first to evaluate the method with lower radiation levels.

To determine whether manual cutting would be possible surveys were made of radiation on the outside of the outer duct and also of the levels in the annulus between inner and outer ducts by drilling a hole (Table 1). The surveys indicated that manual cutting could be considered.

Table 1

External Radiation Readings

Co-axial Duct	'B'	'D'
Radiation at Contact OUTER DUCT	400 μSv/hr γ 400 μSv/hr βγ	800 μSv/hr γ 800 μSv/hr βγ
Radiation 2 ft from duct	150 μSv/hr γ 150 μSv/hr βγ	200 μSv/hr γ 200 μSv/hr βγ
Radiation at Contact INNER DUCT	4.5 mSv/hr βγ 1.2 mSv/hr γ	6 mSv/hr βγ 2.5 mSv/hr γ

The choice of cutting tool was limited very much by the working space available and after consideration of orbital cutting tools the decision was taken to use oxy-acetylene.

Hazard Assessment

All new procedures and equipment for use on decommissioning are authorised for use by the WNL Decommissioning Safety Assessment Committee. For these procedures the main hazards were considered to be radiation, air-borne contamination, fire and material handling hazards.

Radiation uptake was controlled by personal alarm dosimeters and recorded by film badge. Lead sheeting was used locally to minimise radiation levels.

Air-borne contamination was controlled by containment and ventilation by two independent systems.

Fire hazards were dealt with by using flame retardant materials for the local containment and by the provision of suitable sand trays. Fire extinguishers were located at the work place.

Finally it should be said here that importance was attached to the drawing up of rigid operating procedures, the right level of supervision and health physics monitoring, and above all to the full discussion of all aspects of the job with the work force. This last point contributed to a smooth operation with each member of the team knowing his job and able to rely on his colleagues to perform theirs reliably.

PREPARATION

Ventilation

Oxy-acetylene cutting produces a large amount of dross in aerosol form. It was beneficial to remove as much fume as possible close to the burning region utilising existing installations. The reactor vessel and co-axial ducts were ventilated internally using the Ventilation Chemical Plant (VCP) (Fig 2). Holes drilled in the outer duct used for radiation survey were also used to measure the airflow ingress (See Fig 5). An extractor hood was placed above the duct fed to the Gas Discharge Treatment Plant (GDTP). Smoke tests were carried out prior to operations to show the effectiveness of the ventilation and an audible warning system installed in case of plant failure.

Tenting

A fire-proof PVC tent supported on a steel frame formed the primary containment with enough working space for two men. A tented changing area was also constructed, ventilated by the GDTP. The integrity of the tent was checked daily (Fig 3).

Protective Clothing

The operator and supervisor were dressed in remote air-supply pressurised suits for protection against airborne contamination with a qualified dresser in attendance. A full set of welding leathers were worn over the pressurised suit by the cutting operator and were changed when contamination levels on them became unacceptable.

DUCT CUTTING

E Duct

A Clearance Certificate from the Building Foreman was required for drilling and cutting operations. The morning period was used for cutting operations and the afternoon for cleaning. A 600 mm section of the outer duct was removed from the heat exchanger end. The inner duct was severed 100 mm shorter to allow easy removal. The edges were trimmed back to allow blanking plates to be welded onto the open ends (See Fig 6). These plates shielded the workers from radiation from within the heat exchanger and inner duct and contained the contamination on the inner surfaces. Lugs were welded onto the duct to allow the hoist to move the large pieces to a sand tray for cutting up (see Fig 7). The upper half of the outer duct was severed into 300 mm sections and hoisted out into the sand tray. This procedure utilised the lower half of the outer duct as a dross collection tray to reduce fire risk. An end cap was tack-welded onto the inner duct after each severing operation. Once the inner duct

had been removed the lower half of the outer duct was cut into 300 mm sections and transferred to the sand box. The open end of the reactor was sealed with a blanking plate.

D Duct

Internal radiation surveys confirmed the levels on 'D' duct were approximately twice the levels on 'B' duct. Operator dose would be a more significant factor in determining working times. To reduce dose uptake a semi-remote profile cutter was used. The tool ran on tracks formed to the duct contours held on by magnets. An oxy-acetylene torch was located on a boom away from the drive mechanism and track (see Figs 8 and 9).

The procedures used to sever 'B' duct were deployed on 'D'. For 'B' duct two burning operatives were used working alternate days. In the case of 'D' duct, five were used in turn with the working time for each man regulated by the prescribed dose limits.

It will be seen that the operation to sever both ducts resulted in a tonne absorbed dose of 34 man mSv.

The highest air sample result was from 'D' duct cutting. The filter paper measured 400 μSv/hr βγ. This is equivalent to:

3×10^6 Cpm 70 DAC hrs β

Waste Disposal and Clean-up

The cut pieces of duct were regularly removed from the working area by process staff clad in self-powered pressurised suits. Clearly if the pieces had been allowed to accumulate they would have constituted an unnecessary radiation source. All the pieces were surveyed by Health Physics staff. Those falling within shallow land burial limits (<7.5 $mSv.h^{-1}$) were wrapped and sent for disposal to Drigg in sealed skips. Those pieces exceeding this limit were dispatched in a purpose made shielded container to the Sellafield Site Decontamination Centre.

The working area was vacuum cleaned daily and the tent checked and repaired where necessary. Immediately prior to dismantling it the tent inner walls were sprayed with a mixture of glycerine and water and then packed in PVC bags for disposal.

The figures recorded for man dose in mSv are shown in Table 2 overleaf.

Table 2

Dose assessment (results in mSv)

'B' Duct		'D' Duct	
Supervisor	1.51	Supervisor	3.83
Burner 1	2.00	Burner 1	4.12
Burner 2	1.86	Burner 2	2.73
Fitter/Mate 1	1.32	Burner 3	1.96
Fitter/Mate 2	0.64	Burner 4	3.02
Fitter/Mate 3	0.09	Burner 5	2.99
Process 1	1.42	Process 1	0.25
Process 2	0.05	Process 2	0.22
Process 3	0.15	Process 3	0.48
Process 4	0.17	Process 4	0.19
Process 5	0.03	Process 5	0.09
Process 6	0.17	Process 6	0.40
Process 7	0.27	Others	2.00
Others	2.16		
	11.8 man mSv		22.3 man mSv

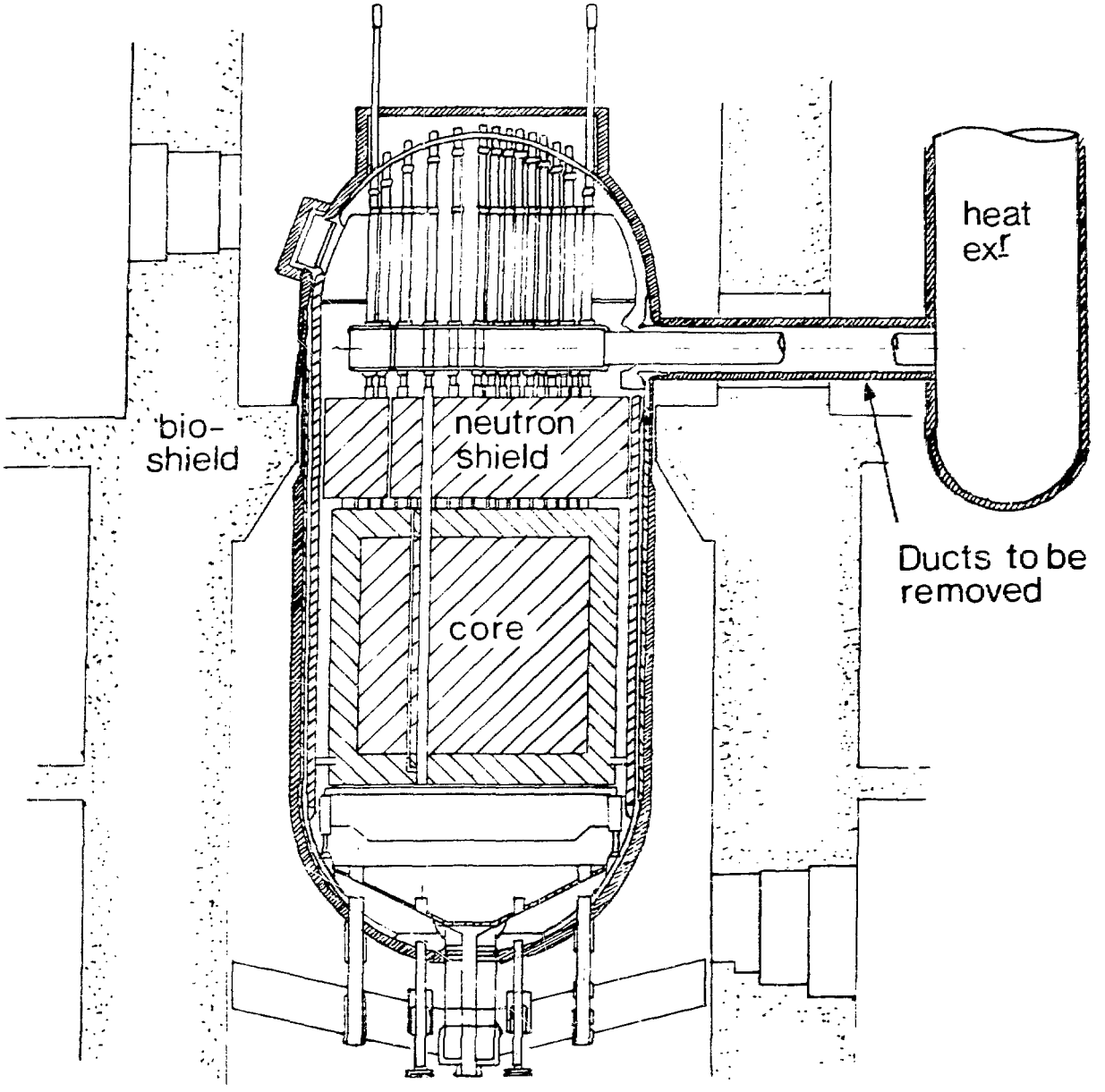
CONCLUSION

The successful completion of the duct removal work was a significant step in the decommissioning of WAGR as it allowed a number of other project activities to proceed.

Some 3 tonness of contaminated material was cut and disposed of with a total absorbed dose of 34 man mSv. This is regarded as a reasonable dose given that the material had surface radiation levels in the 1-10 $mSv.h^{-1}$ range.

It was necessary to carry out the removal work on the first duct (B) in two stages in order to fit in with other project work. Clearly this meant that this first operation was going to be the more expensive of the two. However, the 30% cost reduction achieved in the second operation on the 'D' duct is due in part to the lessons learned on the first job (Fig 4). In particular the lessons are those of good preparation and planning, the provision of comfortable working conditions within the limits of what is possible, and especially the encouragement of team spirit and involvement.

The significance of this work goes well beyond the removal of the contaminated ducts. More importantly it allows the planned removal of the top dome region of the reactor pressure vessel to proceed with confidence.



Reactor Vessel Internals

Fig. 1

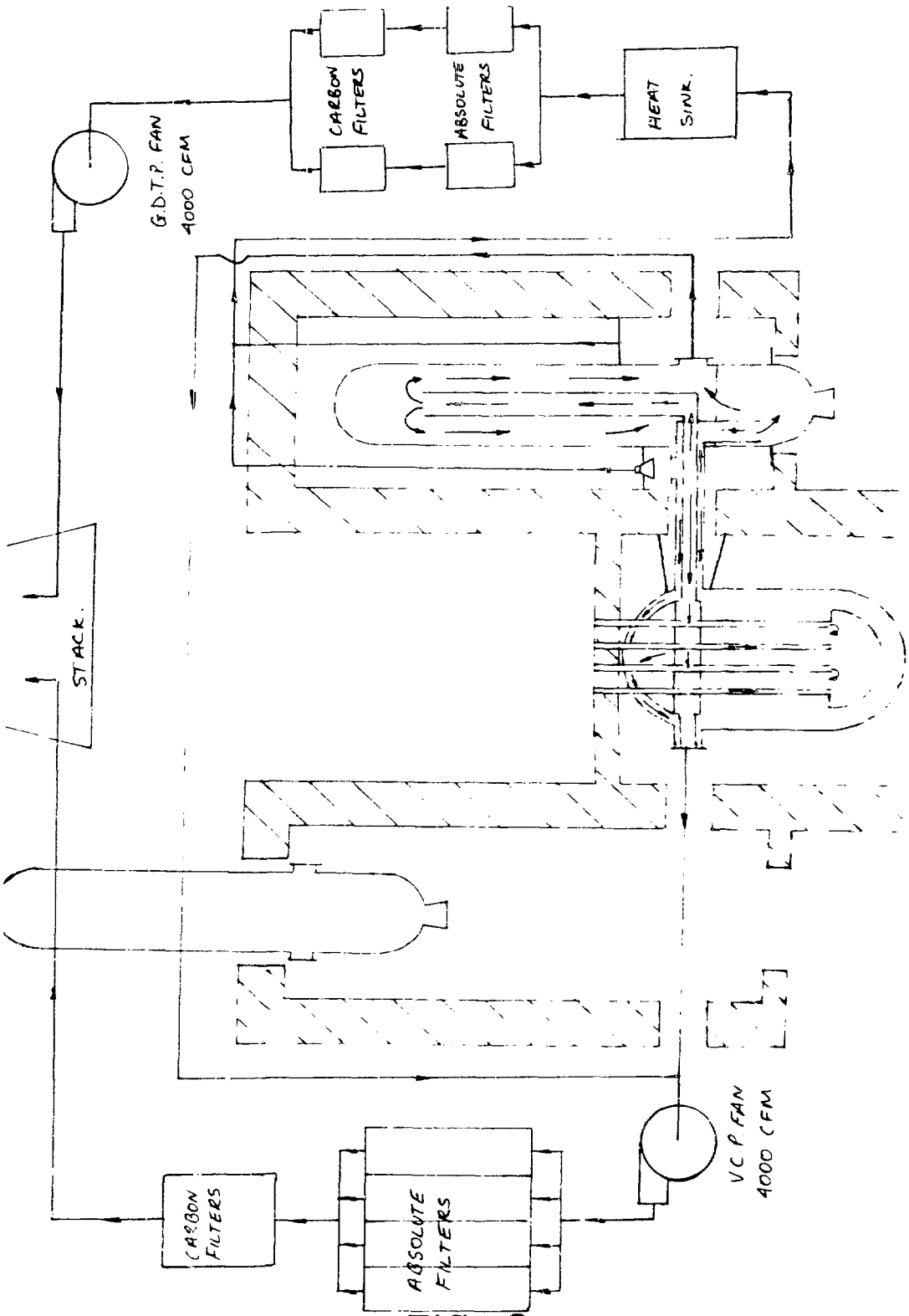

CARBON FILTERS
ABSOLUTE FILTERS
HEAT SINK.
G.D.T.P. FAN
4000 CFM
STACK.
CARBON FILTERS
ABSOLUTE FILTERS
V.C.P. FAN
4000 CFM

Fig. 2

Fig. 3

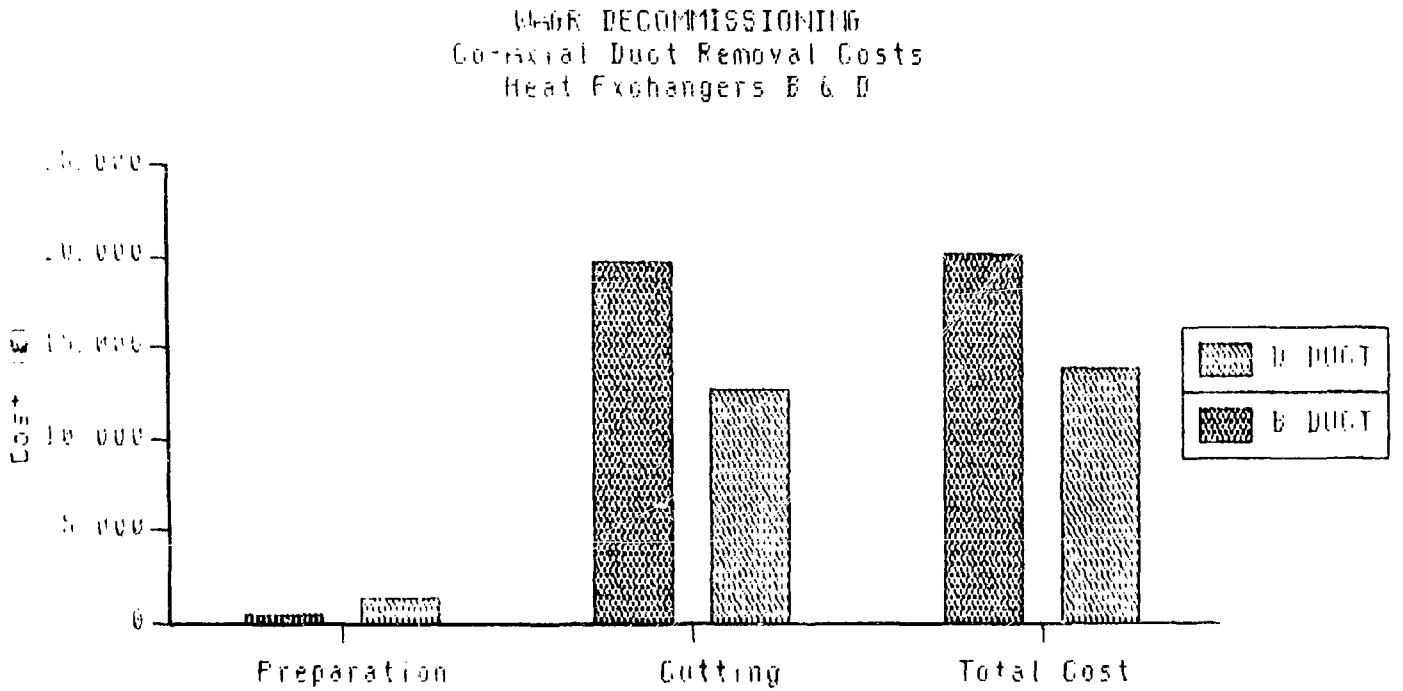
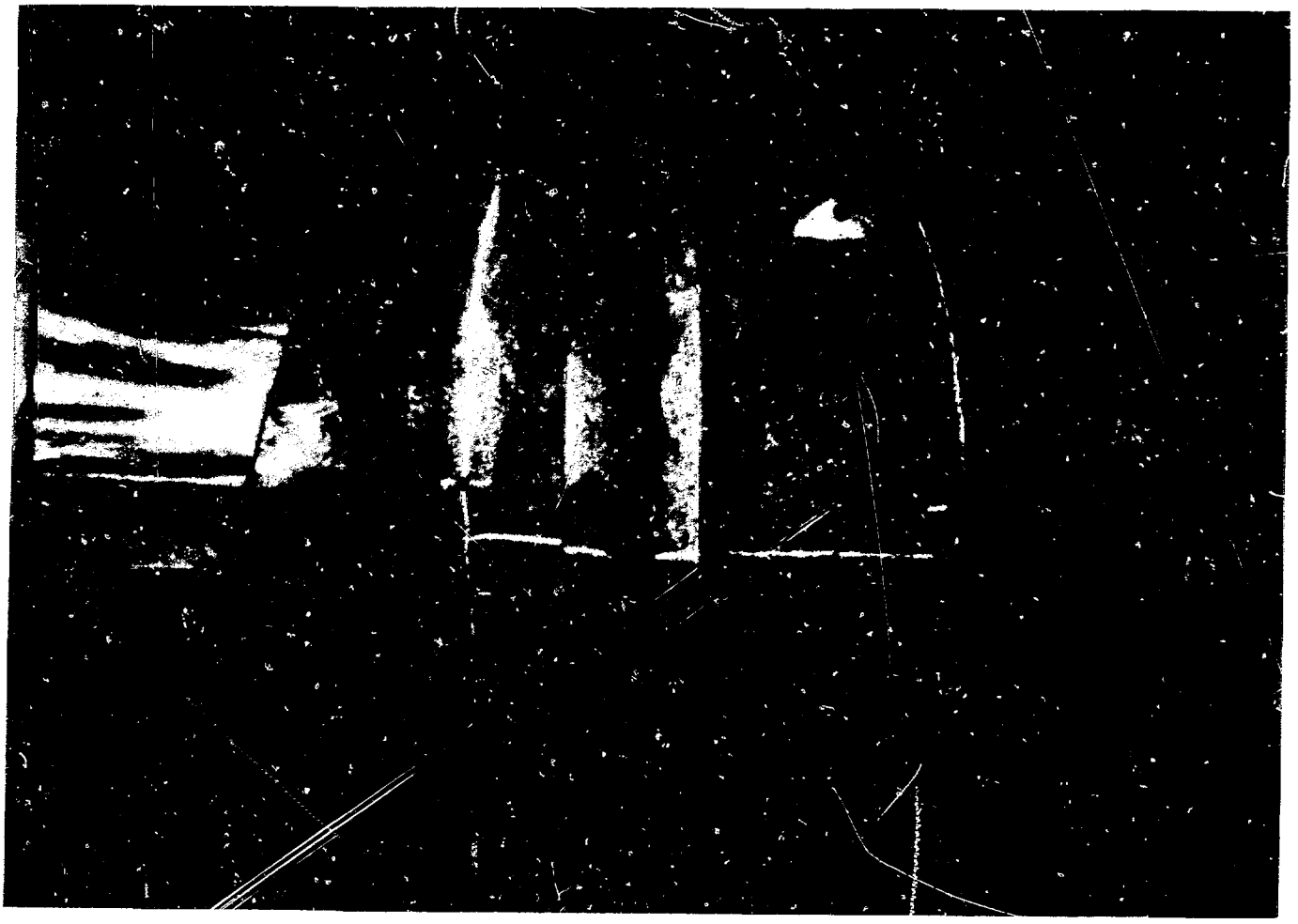


Fig 4

PF: 1/13=Help 2/14=Save 3/15=End 4/16=Print 6/18=Show Errors 12/24=Home

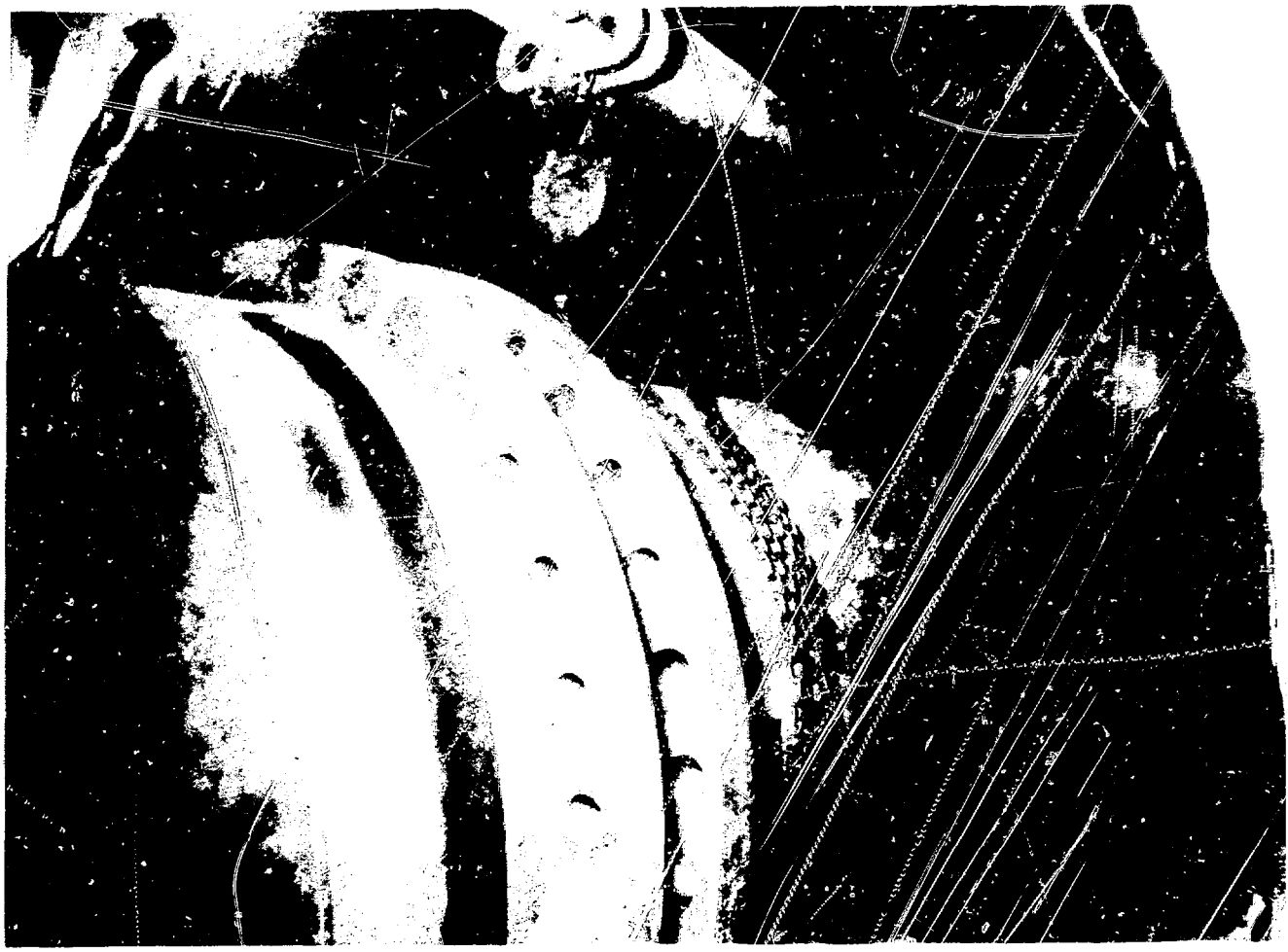


12mm holes prior to cutting Heat Exchanger 'B' duct

Fig 5

End cap Heat Exchanger 'B'

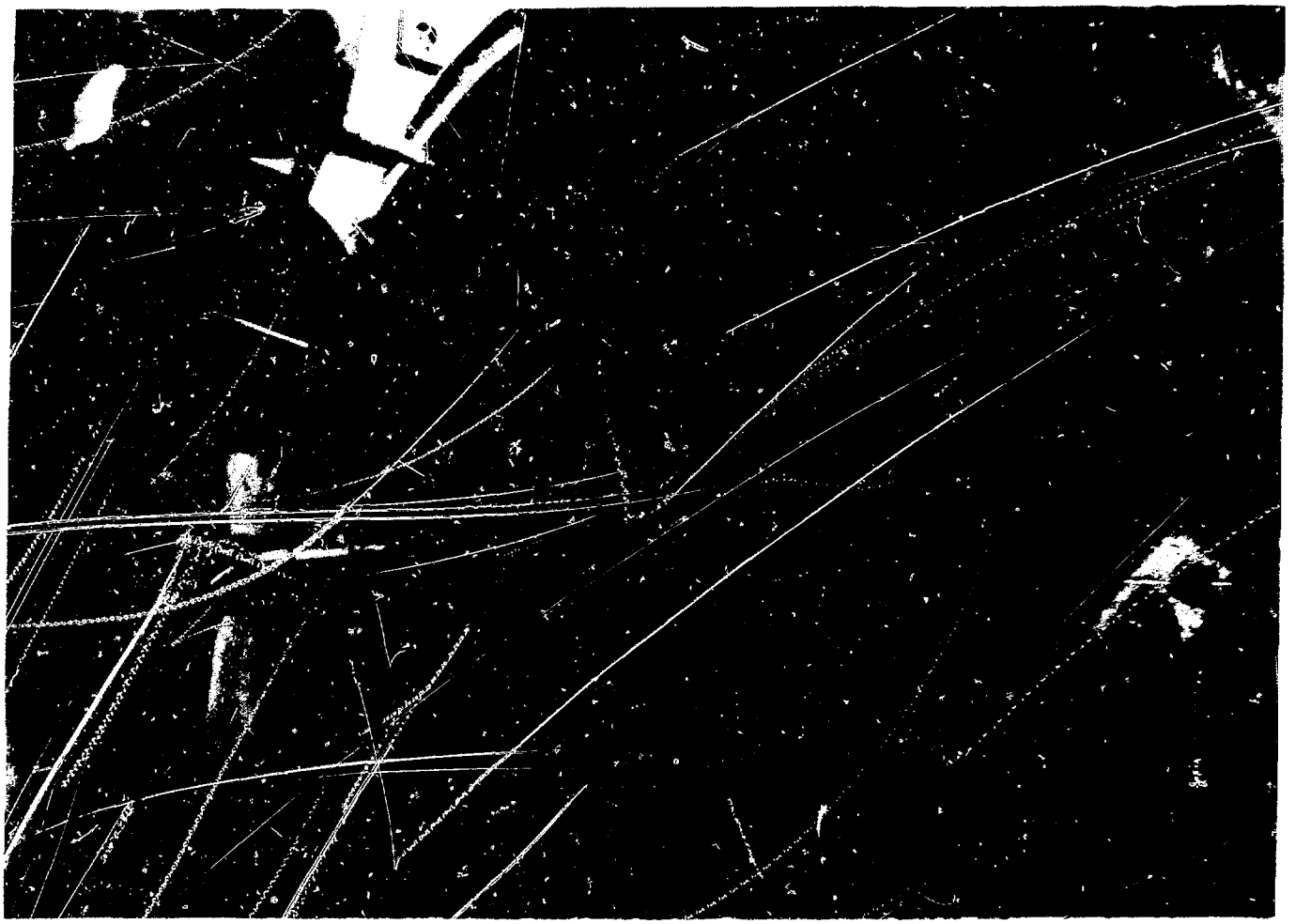
Fig 6



Preparing to cut and remove the co-axial duct, Heat Exchanger 'B'.

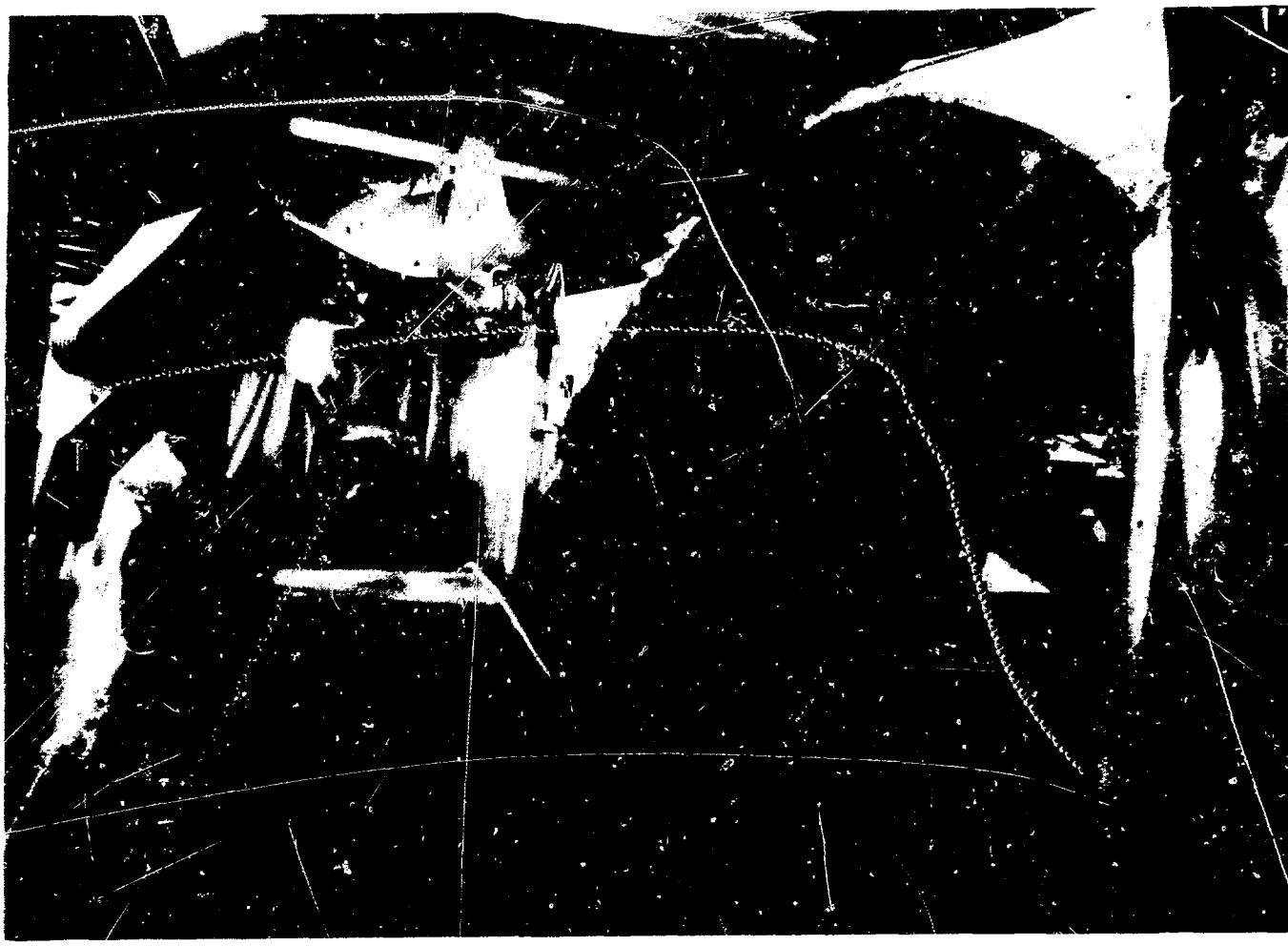
32104 3C

Fig 7



Semi-remote profile cutter - Heat Exchanger 'D' duct

Fig 8



Operator controlling profile cutter Heat Exchanger 'D' duct

Fig 9

AN OVERVIEW OF PLUTONIUM-238 DECONTAMINATION & DECOMMISSIONING (D&D) PROJECTS AT MOUND

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ABSTRACT

Monsanto Research Corporation (MRC), which operates the Mound Site (in suburban Dayton, Ohio) for the Department of Energy (DOE), has been decommissioning radioactively contaminated facilities since 1949. Mound is currently decontaminating (for restricted reuse) and/or decommissioning (for conditional release) four major plutonium-238 contaminated facilities (approximately 75,000 ft^2) that contained 1700 linear feet of gloveboxes and associated equipment and services. In addition, several thousand linear feet of external underground piping, associated tanks and contaminated soil are being removed. Two of the facilities contain ongoing operations and will be reused for both radioactive and nonradioactive programs. Two others will be completely demolished and the land area will become available for future DOE building sites. Currently, over 30,000 Curies of plutonium-238 have been removed in waste and scrap residues.

An overview of the successful techniques and equipment used in the decontamination and decommissioning of individual pieces of equipment, gloveboxes, services, laboratories, sections of buildings, entire buildings, and external underground piping, tanks, and soil in a highly populated residential area will be described and pictorially presented.

*Mound is operated by Monsanto Research Corporation for the U.S. Department of Energy under Contract No. DE-AC04-76-DP00053.

INTRODUCTION

Mound is a Department of Energy installation located in Miamisburg, Ohio. The site (306 acres) is located in a residential/agricultural area of suburban Dayton, Ohio. The facility is operated by Monsanto Research Corporation for the DOE in support of weapons and nonweapons programs.

Monsanto has been involved with radioactive operations since 1944 and the resulting decommissioning operations since 1949. We are currently decommissioning four facilities that were used primarily for the processing and encapsulation of plutonium-238 heat sources for various programs, such as the heat sources used in space applications (SNAP, PIONEER, TRANSIT, VIKING, and VOYAGER).

The current multimillion/multiyear project involves the extensive decontamination and decommissioning (D&D) of inactive areas of four facilities: Plutonium Processing (PP) Building, Research (R) Building, Special Metallurgical (SM) Building, and Waste Transfer System (WTS). The project was initiated in 1978 and is expected to be completed in 1996.

The PP Building is a two-floor (42,000 ft^2) reinforced concrete and concrete block building built in 1967 to process plutonium-238. Approximately 90% of the building will be decommissioned for potential future reuse with the remaining 10% staying operational.

The R Building is a one-floor (54,000 ft^2) concrete block and brick building built in 1948 to research, develop, and process various isotopes. Approximately 10% of the building is being decommissioned for current and potential reuse with the remaining 90% staying operational.

The SM Building is a one-floor (17,300 ft^2) steel-frame-with-curtain-walls building built in 1960. A one-floor (5,700 ft^2) concrete block addition was built in 1965. The building was used for process development, processing, and recovery of plutonium-238. Complete removal of the building, auxiliary buildings, and surrounding contaminated soil is planned.

The WTS was an underground liquid waste transfer system built in 1967 and consisted of two underground lines (2,565 ft each) buried from 6 to 17 ft below ground and a one-level 600 ft^2 concrete block lift station (Building 41) with two underground tanks. Complete removal of the building, tanks, underground piping, and surrounding contaminated soil is in progress.

The total project (PP, R, SM and WTS) involves the removal of 1,700 linear ft of gloveboxes; 930 linear ft of conveyor housing; 2,565 linear ft of dual underground liquid waste lines; and associated contaminated piping, services, equipment, structures, and soil. Estimated waste volumes generated by the decommissioning project are: 1,256,000 ft^3 <100 nCi/g of Pu-238 and 111,000 ft^3 ≥100 nCi/g of Pu-238.

Extensive D&D includes cleaning and removal of internal glovebox equipment and services, removal of gloveboxes, removal of associated laboratory equipment and services, structural decontamination, and disposal of wastes from the PP and R Buildings. Contamination in the inactive areas is reduced to an "as low as reasonable achievable" (ALARA) level, and remaining contamination is permanently sealed so that the areas can be reused with minimal restrictions (restricted release).

The final exposed average contamination levels in these facilities after decontamination and sealing will be:

Wipe	- ≤20 dis/min/100 cm^2
Direct	- ≤1500 dis/min/100 cm^2
External Radiation	- ≤1 mr/hr at surface

These levels are being normally met (except for cracks and crevices) before sealing. Unrestricted release was not considered since this would require demolition of the PP and R Buildings which are used for ongoing programs and could be reused for future DOE programs. Demolition would be required because of known and potential contamination (in structural members, underneath the facilities, and in cracks and crevices) and the inability to detect and remove this contamination without destroying the integrity of the structure. However, complete D&D was considered the only feasible approach for the Waste Transfer System, which is unusable because of previous leaks, and SM, which is not reusable because of below-floor contamination and the curtain-wall construction.

As complete removal is planned for SM and WTS and respective surrounding contaminated soil, the soil contamination level should be able to be reduced to near unrestricted levels* (conditional release).

*A level of 100 pCi/g is the goal of remedial action of near surface (first 12 inches) soil. A lower limit of 25 pCi/g is the goal for ALARA evaluations. DOE Order 6430 "General Design Criteria Manual".

To date, all (1,700 linear ft) gloveboxes (with associated external glovebox equipment, piping, and services) have been cleaned, stripped of piping and equipment, sectioned if required, packaged, and removed from the site. Also, laboratory areas have been completed and are being reused by other DOE programs. In accomplishing this, there have been no significant radiation exposures or environmental releases.

There are several unique characteristics of the current project.

- The site is in a residential/agricultural area; thus outside decommissioning activities are restricted. (Site boundary is as close as 300 ft).

- Normal operations continue in the PP and R Buildings, and in close proximity to SM and WTS; thus decommissioning activities are restricted.

- Most gloveboxes were two-level (operating level and equipment level) and are larger than a standard 4 ft x 4 ft x 7 ft shipping container.

- Although oversized shipping containers were used, some large gloveboxes required sectioning before packaging.

- All equipment, piping, and services must be removed from the gloveboxes before packaging (burial storage facility requirement).

- Three separate funding agencies within DOE (weapons and nonweapons) require coordinated funding and planning.

- Contamination involved is primarily plutonium-238, a high-specific-activity transuranic isotope (16.8 Ci/g) requiring special personnel protection and waste packaging.

- The WTS underground lines (and associated contaminated soil) were located up to 17 ft below the surface on hilly (up to 140-ft elevation change within a distance of 600 feet) terrain which required special excavation procedures.

- Weather conditions (wind, rain, snow) and temperatures (typically $-10^{\circ}F$ to $100^{\circ}F$) restrict outside decommissioning activities.

- All radioactive waste must be shipped for off-site burial.

As a result of these unique characteristics and our previous experience in decommissioning facilities, several techniques were developed in each of the following areas: planning, exposure control, contamination control, equipment removal, structural decontamination, and waste packaging.

PLANNING

Direct management involvement and commitment in the decommissioning project starts at the director level and continues down to the D&D Management Team. The interdepartment Management Team consists of a representative from each of the major D&D functions: Program Management, Operations, Project Engineering, Maintenance, Technical Support, and Environmental, Safety and Health Physics. This matrix Management Team formally meets with the involved Directors on a monthly basis to discuss status, accomplishments, problems, and plans. The Team also meets quarterly with the DOE Area Manager and his staff. This is in addition to the normal weekly and monthly written reports sent to management and the DOE.

In addition to the formal monthly, quarterly, semiannual, and annual reviews of D&D plans by MRC directors and DOE management, plans are formally reviewed weekly in each of the major D&D areas by direct supervisors; as well as special planning sessions held for unusual decommissioning activities along with prejob conferences with the personnel who will be performing the work.

Decommissioning activities are controlled by using special work permits such as the "Radiation Control Area Maintenance Permit." This permit requires interdepartmental review and approval to ensure that jobs are thoroughly preplanned, adequate training and safety analysis have been performed, and proper precautions are being taken.

Special procedures are required for any unusual decommissioning operation not covered by existing work procedures. These procedures require interdepartmental review and approval. In addition, special training is required for any new complex techniques employed. For critical operations, a "mock-up", or nonradioactive test, is made prior to actual operations.

The D&D planning efforts include quality assurance and other control methods to ensure adequacy, consistency, change approval, and reporting. Operations are routinely audited by Environmental, Health and Safety, Quality, Financial, and Management representatives within the company, by an independent DOE contractor, by DOE Area and Field Offices, and by DOE Headquarters.

The use of computerized project management and scheduling programs ("Quicknet" and Project/2") facilitate activity and resource scheduling for complex projects.

Another planning aid has been the use of exception and trend reporting to increase management awareness and response to potential problem areas. These reports cover such areas as radiation exposures, effluents, safety performance, milestone status, and cost versus budget.

One area of planning has been personnel resources. Whenever possible, use is made of personnel with prior operations experience, personnel experienced with decommissioning operations (including consultants and offsite contracts), and personnel experienced with plutonium-238 and other radioisotopes. For new personnel, intensive training and certification are required. Frequent retraining orientations and seminars are presented to operations personnel to reaffirm established techniques and demonstrate new techniques.

EXPOSURE CONTROL

Again, as in planning, direct management involvement and commitment in exposure control start at the director level in the Executive Safety Committee's commitment to keeping exposures and effluents "as low as reasonably achievable" (ALARA). ALARA goals are set yearly after an interdepartmental review and evaluations. Monthly trend and exception reports indicate potential problem areas for management review and follow-up.

A key part of planning for any D&D activity is exposure control. This planning for exposure control includes training and selection of experienced personnel for critical activities, communication of known or suspected hazards, analysis of hazards, procedure review, contamination control precautions, work permits, adequate monitoring, and reporting.

Specific exposure control equipment and techniques include remote operations (including long-handled or remotely operated tools and equipment); portable and personnel shielding (including lead-loaded gloves and aprons); respiratory protection (full-face mask and supplied-air suits); protective equipment (clothing, portable enclosures, local exhausters, contamination fixatives); and special techniques for contamination control, equipment removal, structural decontamination, and waste packaging (see appropriate section for additional details).

Exposure monitoring is accomplished with both in situ and personnel Thermoluminescent Dosimetry (TLD) (including extremities), industrial hygiene monitors, selective plutonium-238 air monitors, fixed position and personnel air samples, alpha/gamma/neutron instrumentation and measurements, and personnel bioassay samples (nosewipes, urine, blood, sputum, fecal, and whole-body counting).

CONTAMINATION CONTROL

After as much of the radioactive material as possible is removed from gloveboxes, equipment, and piping by standard cleaning and flushing techniques; temporary enclosures, fixation, and ventilation become the primary means of contamination control during subsequent removals.

Temporary enclosures are constructed for containment around all separations of gloveboxes, equipment, and piping with a high potential for contamination release. These enclosures range in size from a sealed plastic bag to a series of large rooms or a building with separate HEPA (high efficiency particulate air) filtered exhaust systems (when personnel access is required). The large enclosures are constructed with either fire retardant wood or sheet metal framing covered by heavy clear plastic. In some cases, permanent airlocks are added to building corridors to increase contamination containment and increase air pressure differentials.

Contamination fixation is used during glovebox, equipment, and piping removal after conventional decontamination methods cease to further reduce contamination levels. The fixation agent used depends on the application and includes light water misting, strippable paint, and urethane foam.

A light water misting is used to contain dusting or to clean in areas which could not be cleaned prior to breaching.

Strippable paint is used primarily on contaminated building structures as a temporary fixation until final decontamination can be accomplished.

Fire-retardant urethane foam is used as a fixative inside gloveboxes after equipment, piping, and services are removed, and the interior surfaces are cleaned. This fixative minimizes potential contamination spread during later glovebox separation, packaging, and shipment. Strippable paint is not used because of potential long-term radiolysis of the paint inside gloveboxes. In addition, only a thin layer (1-2 in.) of foam is used to minimize future potential waste-reprocessing problems at the burial, or storage, facility.

Urethane foam is also inserted at separation points in large diameter piping to provide a contamination barrier during subsequent cutting.

A third use for urethane foam is as a shoring material in waste packages (primarily at the four corners, middle of the side, and top). Again, a small amount of urethane foam is used to minimize future potential waste-reprocessing problems at the burial, or storage, facility.

Existing building and/or portable ventilation systems are used to contain contamination. Portable HEPA-filtered ventilation systems are used primarily for temporary enclosures and range in size from 25 to 1500 ft^3/min.

A minimum three-zone concept is also used in contamination control. Each zone represents a certain level of contamination and resulting protection. The first zone is the immediate work area that needs the highest level of protection (air flow, personnel, and controls). The second zone is a buffer zone or airlock, and the third zone is the noncontaminated or low-potential zone. The first zone is normally the enclosure being breached, and the second and third zones are plastic enclosures with the room area being an additional zone.

Administrative control levels are set for airborne and removable contamination, and gamma/neutron radiation in work areas. If the control levels are exceeded, then work is stopped until the levels have been reduced.

EQUIPMENT REMOVAL

Equipment, piping, and services must be removed from the inside of all gloveboxes (burial facility requirement). Because most of the gloveboxes contained two levels (standard operating level and an isolated lower equipment level in the glovebox well), equipment removal required personnel to enter the glovebox well (a highly contaminated atmosphere) from the rear to clean, disconnect, and remove equipment. A plasma cutting technique was developed to cut out sections of the operating glovebox floor to gain access to the equipment well to eliminate the need for initial personnel entrance. All plasma cutting was performed using the standard glovebox gloves.

Plasma was chosen for cutting because the resulting smoke generation is much less than that generated by a standard cutting torch (thereby reducing the particulate accumulation and eventual plugging of the glovebox exhaust HEPA filters). In addition, there is not as much heating of surrounding metal (because of the faster cutting), and the resulting cut edges are not as jagged.

Once access is gained to the equipment well, piping and services are disconnected using long-handled tools, and equipment is moved (with glovebox hoists) in order to clean the equipment and glovebox well. Personnel then enter the well (via a ventilated enclosure) to remove the equipment.

The equipment is then loaded and secured on a wooden platform (pallet) outside the glovebox (and inside a plastic ventilated enclosure which is collapsed around the platform for containment, thus eliminating the need for bagging). The equipment pallet is then loaded into a waste container. This precludes personnel from having to physically carry equipment to the waste container.

After the equipment is removed from the glovebox, the glovebox itself is loaded into a waste container. Occasionally, however, a glovebox is larger than the waste container and must be reduced in size by sectioning. Initially a foam wall was applied where the glovebox was to be sectioned, isolating contamination during the cutting operation. In addition, the safety plate glass glovebox windows were replaced with Plexiglas (methyl acrylate plastic) in areas where the cut was to be made. The outside was cut with a reciprocating saw (inside a temporary enclosure), and the foam wall was cut with a piano wire. The exposed end pieces of the glovebox were capped with sheet metal, and the individual glovebox sections were then packaged into the waste container.

This sectioning technique was very successful, but time consuming. In a modified technique, the foam wall was eliminated from sectioning. The interior surfaces of the glovebox are coated with a 1 to 2 in. layer of urethane foam while the normal glovebox exhaust is maintained. Then, before the stainless steel glovebox is cut, a layer of decontamination soap foam is applied to the cut area and cutting blade to contain metal filings. As the cut is made, the area previously cut is cleaned with a damp rag and sealed with a plastic sleeve and tape. The glovebox is then turned on its side to complete the cut. After cutting is completed, the sections are separated within the plastic sleeve, and the sleeve is crimped, cut, and sealed. Then sheet metal caps are placed over the cut and bagged ends. Replacement of the safety plate glass glovebox windows in the cutting area was also unnecessary, as development showed there is no breakage if the outside of the window is taped with plastic tape, the inner surface is foamed, and a special glass cutting blade is used.

To separate piping external to the glovebox, several techniques are used depending on the potential for the spread of contamination. In low-potential cases, the pipes are cut, with damp rags and/or plastic bags used for containment. In high-potential cases, copper pipes less than 1 in. and stainless steel pipes less than 1/2 in. are cut, using a crimping tool, and capped. For larger pipes, a small hole is drilled, urethane foam is injected into the hole, and then the pipe is cut after the foam has cured. The cut ends can then be capped. Piping and other materials are reduced in length to fit in standard 4 ft x 4 ft x 7 ft waste packages. The internal surfaces of larger ductwork are painted (to contain contamination), separated, and cut in half (diagonally, usually) to efficiently fit inside standard waste containers (boxes).

For transporting equipment, a variety of lifting devices, moving dollies, and hoists is used. In addition, the previously discussed equipment platform is used for transporting glovebox equipment to the waste container.

STRUCTURAL DECONTAMINATION

In the first step of structural decontamination, all unnecessary services (pipes, ducts, conduit) are removed (back to operating headers), and the resulting wall or ceiling opening is monitored, decontaminated, and sealed. In contaminated areas, false ceilings are removed, and concrete ceilings are sandblasted.

In decontaminating walls, the first choice is removal because of void spaces. For poured concrete walls (especially bearing walls), the paint is removed using paint remover, general contamination can be mechanically removed, and isolated spots can be scabbled.

In decontaminating floors, removal of the floor covering (and mastic) or paint removes most of the contamination. If the floor is still generally contaminated, a floor scabbler is used. If it is highly contaminated, removal of 1 to 2 in. of concrete is usually more efficient. Isolated contamination spots are then removed with a hand scabbler and vacuum sweeper. Total removal of the floor is used as a last resort.

Door frames and doors are removed in highly suspect areas. If there is not a double door into large areas, a temporary one is installed to allow waste containers to be moved in and out. Floor drains are also removed.

Since this type of work is very dusty, contaminated dust control is important. This dust is controlled using local exhausters, light water sprays, and immediate vacuum cleaning.

Core samples of soil under first floors are taken to verify the condition of soil and remaining underground drain systems.

If isolation of remaining contamination in cracks, crevices, and structural members is permitted, the first step is the documentation of the levels and location. The surface is permanently sealed, and a sign is posted on the exterior surface; again as a future reminder.

A remotely-controlled, electrically-powered, robotic excavating machine (Brokk 250 "Mini-Max") was used to remove a reinforced concrete room in the SM Building and to remove the below-ground soil and concrete from Building 41.

The use of an independent contractor to verify remaining contamination provides assurance of Mound's monitoring results and documentation for future reference and questions.

WASTE PACKAGING

A variety of waste containers is being used for the estimated 1,367,000 ft^3 of waste to be generated by the project. Low level (LSA) waste is packaged in either 55-gal steel drums or plywood boxes. Most plywood boxes are 4 ft (W) x 4 ft (H) x 7 ft (L). If the waste has a high density, a half-box 4 ft (W) x 2 ft (H) x 7 ft (L) is used to lower the package weight.

For transuranic (TRU) waste, a 20-yr retrievable package is used. Again, either boxes or 55-gal steel drums are used. The TRU 55-gal dr[illegible]s fabricated of thicker steel than the LSA drum and cont[illegible]s a 90-mil high-density polyethylene liner (with a press-fit lid sealed with adhesive).

The boxes used prior to 1986 were fiberglass-coated, heavy-construction, plywood boxes. In 1985, the use of steel boxes with a bolted lid, was initiated.

The standard fiberglass/plywood boxes were 4 ft (W) x 4 ft (H) x 7 ft (L). However, a limited number of larger boxes were previously used to preclude the size reduction of many large gloveboxes. Three larger sizes were used with the largest being slightly less than 6 ft (W) x 9 ft (H) x 12 ft (L).

A gamma scan and final fiberglassing facility (with a common turntable) was constructed to determine isotopic content and apply the final fiberglass seal on the box lid or sections of the box that were used for loading access.

CONCLUSION

Progress to date on the project has verified the importance of adequate planning (with flexibility for the unexpected), matrix organization for effective implementation and control, experienced and trained personnel with innovative abilities, frequent communications at all levels of management, management commitment to safety and ALARA exposures, contamination control techniques and equipment, variety of waste container sizes, adequate dollar and time contingency and independent verification of radiological conditions.

ABRASIVE WATER JET CUTTING TECHNIQUE FOR BIOLOGICAL SHIELD CONCRETE DISMANTLEMENT

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ABSTRACT

The Japan Atomic Energy Research Institute (JAERI) is developing the Abrasive-Water Jet Cutting System to be applied to dismantling the biological shield walls of the JPDR as a part of the reactor dismantling technology development project.

This is a total system for dismantling highly activated concrete. The concrete biological shield wall is cut into blocks by driving the abrasive-water jet nozzle, which is operated with a remote, automated control system. In this system, the concrete blocks are removed to a container, while the slurry and dust/mist which are generated during cutting are collected and treated, both automatically. It is a very practical method and will quite probably be used for actual dismantling of commercial power reactors in the future because it can minimize workers' exposure to radioactivity during dismantling, contributes to preventing diffusion of radiation, and reduces the volume of contaminated secondary waste.

This work was performed by Japan Atomic Energy Research Institute under contract from the Science and Technology Agency.

INTRODUCTION

The biological shield wall surrounding the reactor pressure vessel is composed of very thick reinforced concrete with high-density steel bars. It becomes highly radioactive due to irradiation by neutrons during operation. This radioactivity is generally higher at the reactor side concrete but it decreases inside the concrete and becomes negligible at approximately 1 m inside the wall from the surface.

For dismantling these biological shield walls, both remote controlled dismantling techniques for highly activated concrete and efficient demolition techniques for less activated concrete are required. Also, there must be the capability to cut concrete and steel bars at the same time and to minimize the volume of secondary radioactive waste.

To meet these requirements, JAERI has been developing four systems since 1981: the mechanical cutting technique, abrasive-water jet cutting technique, thermal cutting technique, and controlled blasting technique.

As the first stage of the development, test data were gathered on cutting efficiency of the equipment, characteristics of secondary waste, handling and workability of the equipment, etc. through cutting tests using reinforced concrete structures simulating the biological shield wall. Based on these data, an appropriate dismantling method for the JPDR was discussed. Then abrasive-water jet cutting and mechanical cutting techniques were selected for dismantling highly activated concrete. Actual equipments for using these techniques have been manufactured. Controlled blasting was selected for less activated concrete for the JPDR decommissioning. However, development of the thermal cutting technique was suspended owing to the problem of large quantities of dross and gas generation, etc.

In 1986, the development work necessary for JPDR decommissioning was almost completed. In this paper, the characteristics of the abrasive-water jet cutting system and the dismantling procedure for the biological shield wall of the JPDR is described.

DEVELOPMENT OF ABRASIVE-WATER JET CUTTING SYSTEM

General Requirements and Preliminary Tests Results

The abrasive-water jet cutting system can cut steel bars together with concrete by mixing abrasive with a high-pressure water jet.

An outline of the abrasive-water jet generating unit is shown in Fig.

1. It consists of a nozzle holder which jets a mixture of water and abrasive, a high-pressure water generator, and an abrasive supplying unit. This technique is used in industry to cut composite materials for which ordinary cutting techniques are not applicable, fragile materials like glass, concrete, or rock.

When this technique is used for dismantling activated concrete structures, the following are required to prevent unnecessary radiation exposure of workers and to minimize the diffusion of radioactivity: (1) remote operation for dismantlement, (2) reduction of cutting time, (3) reduction of secondary products' volume, (4) prevention of dust/mist diffusion, (5) efficient collection of slurry, etc.

In order to solve these problems, cutting tests were performed using reinforced concrete specimens simulating the biological shield wall. The equipment used in these cutting tests has the capacity to supply 50 l/min of water with a pressure of 196MPa and 3 to 7 kg/min abrasive. In these tests, cutting properties were studied with the parameters of abrasive material, particle diameter, and nozzle traverse rate. Generation of secondary products and influence on the surrounding environment were also investigated. Thus necessary data for the development of an abrasive-water jet cutting system for dismantling the biological shield wall was obtained.

The results of the cutting tests can be summarized as follows:

1) Regarding the relationships between cutting properties and the material/particle diameter of abrasive, there was no significant difference in cutting properties by abrasive material (garnet, steel grit or Morundum--a brand name for an alumina product), while it was found that #40 particle diameter is appropriate as shown in Fig. 2. But comparing the cutting properties in terms of the volume of abrasive, the specific gravity of the steel grit is the densest of the three types of abrasive, being almost twice that of the other two. This means that the volume of abrasive when using steel grit is half that when using the others.

2) To reduce the volume of secondary products, it is desirable to reuse the abrasive. Fig. 3 shows a comparison in particle diameter between the abrasives before and after use. According to these results, there was the least change in particle diameter in the case of #40 steel grit.

3) Based on the above results, #40 steel grit was selected as the abrasive to be used in actual dismantling.

4) To determine the number of recycle times of abrasive, the cutting test was performed using steel grit to investigate the cutting properties

when the abrasive is reused. It was found, as Fig. 4 shows, that up to the third recycle there is no change in the cutting properties of this abrasive.

5) Fig. 5 shows the cutting properties of the abrasive as relates to the number of repeat cutting passes with parameter of nozzle traverse rate. According to these results, the cutting depth per unit time is increased if the nozzle traverse rate is made faster, with repeated cutting. The cutting speed, i.e. the nozzle traverse rate, was then determined to be 30 cm/min in order to reduce the cutting time in actual dismantling.

6) The amount of slurry which was generated during cutting was 54 to 58 kg/min; the composition was about 90% water, about 9% abrasive, and about 1% concrete dust. Most of the volume is water, and it was found that it is easy to separate the constituents due to the difference in specific gravity of each.

7) Generation of dust was 180 to 630 mg/min, and 80% of the particles have a diameter of above 10 μm, and can easily be collected with a filter.

8) Generation of subsident mist droplets was between 14 and 20 $g/min \cdot m^2$ and the diameter was above 30 μm. This necessitated the mist eliminator in the ventilation system.

Abrasive-Water Jet Cutting System for Dismantlement of JPDR Biological Shielding

This system has been manufactured since 1986 based on the results of the cutting test. As Fig. 6 shows, the system consists of high-pressure pump, abrasive supplying unit, nozzle drive assembly, lift assembly, block bucket, slurry treatment unit, ventilation system, and operation console. In order to minimize radioactive contamination of the system, only the minimum of necessary equipment is placed near the biological shield wall, and the system is operated remotely by the operation console which is placed outside the reactor containment. The specification and performance of this system are shown in Table I.

Outlines of each constituent unit are introduced in the following:

1) The high-pressure pump generates high-pressure water of 196MPa, which is introduced to the nozzle through the high-pressure resistant pipes and a hose. From the nozzle, 50 l/min high-pressure water is jetted mixed with the abrasive at 196MPa.

2) The abrasive is introduced to the nozzle through the abrasive hose by the abrasive supplying unit which supplies the specified quantity.

3) The nozzle drive assembly is hung down inside the biological shield wall by the lift assembly, and is fixed by stretching the outrigger to cut concrete into blocks using the high-pressure abrasive-water jet. The movement of the nozzle is controlled with a cylindrical coordinate system with three axes--rotational, vertical and lateral, with range of movement $\pm 190^{\circ}$, 700 mm, and 750 mm, respectively. Continuous cutting is, therefore, possible within this range without changing the fixed position of the equipment.

The sequence of cutting concrete into blocks is as follows: cutting on the back side, vertical side and horizontal side. In the case of cutting from the back, the nozzle is rotated upward and the abrasive jet is directed upward for cutting. The nozzle traverse rate can be fixed at any value in the range 0 to 120 cm/min. Driving of the nozzle is made by automated remote control operation from the control unit consisting of a control panel and operation console. The location and shapes of blocks to be cut are programmable.

4) The dismantled blocks are collected and carried away with the block bucket and transferred to the transfer vehicle and stored in the containers by the block handling device.

5) Water, abrasive, and concrete slurry coming from the cutting kerf flow down to the funnel and are treated by the slurry treatment unit. As shown in Fig. 7, the slurry treatment unit consists of a funnel, abrasive separator, flocculator and flocculant feed unit, thickener, and effluent unit, where abrasive particles used are separated from the slurry by the magnetic separation method for reuse. For separation from the abrasive of concrete dust, etc. which adhere to the abrasive, showering is also used during magnetic separation. After separation of the abrasive, the slurry is separated into concrete dust and water with the flocculator and the thickener. The concrete dust is stored in the sludge tank. The water is introduced into the drainage system of the JPDR facility.

6) The dust/mist generated is filtrated through a local ventilation system, which consists of a heat exchanger for cooling, mist eliminator, electrical heater, filter, and exhaust unit, as shown in Fig. 8, and is then introduced to the JPDR ventilation system. The dust is filtrated by the filter after the water content of the mist has been eliminated separately.

DISMANTLING PROCEDURE FOR THE JPDR BIOLOGICAL SHIELD WALL

The biological shield wall of the JPDR is a cylindrical reinforced concrete structure surrounding the reactor pressure vessel, with an inner diameter of 2.7 to 3.5 m and with a thickness of 1.5 to 3.0 m. Inside the wall, there are 13 mm-thick liner steel plates.

Dismantlement of the biological shield wall will be performed after the reactor pressure vessel is removed. Although the radioactive inventory values are lower than those inside the reactor or for the pressure vessel, it was estimated to be approximately 8 Ci as of the end of March, 1989. The radiation dose rate inside the biological shield wall was then estimated to be approximately 250mR/h, even after the pressure vessel was removed. However, if highly activated concrete about 40 cm in depth is removed from the inner surface of the wall, the value declines to below 5mR/h and workers can approach it.

Therefore, it was decided to dismantle highly activated zones of 40 cm concrete depth from the inner surface of the wall using two remote controlled techniques: the abrasive-water jet cutting system and the mechanical cutting system. These two techniques will be used to demonstrate the effectiveness of the cutting systems developed for actual dismantling.

After dismantling the highly activated concrete, the remaining concrete of the biological shield will be demolished using the controlled blasting technique.

As shown in Fig. 9, the dismantling will be performed firstly from the upper part of the highly activated zones using the mechanical cutting system, secondly the lower part of the highly activated zone using the abrasive-water jet cutting system and lastly the remaining part of the biological shield wall using the controlled blasting technique. The highly activated concrete will be cut into large blocks to prevent the diffusion of radioactivity and to reduce radioactive waste. Considering the capacity of the steel container and block handling, the size of the blocks was decided to be 40 cm x 40 cm x 80 cm for the abrasive-water jet cutting. The cut-away blocks are handled by remote control and contained in 1 m^3 steel containers and removed. One container can accommodate 4 blocks. In these dismantling operations, concrete of weight 1400 tons with radioactivity 8 Ci will be generated and radiation exposure of workers is estimated to be 7.9 man-Rem (see Table II).

CONCLUSION

The characteristics of the abrasive-water jet cutting system developed

as the dismantling technique for biological shield walls can be summarized as follows:

1) The system is capable of dismantling the highly activated concrete into blocks under remote controlled operation and preventing radiation exposure of workers.

2) The nozzle drive assembly is compact because the cutting reaction of the nozzle is small, and handling is easy with remote controlled operation.

3) Concrete can be cut at any position and into any shape, so cutting and removal of the required parts only (e.g. radioactive parts) is possible.

4) The volume of radiation waste can be decreased by reuse of the abrasive.

5) Most of the dust generated during cutting is collected in the form of slurry and dust/mist generation can be minimized, which is an advantage in terms of preventing the diffusion of contamination by radioactivity and radiation exposure of workers.

The abrasive-water jet cutting system can be applied not only to concrete structures, but also to activated steel structures. The efficiency of the system does not deteriorate even for under water cutting. Without abrasive, i.e. with water jet only, the system can be applied to cleaning contaminated equipment or exfoliated concrete surfaces. The range of application could be extended, depending on future developments.

For the future, JAERI will start a full-scale mockup test on the JPDR biological shield wall using this system in 1988. Its comprehensive performance, workability, etc. will be confirmed for actual dismantling at that time. It is firmly believed that the data and know-how obtained through the dismantling of the JPDR biological shield wall will prove the usefulness of the abrasive-water jet cutting system for decommissioning commercial power reactors in the future.

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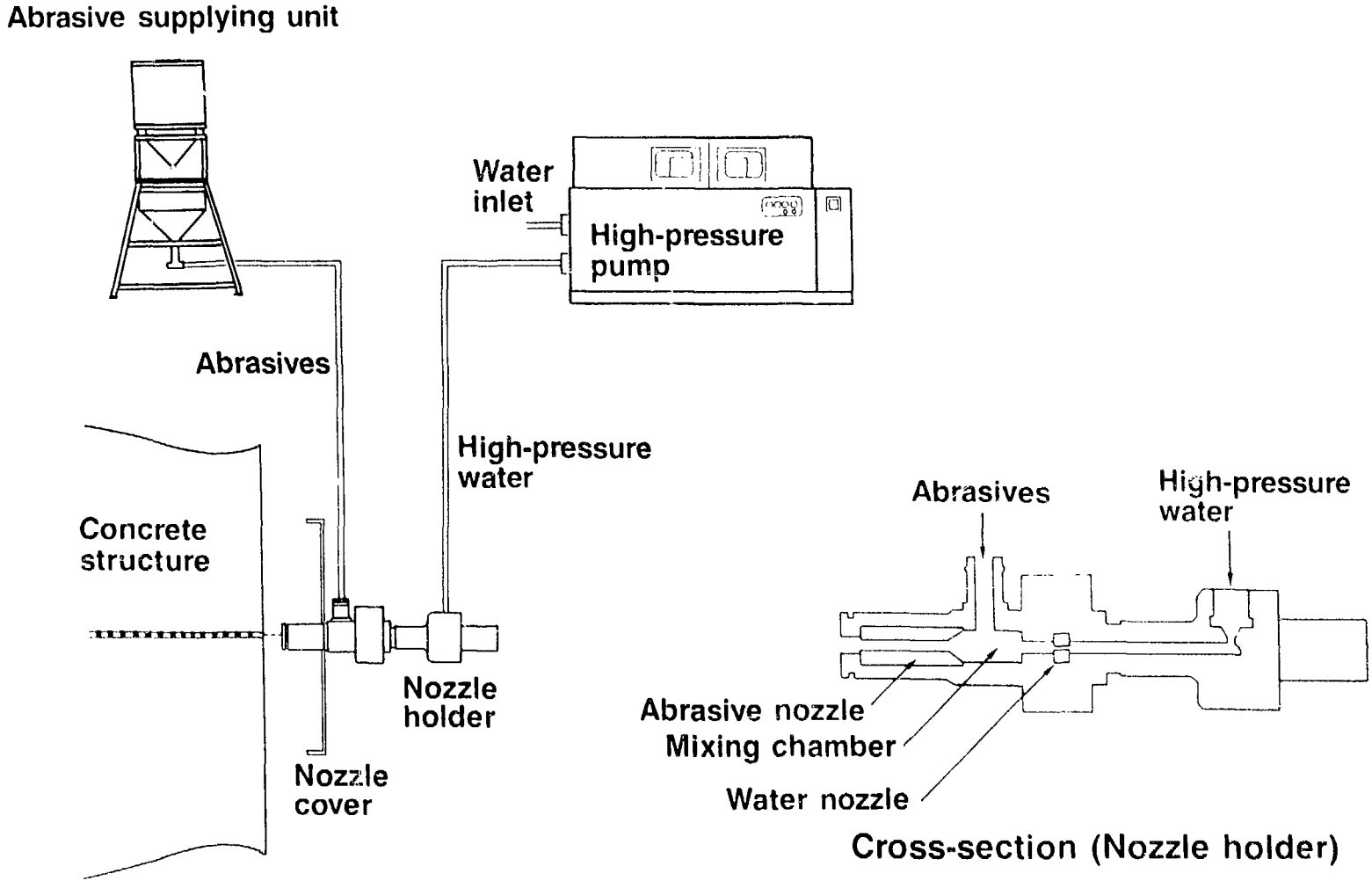
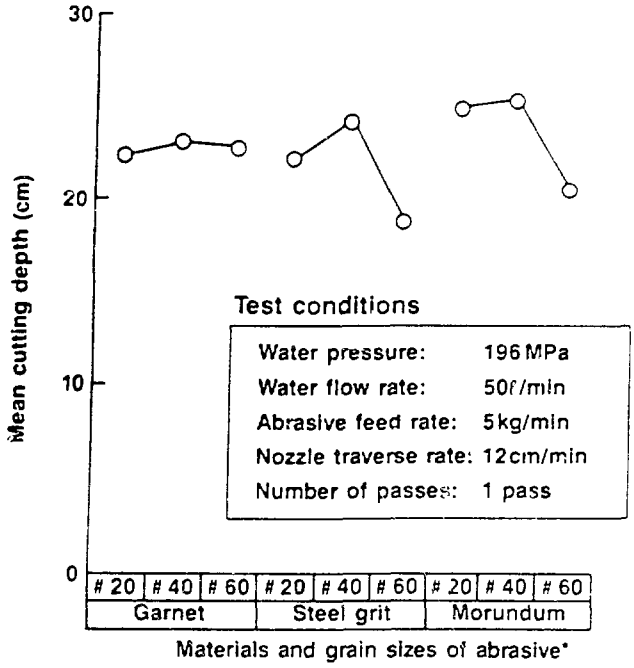


Fig. 1 ABRASIVE-WATER JET GENERATING UNIT



*Specific gravity and hardness of materials

	Specific gravity	Hardness (Hv)
Garnet	4.0	1,350
Steel grit	7.65	800
Morundum	3.97	2,070

Particle diameter of grain sizes of abrasives

Grain sizes	Particle diameter (μm)	
	max.—min.	Mean
# 20	2,000—840	1,300
# 40	840—250	600
# 60	420—105	300

Fig. 2 COMPARISON OF CUTTING DEPTH OF ABRASIVE

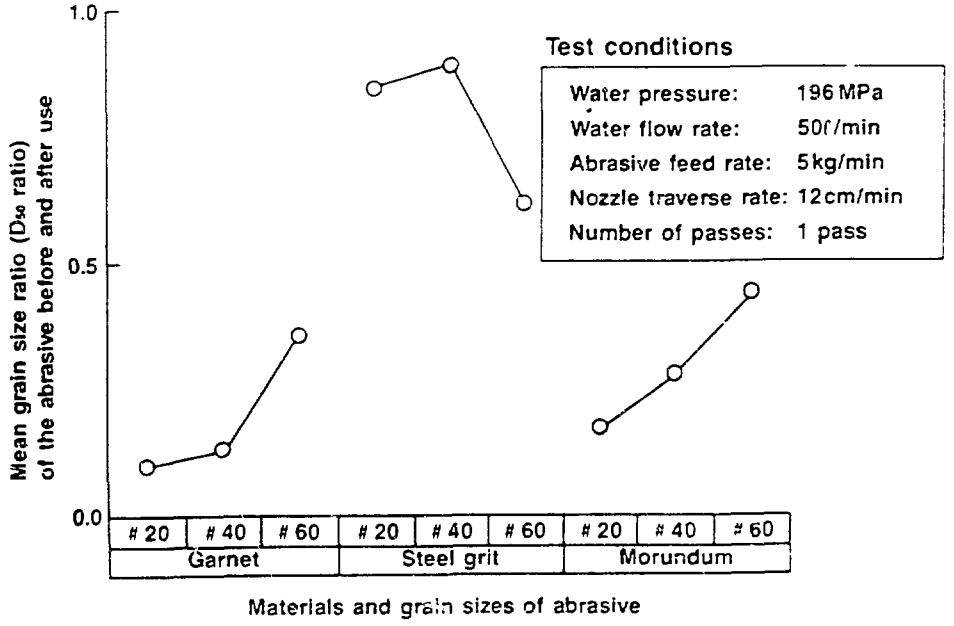


Fig. 3 COMPARISON IN PARTICLE DIAMETER BETWEEN ABRASIVES BEFORE AND AFTER USE

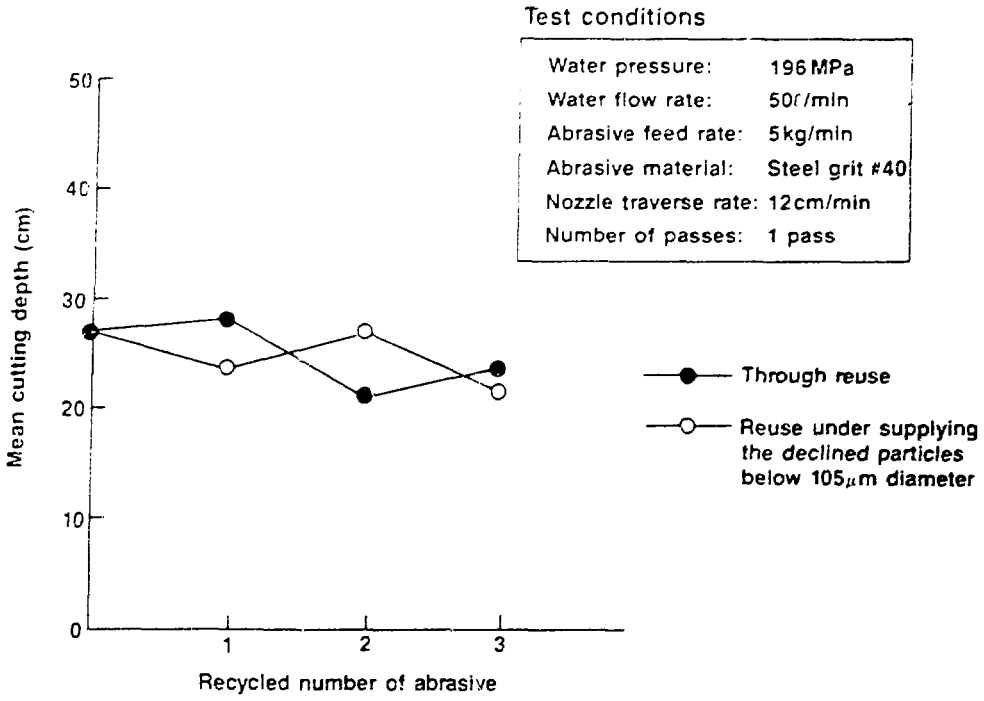


Fig. 4 CUTTING DEPTH OF RECYCLED ABRASIVE

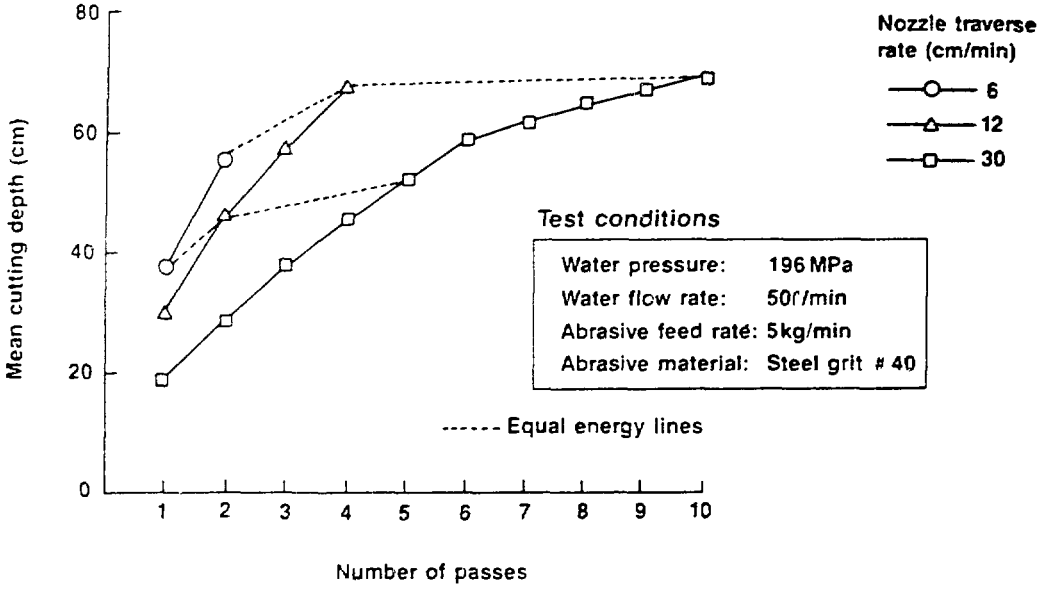


Fig. 5 CUTTING DEPTH RELATING WITH NUMBER OF CUTTING PASS

Table I SPECIFICATION AND PERFORMANCE OF ABRASIVE-WATER JET CUTTING SYSTEM

Specification	Water pressure	196 MPa
	Water flow rate	50 l/min
	Abrasive feed rate	3–7 kg/min
	Nozzle traverse rate	0–120 cm/min
Performance	Cutting time*	60 min
	Concrete dust generation*	33 kg

* Cutting time and concrete dust generation per one block cut off
Block size: 40 cm (depth) · 40 cm (height) · 100 cm (length)

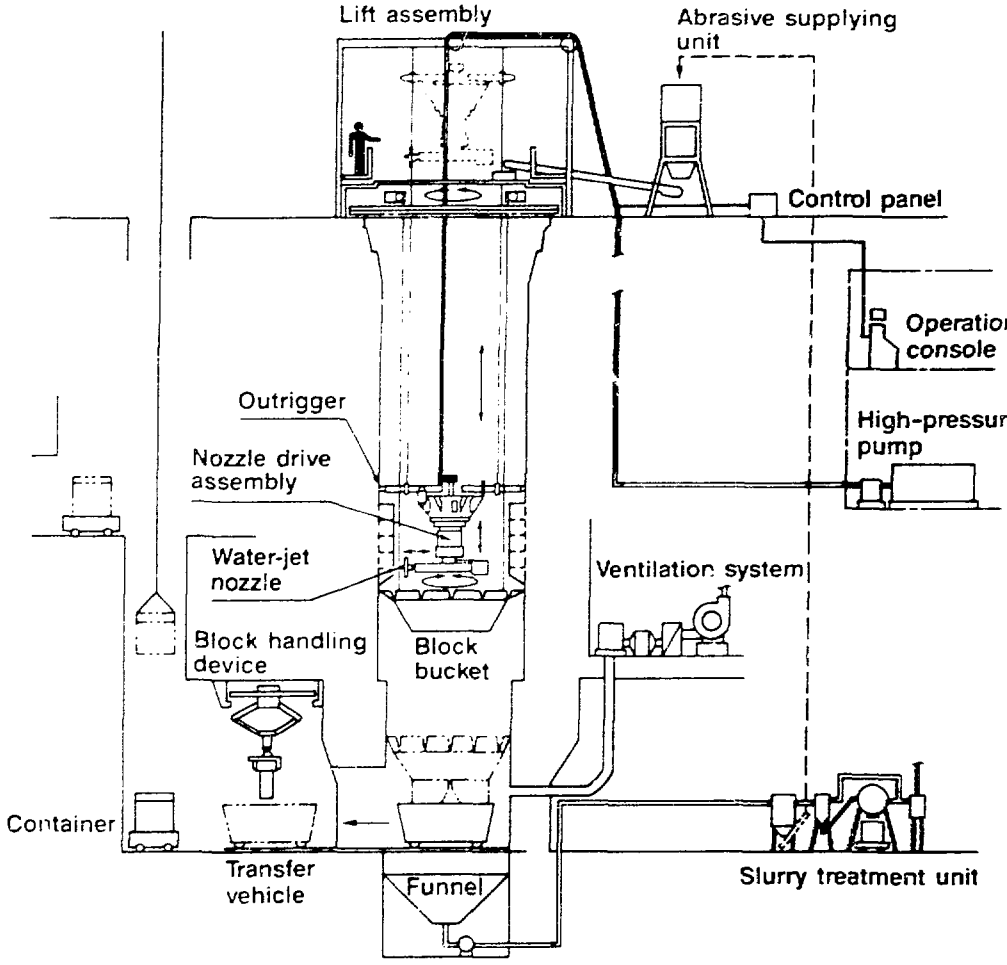


Fig. 6 ABRASIVE-WATER JET CUTTING SYSTEM

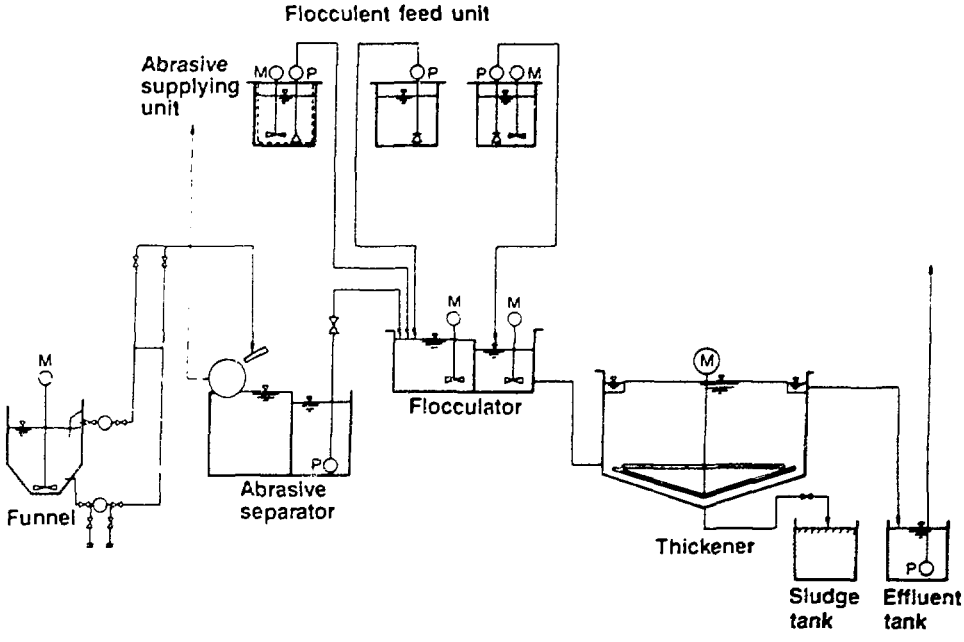


Fig. 7 SLURRY TREATMENT UNIT

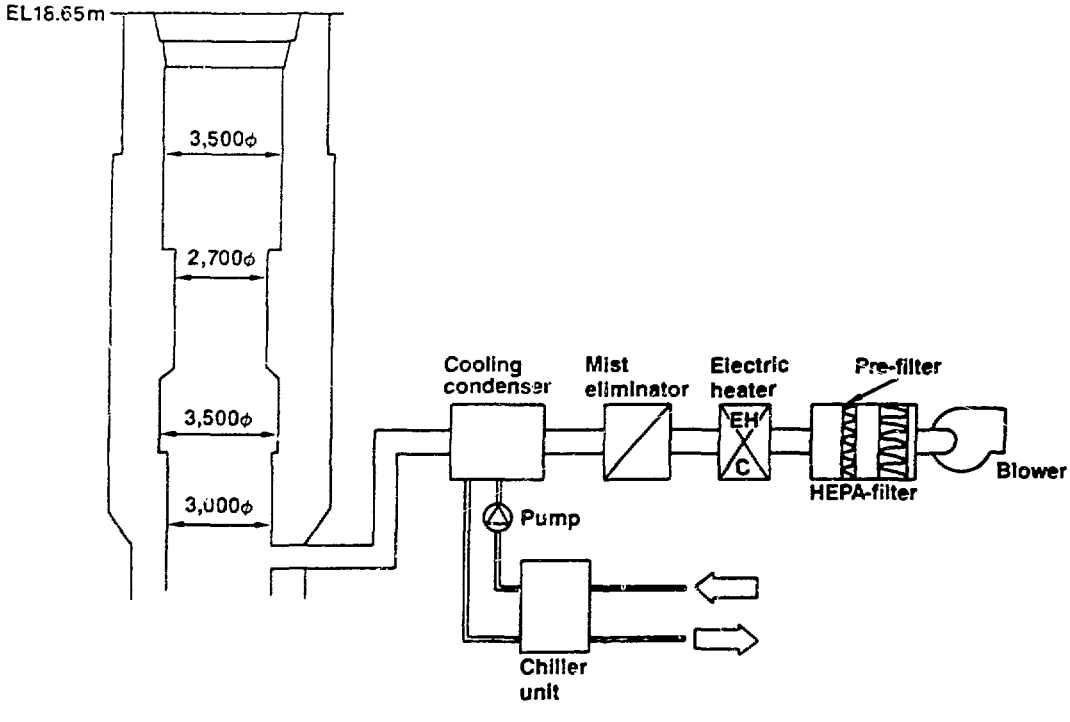


Fig. 8 VENTILATION SYSTEM

Table II ESTIMATION VALUE OF DISMANTLING PLAN FOR JPDR BIOLOGICAL SHIELD WALL

Estimation items \ Dismantling techniques		Highly activated zones		Remaining zones	Total
		Mechanical cutting	Abrasive-water jet cutting	Controlled blasting	
Dismantling concrete volume	(m^3)	5	7	548	560 (1400 t)
Number of workers	(man·day)	890	580	3030	4500
	(man/m^3)	170	80	5.5	8
Radiation exposure for workers	(man-Rem)	1.3	0.6	6	7.9
Concrete sludge volume	(in number of 200l drum can)	30	100	By cutting for slit 30	160
Articles of consumption		Water 50m^3	Water 240m^3	Water 90m^3	
		Saw blade 2 blades	Adrasive 10t	Abrasive 4t	
		Core bit 3 bits	Nozzle 160 pieces	Nozzle 60 pieces	
				Powder of explosive 195kg	

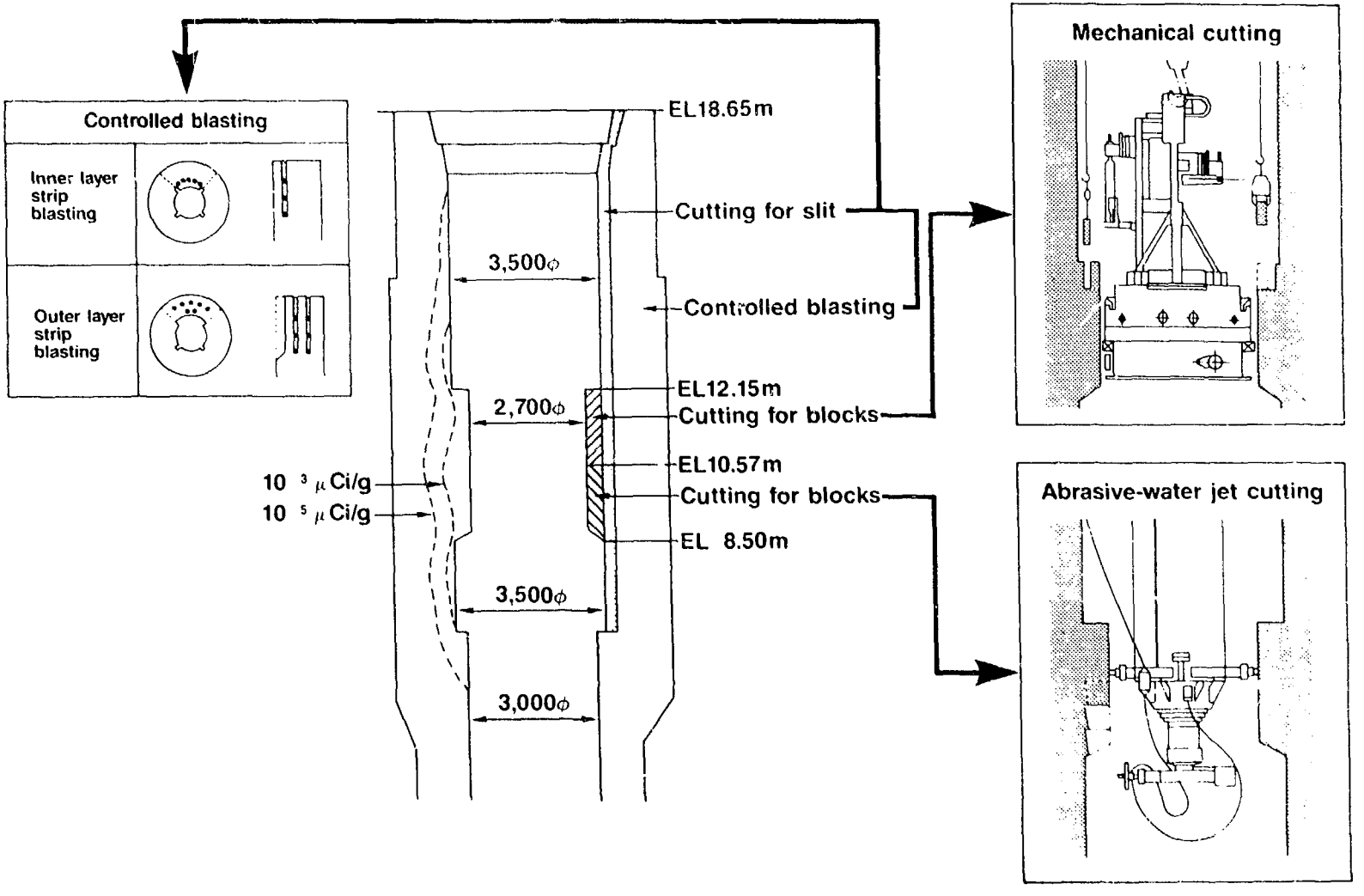


Fig. 9 DISMANTLING PLAN FOR BIOLOGICAL SHIELD

Testing the demolition of concrete and pipes with explosive charges within a nuclear power plant.

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1. Description of HDR and goals of the HDR-ZER-program

At the HDR a former nuclear power plant an extended nuclear safety research program was solved since 1974. Encl. 1 shows the test facility and the applicated accident simulations. Within this program the concrete and pipe demolition by explosive charges (ZER) is one test group running since 1983.

For the purposes of disassembly techniques, the HDR is a facility on which parameters known from single tests can be checked under realistic boundary conditions for their validity within a closed containment.

The ZER test group is subdivided into tests on

- concrete demolition, ZER B
- pipe disassembly, ZER R.

ZER B

When nuclear power plants are decommissioned, it is important to disassemble radioactive plant components (biological shield) in such a way that only the radioactive portions must be put into final storage.
Breaking up activated and contaminated concrete structures by blasting seems to be a suitable disassembly technique from the points of view of radiation protection and economy.

The possibility of such disassembly without any hazard to adjacent structures and to the reactor containment is to be demonstrated on the HDR in a real dry containment; such phenomena as dust emission, shock wave propagation and activity release as well as necessary protective measures and the time required for this type of decomposition technique are to be analyzed.

ZER R

The mechanical equipment components of decommissioned nuclear power plants must be dismantled and conditioned under the most economic boundary conditions possible and so as to involve a minimum of radiation exposure to the decommissioning personnel. Also in this case, the separation of large pipe systems and components by blasting could be a possibility.

Information is to be obtained on the HDR under realistic conditions to allow this method to be evaluated and between compared with other techniques of pipe separation and removal.

2. TEST EXPERIENCE

ZER B experimental setup

Tests conducted on slabs connected with the original HDR structure allow information to be obtained on

- the different ways in which shock waves are introduced into the HDR building structure as a function of blast intensity
- the introduction and behavior of blast waves in the containment
- the vibrations produced in the HDR and in adjacent installations,
- max. possible shock waves and blast intensities in the plant.

Thus two series of Experiments with slabs were performed with different coupling to the structure; different reinforcements and stepwise increased loadings up to 2,5 kg of explosives.

Encl. 2 shows the organization of HDR experiments within the program; encl. 3 shows the experimental setup; encl. 4 presents the experimental matrix. Further planning includes the dismantling of an simulated biological shield within the original HDR-structure, encl. 5.

Experimental results ZER/B

The blasting load must be subdivided into local and global loads acting on the HDR containment.

Local loads are characterized by
- the forces occurring at the support points of the steel

reinforced concrete slab,
- slab deflection and deformation,
- slab acceleration,
- shock wave pressure in the concrete

and thus refer to the load parameters directly acting on the slab.

Forces rise in a linear fashion with increasing amounts of explosive in the case of coupling at four points. As a result of the damage to the slab rising by the same degree, forces reach a kind of saturation, and the rigidity of the slab decreases, i.e., subsequent blast make the slab more and more brittle, encl. 6.

The slab accelerations reflect the shock type load produced by the detonation of the blast hole charges even before there is any sizable deformation. These shock waves result in high-frequency loads acting on the slab; in the area contact mode also the rotunda is excited to high-frequency vibrations. However, these high-frequency excitations decay very quickly (after some 10 ms) and are of no relevance to the damage caused, because their energy is dissipated over large areas by reflection and dispersion, encl. 7.

Assessing the distribution of stresses and deformations in the environment of the blast hole was found to be very difficult, the pressure drop could be measured only up to a distance of about 15 cm from the blast hole.

The global impulse type load acting on the HDR in reinforced concrete blasting can be subdivided into three contributions:

- The radial impulse from the slab to the rotunda
- The axial impulse tc the HDR by the mass load dropped on the floor and its dropping velocity.
- The explosive cloud impulse (contained in the radial impulse).

The shaking of the plant by this impulse type load resulting from the explosion mainly occurs at very high frequencies, which are not related to any damage, of the HDR-structure, and the excitations are of no harm to the neighbourhood too blast waves decrease within short distances around the slab, encl. 8.

Screening of the crushed material and dust measurements were performed during the experiments permitting the following statements:

The specific quantity of crushed material is between 1 and 3.8 kg of crushed concrete per g of explosive, encl. 9
This is in agreement with the results of earlier experiments.

- The total quantities of dust are on the order of 10^{-4}mm, relative to the total amount of crushed material, which is approximately one order of magnitude higher than determined in earlier experiments in the open air.
- The dust contains material from the sealing plug with a fraction larger than 10%.

The follow-on experiments are to predict transferibility of HDR results to realistic concrete structures of NPP'S.

ZER R

In the tests conducted on the HDR, reference values in a realistic environment were determined for the method of separating active pipes blasting compared to other methods.

The following points were studied:

- optimization of the necessary charge volumes and configurations,
- testing protective shieldings against fragments,
- pipe- and plant behavior during explosive cutting procedure

The optimized charges are able to cut pipes up to 600 mm Diameter and 20-30 mm thickness.

Fragments of the charges can be handled by the used shielding method, evenwell the used shielding cave is not suitable to all pipe systems and needs optimization.

First experiments show high loads with high-frequency vibrations to the containment when cutting a short pipe fixed to the wall, and a very low transmission of vibration when cutting a real loop portion of HDR, encl. 10.

The applied measuring technique needed a lot of evaluation especially for the very short load duration time and the impulse load measurement during area coupling.

3. CALCULABILITY OF TESTS

ZER B

Precalculations and recalculations of the tests were

carried out.
Information was sought on

- the short-range effects,
- the introduction of the blasting excitation into the ambient structure and the building,
- the global behavior of the HDR facility (safety statement),
- the transmission of this excitation to adjacent buildings,
- the transferability of HDR experiments to present-day nuclear power plant structures.

Four possible steps in computation are envisaged:

- Demonstration of the safety of the HDR and of the technical process and measurement designs,
- demonstration of the possibility to determine local effects and the chain of transmission in the HDR by precalculations and recalculations of specific tests,
- demonstration of the observance of licensing conditions during disassembly by blasting,
- transferability of HDR findings to other plants.

A comparison of global load predictions with the computer models (simple beam model and shell model) shows good agreement with the filtered acceleration measuring points, after a modified time - vs. - load function was set in which the axial impulse derived from the dropped load was entered into the computer models as an input parameter. The global HDR behavior can be determined with sufficient accuracy by the existing methods of computation, Encl. 11.

The blast wave peak pressures in the short range blast area are roughly one order of magnitude below theoretical data.

Code prediction of local load distribution and the short-range behavior need further evaluation efforts.

4. ENVIRONMENTAL IMPACT AND LICENSING ASPECT

Measures were taken in the neighbourhood of the plant to learn about the environmental impact to another near by situated NPP.

No sign were found, that the high frequency loads lead to vibrations outside the plant.

Within the plant it could be shown that the high frequency impuls leads to no damage of components and structures. The primitive dust protection installation (curtains and venti-

lation) protected the plant level 1.500 from dust and toxic gases.

The blastwaves are bearable by the plant structures up to 10 times higher charge loads, even in the narrow compartments of the HDR. Blast waves data will be used by german licensing authorities to check future decommissioning activities in real plants.

The load bearing capacity of HDR is about 10 times higher than precalculated.

5. References

1. Freund, H.U.; Wegener, H.; Mueller, K.: Quick Look Report Nr. 62-85; Versuchsgruppe ZER, Juli 1986

2. Grimm, R.: Versuchsprotokoll T54.2.1
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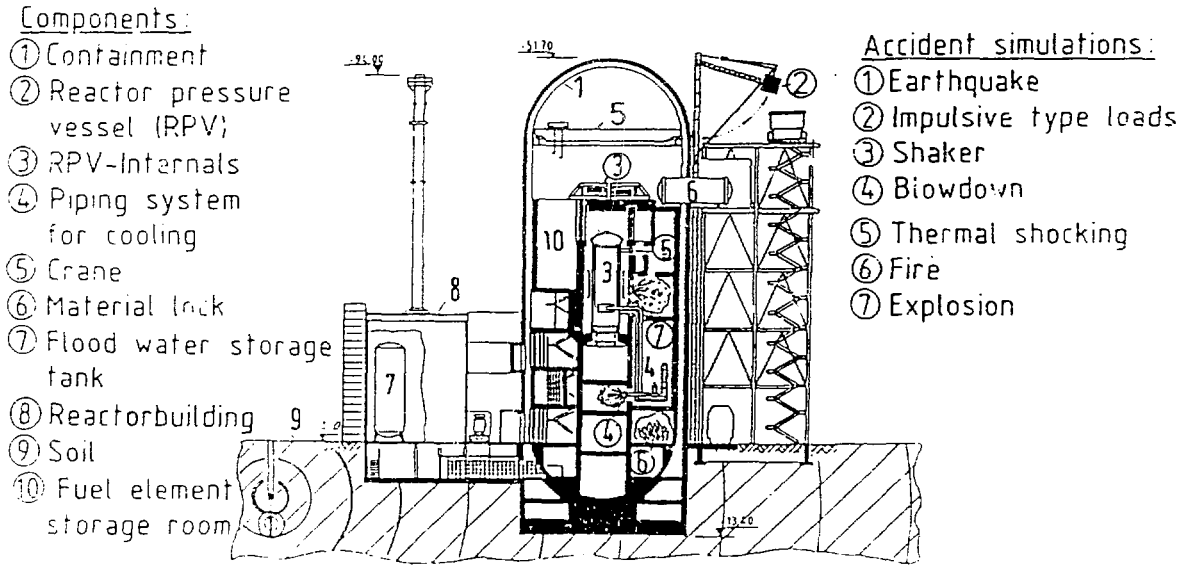
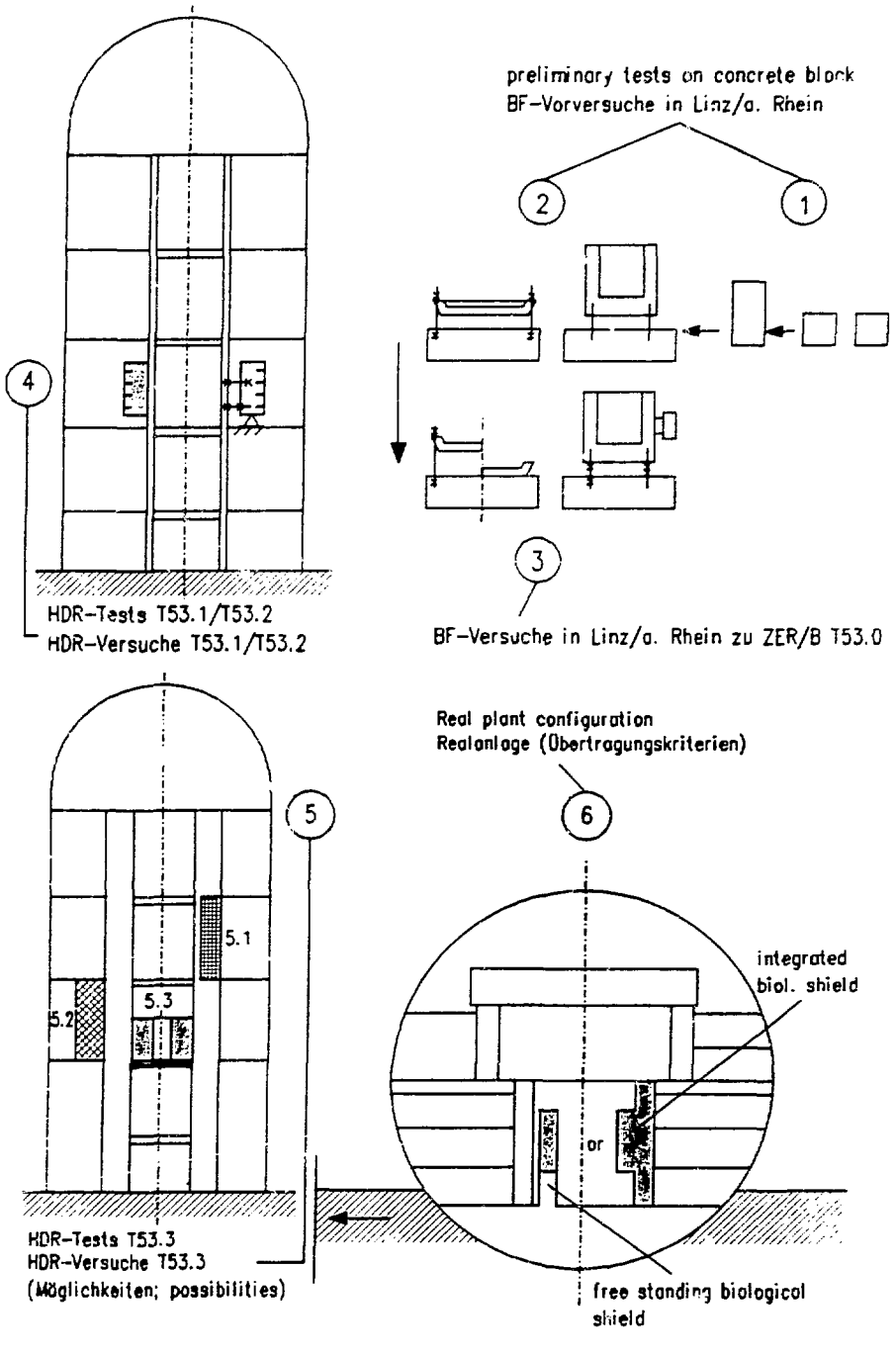
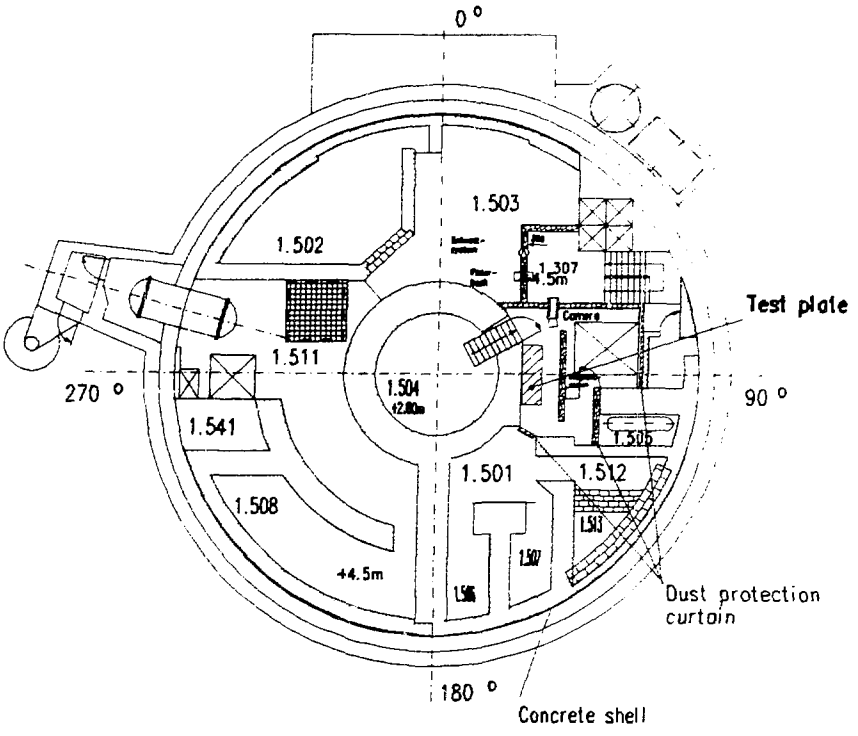


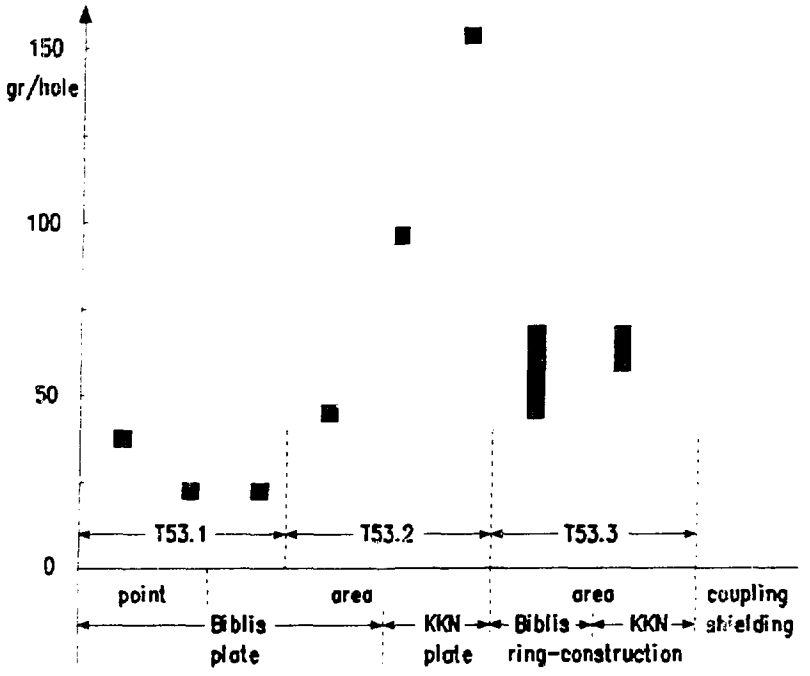
Encl. 1 HDR test facility and applicated accident simulations



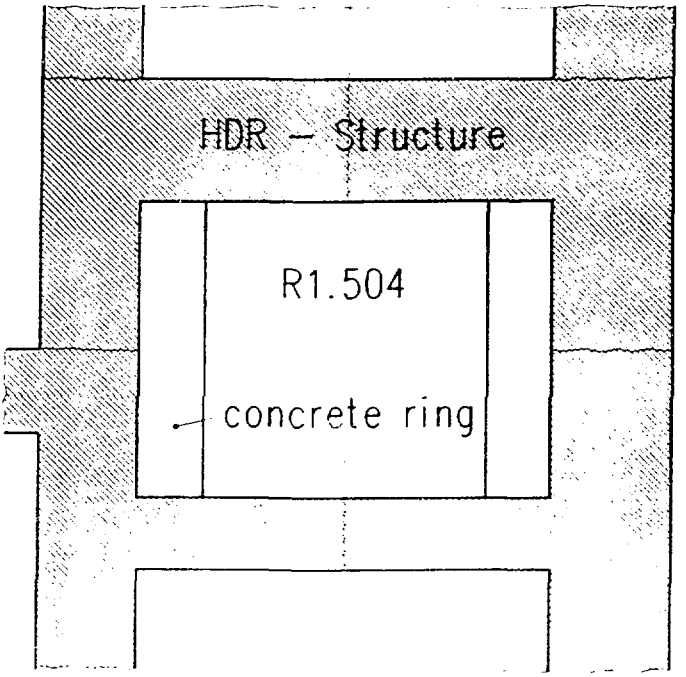
Encl. 2 ZERB program construction



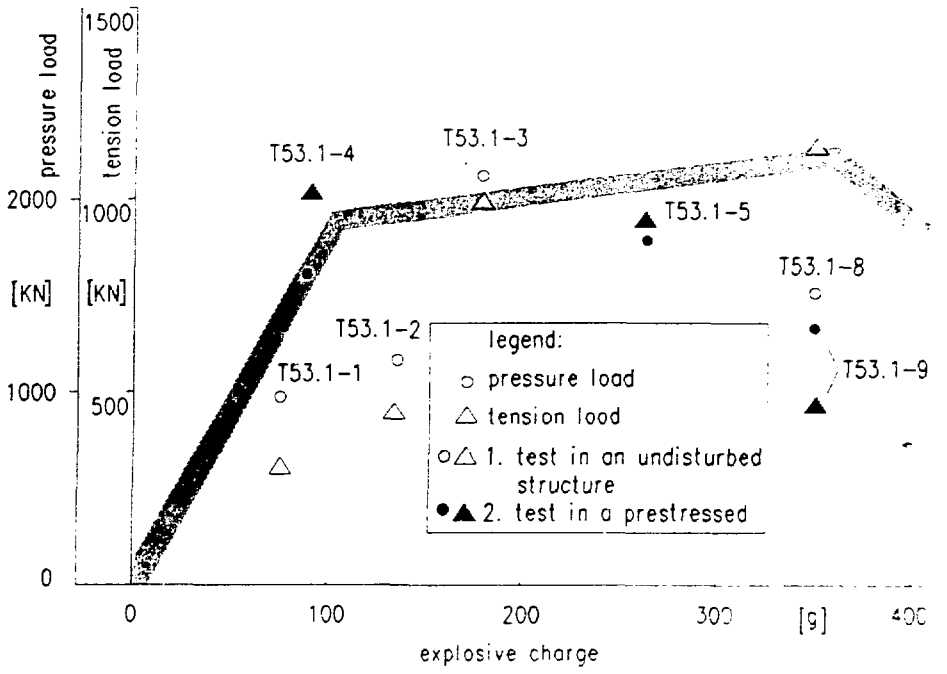
Encl. 3 Test setup T53.1/2



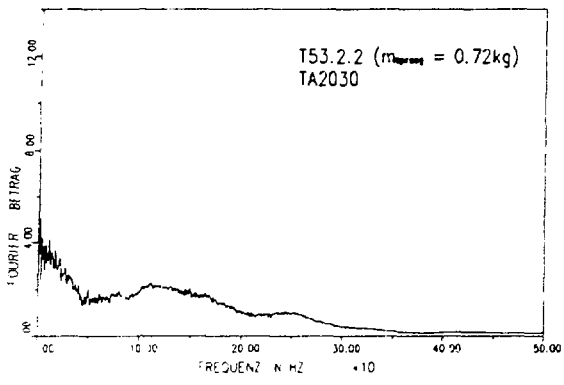
Encl. 4 Experimental matrix

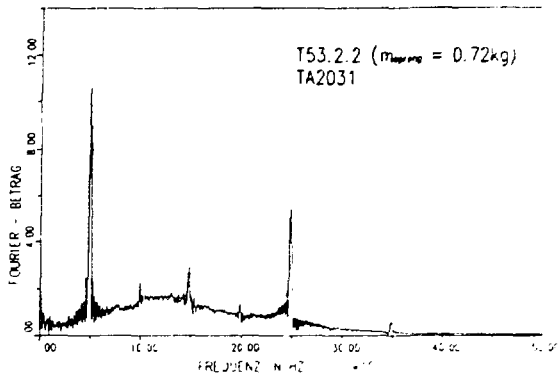


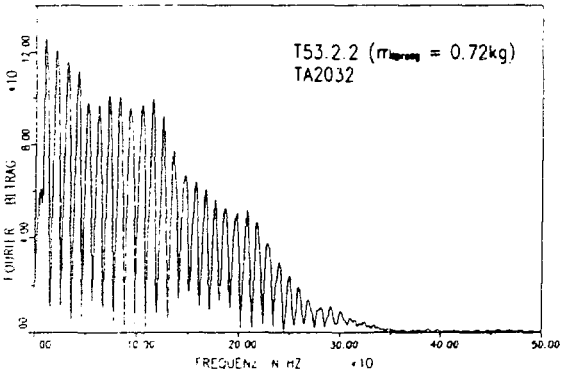
Encl. 5 Test setup T53.3

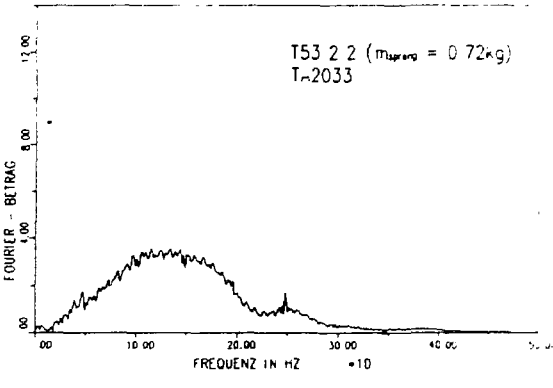


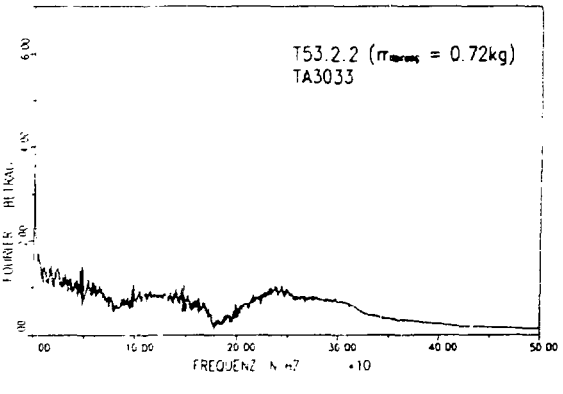
Encl. 6 Slab forces point coupling

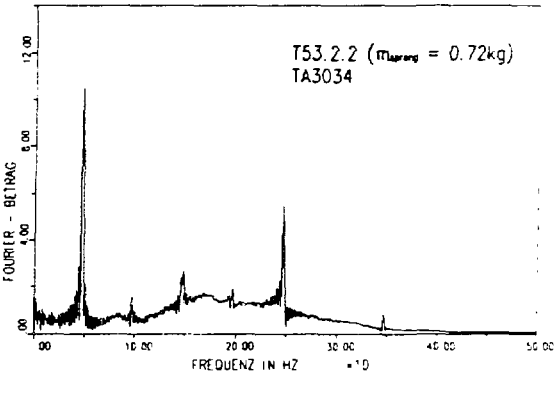




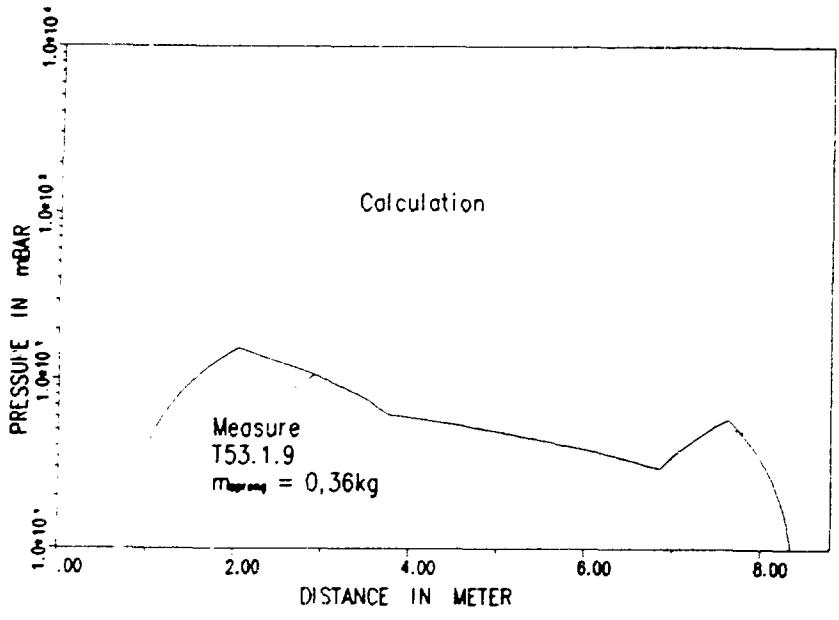




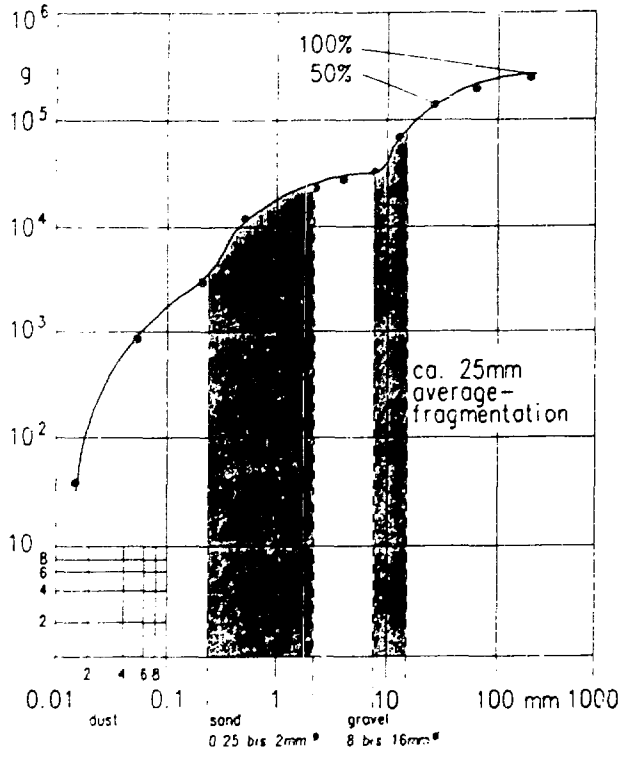




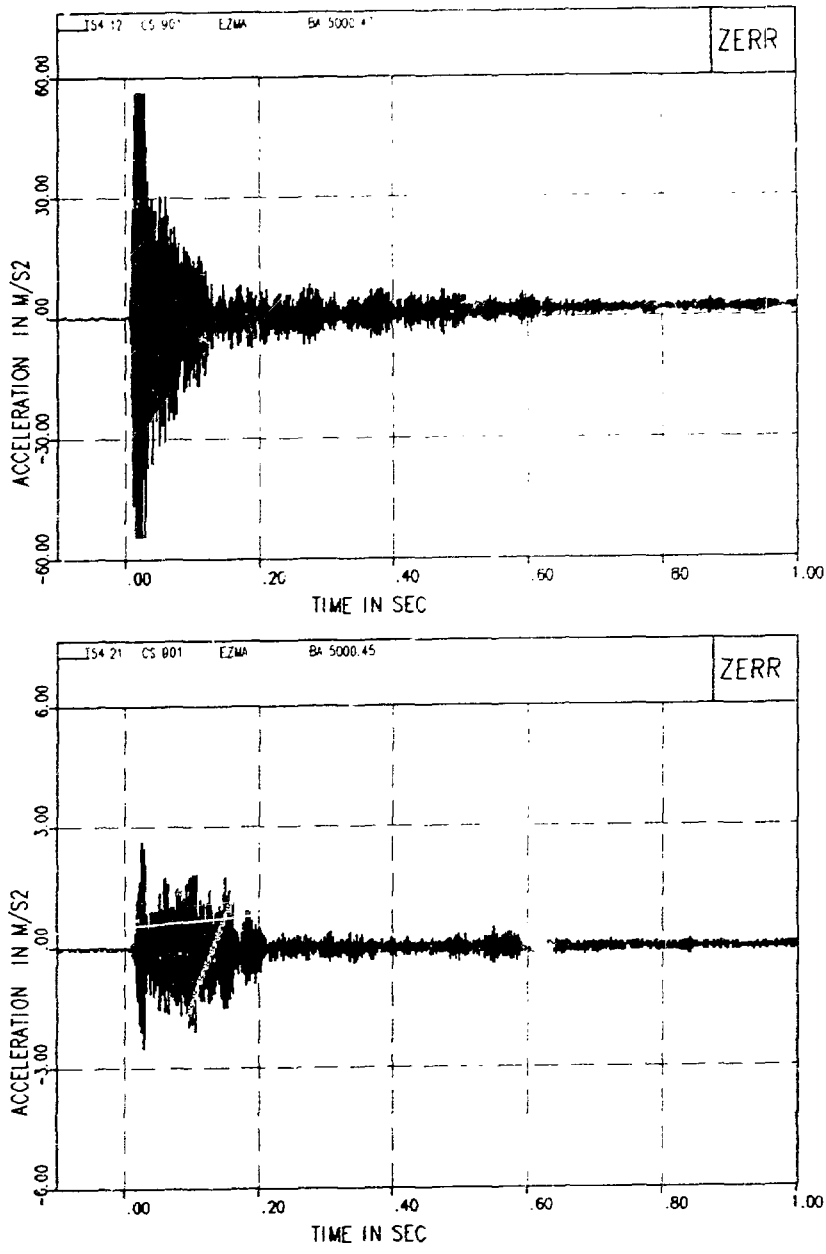
Encl. 7 Slab acceleration analysis



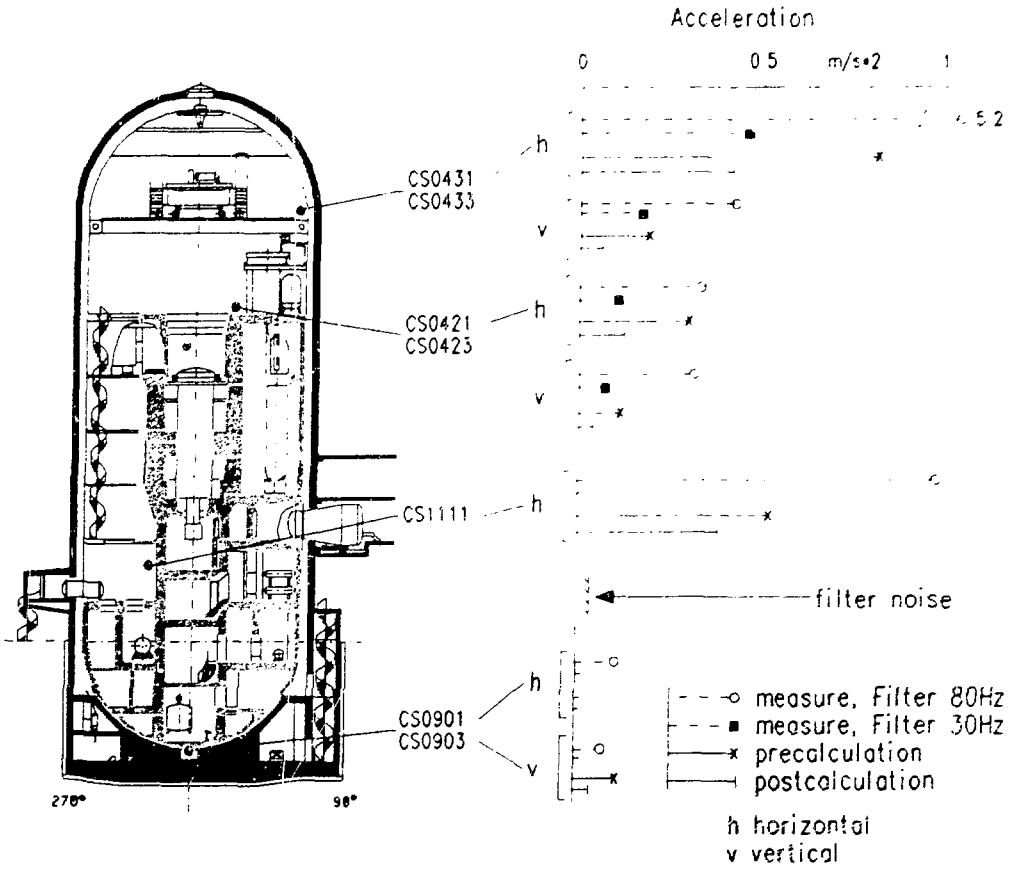
Encl. 8 Blast wave behavior



Encl. 9 Crushed material pattern



Encl. 10 Comparison of accelerations on pipes



Encl. 11 Global HDR-behavior Me/Re

EXPLOSIVE PIPE CUTTING BY SHAPED CHARGES IN AN ANNULAR CONFIGURATION *)

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ABSTRACT

For the decommissioning of nuclear power stations it is necessary to develop dismantling techniques to minimize the dose of radioactivity for the personnel. Concerning the primary cooling system the separation of contaminated steel pipes is solved up to now only unsatisfactorily by usual cutting techniques like sawing or thermal cutting if there is impeded access to the tubing by narrow constructions. An explosive pipe cutting technique was developed to separate thick walled tubes. The simplicity of the arrangement and its easy applicability are the advantages of this dismantling methode. The developed sleeve charge is assembled from linear modules surrounding the tube wall as a polygon. When initiated the projectiles formed from the metallic liner of the charge hit the tube wall simultaneously and pipes up to 610 mm diameter and 36 mm wall thickness were cut. Such pipes comply with the fresh steam pipe of a recent German boiling water reactor. The impulse load to the tube wall outside the cutting region is far below the value which would lead to destructions at the remaining pipe system. Strong confinement of the charge reduces the blastwave pressure and the damage boundary value of 1 bar is exceeded only within a distance of 3 m to the centre of the tube. To avoid fragment damage it is necessary to apply a heavy protective construction. Using such a construction the developed sleeve charge technique was tested when cutting a section of the cooling system of a hot steam reactor (HDR).

INTRODUCTION

For the separation of steel pipes, both mechanical (sawing, milling) and thermal cutting processes (welding) are usual. The separation of steel tubing by explosive techniques comes into consideration in cases where the apparatus for separation has to be simple because the access to the pipe is impeded by narrow construction and the prevailing conditions restrict or even prevent the on-site presence of personnel. An example for such an application is the dismantling of tubes in areas of high radiation density inside a nuclear power station to be decommissioned.

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*) Supported by the Federal Ministry for Research and Technology, Bonn

Explosive pipe cutting by shaped charges may be carried out in two different ways: from the outside using sleeve charges, or from the inside using catheter charges. The cutting of tubes from the outside using sleeve charges and the resulting effects on the environment are discussed.

CUTTING CHARGE DEVELOPMENT

Explosive Cutting Technique

In order to separate steel tubes, so-called cutting charges (linear shaped charges) are used. With these, the energy of the explosive used is concentrated into a specific direction by suitable shaping of the charge /1/. The principle of action of a cutting charge is shown by the sketch in Fig. 1. When the explosive is detonated, the so-called cutting liner (metals such as Al, Cu, Ag, Pb or their alloys are suitable materials) is accelerated in the direction of the material to be worked upon (in this case the wall of a steel tube which has to be dissected), thus forming a sword-shaped projectile. Although the strength of this "sword" is low, the velocity it attains (typical values are in the region of > 2000 m/s) is large enough for it to penetrate the material hydrodynamically.

The penetration depth and the crack propagation, which provide the resultant cutting depth, depend to a high degree on the properties of the material to be penetrated. The cutting capacity of a linear shaped charge for a particular material (e.g. St37 steel) is principally determined by its specific content of explosive material (mass of explosive per unit of length), the cross-section geometry and the material of the liner /2/. Normally, the metal confinement and the liner of an linear shaped charge consist of the same material and are made in one piece. However, only lead and its alloys are suitable for the enclosure of flexible cutting charges. All the other metals mentioned above and their alloys are superior to lead as regards their actual cutting performance; however, they can only be molded during actual manufacture of the charge.

The spacer element providing the proper standoff necessary to obtain the optimum cutting depth is integrated within the metal confinement of the cutting charges as shown in Fig. 1. Attachment and fixation onto the structure to be cut (the tube wall) is generally achieved by adhesion or with magnetic attachment. Initiation of the cutting charge may, if required, be undertaken simultaneously at several points.

Test of the Method with Flexible Cutting Charges

First trials were carried out on the separation of steel tubes using commercially obtainable, flexible explosive cutting charges to find out the cutting performance and the effects on the environment incurred during separation. Tubes made of mild steel having 220 mm diameter and 10 mm wall

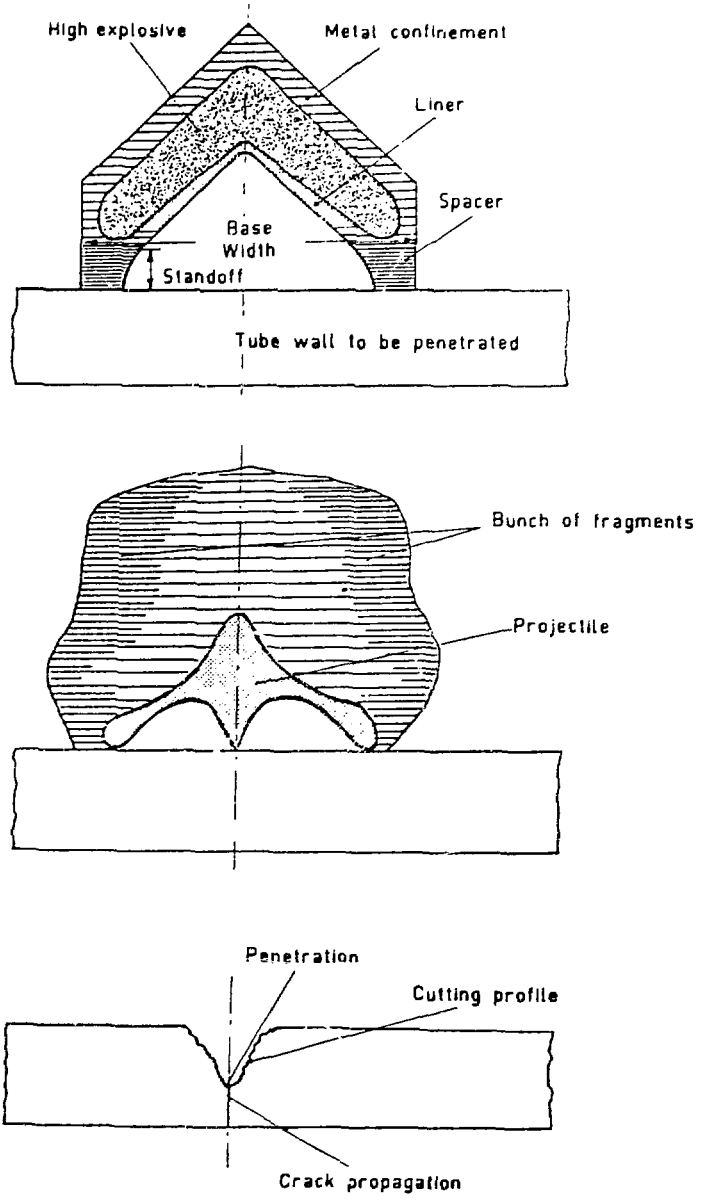

Figure 1: The principle of action of cutting charges

thickness as well as tubes consisting of high-tensile steel with a diameter of 273 mm and 12.5 mm wall thickness were used for this purpose.

A pipe section > 600 mm in length was welded onto a base plate, which was in turn fixed onto a massive steel base by means of 3 force transducers to measure the axial impact force.

For protection against fragments from the metal confinement the trial setup was concentrically surrounded by an open-top sheet steel cylinder. In this surrounding cylinder, three marker boards consisting of 50 mm thick Styropor layers sandwiched between 3 mm light metal plates were positioned to record fragmentation effects. It was thus possible to obtain information on the size, direction and penetration power of the fragments.

In the trials, measurement of the blastwave pressure was carried out with quartz pressure sensors located at different distances from the charge.

It was possible to cut both kinds of the above mentioned tubes accurately and neatly with the charges used. However, the projectiles formed during the detonation process are comparable slow and not compact, for which reason the cutting performance of these commercial charges used is not optimal.

Measurements of the environmental effects conducted during the trials show that the incidental effects of explosive tube cutting which are introduced by fragmentation, blastwave and projectile impact on the tube line are controllable. The trials carried out with flexible linear shaped charges have figured out a considerable potential for improvements which led to the development of strongly confined shaped charge modules.

Development of Charge Modules and Improvement of the Cutting Performance

A very useful experimental setup to determine the cutting capacity of a linear shaped charge can be seen from Fig. 2. The shown arrangement allows to measure the cutting or penetration depth profile e.g. in steel as a function of the charge standoff.

In order to reduce the amount of parameters to be investigated for the improvements from the knowledge about projectile forming linear and conical charges we were able to make the following statements:

- To cut steel the best form for the liner is a triangle, the best opening angle has to be determined (90° is a very common value).
- The best material for the liner is copper, the liner thickness has to be investigated.

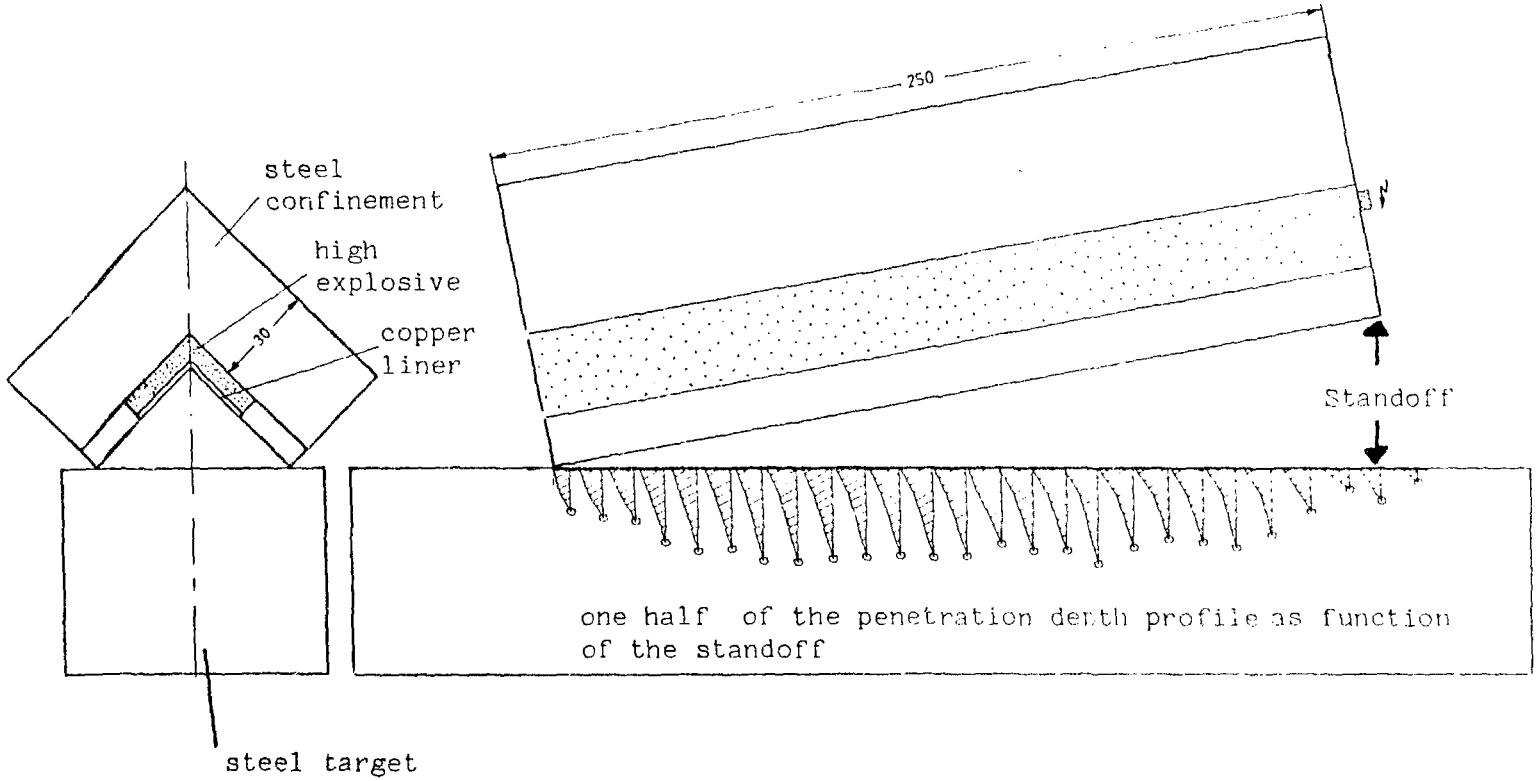

Figure 2: Experimental setup to determine the cutting capacity of a linear shaped charge

- To minimize the amount of nigh explosive, which is very important for the indoor application it is necessary to have a strong confinement. From data of explosively driven flyer plates /3/ a thickness of about 30 mm could be estimated to be sufficient.

So we started our investigations with the 90°-linear shaped charge shown in Fig. 2. One half of the obtained cutting profile is outlined in Fig. 2 also.

In contrast to the behaviour of conical shaped charges the reduction of the opening angle to 60° reduces the cutting capacity by approximately 30 %. A reason for this result is the lower degree of symmetry which leads to a very strong particulation of the 60°-linear shaped charge jet. The x-ray flash pictures in Fig. 3 illustrate how the opening angle changes jet velocity and particulation.

For three charges having opening angles of 60°, 90° and 120° at different times two pictures in direction perpendicular to each other were taken. So beside the degree of particulation for the projectiles the jet velocity could be determined.

While the projectile becomes even more compact, the increase of the opening angle decreases the jet velocity (3550 m/s for 60°, 2630 m/s for 90° and 1840 m/s for 120°).

Concerning the 60°-charge, the cutting profiles measured in other trials allow the conclusion that the destortion of the beam running to the left side in Fig. 3 ist not typical for this 60°-charge and so does not explain the bad cutting performance. Beside our experiments we have carried out one dimensional calculations to simulate the process of jet formation. The obtained velocities and mass ratios are in good agreement with the experimental results; these calculations also show that a degressive coverage of the liner with high explosive reduces the spread in jet velocity. According to this result we have obtained the best cutting performance for linear shaped charges having flat opening angles for the liner and a 90°-confinement.

Performance Test: Thickwall Pipe Cutting

The good cutting performance of the developed linear shaped charge over a large region of standoff leads to a sleeve charge configuration which is assembled from linear modules surrounding the tube wall as a polygon. Fig. 4 shows in the upper part the octagon used to cut a pipe of 610 mm diameter and 36 mm wall thickness corresponding to the fresh steam pipe of a recent German boiling water reactor. The experimental result is shown in the lower part of Fig. 4; the tube was cut accurately.

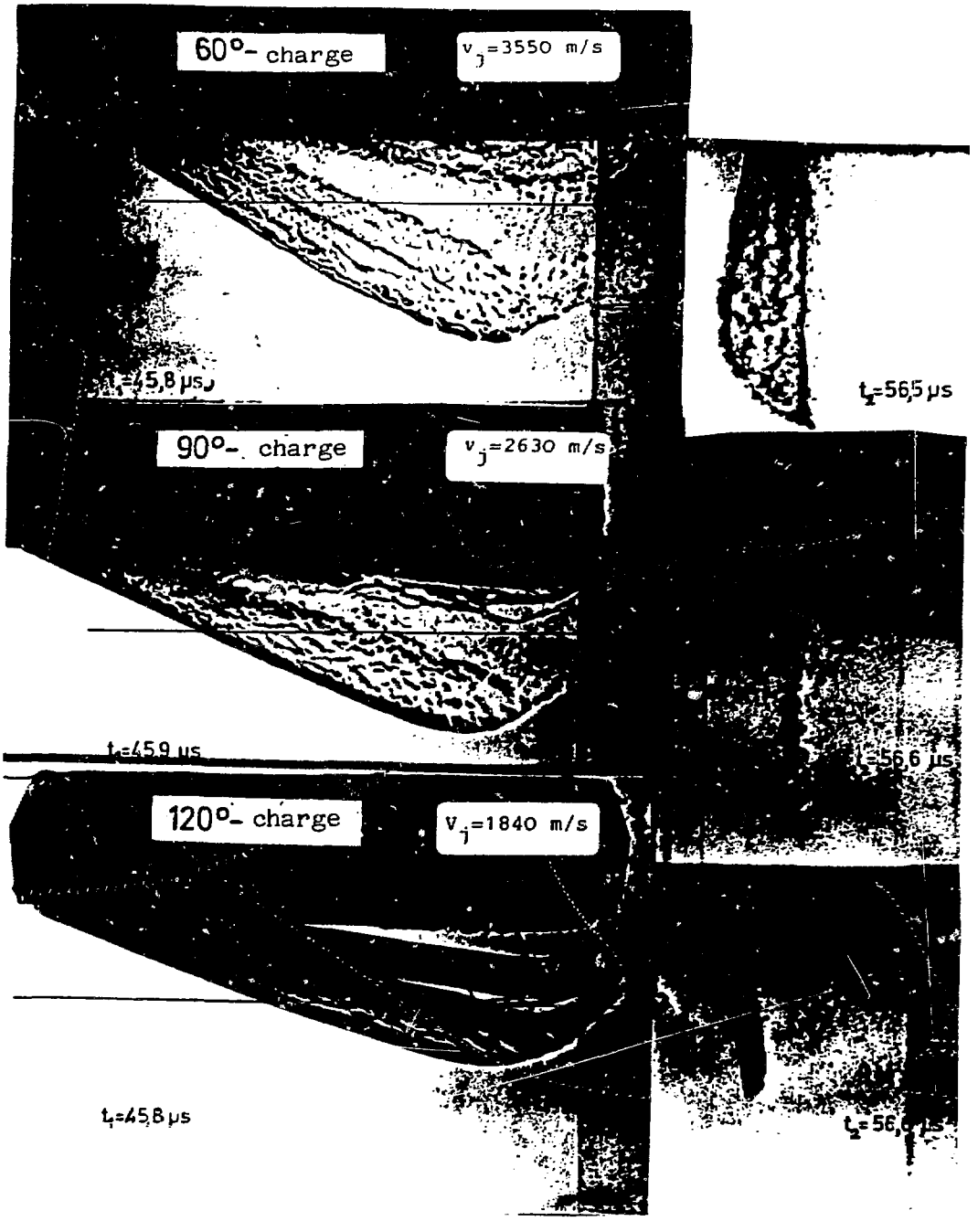

Figure 3: X-ray flash pictures of linear shaped charge projectiles for different opening angles

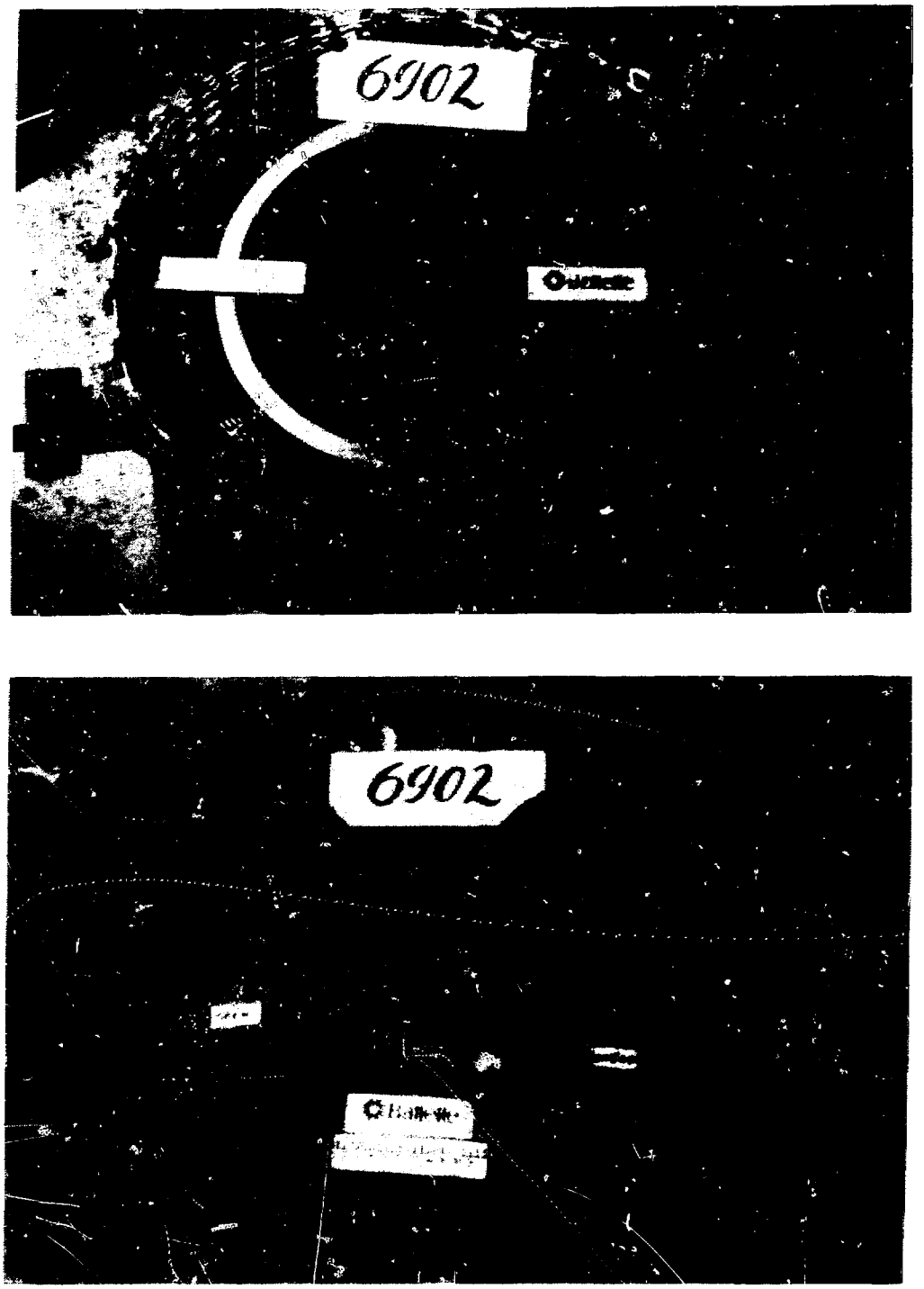

Figure 4: Experimental setup (upper part) and result (lower part) of the trial cutting a 610 mm Ø x 36 mm St52 tube

The measurements of environmental effects like blastwave, impulse load to the tube wall or fragment damage from the confinement, which will be discussed next, lead to the conclusion, that it is possible to carry out pipe cutting trials within the containment of the hot steam reactor HDR under realistic conditions but without high radioactivity in the piping. Two different types of experiments were carried out within the HDR to allow the developed pipe cutting method to be evaluated and to be compared with other techniques.

First, to improve the method, two trials were carried out cutting tube segments of 830 mm length which were mounted from the outside to the concrete structure "Rotunde". The tubes of slightly different dimensions (435 mm, 402 mm diameter and 19 mm, 22 mm wall thickness) had different steel qualitiy (W36-and X10CrNiNb189-steel). Secondly, the final trial was carried out to test the method when cutting a real pipe of the cooling system. The setup is shown in Fig. 5. The W36-steel tube cut had 438 mm diameter and 20 mm wall thickness. To avoid fragment damage it is necessary to catch the fragments of the charge confinement. The results of the trials carried out at the HDR can be summarized as follows: the pipes were cut in the desired way except for the first trial, where one of the four modules was not initiated. This happened because of a detonator misfunction and the fact that in this only case the ring of high explosive around the tube was not closed completely in order to reduce the amount of high explosive.

Environmental Effects of the Method

Fragment protection. Two different types of fragments are obtained depending on the kind of charge used. In the case of the cutting of thin walled tubes with flexible linear shaped charges small and fast fragments are produced. Marker boards were used to detect the effect of these fragments. For the charges used (80 g/m or 250 g/m high explosive load) a fragment size similar to bullets with 2 mm up to 3 mm diameter and a velocity of 680 m/s up to 940 m/s could be determined. In the case of the cutting of thick walled tubes with the developed, strongly confined charge slow and heavy fragments are obtained. Using the x-ray flash picture technique a fragment velocity of 140 m/s was determined. These fragments are steel plates with a mass of approximately 850 g.

The attempts to catch the fragments of the charge confinement using a truck tire or a net built from steel ropes (s. Fig. 4) failed. It was found, that a very heavy protective construction is necessary to avoid fragment damage. Basing on these results and the determined kinetic energy of the fragments of approximately 8.3 kJ, the company Kraftanlagen Heidelberg built the protection construction having the shape of an octagon like the charge. The applied steel/rubber sandwich structure and a total mass of more than 800 kg resulted in a perfect fragment protection during all three trials carried out at the HDR.

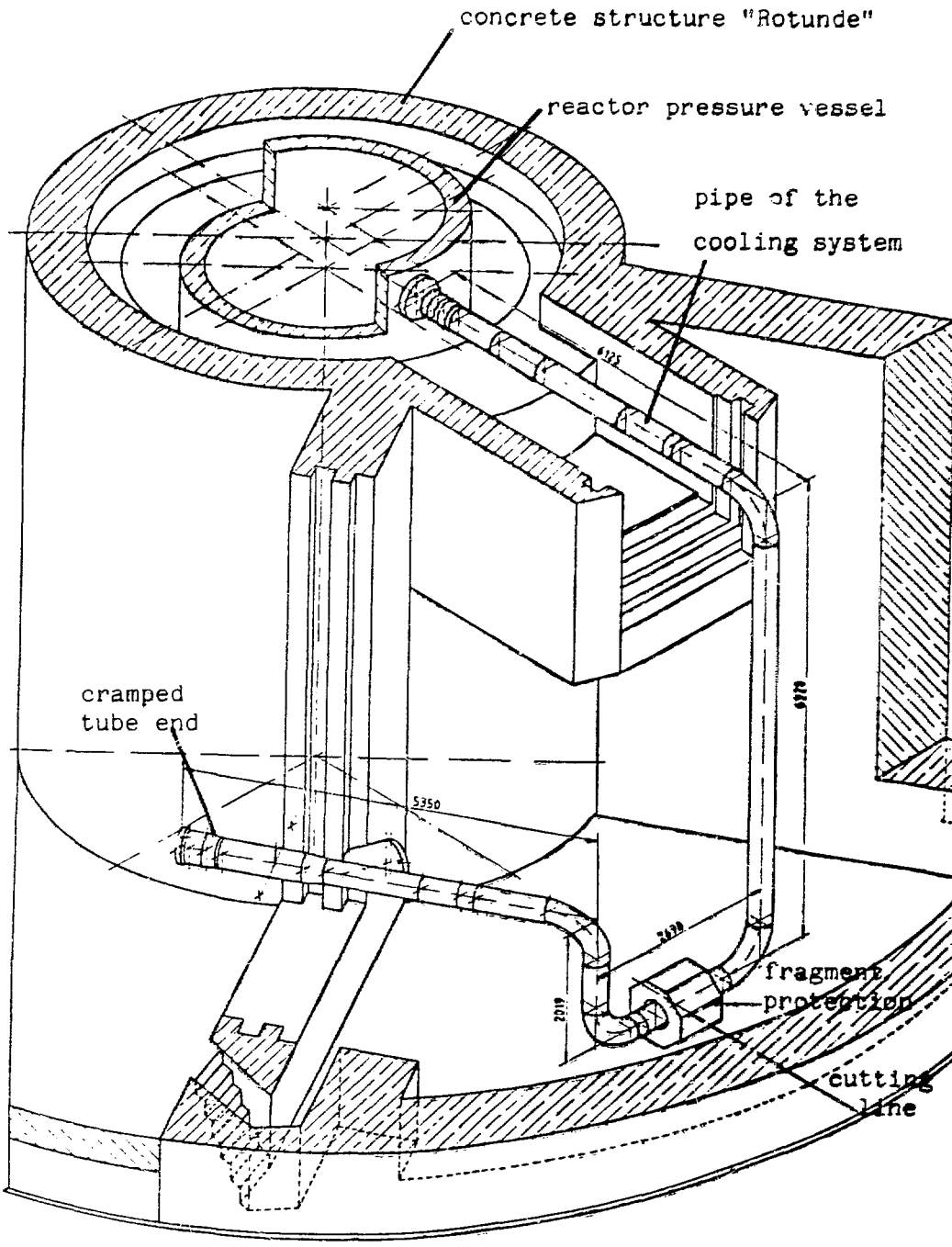

Figure 5: Experimental setup (dimension in mm)

Impact force on the pipe during cutting. With an arrangement similar to the experimental setup used for the test of the method the impulse to the tube wall in axial direction during the separation was determined. Fig. 6 shows the measured forces as a function of the high explosive load necessary to cut tubes of different wall thickness and diameter. Even for the experiment carried out cutting the St52-steel tube the stress outside the cutting region is far below the material strength of steel; the duration of the impulse load is very short, depending on the high explosive load it varies between 0.05 ms and 1.20 ms.

Blast wave effect on environment. The blast wave released is a characteristic feature in every type of explosion. This is characterized by an instant rise in pressure at the wave front and a drop from positive to negative pressures immediately following it. For a large range of scaled distance $Z = R/W^{1/3}$ (distance R scaled with the explosive mass W), the peak overpressure (P) can be expressed by the empirical relation $P = A/Z^2$ /5/. The proportional factor (A) takes the amount of confinement and screening effects e.g. by the tube itself or by the fragment protection into consideration. The propagation of the blastwave overpressure of a spherical, unconfined charge is fairly well described by A = 20 /5/. The comparison of the theoretical predictions with experimental results shows, that the partial screening by the tube and the confinement of the sleeve charge lead to A = 6. A further reduction to A = 2 is introduced by the screening of the very massive fragment protection.

This led to the fact for the cutting trials carried out at the HDR the damage boundary value of 1 bar only was exceeded within a distance of approximately 1 m to the axis of the pipe cut.

Evaluation of Results

The suitability of a cutting charge for the separation of steel tubes can be assessed using the following criteria:

- penetration depth of the projectile
- thickness of fracture zone
- reduction of tube diameter at the location of cutting
- spalling of steel particles at the inner tube wall
- impact force on tube
- effect on surroundings through projection of fragments
- extent of blastwave in the surroundings.

The application of these criteria on the test results obtained here shows that sleeve charges are basically suitable for the separation of steel tubes, e.g. in the nuclear field for the dismantling of power stations. However, this also applies to fields in which access with large tools (e.g. coping saw or circular cutting equipment) is either difficult or not possible at all.

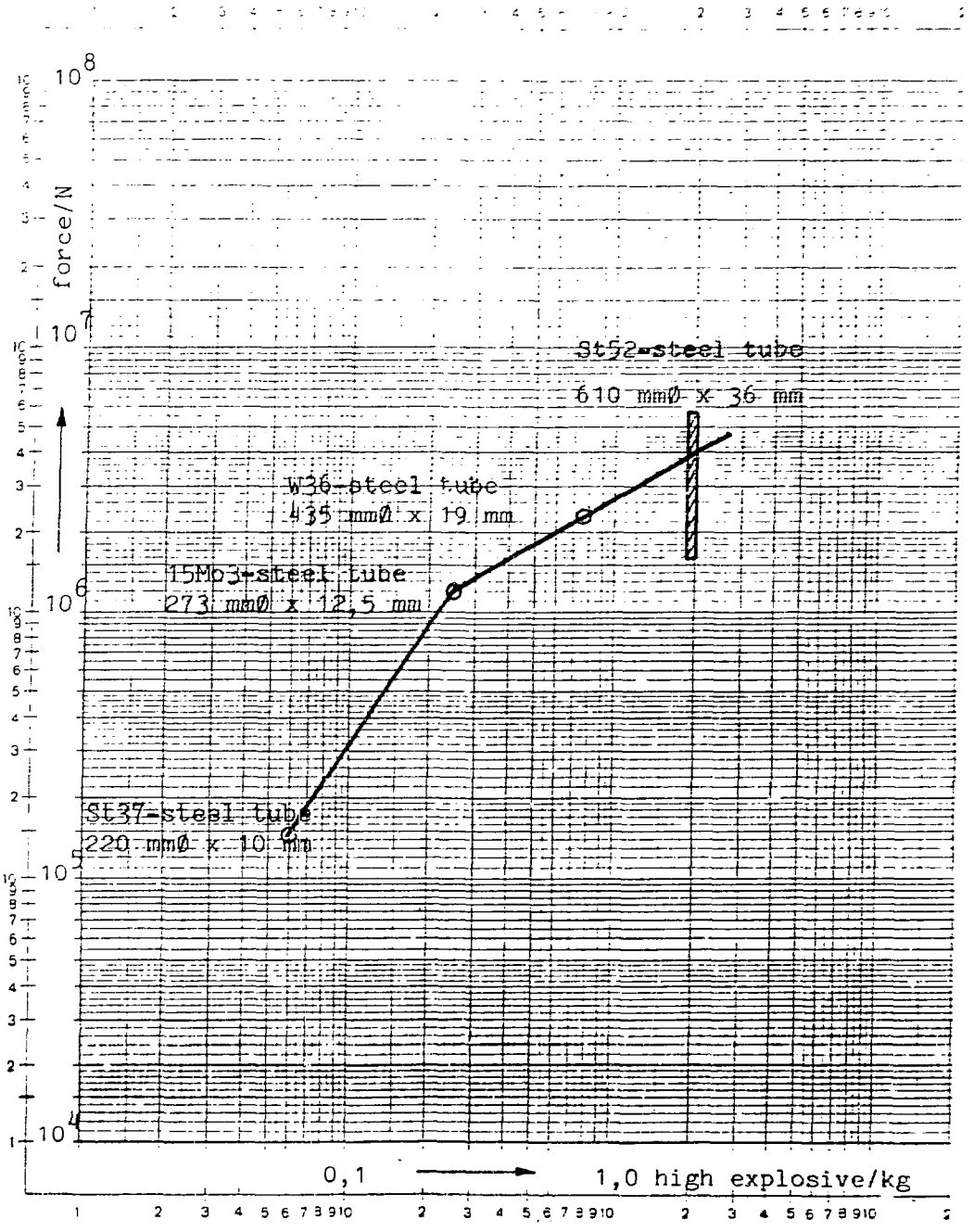

Figure 6: Force to the tube wall for different pipe cutting trials

The effects on the surroundings due to fragmentation and blastwave can largely be reduced by the strong confinement of the charge and the shielding described.

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A STUDY ON TREATMENT OF DUST BY DISMANTLING

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ABSTRACT

In dismantling of nuclear reactors, various kinds of treatment of dust generated by cutting or dimantling concrete structures or components of reactors are evaluated for safety, cost and performance comparing the work in air with in water. A method of dust treatment for work in air is discussed in this study. The dry method has an easy operation in practice and a good performance in the equipment, but has a problem on the prevention from radioactive contamination by diffusion of dust in air. For the purpose of advancing the strong points and eliminating the weak points in dry method, improved venturi scrubber system is proposed for dismantling work as a dust collecting system. The system consists of dust absorbing pipe, dust collector, separator of dust and water and dust transfer equipment to a storage of waste. This system would be expected the better performance and the lower operating cost in decommissioning nuclear reactors, especially, the number of dust filters, for example, HEPA filters, will be considerably saved.

INTRODUCTION

In decommissioning nuclear plants, a great amount of wet and dry dust are generated by dismantling work of metallic or concrete structure of reactors. If water is used in cutting the structure, particles and dust by cutting are mixed with the water, then the dust becomes a slurry. For the separation of water from slurry, an evaporator etc, is employed. But if the concentration of solid particles in slurry is considerably high, the

slurry is directly vetrificated as a waste material. In practice, water is easily spread out and the surroundings is widely contaminated with radioactive dust. So, the decommissioning of nuclear reactors may involve much generation of radioactive waste which should be evacuated in special and suitable containers. For the case of separating water and dust from slurry for disposal of waste, some kind of filter in vacuum and screen system, like a Oliver filter as an example, can not perfectly eliminate dust in separated water. Therefore, a good filteration of water is strongly desired in a radioactive waste treatment facilities unless separarted water from slurry is enclosed by recirculation means. The use of water is not so good means except the decontamination in primary system of nuclear reactors and cutting of high level radioactive materials by access to them.

In general practice of dismantling work , irradiated components and radioactive waste are usually cut and broken in atmospheric enviroment. These dismantling and demolution process produce a large amount of irradiated dust and the most of dust are easily collected as a rule because the large particles of the dust precipitate in the bottom of containers. But fine particles and aerosol in the dust are not so easily collectable. If an intake pipe system of air and generated dust by cutting are provided, those dust and air need to be separated by a separator and prefilteration is required. Generally, HEPA filters are employed for the prefilteration and considerable many number of HEPA filters are expended because the perfomance of HEPA filters drops in a short time due to clogging. Prefiteration system itself have been developed before nuclear age, but a few suitable prefilteration system (1) are provided for decommissioning of nuclear reactors. Of course, those method are possible to be improved for decommissioning of nuclear reactors as well as for other industrial equipments.

Then, at first, a practical dust treatment method is considered in this study. The method has three processes as shown in Fig. 1. The 1st stage has a dust collecting tube, the 2nd has a cyclone separating system

and the 3rd has a filterating system consisting of HEPA filters. In this system, the HEPA filters can not be expected to be saved.

As next consideration, a water spray system is studied. the dust is absorbed with water particles and removed togather with the mist. This system consists of dust collecting system and precipitator by water mist. The mist is poured into a vessel, if it is neccessary to use it and no HEPA filters. According to this system, the number of HEPA filters assembled in the 3rd stage is expected to be considerably decreased because the dust in air before the inlet of HEPA filter is hardly released.

Then, a dust separating system without HEPA filter and prefilteration is considered and studied. That is an improvement and development of venturi scrubber system.

ASSESSMENT OF DISMANTLING TOOLS

Cutting tools for dismantling of nuclear reactors are used in water and air.

Assessment of cutting tools in water

The cutting tools used in water make a lot of cutting trash as slurry state during dimantling work. The dust particles and water need to be separated from the slurry in this disposal.

Equipments or tools used in water consist of sinks, filters, centrifugal arrangements, presses, screws and liquid cyclones etc. Sinks have a large surface area of water in the vessel. Two types of filters are employed in the equipments, one is a periodically pressurized type and the another is a continuously vacuumed type. Anyway, these filters are not exchangeable and not easy handling, and furthermore, the costs are considerably expensive for dismantling of nuclear reactors. Centrifuge consume much power, and it is difficult to move them because of heavy

weight as shown in Fig.2. Presses and screws are also heavy arrangements and not movable. Liquid cyclones and screens are simple mechanism, but those perfomances of separation are not so good. Generally speaking, separation of powderized sclid waste and water causes high pressure loss of fluid flow, and long endurance is desired for the equipments of those disposal in general industry. Then, as explained above, those equipments and tools are heavy and not easy handling as movable tools for dismantling of nuclear reactors in field work.

Easily spreading contamination is most severe problem for cutting tools used in water and the use of water. However, these problems are possibly improved to be compact and easy handling for special conditions.

Assessment of cutting tools in air

Cutting tools used in the air generate fine particles or aerosol which are floating in the air. Generally speaking, these particles are separated from the air in waste disposal and the fine particles are eliminated from the exhaust of air by filters in dismantling of nuclear reactors. Many kinds of devices are employed, for example, cyclones, scrubbers, bagfilters, electrostatic filters, HEPA filters and venturi scrubbers etc.

Low speed type shown in Fig.3, high speed type and multi type are employed in cyclones. Pressure loss of fluid flow is 100 - 200 mmAq at velocity range of 10 - 30 m/s for particles of the diameter of 2 - 10 μm and the dust collecting efficiency is 50 - 99 % as a perfomance of multi type cyclone . High speed cyclones are employed for the collecting of particles of 8 -20 μm in diameter. The cost of cyclones as equipments are not expensive, and they have simple mechanisms and also can be operated with a few troubles. Pressure loss of two phase flow of air and dust in the scrubbers are 100 - 300 mm Aq at velocity of 2 - 30 m/s for particles of the diameter of 0.2 - 2 μm and the collecting efficiency is 50 - 99 %.

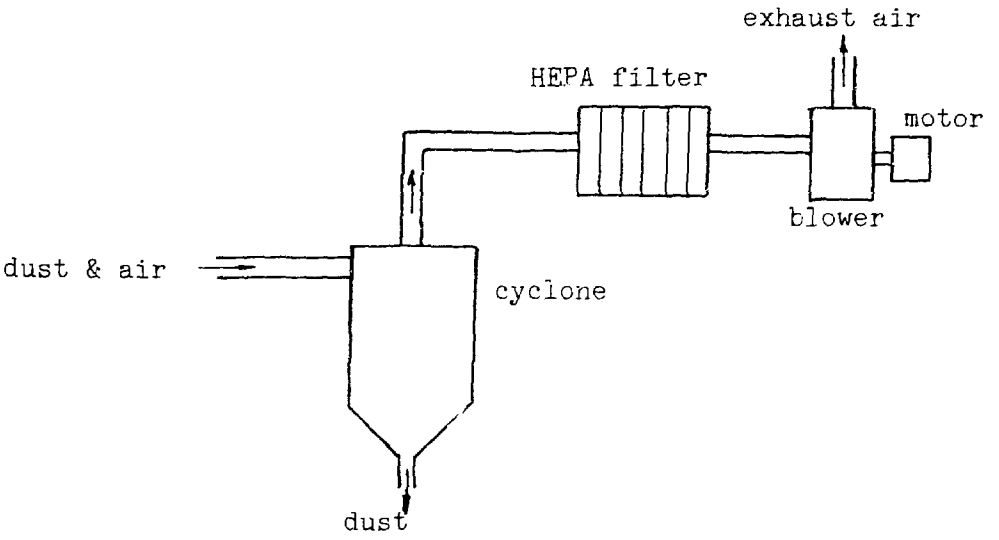


Fig. 1 Dust collecting system

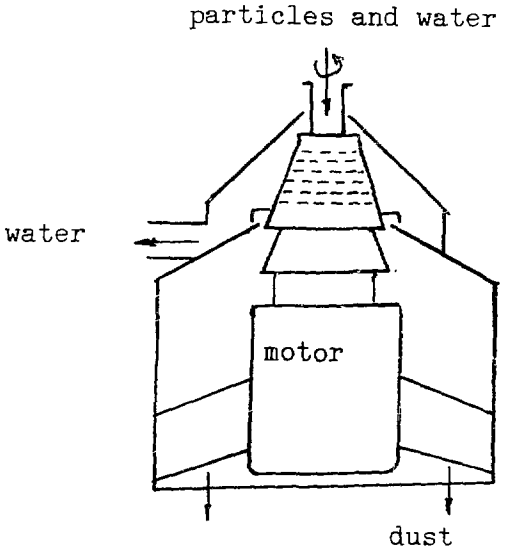


Fig. 2 Centrifuge

Bagfilters , as shown in Fig. 4, are expensive and are not so easy in operatability of equipments, but those equipments have fairly good performances for particles of the diameter of 0.1 - 0.2 μm at the air velocity of 0.01 - 0.05 m/s and the collecting efficiency of 95 - 99 %. Pressure loss of fluid flow through scrubbers are 100 - 300 mmAq. Electrostatic filters, as shown in Fig. 5, must have protecting arrangements for high voltage circuit. They are not so easily movable for safety, and the cost is highest in any other filters, but they have very good performance. The electrostatic filters catch the dust particles of diameter of 0.2 ~0.5 μm at the flow velocity of 1 - 2 m/s and the pressure loss of flow is 20 - 30 mmAq. Air filters and blocks, HEPA filters as an example, have a good perfomances . The diameters of collectable particles are 0.1 - 30 μm at velocity of 0.1 - 3 m/s , and the pressure drop of fluid flow is 10 - 200 mmAq in the collecting efficiency of 90 - 99 %. But the cost is considerably expensive. In case of using HEPA filters as dust collectors, the filters need to be exchanged frequently to keep a good performance and venturi scrubbers have been developed for the purpose of eliminating dust in blast furnace gas. These equipments are not so expensive and suitable for dust collecting system in good performances. The diameters of collectable particles are 0.2 - 0.5 μm at high velocity of 60 - 100 m/s. The pressure drop of fluid flow is as very high as 500 - 2000 mmAq in the collecting efficiency of 90 - 99 %.

Results of assessment of various tools

It is a matter of dismantling of nuclear reactors that the tools used in water bring a wide radioactive contaminatic y spread of water or slurry. It is impossible in practice to enclose completely the radioactive slurry in the dismantling working space because nuclear reactor plants have many various holes or floors at different height. Therefore, it is not so good method to use water for dismantling. However, the wet method , work in water or use of water, is suitable for inside of primary system of light water reactors(LWR) in practice, because water has been already used for LWR. In the case of dismantling in

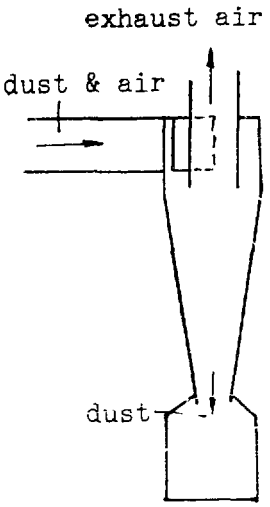


Fig. 3 Cyclone

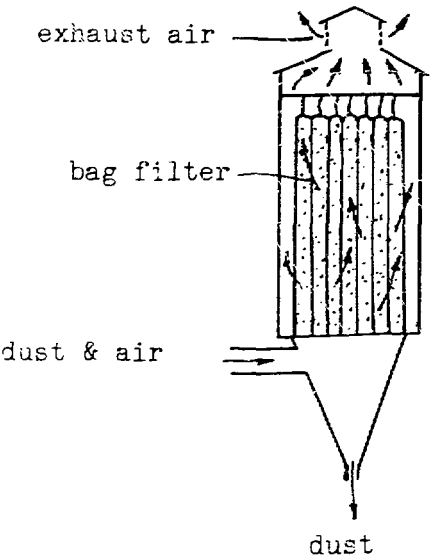


Fig.4 Bagfilter

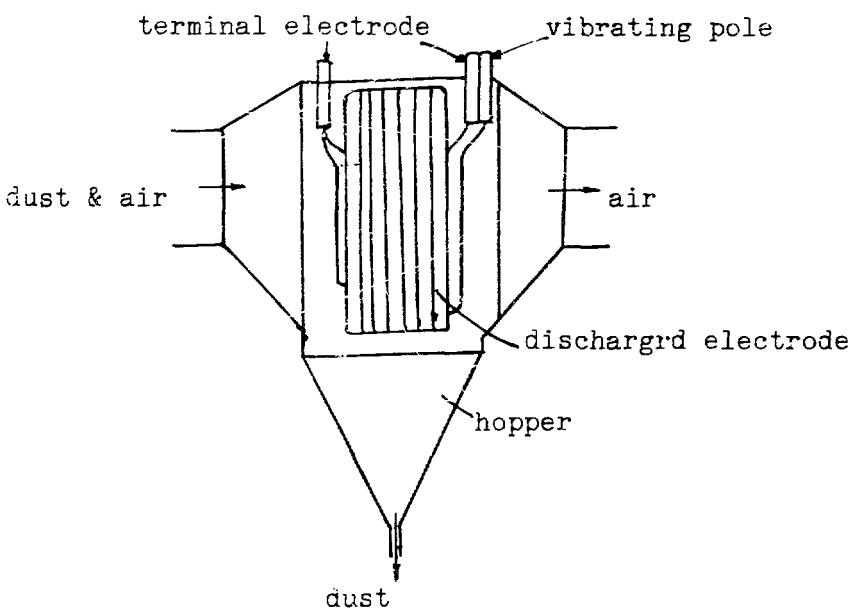


Fig. 5 Electrostatic filter

air, it is considered that the generated dust by cutting the concrete structures and components diffuses widely in air. But if the cutting parts or places are enclosed and filtered, the radioactive dust is prevented from the wide diffusion in air. In this case, if the pressure of inside area is kept little lower than the atmospheric pressure (outside enviroment), it is not neccessary that the enclosure is completely tight for the dust leak. Then, if some good dust collector is employed for dismantling in air, the adoption of dismantling tools in air is exactly a good method.

DUST COLLECTING SYSTEM FOR WORK IN AIR

Requirements of dust collecting system

Dust collecting system for dismantling of nuclear reactors required safety, good performances and low cost. The requirements for the safety are, first of all, no spread of contamination to other spaces and no leak of dust and radioactive materials from vessels or pipes. The requirements for the good performance are easy operation which are compact type or handy type equipments and easy transfer of the equipments for dismantling. The requirements of the low cost mean, of course, that the cost of equipments is low and the maintenance cost is economical. The equipments for dismantling of nuclear reactors in air equip cutting or demolution tools and transfer device which transfers the dismantled components and the dust from the work field to the storage place of dismantling waste. Now in this paper, the dust collecting system is evaluated on the basis of above mentioned assessment of tools for dismantling of nuclear reactors.

Dust collecting system

The dust is separated by the separator as above mentioned manner. Two types of the system are proposed in this study. One is a HEPA filters with a cyclone for prefilteration as shown in Fig.9, another one is an improved venturi scrubber with wet dust separator by vacuum as shown in

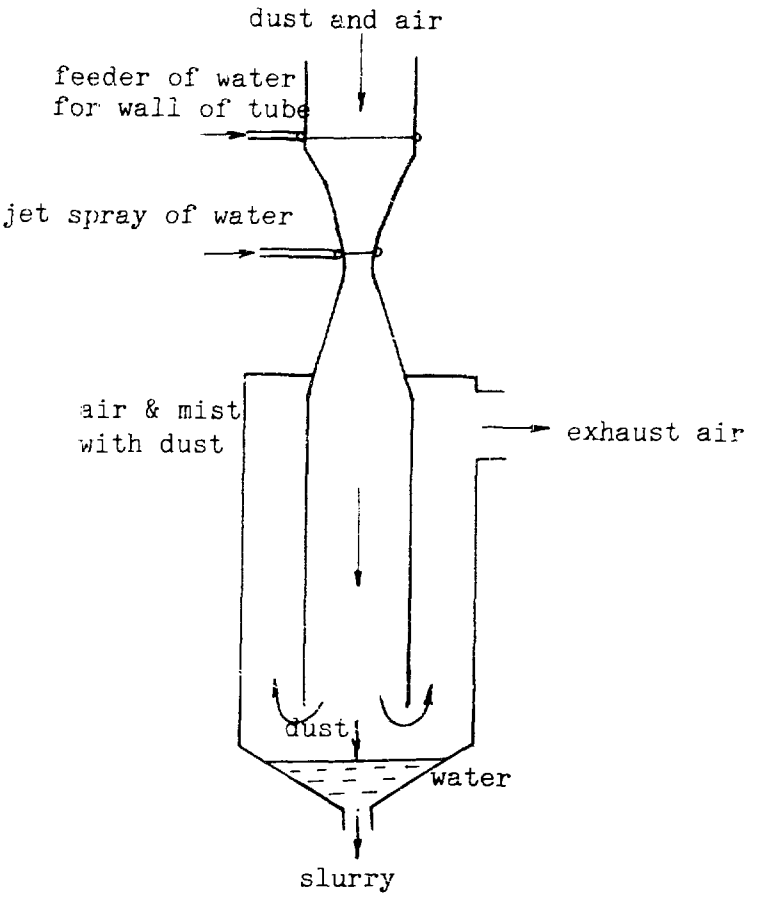


Fig. 6 Venturi scrubber

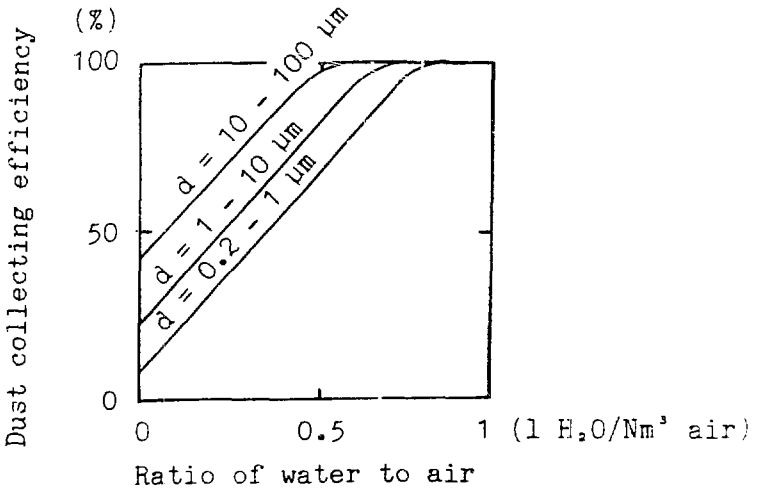


Fig. 7 Performance of a venturi scrubber

Fig.6. If a great many large particles exists in the dust, a cyclone is suitable for the prefilteration because of an economical consideration. Of course, if the dismantling is operated at the inside of a reactor container, operators or workmen will use suitable masks, and HEPA filter is recommended to be used as a similar reason for health physics in usual nuclear reactor operation. Conventional venturi scrubber expends a large amount of water which is exhausted to the outside, but this improved venturi scrubber utilizes an effect of recirculating water which separates the dust in a vacuum filter like a small Oliver filter and the performance of venturi scrubber is shown in Fig.7. The separated wet dust is stripped off from the filter, and the separated water is lifted up to the above mentioned venturi scrubber by multi-stage turbopump (for example of perfomance; 0.4 m /min, 20 - 100 mAq, 5 KW) which is generated vacuume suction (0.05 - 0.5 MPabs.) inside the screen as shown in Fig. 8 for details.

The dust collector system in dismantling equips the dust collecting pipes which intake the dust and carry from cutting work field to the dust collector for packing into drums and the transfer device which transfer the drums to the storage of waste else. In this case, two methods are employed, one is a blower type, and another one is an injector type as shown in Fig. 9. The mechanical efficiency of an injector is not so good but very simple and portable device. The performance and efficiency on an air injector have been investigated by one of the authors (2)(3) and are shown in Fig. 10 as an example of the experimental results. According to the studies, an appropriate injector is possible to be equipped for the collector system.

The improved venturi scrubber

If dismantling dust is vetrificated in waste disposal, wet dust is well disposed by this treatment. The improved venturi scrubber produces slurry with dust and jet spray of water which the many fine water drops catch the dust particles. The size in mean diameters as global

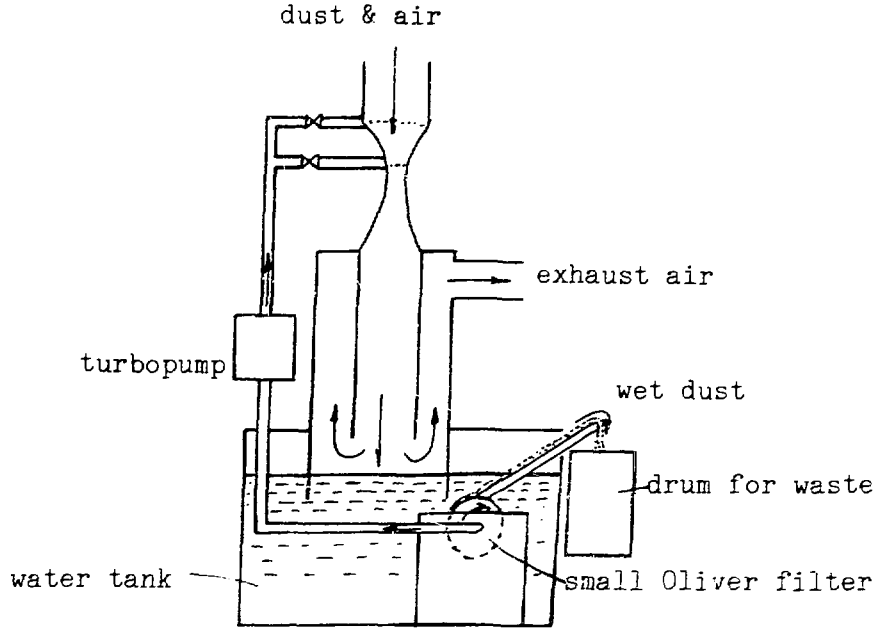


Fig. 8 Separation system of air in venturi scrubber system

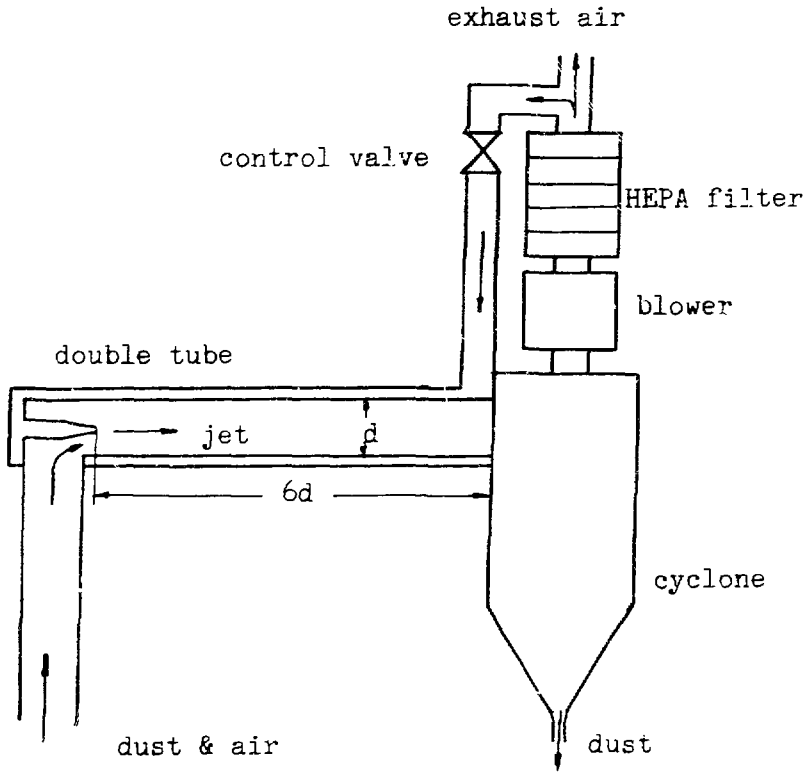
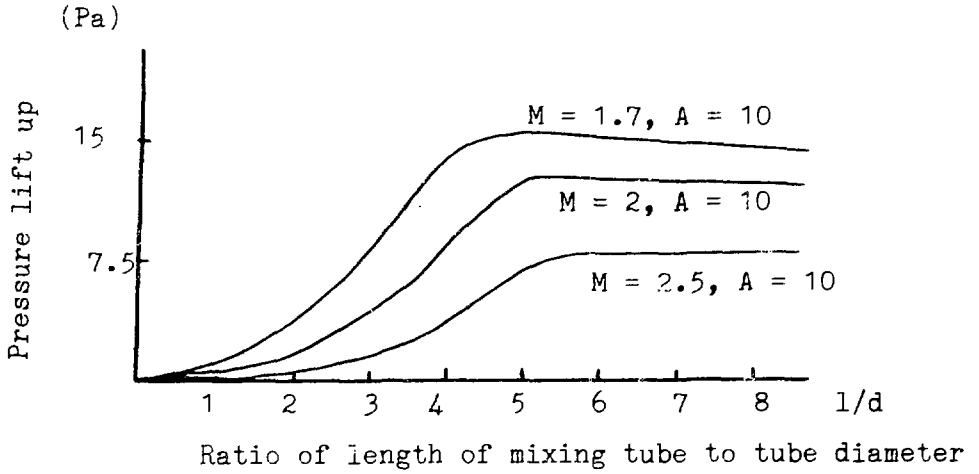
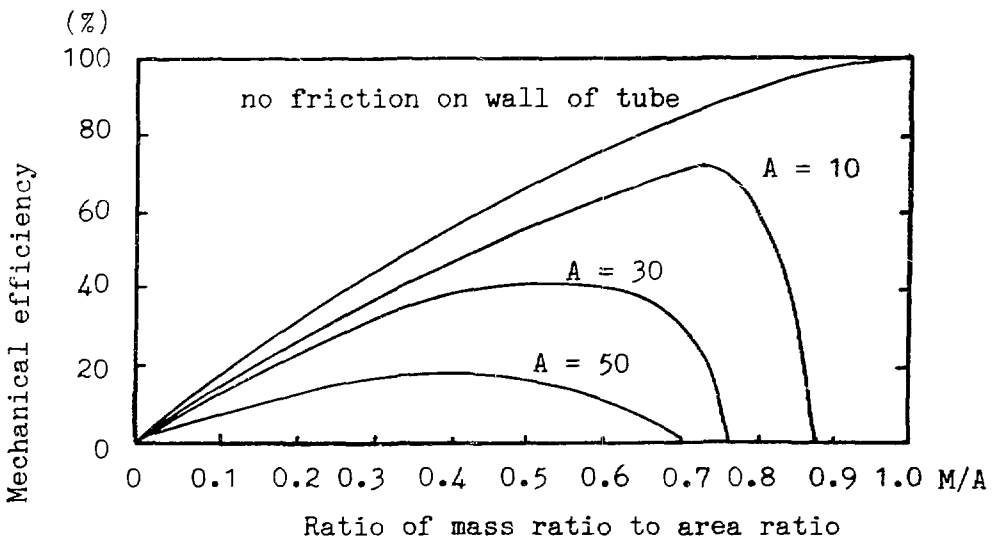


Fig. 9 An injector for cyclone dust collecting system



(1) Pressure lift up in an air injector

$$M = \frac{\text{total flow rate in tube}}{\text{flow rate of jet}} \qquad A = \frac{\text{area of mixinfg pipe}}{\text{area of jet}}$$



(2) Mechanical efficiency in an air injector

Fig. 10 Performance of an air injector

equivalent of almost dust in dismantling is 1 - 10 μm for mechanical cutting machine, 0.1 - 0.5 μm for thermal cutting machine, and the concentration of the dust before dust collector is globally 10^4 - 10^7 particles for unit volume (1 cm^3) in nuclear reactors. The reduction of the dust concentration by venturi scrubbber is 100 - 1000 times of the one that the collecting efficiency is very high as well as HEPA filter. Therefore, the purpose can be attained without HEPA filter for each cutting system by the improved venturi scrubber. HEPA filter has a good performance for collecting of dust, but it must be frequently exchanged because of rapidly clogging by the dust of cutting materials in nuclear reactors. Unless HEPA filter is frequently exchanged in operation, the radioactive waste increases considerably as the result, and many new HEPA filter which is much high in cost must be exchanged frequently. The stripped wet dust from the vacuum filter contains a little water which is more easy vetrification for disposal than the case used HEPA filters waste disposal.

The recirculated water in a diluted slurry increases in concentration, if the volume of water in a venturi scrubber system is constant, but the dust concentration of recirculating water is in steady state because the wet dust is separated and released to the storage of waste. Then a turbopump is different from the conventional one and is the special one for this slurry.

Centerized dust collecter system

The cost for unit volume of the dealing dust in the dust collector and the capacity of transfer equipment of the waste of dust are able to be decreased if the capacity of the dust collector is increased. The reactor components and the concrete structures must be dismantled in many parts for decommissioning of a nuclear reactor. Then, if the dust separation by venturi scrubber system is centerized as shown in Fig.11, many parts of a decommissionig area are possible to be dismantled at the same time. And furthermore, if there are many centers of those venturi scrubber system,

decommisioning of a nuclear reactor will be completed more speedy.

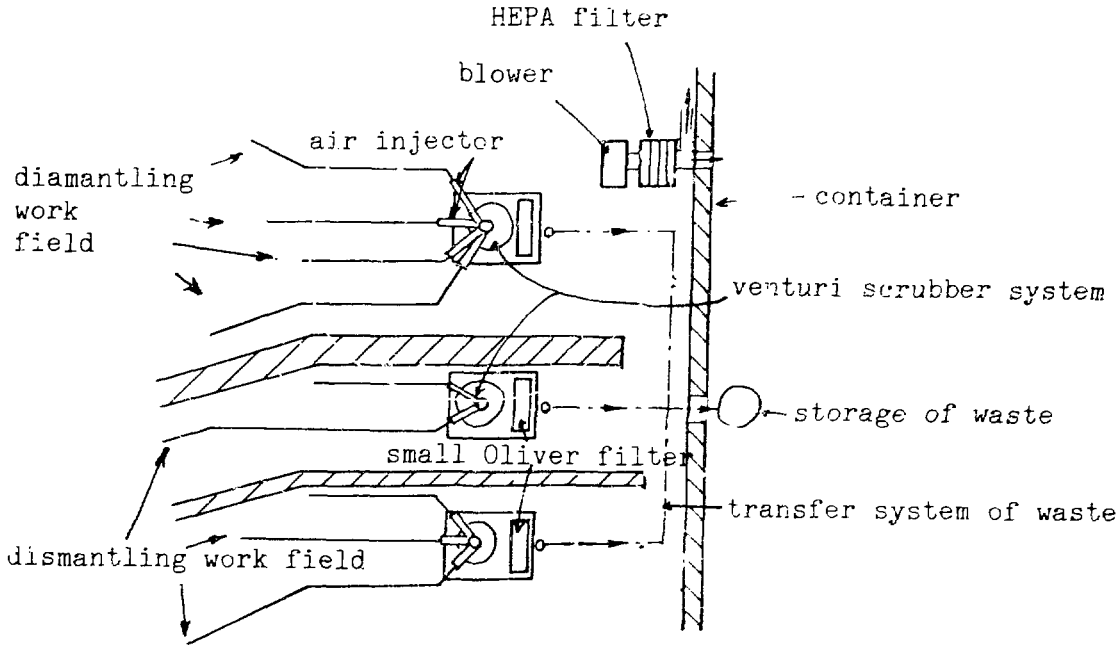


Fig. 11 Centerized dust treatment system

CONCLUSION

Dismantling of nuclear reactors in water and in air are evaluated for safety, performance and cost etc. Generally speaking, dismantling work in air is , anyway, better than dismantling work in water for above mentioned items.

Nevertheless, some problems exist in the treatment of generated dust by cutting of components or concrete structures. And two types of dust collecting systems, they are, a system of HEPA filter with cyclone type prefilter and a system of improved venturi scrubber, are evaluated for performance, operatability and cost, respectively. As a result of the discussion, the dust collecting system of the improved venturi scrubber indicates the similar performance to the dust collecting system of the

HEPA filter with cyclone. And the former system can be operated continuously for a long time without drop of the performance, but the performance of the latter system is gradually decreased because of clogging by dust and it must be exchanged with new one to prevent the increase of radioactive waste . Furthermore, the operating cost of the latter system is considerably high because HEPA filter is very expensive.

We greatly appreciate the Japan Technology Transfer Association for help in technology and the Yamada foundation for help in finance.

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DIAMOND SAWING AND CORING TECHNIQUE FOR BIOLOGICAL SHIELD CONCRETE DISMANTLEMENT

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ABSTRACT

A diamond sawing and coring technique has been developed to dismantle a biological shield in the JPDR decommissioning program. Preliminary cutting tests were conducted using a prototype machine equipped with sawing and coring units to obtain the data concerning the characteristics of the cutting machine as well as to confirm the applicability of this technique for reinforced concrete. Many data including cutting rates under various cutting conditions and characteristics of by-products were obtained in the tests. Based on the results of the cutting tests, the cutting system was designed for safe and efficient dismantlement of the JPDR biological shield.

INTRODUCTION

The biological shield surrounding the reactor pressure vessel is composed of massive and heavily-reinforced concrete and is highly activated by neutron irradiation. By removing an inner activated layer from the biological shield, ordinary dismantling techniques such as a hydraulic nibbler and a breaker could be applicable to the dismantlement of the remaining concrete structure. Japan Atomic Energy Research Institute (JAERI) has developed a diamond sawing and coring technique for dismantling a biological shield in the Japan Power Demonstration Reactor (JPDR) decommissioning program.[1),2)*] The sawing and coring technique is advantageous with regard to remote control of the machine, processing of the by-products and reducing airborne particles compared with the other dismantling techniques such as an abrasive water jet and controlled blasting. The diamond sawing and coring technique, therefore, will be used to dismantle the inner activated layer of the JPDR biological shield to demonstrate safe and efficient performance for dismantlement of the reinforced concrete structure.

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* This work was performed under contract with the Science and Technology Agency in Japan.

Preliminary cutting tests using a prototype machine were conducted in order to obtain data for the biological shield dismantlement machine and system design. The cutting system was then designed for the actual dismantlement of the JPDR biological shield using the data obtained in the cutting tests.

This paper describes the results of the cutting tests and the system design using the diamond sawing and coring technique for dismantlement of the JPDR biological shield.

BIOLOGICAL SHIELD OF THE JPDR

Figure 1 shows a cross-section of the JPDR biological shield including activity levels and dismantling techniques to be applied. As shown in the figure, the biological shield has a cylindrical shape and has an inner diameter of 2.7 to 3.5 m, a wall thickness of 2.5 m maximum and a height of 21 m. The shielding wall is composed of ordinary concrete having a compressive strength of 300 Kgf/cm^2. The concrete is reinforced by 29 mm diameter steel rods on a grid of 15 cm from the region of 8.0 cm from the inner surface and at an elevation between 12.15 and 9.15 m. The inner surface of the biological shield is lined with 13 mm thick carbon steel plate. Cooling pipes, incore monitor tubes and ion chamber guide tubes are contained in the concrete structure.

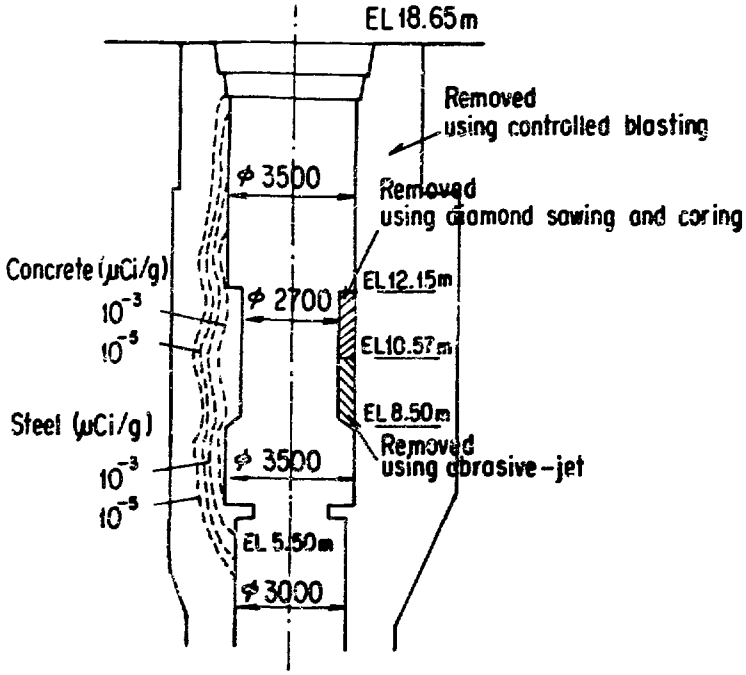


Fig. 1 Cross-Section of the JPDR Biological Shield

CUTTING TESTS BY PROTOTYPE MACHINE

Prototype Cutting Machine

Figures 2 and 3 show the prototype cutting machine and concrete structure used in the cutting tests, respectively. The blades and core bits are tipped with segments manufactured by molding diamond abrasives and metal powders under high temperature and pressure. The diameter and width are 1070 mm and 5.5 mm for the saw blades and 150 mm and 4 mm for the core bits. The length of the core bit is 1100 mm. A special 15 hp. high-frequency motor is used for driving the blade and another 7.5 hp. is used for driving the core bit.

Test Results

Many tests were conducted with different reinforcing steel-concrete ratios and cutting parameters such as cutting direction and flow rate of cooling water. The test results are summarized in Table I. The main results are described as follows.

Table. I Results of the Preliminary Cutting Tests

Test items				Test results
Cutting speed	Sawing			788 rpm
	Coring			436 rpm
Traverse rate	Sawing	Vertical	R/C	212 mm/min (axial direction)
		Horizontal	R/C	74 mm/min (circumferential direction)
			Lined concrete	50 mm/min (circumferential direction)
	Coring		R/C	40 mm/min (axial direction)
Wear rate	Sawing			max. 0.57 mm/m²
	Coring			max. 0.19 mm/m²
Flow rate of cooling water	Sawing			4,000 cm³/min
	Coring			3,000 cm³/min
Diameter of airborne particles	Sawing			max. 16 μm (< 2 μm : 90%)
	Coring			max. 20 μm (< 4 μm : 90%)

R/C : reinforced concrete

Cutting depth is 100 mm for R/C and 75 mm for lined concrete.

Fig. 2 Prototype Cutting Machine

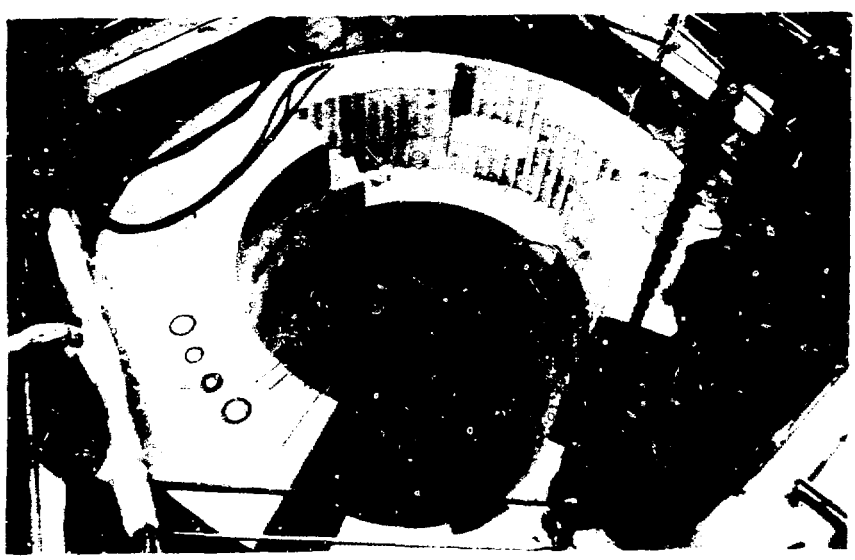
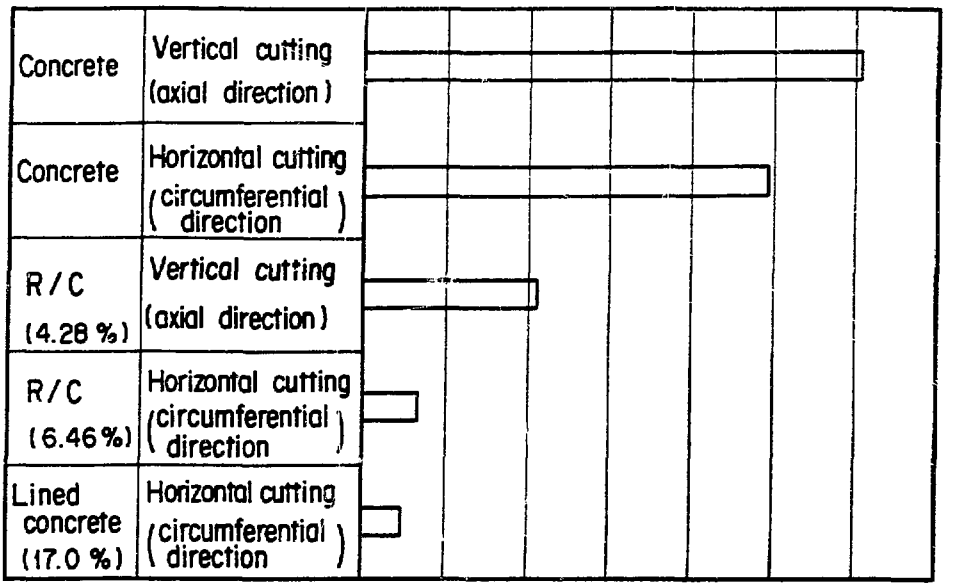


Fig. 3 Concrete Structure Used in the Preliminary Cutting Test

Cutting speed. The average traverse rate for cutting was measured changing cutting directions and the reinforcing steel/concrete ratio with a stable current supply to the motor. Figure 4 shows the measured traverse rates for cutting under various conditions. As shown in the figure, the traverse rate for cutting decreases with an increase in the steel to concrete ratio. Since the blades were moved in both radial and circumferential directions with horizontal cutting of the cylindrical concrete structure, the contact region of the blade with the concrete surface is larger with horizontal cutting than vertical. Therefore the cutting speed was slower with horizontal cutting compared with vertical cutting because of larger friction between the blade and the concrete surface.



R/C : reinforced concrete

Cutting depth is 100 mm for R/C and 75mm for lined concrete.

Fig. 4 Measured Traverse Rate for Various Contained Steel Ratios in the Concrete

Characteristics of by-products. It is important to characterize airborne particles produced by cutting. Airborne particles were sampled close to the material being cut. The particle size distribution was analyzed. The particle diameters were mostly less than 2 μm for coring and 4 μm for sawing, and the maximum diameters of the particles were 16 and 20 μm for sawing and coring, respectively. These data were used for designs of filters in the local ventilation system.

Working time. To evaluate the operation of the cutting and handling machines, the whole dismantling performance was tested by sawing, coring and removing the separated concrete pieces, thus simulating the actual dismantling of the JPDR biological shield. The total sawing area and coring length were 0.8 m^2 and 10 m, respectively. Working time was measured for each activity. The total working time and the ratio of each activity is shown in Fig. 5. Approximately 75 % of the total work time was required for coring and separating and removing concrete cores.

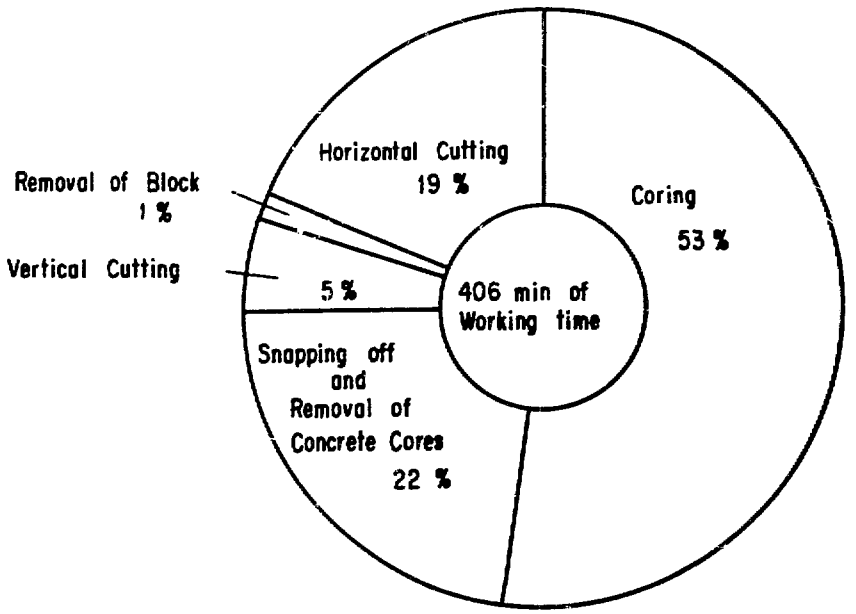


Fig. 5 Working Time Ratio in the Concrete Dismantling Activity

Flow rate of cooling water. Since water used for cooling the blade and core bit is disposed as a radioactive liquid waste, it is desirable to minimize the amount of water used for cooling. The minimum amount of cooling water needed to ensure good cutting ability was evaluated. Though the cutting ability was not affected by a water flow rate of 1 l/min, a slurry was deposited on the concrete surface being sawed. However, the traverse rate for coring decreased to 70 % of the usual value with a flow rate of 1 l/min. Through the study, practical flow rates with sufficient margin for the system design were determined to be 4 and 3 l/min for sawing and coring, respectively.

Wear rate of cutter edge. Wear rates of the blade and core bit were evaluated in the cutting tests. Figure 6 shows measured wear rates as a function of the reinforcing steel/concrete ratio. As shown in the figure, the wear rates of the blade and core bit approximately increase in proporting with an increase in the steel/concrete ratio. Wear rates of 0.57 mm/m^2 for blades and 0.19 mm/m^2 for core bits were selected for fabricating the necessary number of saw blades and core bits. The area capable of being cut by a saw blade and a core bit was estimated to be 12 m^2 and 32 m^2, respectively.

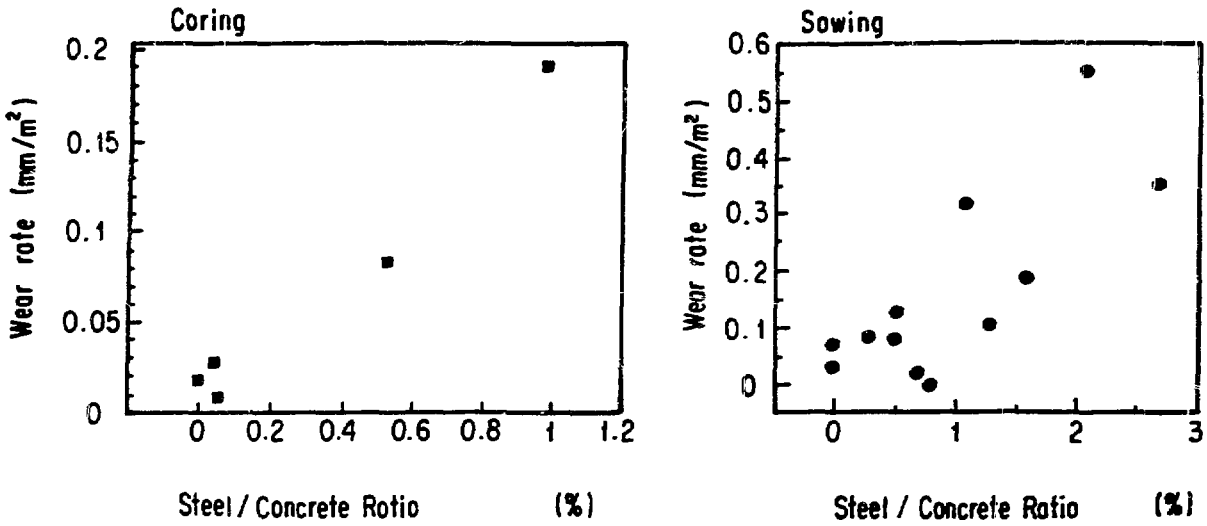


Fig. 6 Measured Wear Rate vs. Steel/Concrete Ratio

DESIGN OF THE JPDR BIOLOGICAL SHIELD DISMANTLEMENT

Concept Of The Dismantling System

The dismantlement of the radioactive biological shield must address the prevention of spread of radioactive contamination and the safety of workers. The dismantling system design carefully considered the experience of the one piece removal of a biological shield in JRR-3 (Japan Research Reactor 3)[3]. Figure 7 shows the equipment and a dismantling flow diagram for the JPDR biological shield. In addition, following concepts will be implemented to minimize worker exposure.

- Cutting and removal will be conducted by remote control.
- A local ventilation system will be used to prevent spread of airborne particles.
- The Dismantling machine is designed to be easily decontaminated.

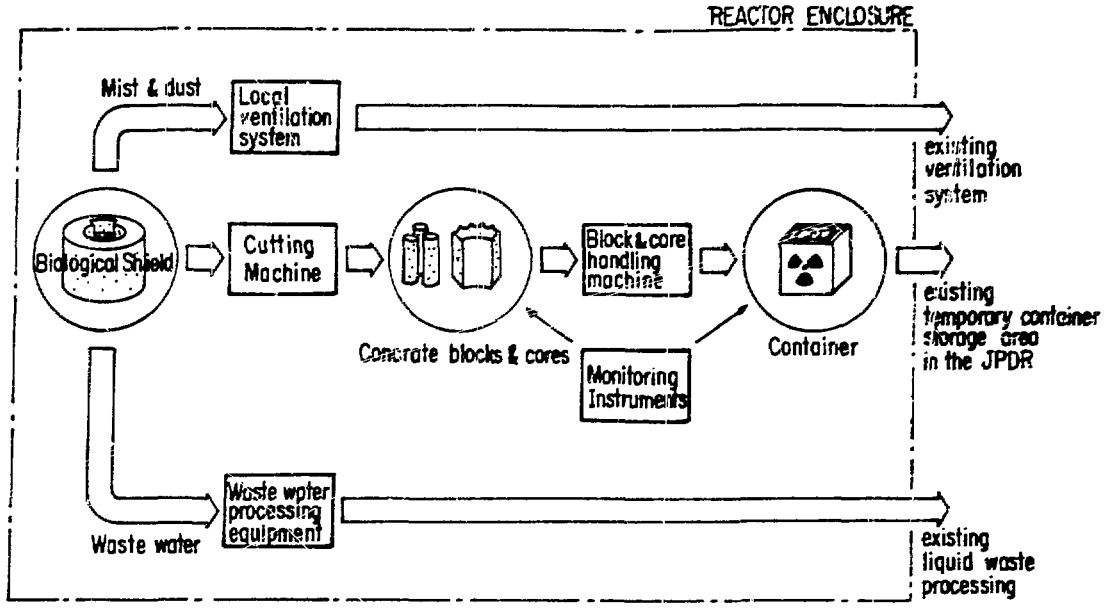


Fig. 7 Flow Diagram for the JPDR Biological Shield Dismantlement

Dismantling System

The dismantling system uses sawing and coring techniques and consists of a cutting machine, a concrete block handling machine, a local ventilation system, water processing equipment, monitoring instruments, etc. Figure 8 shows a schematic of the dismantling system. The components of the system are as follows.

The cutting machine is composed of five main parts as shown in Fig. 9. These are sawing and coring units, actuators for moving the sawing and coring units, a frame for the actuators, blade and core bit grinding unit and outriggers for positioning the frame. The sawing and coring units are installed in the cutting machine facing in opposite directions. This allows the exchange of the units for sawing or coring by rotating the units. This is done remotely without a worker's direct assistance. The cutting machine has mobility in three directions, that is, axial, radial and circumferential, all while confined in a cylindrically shaped space. Table II shows the physical characteristics of the cutting machine. Ball-screws are used in the actuators for axial and radial movements, and a harmonic drive is used for the circumferential movement. These transmissions provide the low machine vibration, precise feed rates and large physical forces necessary to cut reinforced concrete.

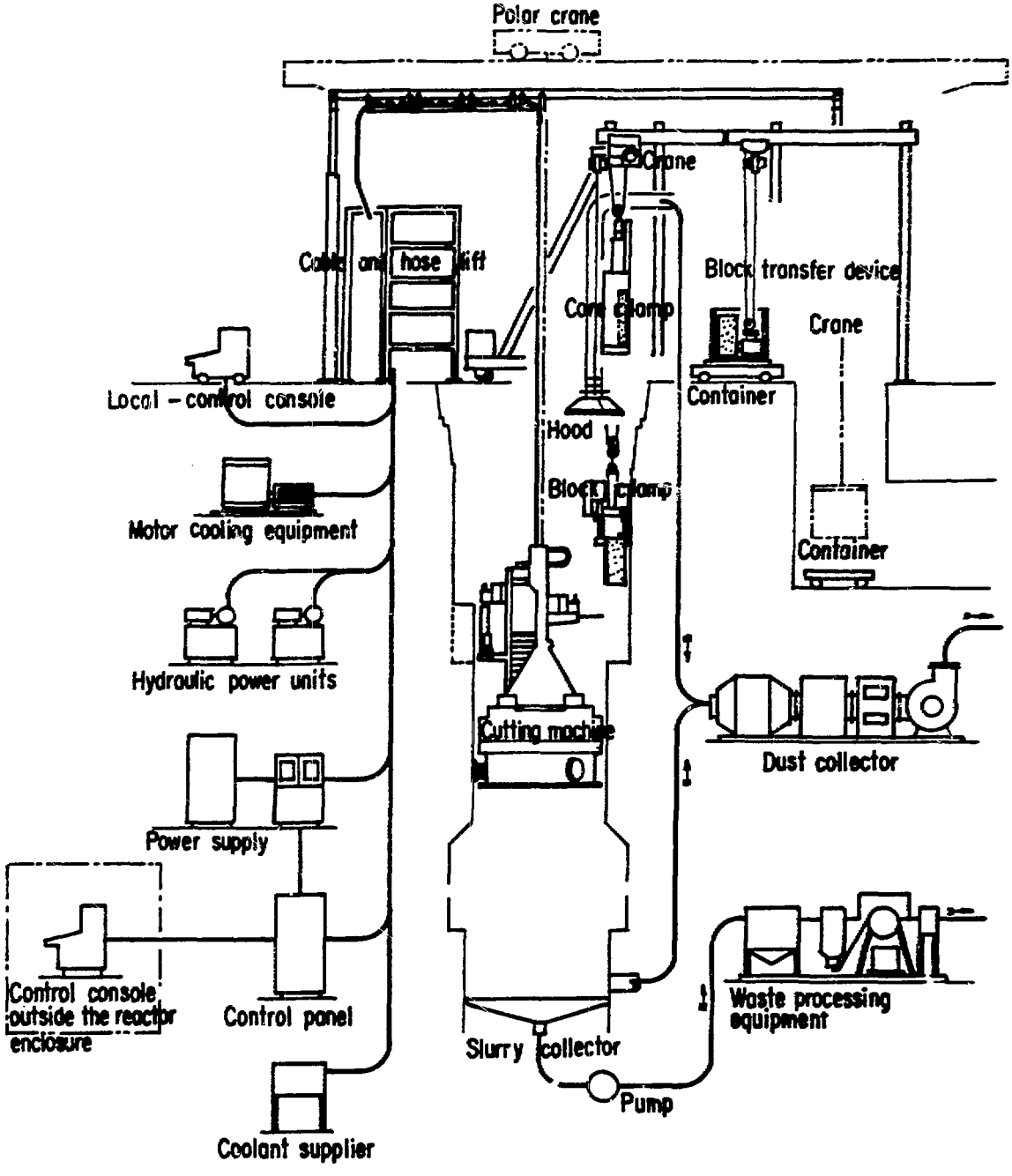


Fig. 8 Schematic of the Dismantling System

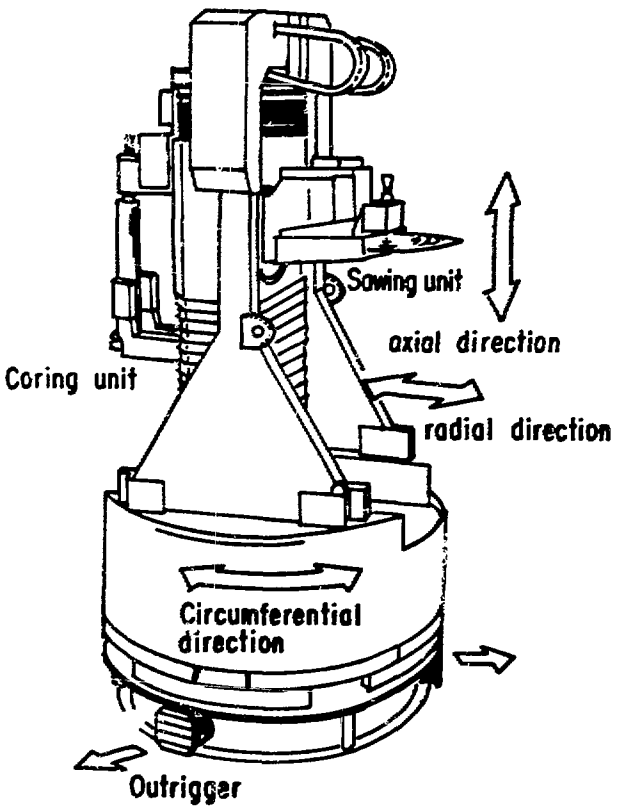


Fig. 9 Cutting Machine for the Dismantlement of the JPDR Biological Shield

Once separated, concrete pieces are handled by one of two clamps, one for prismatoidal and another for columnar shapes. The clamps have the ability to hold a 1200 kg prismatoidal shaped concrete piece or a 50 kg columnar shaped piece even if the power supply fails. The local ventilation system, composed of a hood and a dust collector, collects mist and dust close to the cutting machine. The local ventilation system reduces the load on the existing ventilation system as well as keeping the working area clean. The air drawn into the hood is exhausted through the dust collector to an existing duct of the building ventilation system.

Table II Physical Characteristics of the Cutting Machine for the Dismantlement of the JPDR Biological Shield

Height		5.0 m
Outer diameter		2.55 m
Weight		15 ton
Cutting load		300 kg
Number of outriggers		3
Mobility	axial	1050 mm
	radial	945mm~-485 mm
	circumferential	380

Dismantling Procedure

The JPDR biological shield will be dismantled after removing the reactor pressure vessel and all pipes and components from the cavity. Preparation work will be conducted to prevent the spread of contamination during the dismantlement of the biological shield concrete as follows.

- Openings in the biological shield will be closed by attaching steel plates to cover the openings.

- The pipes in the biological shield will be filled with grout.

- A slurry collector will be installed in the bottom of the cavity.

The radioactive biological shield concrete is then dismantled by repeating the following steps:

- The cutting machine, suspended from the polar crane is lowered into the cavity of the biological shield. It is then rigidly positioned against the wall with its outriggers.

- A portion of the concrete structure is then dismantled by following sequence: circumferential sawing, coring, removing the concrete cores, then vertical sawing as shown in Fig. 10. The prismatoidal shaped piece is then removed and placed in a container.

- The radiation level of container surface is monitored.

- The containers are shipped out of the reactor building.

These activities are repeated and conducted continuously. Approximately 15 tons of concrete will be dismantled by sawing and coring during the 3 month operation.

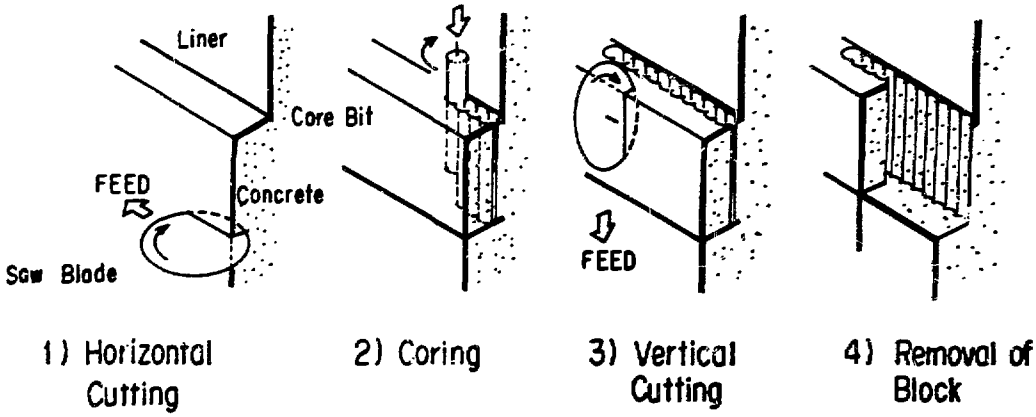


Fig. 10 Concrete Cutting and Removal Sequence

CONCLUSIONS

JAERI has developed a diamond sawing and coring technique for dismantling the inner layer of the biological shield in the JPDR decommissioning program. Preliminary cutting tests by a prototype machine showed that the diamond sawing and coring technique is applicable to dismantling the reinforced concrete structure. Cutting speed and working time were evaluated under various cutting conditions. The wear rates of blades and core bits and characteristics of the by-products were also evaluated to design the dismantling system for the JPDR biological shield. Based on the results obtained in the tests, the actual cutting system was designed and fabricated. It is expected that this system will provide safe and efficient dismantlement of the JPDR biological shield.

The actual dismantlement of the JPDR biological shield will be conducted in 1989 after confirming the applicability of this technique by a mock-up cutting tests.

ACKNOWLEDGMENT

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POSTER SESSION

PROJECT PLANNING, MANAGEMENT, AND COST ESTIMATING

RADIOLOGICAL CHARACTERIZATION - STRATEGY TO AVOID SURPRISE

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ABSTRACT

Radiological characterizations are designed to gather information about the current status of a property, or site. Site, as used here, represents a parcel of open land, or one with improvements. This paper addresses radiological characterizations, specifically those which are a part of major remedial action programs. The principal objective of this work is to determine the distribution of contaminants within and around a site boundary. Results may be required as input for engineering assessments and for remedial action design. Inadequately planned and executed characterizations can lead to inacurate estimates of waste volumes, to expanded field measurements, and to delays in construction. Since funding for all remedial activities is shared, there is a strong incentive to make efficient use of available resources. TMA/Eberline Analytical has over 15 years experience in providing radiological support for remedial actions including the clean up of nuclear weapon test areas, uranium processing facilities, and other radiologically contaminated structures. The authors believe that careful consideration of the discussions presented below will help to minimize surprises once remedial action begins.

INTRODUCTION

In the organizational phases of remedial action programs, preliminary studies were carried out to define an overall program scope and to establish minimum funding requirements. These studies included an evaluation of the history of sites anticipated to be included in the program. Once this was done, sites were categorized, and visits were made in order to determine each site's current use and make exploratory radiation measurements. Based on findings of these visits, further studies were performed. These had a goal of determining whether existing radioactivity could be controlled through institutional arrangements. In many cases, it appeared that an engineering solution would be required to stabilize or otherwise control the spread of radioactivity. Sites were then designated as candidates for remedial action, and plans were made to implement the program.

The development of alternatives for remedial action can only be based on knowledge of the distribution of radioactivity on an individual site. Therefore, site designation established the need for a comprehensive characterization. Remedial action at some sites will include the stabilization or cleanup of potentially hazardous chemicals as well as radioactivity. While this paper does not address chemical characterization activities per se, it is felt that techniques discussed here will be helpful in both cases.

Site investigations are designed around a series of measurements and sample analyses which should describe the distribution of contaminants within a defined area of concern. Small sites with minor previous activity may be characterized in accordance with a generic plan[1]. Work under this type plan includes in-situ radiation measurements at regularly spaced intervals, and the analysis of water, soil, and sediment samples. A comprehensive site specific plan[2] is needed, however, for the characterization of sites used to house production facilities or for the long term storage of material such as process residues. In either case, it is essential that characterization plans be based on known or obtainable information regarding the site. To accomplish this, one must review a variety of information sources including historical site files and reports, observations of current physical features, results of exploratory measurements (from preliminary visits), and interviews with operational personnel and owners of vicinity properties. Even so, characterizations are sometimes foiled by past events and practices which have been forgotton or which were not well documented. It is this type situation which can lead to costly expanded characterization activity and delays in planned remedial action. The most time consuming and costly situations are those requiring the collection of numerous soil, water, and sediment samples which must be analyzed using radiochemical techniques not normally available in a field laboratory, and the drilling of boreholes in addition to those originally planned. Others include the re-analysis of samples for unanticipated radionuclides, and addition in-situ radiation measurements in areas beyond those included in original survey boundaries.

Where surprises have occurred, they have done so in spite of plans which were believed to be adequate. Characterization planning is a technology which, as in other cases, improves with expanded experience. Methods which may be utilized to reduce the occurrence of unanticipated situations will be introduced in the following sections.

OBSERVED SURPRISES

Project Managers lead an uneasy life. The last words they wish to hear from the field are "guess what we found", or "you need to sit down for this one". On several occasions, the need for expanded characterization activity was discovered during remediation. Cost impacts are not limited to those associated with additional measurements and sample analyses. For remedial action in progress, additional costs result from construction delays and re-work. This is particularly troublesome if cleaned and backfilled areas must be re-excavated or the boundaries of construction must be enlarged. The result of this type situation is the substantial cost associated with handling, temporary storage and ultimate disposal of an "extra" volume of contaminated soil. Examples of unusual developments which have resulted in additional work include those presented in Table I.

TABLE I

EXAMPLES OF UNUSUAL OCCURRENCES

- o Radionuclides with weakly penetrating radiation, and which escaped detection during preliminary surveys (ie: Th-230).
- o Unknown haul routes for material when moved between sites.
- o Unknown improvements of haul routes included in characterization or remedial action.
- o Unknown utility construction during periods of inactivity at an individual site.
- o Use of soils containing radioactivity around and within sewers and other underground utilities.
- o Unsuspected deep deposits of contaminated material in areas thought to be devoid of previous site activity.
- o Discovery of materials containing radioactivity not associated with site operations (for example, slag from the processing of phosphate rock and fly ash).
- o Failure to include radionuclides required under state regulations.
- o Radioactivity in areas outside planned survey boundaries, and inhomogeniety of on-site radioactivity.

Other investigators have profited from their experiences. A useful and detailed discussion of "lessions learned" during characterization and cleanup is given by Hynveen, et al[3].

RADIOLOGICAL CHARACTERIZATION PLAN DEVELOPMENT

Radiological characterizations require guidance from a well organized work plan. This document must lead the investigators through a series of tasks to determine the distribution of radioactivity in open land and on buildings and equipment. Use of a generic checklist is very helpful in developing a plan. Items in the checklist should be inclusive of all potential characterization activities. As one works through the document, decision points will permit the elimination of certain activities based on previous site operations. Preparation of the work plan should not begin until there is a clear understanding of what is to be accomplished. Meetings with project managers, engineers, and construction managers are the best medium for this task. Although the initial scope of a characterization will depend on previously forcasted funds, the final scope of work and associated cost estimates will result from the planning process itself. To be successful, the plan must include a statement of objective and technical approach, a discussion of the probable type and location of radioactivity to be encountered, instrumentation to be used, a description of each element of field activity, a discussion of standards, guidelines, and regulations, a description of Quality Assurance program, and a description of data evaluation techniques. A copy of the completed plan development checklist should be presented in an appendix.

Review of Historical Information

In this step of the planning process, it is desirable to review documents concerning the subject facility itself. Other items needed for review include a description of processes involved, types of feed material and product, time periods for operations, accounts of accidents, spills or other unusual releases of radioactivity, descriptions of liquid and solid waste streams, type and quantity of materials handled and shipped, etc. This material is not always readily available. In some cases, documents have been retired to federal repositories, or destroyed.

When an entire facility is the subject of characterization (such as a uranium mill or refinery), it is helpful to review information which pre-dates construction. This is especially useful in determining drainage patterns, areas which may have been filled to accomodate new construction, specific geological features, previous land use, and pre-existing utilities. To join this information to the current site, it is necessary to study as-built drawings and construction completion reports to learn to what extent the existing property was altered and to predict where radioactivity will most likely be found. Such detail is less important for the case where only a small portion of a facility was

used to process radioactive material. It is desirable, however to review facility drawings to determine the extent of overall facility involvement.

Prior to the actual collection of documents, it is recommended that literature searches be conducted using information stored in computer data bases. A listing of valuable reference material (authors, titles, and summaries) can be obtained. There are several data bases available for this type literature search. Representatives from the Environmental Protection Agency, the Department of Energy, the Nuclear Regulatory Commission, and the Department of the Interior (Geological Survey) can provide information on firms with search capability. Once data base searches have been completed, references should be placed in categories, and prioritized according to ease of availability.

Maps and Reports. Very useful initial information can be found on USGS quadrangle maps and in Geological Survey reports. USGS maps dating from pre-construction to the present provide information on land use growth over the years. These maps also reveal changes in roads and highways, ponds, lakes, streams, and rivers. Geologic reports are useful in providing information on the geologic strata (glacial till, clay, sand, gravel, and bedrock) and groundwater. City and county tax maps may be used to determine land use patterns in the immediate vicinity of the facility. All of this type information is needed to assess potential pathways for radiation exposure.

Aerial Photographs. Aerial photographs dating to the 1930's are available for most areas of the country. For example, the Department of Agriculture maintains files of photos for all rural areas. These were taken at intervals ranging from 5 to 7 years, and may be obtained in the form of copy negatives, contact prints (in stereo pairs or triplicates), and in enlargements. Photos are also available from the USGS, from military records through the National Archive in Washington, D.C., and from EG&G, Inc. which operates an aerial radiation measurements service for the Department of Energy. The authors have made beneficial use of aerial photos in the development of characterization plans, and as an aid in explaining data conflicts. In the late 1970's, a site in New Jersey was surveyed and found to contain radioactivity above guideline values in two small areas. Because of unexpected site development by the owner, cleanup was initiated. During the project, uranium contamination was found on a portion of the site several hundred feet from the remedial action and in an area not thought to have been involved in the original operation. After a diligent search, a view of the site (US Army, circa 1943) was located by National Archive staff. Six men who worked on the site in 1943 viewed the photograph. Each of them pointed to a single building as their work location, which corresponded to the recently identified contamination. The construction contract was modified and this much larger volume of contaminated soil was also removed. Periodic aerial photos have also been used to identify re-routed major stream beds in areas likely to have received radioactive effluent, the location of sink holes, the status of original

construction of at least two candidate remedial action sites, patterns of wind blown uranium mill tailings, patterns of erosion around site boundaries, the location of former landfill areas and waste water storage ponds within facilities, the location of formerly used buildings now demolished, the location of utilities (sewers and other underground piping) and former roadways, and the original boundaries of stored residues now relocated. One can see from this brief summary that aerial photos are a valuable aid in planning the extent of a field measurements and sampling program.

Site Files and Drawings. Information in this category is a must for review because it tells the site story. Original as-built drawings (if available) indicate where buildings are located, depth and size of footings for foundations and piers, the location, type, and size of underground piping, the location of ponds and pits, the location and size of air exhaust systems including filters or bag houses, as well as details of general construction. Information in site files can be used to identify the type and quantity of major radioactive feed materials, to reconstruct flow diagrams for radioactive material, and to reconstruct material balance diagrams for chemicals used to process radioactive materials. These files can also yield information concerning waste generation, treatment, storage, and disposal, health and safety guidelines in effect during the period of operation, details of major spills or other releases of radioactivity to the environment, and details of site improvements and reconstruction or renovation. One valuable document is a facility construction completion report. It provides a narrative for as-built drawings and helps to integrate other bits of site information. Measures which were taken to reduce construction costs are usually found in this type report. For example, equipment from other plants could have been substituted for new items. If these items had been used to process radioactive materials different from those processed at the current site, additional radionuclides may be present. Other documents to be reviewed include weekly reports from departments with responsibility for material processing and control, periodic reports to funding agencies (if federally owned), annual progress reports, annual environmental reports, and reports to regulatory bodies. These provide information on operational trends, site improvements, time periods for processing each type of radioactive material, the frequency for monitoring environmental radioactivity, and information related to licenses and permits. On-site meteorological data (if available) will permit an estimate of the probable path for releases of radioactivity. Some information may not be found in site documents. For example, feed material for uranium refineries was delivered as uranium concentrate from uranium mills. Specifications for this product[4] required at least 75 percent U_3O_8 along with allowable percentages of impurities. Most of the uranium concentrate contained Th-230 in concentrations from 5 to 7 percent of the uranium activity. The actual significance of this impurity has only recently been realized when areas thought to be contaminated with Ra-226 were found to contain Th-230 at concentrations of several hundred Bq/g. Other information

which may be missing from site files is the location of sanitary landfills which could have received materials containing radioactivity in excess of current guidelines.

Standards and Regulatory Guidelines

In developing the work plan, careful consideration must be given to an evaluation of local, state, and federal standards and guidelines for limits of radioactivity on surfaces and in environmental media such as air, water, soil, sediment, and vegetation. Documents which contain these standards and guides are basic references for the work plan, and serve as an aid in developing elements of the field investigation. Interface with representatives of government agencies will assure that characterization activities are planned in accordance with the latest version of standards and guides. These meetings will provide an opportunity for regulatory personnel to explain requirements under existing regulations and to suggest a schedule for visits during site work.

Observations During Site Visits

Upon completing the review of historical information, prepare a detailed checklist for tasks to be carried out during a visit to the site. This document should include a list of individual tasks, assigned responsibilities, and a schedule. A selection of the more pertinent site related documents should be available. Of particular importance are maps and photographs, drawings, and reports which describe processes and facilities. Video and still cameras, and audio tape recorders should be used to document observations for future reference. Typical of tasks which might be carried out during the visit are presented in Table II. All items on the checklist should be completed before leaving the site. Conduct a short meeting of the participants to verify completion of tasks and review the findings. Assemble all relevant bits and pieces of information in accordance with sections of the characterization plan where it will be utilized. During preparation of the plan, use this information as a tool in refining the locations for measurements and sample collection.

Grid System

Planned measurements and sampling on open land should correspond to locations which are positioned on a site grid system. This normally consists of a series of mutually perpendicular lines placed at uniform intervals and tied to the state grid plane. For simplicity, the grid system can be oriented according to the site layout so long as it is reproducible. Remedial action design drawings will be related to this grid. In order to avoid excavation of areas which are not contaminated, it is important that this grid system be constructed and documented according to ASTM standards for land survey. Characterization activity inside and on buildings must also be tied to a reproducible grid system, however installation of this grid is normally done by the

TABLE II

TASKS TO BE CONDUCTED DURING A SITE VISIT

- o A general tour of the grounds and buildings to compare the current status with historical information.
- o An inspection of site drainages including ditches, swales, and process and storm sewers (to determine points of access for measurement and sampling activities).
- o An inspection of process building floors to locate sumps, hold-up tanks, and other areas where radioactive liquids could have migrated to areas beneath the floor.
- o An inspection of facilities used to load and unload radioactive material.
- o An inspection of process equipment and of air exhaust and filter systems.
- o An investigation of equipment used for waste handling, transfer and storage.
- o Perform exploratory radiation measurements to determine general levels of total and transferable alpha and beta radiation, determine the presence of weak beta radiation, and determine a range of gamma radiation levels expected during the characterization.
- o Collect selected soil/sediment and water samples to identify radionuclides present.
- o Determine presence of radon and thoron daughters if radium and/or thorium was processed.
- o Perform a health and safety inspection to identify potential safety hazards to survey personnel and to identify chemical characterization tasks.
- o Conduct interviews with plant personnel if the facility is still in operation, with former employees if not, with local officials, and with nearby residents. A given set of questions should be asked of each interviewee in a effort to gain common information regarding undocumented spills, waste disposal, construction or renovation which may have masked contamination, etc.

characterization team. The characterization plan should contain a clear description of all grid systems and how they are to be used.

Geophysical Surveys

As a means of identifying underground facilities such as sewers and other piping, and buried items in landfills or in open land in general, goephysical techniques should be utilized. There are three commonly used methods for this type survey:

- o Magnetometer - for use in locating buried metal items.
- o Ground Penetrating Radar (GPR) - for use in defining boundaries of covered landfills and waste pits and to verify indicated underground metals from magnetometer surveys.
- o Electromagnetic Terrain Conductivity (EM) - for use in detecting high conductivity in groundwater.

Information from these surveys can be used to refine the location of boreholes for gamma-ray logging and sample coring, placement of monitoring wells, and to identify potentially hazardous drilling locations. Use of these techniques should be included in the plan and the surveys should be conducted in advance of radiological measurements and sampling.

Radiological Field Activities

The work plan elements discussed above will not in themselves assure a successful characterization. They can, however, play a major role in designing the location and frequency of measurements and samples. Careful consideration of the foregoing will suggest areas where special attention should be placed on certain types of measurements and where particular techniques may be required to obtain samples.

Radiation Measurements. These will be made at regular intervals on all grids (inside buildings and outdoors in open land) and in biased locations where there is reason to believe that closely spaced measurements will be required to adequately describe the limits of contamination. The actual measurements to be performed will be dictated by the type of radioactive material known or suspected to be present. It must be remembered that some radionuclides emit low energy radiation and may not be detectable on land surfaces using portable instruments for in situ measurements. Radiochemical analyses of samples may be the only way to document the presence of these materials. Open land areas

will require gamma-ray scanning surveys to determine those areas with anomolous radiation levels. Results of these surveys can be used to establish biased sampling locations as well as to locate boreholes for gamma-ray logging and core sampling. Gamma-ray logging of boreholes is used to obtain information on the depth and type of radioactivity. Similar techniques may be used inside buildings to determine the extent of contamination on and under building floors.

Instrumentation. The type of individual radiation detection equipment to be used during a survey should be based on site research. In certain cases, special detectors such as thin sodium iodide crystals, gamma-ray spectrometers, and large area gas flow proportional counters may be required. This will depend on the type of material known to have been handled as well as radioactivity which could exist as a co-contaminant. Plans must be made to include an adequate supply of anticipated instruments, and to provide maintenance and calibration support. For major characterization, a field laboratory offers an economical means to perform certain analyses. Typical of instruments included in a mobile laboratory are: gross alpha and beta counters for air and swipe samples (also evaporated water samples), alpha spectrometers to detect specific alpha emitters on air samples and swipes, high resolution computer based gamma-ray spectrometers for determining concentrations of gamma-ray emitters in soil, sediment, and water, and various laboratory equipment for drying/preparing samples, and for handling waste material. Equipment for radiochemical analyses should not be included in a field laboratory unless it can be shown through a cost study that analyses in the field can meet or exceed considerations for cost and quality control.

Environmental Samples. The site characterization plan will include details for the collection of environmental samples. Sampling locations (systematic and biased), frequency, type of sample, and analyses will be governed to a large extent by information obtained during development of the plan (site research and results from site visits). From a practical standpoint, the number of samples to be collected and analyzed will depend on available funds. There must be some flexibility in implementing the plan at the field level. Once sampling equipment is in place, additional samples should be collected if there is a demonstrated need to do so based on observations at the sampling site. A selection of samples for analysis can be made after the fact. Samples which are not to be analyzed initially should be archived in a secure location. Chain of custody records must be initiated at the time of collection and maintained for all samples.

Quality Assurance

An on-going and integral part of the characterization is a series of activities which assure that all work is done in accordance with procedures and standards. Quality assurance measures need to be summarized in the work plan. The plan should also include a discussion

of the type and frequency of quality assurance audits which will be conducted in the field. These audits permit managers to identify and rectify situations before there is a serious impact on the quality of field work.

Data Management

Include in the work plan a description of methods which will be used to manage and evaluate field data. Micro computers and integrated programs offer a powerful tool for remaining current in data review and evaluation. For example, the use of spreadsheets permits field data to be segmented by category and geographical site area regardless of the order in which it was acquired. This type program also allows one to maintain track of schedules and resources. It is also very useful in the preparation of tables and graphs for progress reports. Perhaps the best advantage, however, is the ability to identify problem areas before leaving the field.

SUMMARY

The foregoing discussion of radiological work plan development is presented as a guide. The authors have drawn from their experiences in organizing and conducting field exercises. Considerable work is required to obtain the maximum useful information within available resources. In some cases it may be difficult to persuade managers of the necessity to expend the suggested effort in planning. It must be remembered, however that repeated trips to the field are expensive and consume valuable time.

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DECOMMISSION PLANS FOR LOVIISA 1 AND 2 UNITS IN FINLAND

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ABSTRACT

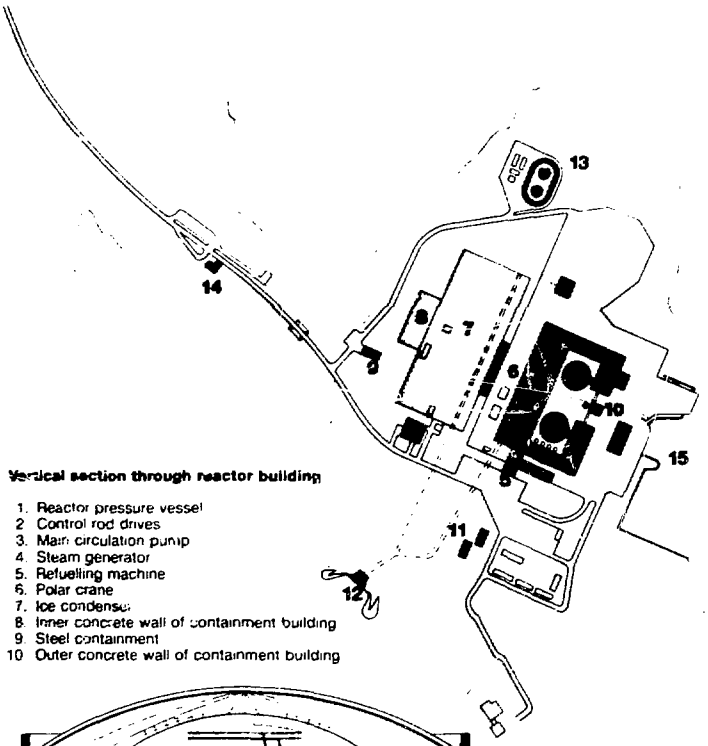
The reactor pressure vessel will be transported in one piece to the final repository. The reactor internals will be stored inside the reactor pressure vessel, which acts as a waste container. The steam generators will be also transported to the final repository intact. The final repository will be a rock cavern at a depth of 100 meters.

INTRODUCTION

The Loviisa Nuclear Power Plant consist of two VVER-440 type PWRs. The first unit started its commercial operation in May 1977 and the second unit in January 1981. The electric power of one unit is 465 MWe. The plant has six horizontal steam generators and six primary coolant pumps. The containment is made of steel and it is equipped with ice condensers. Each unit has two turbines. The building volume of one unit is approximately 500 000 cubic meters. Both units have had excellent operating history with 79 % and 83 % cumulative load factors. The plant buildings and containment section are shown in Figure 1.

DECOMMISSIONING PRINCIPLE

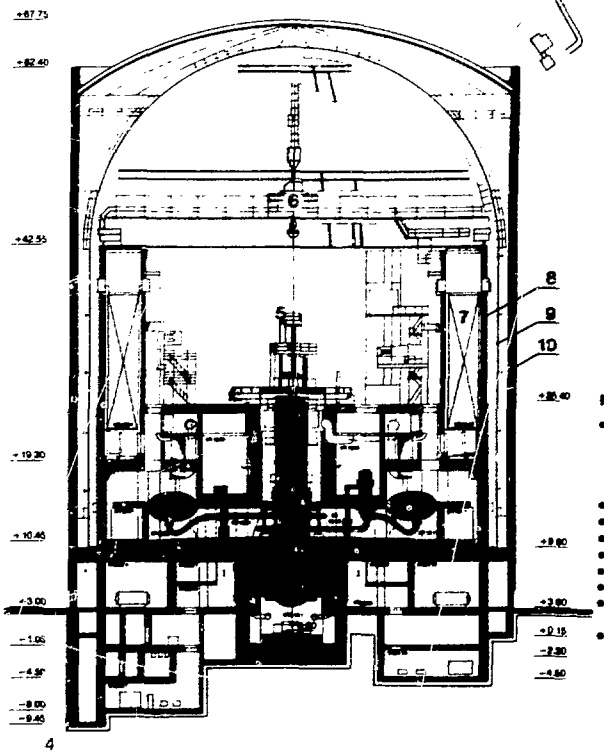
Decommissioning target is the Stage III. The plant will be cleaned from radioactivity but the buildings and structures will not be dismantled if not necessary for removing radioactivity or if the buildings and structures will not prevent further plant units to be constructed on the site. The plant site will be used for power production purposes also in the future.



1 Reactor building
2 Turbine and control building
3 Laboratory building
4 Auxiliary building
5 Office building
6 Diesel building
7 400 kV switchyard
8 110 kV switchyard
9 Water treatment plant
10 Service building
11 Information building and training office
12 Cooling water intake
13 Gas turbines
14 Guard house
15 Cooling water outlet

Vertical section through reactor building

1. Reactor pressure vessel
2. Control rod drives
3. Main circulation pump
4. Steam generator
5. Refuelling machine
6. Polar crane
7. Ice condenser
8. Inner concrete wall of containment building
9. Steel containment
10. Outer concrete wall of containment building



Numerical data	Lo 1	Lo 2
• Total volume of buildings	510 000 m³	390 000 m³
– reactor building	121 000 m³	121 000 m³
– turbine and control building	196 000 m³	148 000 m³
– auxiliary building	90 000 m³	77 000 m³
• Soil and rock excavated	237 800 m³	83 400 m³
• Concrete	86 000 m³	64 000 m³
• Reinforcement steel	9 900 t	8 000 t
• Steel structures	3 000 t	3 000 t
• Pipework	3 300 t	3 600 t
• Number of valves	12 900	12 000
• Instrumentation measuring points	4 500	4 500
• Instrumentation cables	1 200 km	950 km

Figure 1. Loviisa plant buildings. Section of containment.

The basic idea in the decommissioning is to transport the reactor, the reactor internals and the steam generators to the final repository as intact components. The components will be transported in the same way as they were transported into the containment during the installation. The reactor pressure vessel will be used as waste package for reactor internals and dummy fuel elements. The dummy fuel elements were put into the reactor core peripheral area for decreasing the neutron doses of the reactor pressure vessel and the radiation enbrittlement.

Costs and radiation doses can be saved by this way. The same technology will allow also the change the reactor pressure vessel, which will be necessary if extending the plant life.

REMOVAL OF PRIMARY CIRCUIT COMPONENTS

Detailed plans were been made for removing the primary circuit components to the final repository. Figures 2 and 3 show the reactor vessel disengagement procedure.

The reactor pressure vessel will be transported in one piece. Its empty weight is 214 tons. It is necessary to have a radiation shield in the area of reactor core. The concrete shield having thickness of 300 mm and weighing 55 tons will be fixed in the midsection of the reactor pressure vessel, Figure 4.

The maximum transportation weight is 282 tons. The nominal lifting capasity of the reactor hall polar crane is 250 tons, so the reactor pressure vessel can be handled with the polar crane by small overloading . The maximum transportation capacity of available trailer is 320 tons, Figure 5.

The reactor internals will be transported to the final repository with help of a shielding cylinder and the transportation trailer. The shielding cylinder is used during plant operation as a lifting tool and a radiation shield when extracting the reactor internals for refueling or for reactor pressure vessel inspections. The shielding cylinder is made of steel and its weight is approximately 200 tons. The shielding cylinder is equipped with bottom lid and it is used as a radiation shield, when the reactor internals are transported to the final repository.

Steam generators, pressurizer and the bubbler tank will be transported to the final storage also as intact components. No additional radiation shields are needed.

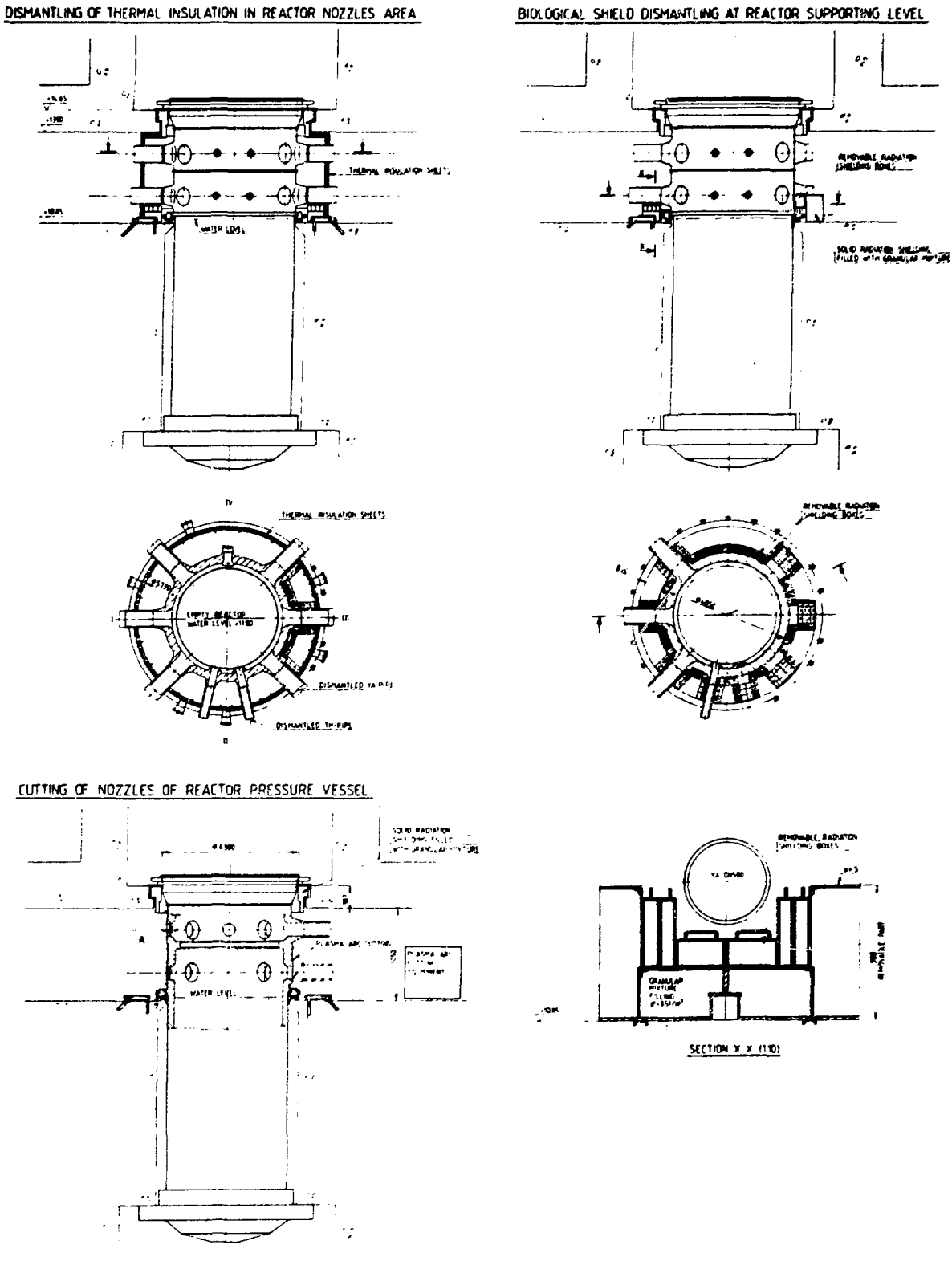


Figure 2. Disengaging of the reactor pressure vessel.

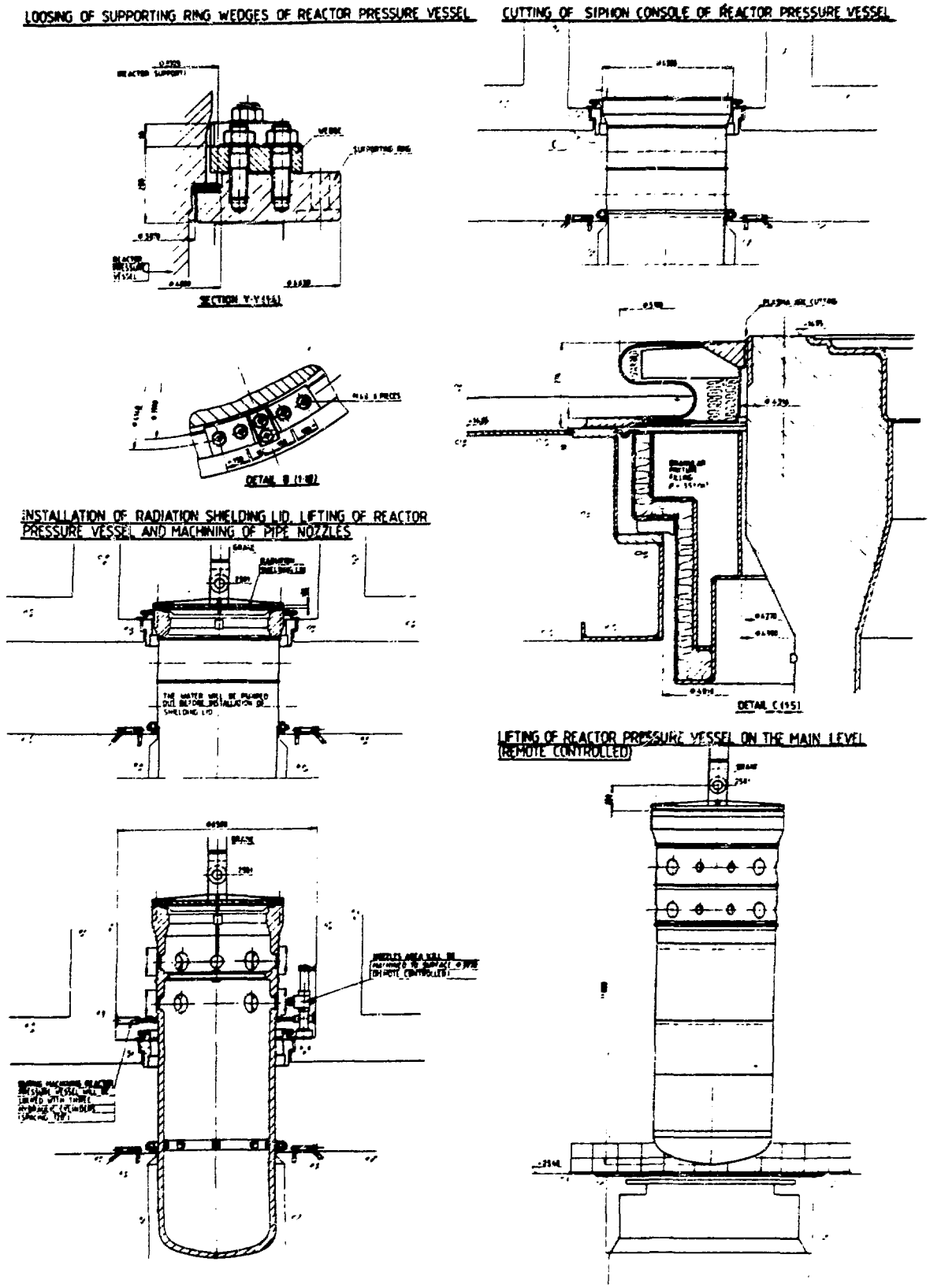


Figure 3. Disengaging and lifting of the reactor pressure vessel.

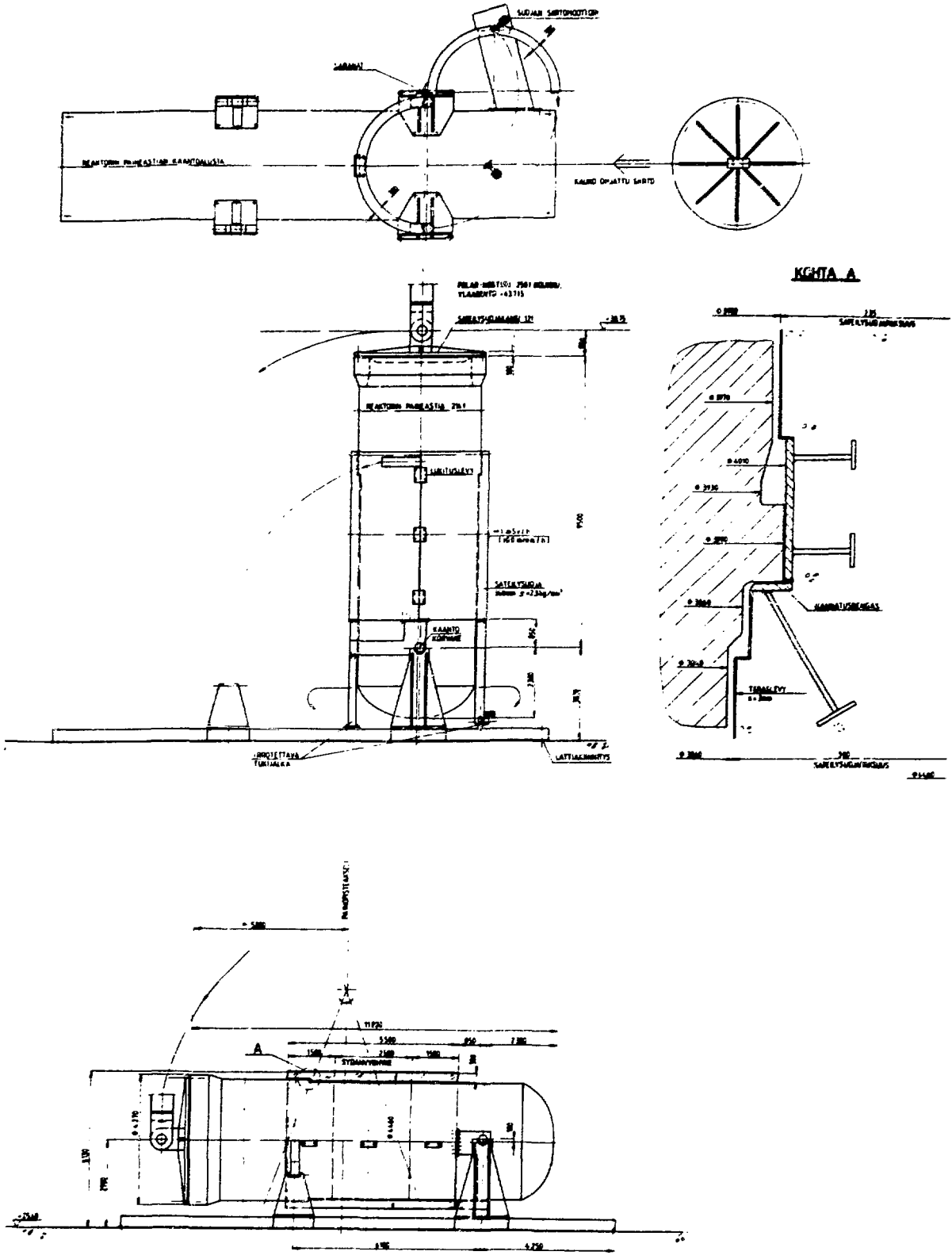


Figure 4. Installing of radiation shield and tilting of RPV

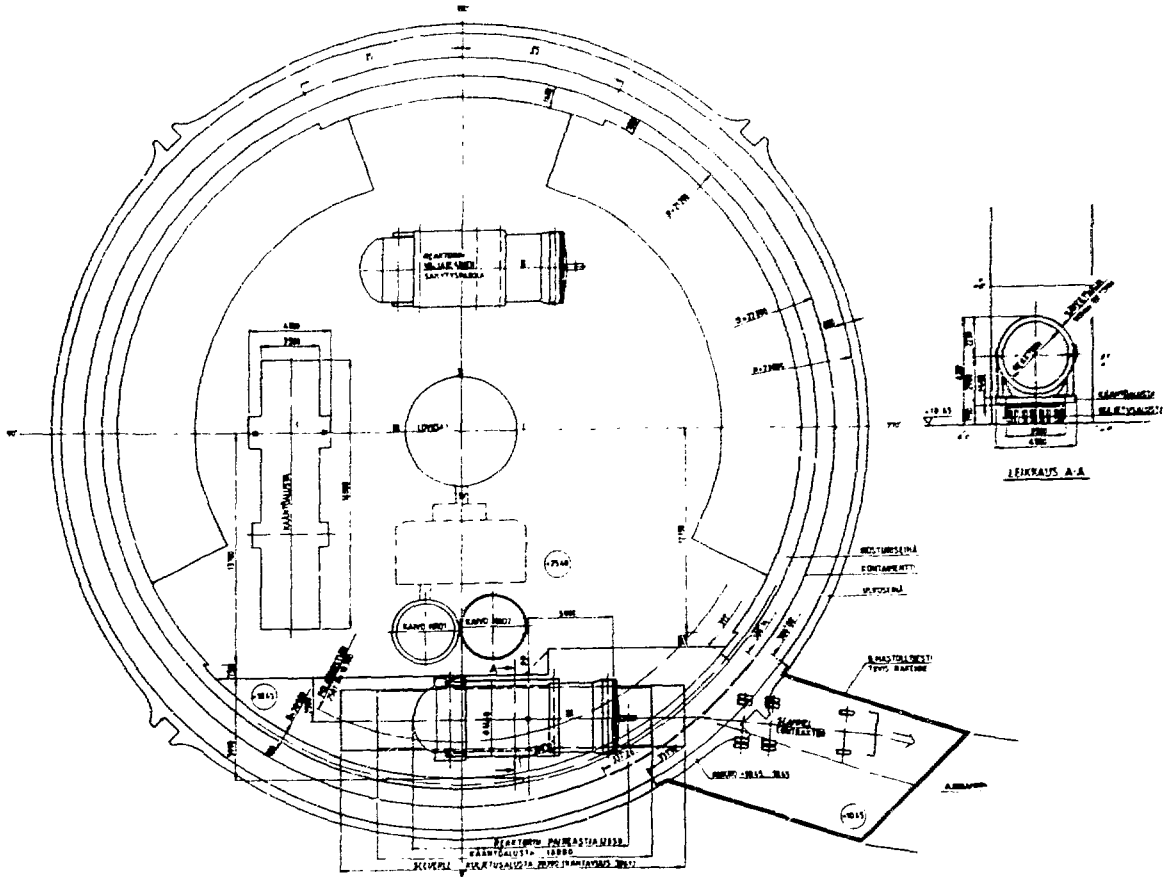


Figure 5. Transporting out the reactor pressure vessel.

FINAL REPOSITORY

The final repository is a rock cavern system, which is located in the plant site at a depth of 100 m below ground level. Transport will take place via transportation tunnel. Its slope is 1:10 and the length one kilometer. All plant wastes except spent fuel will be stored into that rock cavern system.

In the repository a pit for the reactor pressure vessel is reserved, Figure 6. The reactor pressure vessel is in upright position so that the reactor internals and dummy elements can be positioned into the reactor, Figure 7. The reactor pressure vessel acts as waste package for the reactor internals and dummy elements. The cavern is equipped with a crane, so that handling of heavy components and the shielding cylinder is possible, Figure 8.

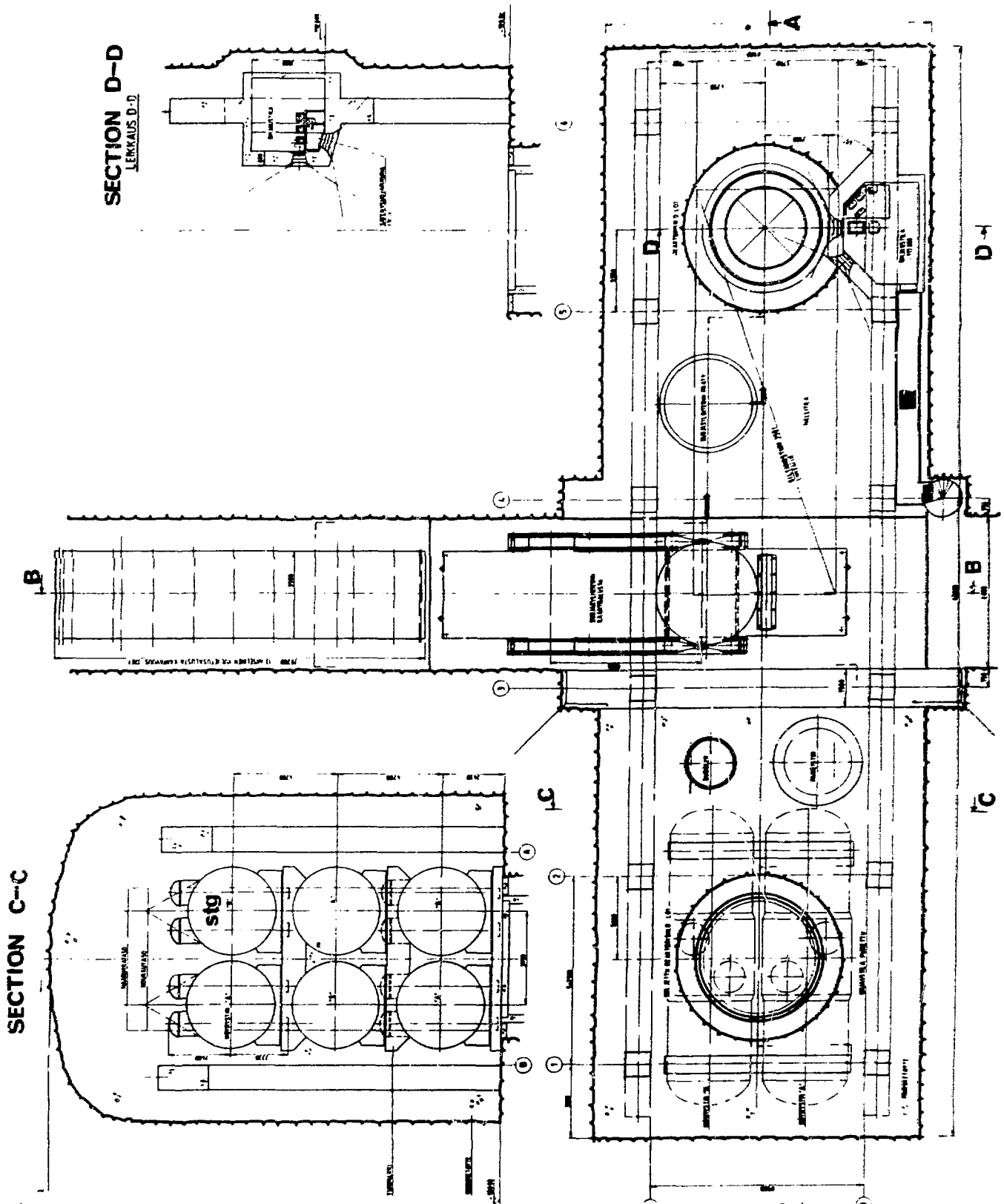


Figure 6. Final repository cavern, plan view.

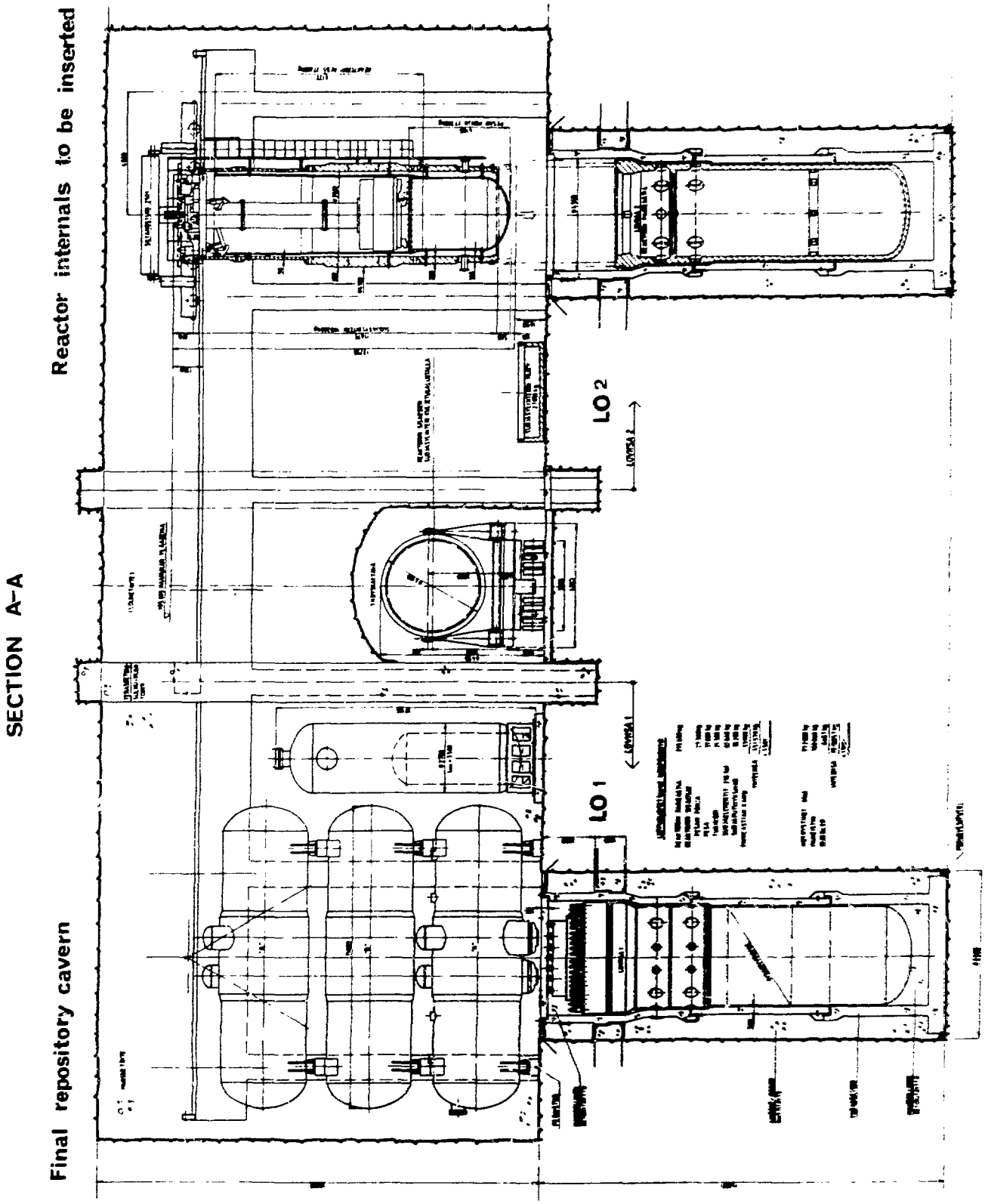


Figure 7. Final repository cavern, section A - A.

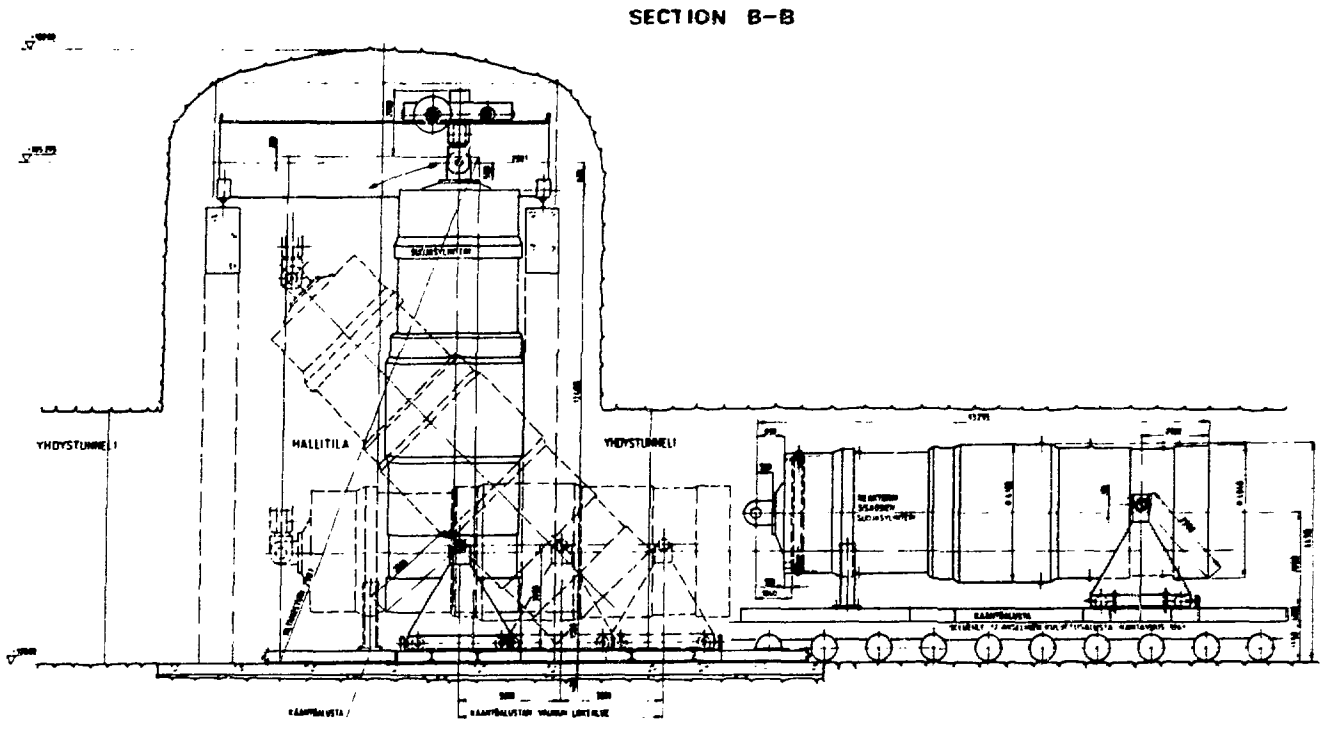


Figure 8. Final repository cavern, section B - B.

The steam generators and the pressurizer as well as the bubbler tank will be positioned into the same rock cavern as the reactor. Three steam generators will be put one on top of another. Because of six steam generators two piles of steam generators will be accumulated per plant unit.

The activity of reactor internals is 14500 TBg an the activity of the dummy elements is 30000 TBg. The activity of the reactor pressure vessel is round about 260 TBg.

The rock cavern will be filled with sand and gravel and shielded with concrete lock. The transportation tunnel will be closed by three or four concrete blocks.

DECOMMISSION TIME SCHEDULE

The time to start the decommission works would be around year 2015. The designed time schedule of decommissioning is shown in Figure 9.

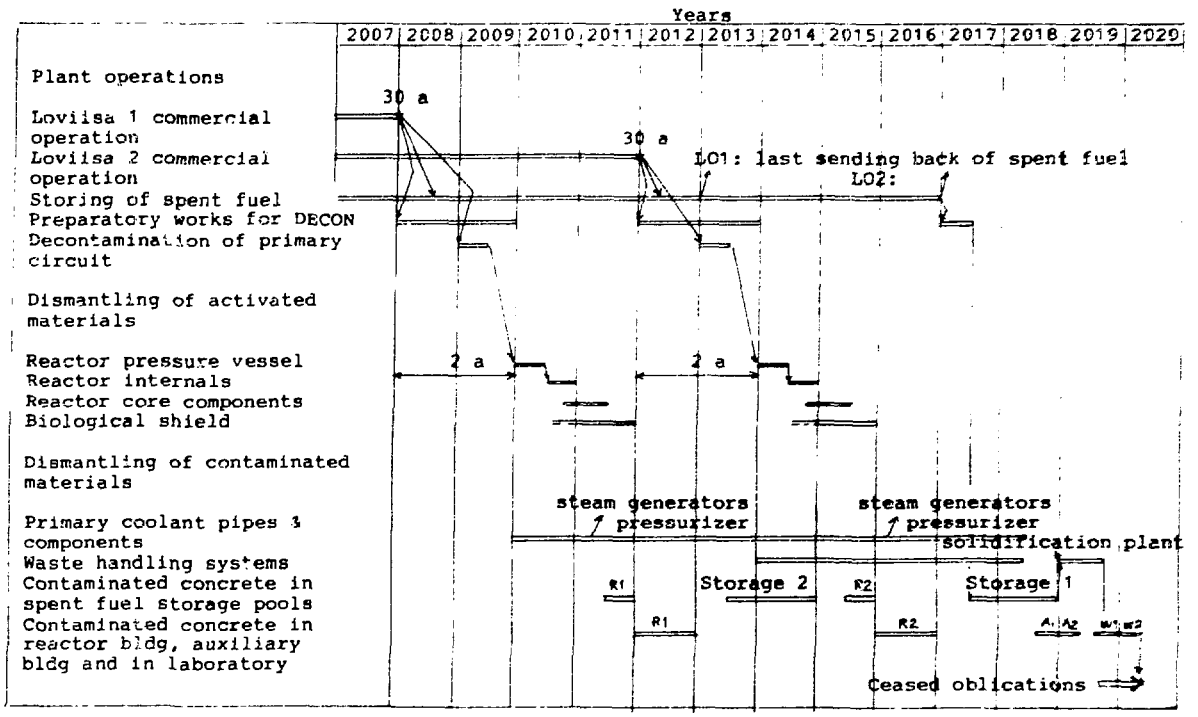


Figure 9. Time schedule of decommissioning works.

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THE IMPORTANCE OF FUNDING IN DECOMMISSIONING COST ESTIMATES

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ABSTRACT

Decommissioning cost estimates have been made by several study groups for the decommissioning of pressurized-water and boiling-water nuclear power stations. The results of these studies are comparable when corrected for inflation and the differences in contingency factors applied by the study groups. The estimated dismantling costs differ far less than a factor of 2 in all cases, despite the design differences found in the plants that were studied.

An analysis of the different methods available for funding the dismantling of these facilities shows the much stronger effect that the choice of funding methods has on the net cost of decommissioning. The total cost of dismantling may vary more than a factor of 4 from one funding method to another, assuming current or recent historical inflation rates.

The funding methods evaluated include sinking funds, deposits, negative-salvage value depreciation, and insurance. These funding methods are taken from the NRC's Notice of Proposed Rulemaking description of acceptable funding methods.

The funding analysis for this paper is performed using the DECOST-86 computer code. DECOST-86 is a computer code designed for decommissioning cost and funding analysis. The Battelle Northwest Laboratories' studies (NUREG/CR-0130 and NUREG/CR-0672) were used as the sources for the engineering costs of decommissioning used in the funding analysis.

The evaluation of funding options for a nuclear facility, and the appropriate choice of the funding method best for that facility, are found to be more important than detailed engineering studies in determining the net cost of decommissioning during the early portions of a plant's operating lifetime. This finding is independent of the difficulty in determining available decommissioning technology up to 30 years in advance of actual decommissioning.

INTRODUCTION

Decommissioning cost estimates have been made for the decommissioning of pressurized-water and boiling-water nuclear power stations by several groups. These studies are primarily limited to calculating the engineering costs of decommissioning a nuclear power station. These studies have reached engineering cost estimates that are comparable for PWR's and BWR's

when inflation is considered and when contingency factors are removed. The differences in the engineering cost estimates for dismantling are small (less than 3% in references 1 and 2) since the assumptions for dismantling are less open to interpretation than other decommissioning modes.

The differences in the actual cost to the ratepayers or company, however, can vary significantly with type of funding used to pay for the decommissioning of a nuclear power station. The use of different funding methods can change the actual cost of decommissioning by up to a factor of four.

ANALYSIS

The engineering costs for dismantling a boiling-water reactor will be used as an example in this analysis. The references also address decommissioning methods other than dismantling; however, the other methods vary in both definition and scope from study to study. As a result, the estimated engineering costs for decommissioning methods other than dismantling are not readily comparable. These alternative methods may be described as variations of the mothballing and entombing methods described in the 1975 Atomic Industrial Forum study. The NRC's Notice of Proposed Rulemaking (NPR) of February 1985 would relegate these alternatives to the status of interim steps prior to final dismantling. This paper will, therefore, concentrate on dismantling as a basis for comparison.

Engineering Costs Used

The engineering costs for dismantling a boiling-water reactor are considered as $51.5 million in 1985 dollars. This value is based on reference 2 figures, adjusted for the known inflation between 1978 and 1985 and without any contingency factor. This engineering cost would be equivalent to $227 million in 2015 for an average inflation rate of 5% over a 30-year operating life between 1985 and 2015.

Funding Methods Evaluated. The funding methods evaluated in this paper include sinking funds, deposits, negative-salvage value depreciation, and insurance. These funding methods are taken from the NRC's Notice of Proposed Rulemaking description of acceptable funding methods. These funding methods are defined below:

1) Constant-fee sinking fund: A funding method where the monies collected are held in an interest-bearing account, and yearly charges are assessed as a constant fee over the life of the fund.

2) Escalating-fee sinking fund: Same as 1, but charges are assessed at a rate that varies with the rate of inflation.

3) Deposit for operational life: A funding method in which a deposit is made in an interest-bearing account such that there will be sufficient funds at the end of the expected operational life.

4) Deposit at startup: Same as 3, but the amount of the deposit will be sufficient at the beginning of the operational life.

5) Deposit with return: Same as 4, but annual adjustments are made to the fund depending on earnings and inflation, with excess funds returned to the customers.

6) Straight-line depreciation: A funding method in which the operator of the facility funds the dismantling through internal bookkeeping. A negative-salvage value is assumed for the facility at the end of the expected operational life. An internal fund is maintained for the purposes of meeting costs. Appropriations are calculated as a constant value each year.

7) Adjusted, straight-line depreciation: Same as 6, but appropriations are calculated at a rate that increases each year.

8) Declining-value insurance: A type of insurance where the value of the surety or policy covers the difference between the current cost of decommissioning and the current value of the fund used.

9) Constant-value insurance: Same as 8, but the surety or policy covers the current cost of decommissioning.

These funding methods are evaluated using the DECOST-86 computer code. DECOST-86 is a computer code designed for decommissioning cost and funding analysis. The generic data built into the DECOST-86 code are taken from the Battelle Northwest Laboratories' studies (references 2 and 3). The generic data are used for this example.

Since insurance is not properly a funding method (it must be coupled with another funding method), it will not be considered in this paper. Insurance costs can affect the relative economy of the basic funding methods evaluated in specific cases. These costs typically affect sinking funds more than deposits and affect depreciation funding most of all.

Results of Analysis. The analysis used a base case of an inflation rate of 6%, an interest rate of 8%, a tax rate of 0%, and a finance rate of 10%. These parameters are varied one at a time in the following tables:

TABLE 1

Actual Cost of Funding (million 1985 dollars) vs. Interest Rate

	Yearly Interest Rate			
	2.5%	5.0%	7.5%	10.0%
Constant-fee Sinking Fund	92.7	61.3	39.4	24.7
Escalating-fee Sinking Fund	80.3	58.9	41.7	28.6
Deposit - Operational Life	218.	106.	52.3	26.2
Deposit - Startup	79.7**	79.7**	79.7	79.7
Deposit - Return	134.	95.1	56.5	17.9
Straight-line Depreciation*	136.	136.	136.	136.
Adjusted, Straight-line Depr.*	106.	106.	106.	106.

*Cost to ratepayers. Cost to operator is same as sinking fund.
**Will not cover decommissioning cost.

TABLE 2

Actual Cost of Funding (million 1985 dollars) vs. Inflation Rate

	Yearly Inflation Rate			
	0.0%	5.0%	10.0%	15.0%
Constant-fee Sinking Fund	14.5	30.5	72.0	182.
Escalating-fee Sinking Fund	14.5	33.5	63.9	102.
Deposit - Operational Life	17.3	38.2	94.6	250.
Deposit - Startup	174.	89.0	54.6**	38.0**
Deposit - Return	42.7	42.2	84.3	138.
Straight-line Depreciation*	54.6	115.	272.	688.
Adjusted, Straight-line Depr.*	54.6	96.6	144.	188.

*Cost to ratepayers. Cost to operator is same as sinking fund.
**Will not cover decommissioning cost.

TABLE 3

Actual Cost of Funding (million 1985 dollars) vs. Finance Rate

	Yearly Finance Rate			
	5.0%	7.5%	10.0%	12.5%
Constant-fee Sinking Fund	35.9	35.9	35.9	35.9
Escalating-fee Sinking Fund	38.8	38.8	38.8	38.8
Deposit - Operational Life	27.9	36.3	45.5	55.2
Deposit - Startup	48.9	63.6	79.7	96.7
Deposit - Return	18.0	32.7	48.8	65.8
Straight-line Depreciation*	136.	136.	136.	136.
Adjusted, Straight-line Depr.*	106.	106.	106.	106.

*Cost to ratepayers. Cost to operator is same as sinking fund.

TABLE 4

Actual Cost of Funding (million 1985 dollars) vs. Tax Rate

	Tax Rate			
	0.0%	10.0%	20.0%	30.0%
Constant-fee Sinking Fund	35.9	41.6	48.0	55.2
Escalating-fee Sinking Fund	38.8	43.5	48.7	54.4
Deposit - Operational Life	45.5	52.2	59.6	67.8
Deposit - Startup	79.7	73.1	66.7	60.5
Deposit - Return	48.8	56.4	61.8	64.9
Straight-line Depreciation*	136.	136.	136.	136.
Adjusted, Straight-line Depr.*	106.	106.	106.	106.

*Cost to ratepayers. Cost to operator is same as sinking fund.

CONCLUSIONS

Tables 1 and 2 show that the differences between the funding methods chosen may exceed a factor of 4. This factor is reached when the interest rate is above or only slightly below the inflation rate. The relation is relatively independent of the finance rate and the tax rate as shown in Tables 3 and 4.

As a result of the above analysis, it is concluded that the evaluation of decommissioning funding options for a nuclear facility is more important than detailed engineering studies in determining the actual costs of decommissioning. This will remain true during the construction and early to middle portions of the plant operating life. Detailed decommissioning plans developed early in the plant operating life suffer from the inability to be certain of the final plant system layout and the status of decommissioning technology and regulations.

Detailed engineering studies are needed prior to actual decommissioning. An operating facility may make many engineering changes over its operating lifetime that invalidate assumptions in detailed studies that have been performed early in a nuclear facility's operating lifetime. The results of this paper suggest that it is more appropriate to utilize generic or modified generic decommissioning studies with periodic review in early planning for decommissioning. Periodic reviews could be made to address changes in costs (inflation), changes in technology, changes in regulations, and additional information from later generic studies. Detailed engineering studies would be postponed until 5 or 10 years prior to the end of the initial license period. Detailed engineering studies could then also address the feasibility of plant life extension at the most logical time for such consideration.

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POSTER SESSION

DECOMMISSIONING WASTES

WASTE PACKAGES FROM DECOMMISSIONING: CHARACTERISATION, CONSTRAINTS AND DISPOSAL RISK

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ABSTRACT

In Switzerland projects have been prepared by Nagra for demonstrating the feasibility of safe disposal for all types of nuclear wastes. The management of the decommissioning waste has been thus in a first approach covered fully from dismantling studies of the 4 Swiss nuclear plants, through waste conditioning and packaging, up to disposal in a specified repository with corresponding safety assessement. The main lessons learned from the links between the different steps are presented and are summarized with emphasis on the heat producing waste packages which are one of the main concerns.

INTRODUCTION

Some years ago conceptual decommissioning studies were required by the Swiss Nuclear Regulatory Authorities for each of the 4 Swiss power plants to demonstrate technical feasibility, to estimate doses and costs and to assess the quantities of wastes arising. In parallel, a project demonstrating the feasibility of safe final disposal for all types of Swiss nuclear wastes including decommissioning wastes was legally required. This project should be based on actual geological field data, describe the repository construction, and include safety analyses. Based on the decommissioning studies, the initial work for a preliminary characterisation, classification and repository allocation of the decommissioning waste was reported at the previous symposium (Ref. 1). This work was continued to define the appropriate waste package configurations within the different waste categories to be taken into account for the repository design, waste package emplacement and safety analyses in the above mentioned project which was submitted in 1985 to the Authorities under the title of "Project Gewähr 1985" (Ref. 2).

A direct link between decommissioning studies and feasibility studies of safe waste disposal involves definition of practical waste packages to be handled. The lessons learned and the problems identified concerning the waste form and packaging and the time strategies for waste package production, storage and disposal are presented in this paper.

DECOMMISSIONING AND WASTE DISPOSAL SCENARIA

Decommissioning

After 40 years of operation the nuclear plants are shut down. Dismantling work takes place between 2 and 9 years after shut down. The alternative of dismantling after 30 years has been investigated in the decommissioning studies but was not considered in the waste disposal studies since it is non-conservative for that purpose.

Waste Package

The basic concept of the nuclear plants and Nagra is to use large concrete containers for the decommissioning waste (2.18 m x 2.08 m x 4.78 m i.e. 21.7 m^3 with 10 cm wall thickness and 17.1 m^3 inner volume) and to condition with cement slurry along with internal metal shielding if required.

Repositories

The repository design in "Project Gewähr 1985" consists of underground caverns in marl with a cross-section accommodating 14 large containers. The remaining voids are filled with special cement. In project Gewähr, this type of repository (called Type B) is proposed for low, intermediate and transuranic wastes. Final allocation of wastes will depend upon safety analyses based on site characterization data; it is considered important to have a disposal facility allowing emplacement of the large decommissioning packages in horizontally-accessed caverns. An alternative engineered facility nearer the surface (Type A) has also been considered for low level wastes alone.

WASTE PACKAGE CHARACTERISATION

For feasibility studies of safe disposal, waste categories must be defined so as to be each represented by a typical waste package. In a first iteration only 8 wastes categories were considered with no detailed requirements being put on the waste package. A preliminary allocation of the waste to the repository A made at this stage had indicated that about 90% of the waste could be disposed of in this repository (Ref. 1).

For "Project Gewähr 1985" more realistic waste packages fullfilling dose rate transport criteria as well as heat production disposal criteria had to be considered. Therefore the lay-out of the filled containers (waste component, shielding and geometrical emplacement) was determined in a first interation step based on the dose rate criteria. The reference time was chosen conservatively as being 2 years after shut-down. As a result the 8 initial waste categories had to be extended by further subdivision of the waste from the reactor vessel and internals into up to 9 subcategories depending on the nuclear plant considered.

THERMAL CONSTRAINTS

Disposal of Stacked Containers

Thermal constraints must be considered only for waste subcategories corresponding to part of the reactor vessel and internals. Container thermal calculation were made for the cavern design of the type B repository of "Project Gewähr 1985". As a first iteration step, the container temperature was calculated as a function of heat production up to 10 W/m^3 for stacked containers in the repository cavern in free air and also after both backfilling of the cavern (Fig. 1). The decay of the heat production is governed mainly by Co-60. Material thermal parameters of the waste packages were conservatively assumed to be those of concrete. The thermal parameters pertaining to the type B repository (Fig. 1) are as follows:

Host rock: 2.0 $W\ m^{-1}\ K^{-1}$ and 3.0 $10^6\ J\ m^{-3}\ K^{-1}$
Concrete: 1.0 $W\ m^{-1}\ K^{-1}$ and 2.6 $10^6\ J\ m^{-3}\ K^{-1}$
Cement: 1.0 $W\ m^{-1}\ K^{-1}$ and 2.2 $10^6\ J\ m^{-3}\ K^{-1}$

Initial temperature of the repository materials was assumed to be 35°C and that of the container 38.5°C.

The results for the maximum temperature in a waste package are given in Fig. 2 for heat loading of 6 W/m^3 and for different delays before cavern backfilling. The larger the delay period, the lower is the maximum temperature. For a delay period of 0.5 year the maximum temperature will reach 94°C after 3.2 years and for a delay period of 3 years 83°C after 4.9 years. The maximum temperature gradient is reached on top of the stack (0.16 °C/cm) after 1.4 years when the cavern has not been backfilled up to this time (for more details see Ref. 3).

The conclusion of this first iteration step is that a thermal loading density limit of about 6 W/m^3 is set on the large container at time of disposal in the described type B repository if one assumes a limiting concrete/cement temperature and gradient of respectively 100°C and 1°C/cm.

Storage of Individual Containers in Free Air

Containers with a heat production greater than the limit for disposal of 6 W/m^3 will have to be stored temporarily. Further thermal calculation were then made to determine the constraint on an individual large container stored in free air (20°C) with the temperature and gradient limits mentioned above.

The resulting heat load limit for an uniform heat distribution is about 140 W/m^3 and is limited by the temperature gradient. For a container clear of the ground surface or standing on a concrete floor the maximum temperature is about 80°C and the maximum surface temperature about 36°C.

For such a high heat load, the assumption of uniform heat distribution cannot be used any longer practically because constraints on dose rate imply that the waste will be centered in the container and thus lead to an increase limit for the maximum gradient. A calculation was made for a source of about 1 m^3 surrounded by 0,75 m of cement in each direction. The corresponding limit for averaged heat load was decreased to about 30 W/m^3 (600 W/container).

When filling the container, the effect of the reaction heat evolution during cement setting must be further taken into account.

Effects of radiation and thermal constraints

The thermal constraints described above were then applied to containers whose configuration (waste component, shielding and their geometrical emplacement) had been first determined with sole respect only to the dose rate constraints. The reference time considered in both cases is 2 years after shut-down; the dismantling (i.e. container filling) period extends from 2 to 9 years after shut-down. The results of this screening can be summarized as follows:

- 45 waste packages with a heat production at 2 years of between 6 and 15 W/m^3 could be disposed of before the end of the dismantling period, in some cases with prior on-site storage depending on their time of production.

- 13 waste packages with a heat production of around 30 W/m^3 could give problems if they are conditioned immediately on waste arising or even before the end of the dismantling period because of the cement setting heat and the concentrated source of decay heat. They will need a few years storage before disposal.

- 70 waste packages with an averaged heat production of more than 120 W/m^3 can not be conditioned during the dismantling period because of the heat source localisation; they may need 1 to 2 decades of storage before disposal. The problem of the packaging for these wastes must be reexamined.

The 128 waste packages mentioned represent 50% of the heat-producing wastes and 6% of the total decommissioning waste packages from the 4 Swiss nuclear plants (3 GWe).

DISPOSAL RISK

In the safety analyses of "Project Gewähr 1985", all the decommissioning waste packages were considered to be disposed in the type B repository. For the key issue of long term disposal safety, the fabrication and storage problem raised by the 70 waste packages mentioned was not relevant. For the sake of simplification in a demonstration project, all the other

wastes packaged in different drum sizes were also assumed to be packed into similar large containers and disposed of the same way (Fig. 1) as the decommissioning waste. From the calculated dose to man given by different radionuclides at different times from the whole nuclide inventory, the contribution the different decommissioning waste categories can be calculated. An illustrative measure of relative disposal risk (R) of the different waste categories can be then estimated by

$$R = \frac{\Sigma DM_j}{DL} = \Sigma \frac{C_j}{CL_j}$$

where j = radionuclide index
DM_j = maximum dose from nuclide j
DL = dose limit (10 mrem/yr in Switzerland)
C = radionuclide concentration of the typical waste package
CL = radionuclide concentration limit

A value R = 1 for given wastes in a specific repository type means that the Swiss dose limit criteria could just be met for this situation.

A similar estimate was made using the concentration limits for the repository type A (near surface) and as a comparison also for the US Shallow Land Burial (10 CFR 61). The results which are used for design optimisation of the projected repositories can be summarized as follows:

Relative disposal risk of Swiss decommissioning waste (R)	10^{-8}	10^{-6}	10^{-4}	10^{-2}	10^{0}	10^{2}	10^{4}
Repository type B		----------	----------				
Repository type A			----------	----------			
US Shallow Land Burial			----------	----------	----------		

The distribution of the waste volume with respect to the disposal risk is very similar for the type A repository and the US Shallow Land Burial. Those wastes which cannot be disposed of in the type A repository comprise parts of the reactor vessel and internals. These are the same wastes for which the heat load per package is greater than the constraint of 6 W/m^3 (see above section), and they represent as has been seen 6% of the total waste packages.

These results are in agreement with those of 2 American studies (Ref. 4 and 5) which quote respectively 0.2% (47 m^3 burial volume) for a BWR and 0.7% (133 m^3 burial volume) for a PWR of the waste which cannot be classified under 10 CFR 61. It must be recalled that the actual volume is smaller and the package volume greater than the burial volume. As an example, 12 m^3 actual waste volume from the Leibstadt BWR of 940 MWe which could not be disposed of in the Swiss repository type A have been packaged in 40 containers i.e. 870 m^3 having an averaged heat production slightly over the disposal constraint of 6 W/m^3.

CONCLUSIONS

Decommissioning studies and waste disposal safety studies have taught the following lessons about decommissioning waste packages and their management:

- It is recognized that a large container is in general more appropriate for decommissioning waste.

- Problems arise for the high radiation and heat-producing waste from the reactor vessel and internals. These problems are due to the combination of all the radiation and thermal constraints which can result in severe requirements on the waste packages.

- As these particular wastes will have to be disposed in an underground repository, the thermal constraint for disposal can be rather restrictive (in this work 6 W/m^3 i.e. 120 W/container of 20 m^3) and can lead to requirements for intermediate storage of the waste packages.

- The production of the waste packages can be also delayed in certain cases because of further thermal constraints (temperature and gradient) for free-standing large containers filled with cement. Although the theoretical constraint amounts to 140 W/m^3 for a container of about 20 m^3 homogeneously loaded (2800 W/container), it reduces to 30 W/m^3 (600 W/container) when the waste is centered in 1 m^3 because of dose rate constraints.

- An indication of the possible impact of the constraint effects is given by the prediction that the management of 6% of the waste packages may have to be modified under the present conditions chosen for decommissioning, packaging and disposal.

- The procedures and timing for packaging the high radiation and heat producing wastes must be set with simultaneous consideration of all the mentioned constraints and will depend upon the chosen dismantling operation (for instance size of the waste components, time of dismantling) and the disposal scenario.

- Due to their radiation and heat properties as well as their rather small component volume, these decommissioning waste would be appropriate for packaging in thick-walled iron containers (as for fuel assemblies) which would allow for an easy cooling before disposal. This option, which could reduce significantly the final volume of these wastes, should be investigated in the future.

- Considerations of the disposal risk of the decommissioning waste have indicated that part of the reactor vessel and internal cannot be disposed of in a near surface repository. For these wastes the thermal constraints on disposal are most important.

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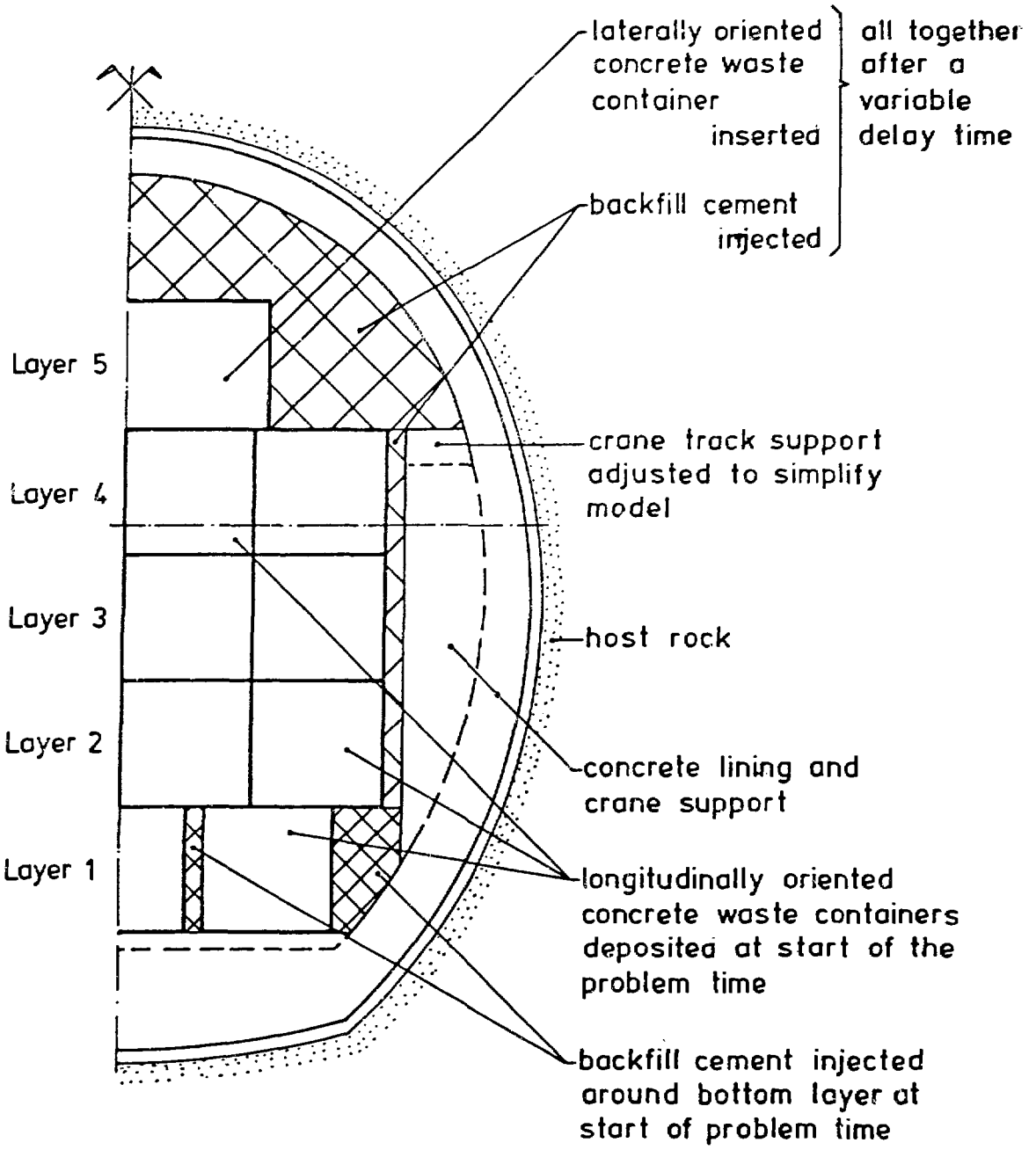

FIGURE 1: ENLARGED SECTIONAL VIEW OF A FILLED STORAGE CAVERN SHOWING THE MATERIAL VARIATIONS

FIGURE 2: MAXIMUM TEMPERATURES IN THE DECOMMISSIONING WASTE IN A TYPE B REPOSITORY WITHOUT BACKFILL AND BACKFILLED WITH CEMENT AFTER A DELAY TIME OF X YEARS. HEAT LOAD 6 W/M^2

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WAGR DECOMMISSIONING - THE PROCESSING OF OPERATIONAL WASTE

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ABSTRACT

In decommissioning the Windscale AGR most of the waste items have to be conditioned in some way to make them suitable for disposal. This paper describes the techniques used to prepare operational waste, a term applied to all of the WAGR's removable channel items, for disposal. These include plug stringers (the upper part of a fuel stringer), control rods and arrestor mechanisms.

A plug stringer is 10 metres in length and includes a 2½ metre long neutron shield plug (NSP). The plug section is separated from the NSP and cut into convenient lengths by 'hands-on' techniques in a ventilated facility. The NSP is cut in a shielded facility using a carborandum disc. Arrestor mechanisms are taken from the core into a modified thimble flask, crushed to a suitable length and returned.

The tools and procedures adopted are described. A key feature of this work has been to use simple engineering techniques and to adapt existing facilities.

INTRODUCTION

At the conclusion of the defuelling operations all 253 reactor channels contained operational waste. Operational waste in this context is the term used to describe all of the removable channel items in the core. These items include plug stringers (the reusable portion of fuel assemblies), control rods and impact absorbing 'arrestor mechanisms' fitted to the bottom of each channel, all of which were designed to be replaceable.

Comprehensive monitoring of the waste ensures that it is separated into intermediate (ILW) and low level (LLW) waste categories. The upper

plug stringer was transferred to an enclosed cutting facility following separation from its neutron shield plug (NSP).

In their original state the NSPs (ILW) are too long to fit into WAGR rad-waste containers and need to be cut into two pieces. The two parts are separately transferred back to the reactor using the Refuelling Machine, where they are stored until the Remote Dismantling Machine system is available to pass them through the rad-waste container packaging route.

The arrestor mechanisms (ILW) are also too long for the waste containers. Their design was based on an energy absorbing 'crumple tube'. The mechanisms therefore lend themselves very readily to size reduction by crushing.

The mechanism is lifted into a modified flask positioned over a channel. A hydraulic ram mounted on top of the flask is used to crush the mechanism.

The processing of operational waste is an important task which can be completed early in the reactor decommissioning project without great expenditure on special handling equipment. Its successful completion is essential to enable subsequent dependent operations like the dismantling of the Refuelling Machine and the installation of the Remote Dismantling Machine to proceed.

Plug Stringer

A plug stringer is some 10m long and from the top downwards consists of a latch mechanism which seals the fuel stringer into the reactor and provides a means of lifting it with the Refuelling Machine; a biological shield section for γ shielding; a tubular section which includes the gas gag valve for controlling gas flow through the stringer; and finally the neutron shield plug (NSP) which at its lower end connects directly to the fuel (Fig 1).

The plug stringer falls into two waste categories. That portion above the NSP was shielded from neutrons in operation and therefore has only surface contamination from spalled activation products and fission products released from fuel failed deliberately in the concluding experiments. In most cases this portion lies within the LLW category. The NSPs however are ILW and to separate the two categories it was necessary to break the connection between the NSP and the rest of the plug stringer. This was done at a convenient dowelled joint. The operation of separating the two parts was carried out under full-time health physics control on the reactor pile cap by withdrawing the plug stringer from the reactor using the building crane. The stringer was withdrawn until the top of the NSP was just exposed. To prevent the spread of contamination the upper plug stringer was drawn into a

polythene sleeve as it was lifted out. Taking the weight of the plug stringer at the top of the NSP, the joint between the NSP and the upper plug stringer was separated, allowing the upper plug stringer to be lifted away and placed into a supporting strongback. A standard lifting adaptor (suitable for refuelling machine handling) was then fitted to the top of the NSP and it was lowered back into the reactor.

The plug stringer in its strongback was transferred to an enclosed and ventilated cutting facility established to one side of the pile cap. A power hacksaw was used to cut the stringer into six handleable pieces which were then monitored, wrapped and dispatched for disposal to the low level waste site. This cutting technique produces low velocity swarf in large fragments thus reducing the spread of contamination. The resulting swarf was collected by an extract system mounted local to the saw.

Operatives were dressed in self-pressurised protective suits during the cutting and waste cleaning operations. These suits, with their re-chargeable battery driven ventilation systems, provide the freedom of movement and breathing protection of respirators but with a comfort level similar to that of an air line pressurised suit.

The cutting and disposal of all of the reactor's 230 upper plug stringers was completed in August 1986. This was the end of the phase of dealing with low level operational wastes. The next phase was to process ILW operational wastes to make them suitable for disposal through he ILW route.

The Intermediate Level Waste Route

Briefly, the ILW route is a means of packaging rad-waste containers by remote handling methods with all of the reactor components and structure that lie within the ILW category. It comprises a Waste Packaging Building([1]) into which empty reinforced concrete rad-waste containers will be brought to be filled with the waste, grouted and finished with a poured reinforced concrete lid so that they end as monolithic containers. The maximum weight of one of these filled containers is 50 tonnes.

The waste will be carried through the route by the WAGR Remote Dismantling Machine([2]) once it has been cut free or released from the reactor.

The constraints on the form of the waste are that it must fit within the internal dimensions of the rad-waste container, that it is within the lifting capabilities of the machine handling system and that a means be provided for grabbing the waste.

In the case of operational waste there is no problem with the weights of the items, but as was mentioned earlier their lengths exceed

the limiting internal dimension of the container (1.86m). To overcome this the items are being processed as described later. As far as lifting features and grab for these items is concerned the same type of grab as is fitted to the Refuelling Machine Special Handling System will be used. This is an electrically motorised ball grab which latches through a 43 mm diameter hole in the item to be lifted. This simple feature was provided in the design of the arrestor mechanism. All other items are being fitted with an adaptor which includes a top plate with this same sized hole.

Neutron Shield Plugs

Neutron shield plugs (NSPs) are made up of two helical sections with bellows in between shrouded in a thin walled stainless steel tube (Fig 2). The bellows section with its wall thickness of 7 mm provides a point where the two sections can be separated by cutting through the outer tube. The central rod is a convenient fe ture on to which to fit the lifting adaptor for the lower half.

The cutting is carried out in the redundant shielded facility previously used to uncouple fuel from plug stringers during the reactor's operating life. The cell, which is of concrete construction with a lead brick front wall, is equipped with lead glass windows and handling tongs. Inside the cell is a multi-purpose six-axis motion machine driven by remote control from the operating platform.

To cut through the outer stainless steel tube a slitting disc cutter was chosen. This is driven by an electrically powered tool mounted on the multi-purpose machine (Fig 3).

The operation commences by lowering an NSP into the fuel uncoupling facility using the Refuelling Machine (Fig 4). The NSP is supported in a rotatable assembly mounted in the floor of the cell.

Four cuts are made, with the NSP being rotated a $\frac{1}{4}$ of a turn between each cut. The disc is enclosed in a swarf box with a vacuum extract connected to a cyclone filter. This reduces the in-cell contamination levels to facilitate regular cell entries for maintenance and cleaning. The cell itself is ventilated via existing systems and a radioactive particle-in-air detector is positioned adjacent to the cell face to monitor the air on the operating platform.

After separating the two halves the top section of the NSP is transferred back to a reactor channel (Fig 5). The bottom half is fitted with its adaptor (from a stock previously placed in the cell through the access door), and the fixing screws of the adaptor are tightened by hand by means of a tong mounted tool. This allows the bottom half to be similarly transferred back to a reactor channel where both halves will be stored until lifted out for disposal by the Remote Dismantling Machine.

Trials using inactive test pieces were carried out to establish disc speed, type of disc and the feed rate for cutting.

Active commissioning trials on the first two NSPs have shown that the equipment and procedures are adequate for the task in hand and the cell contamination levels have remained encouragingly low.

Arrestor Mechanism Crushing

Arrestor mechanisms are fitted to the bottom of all WAGR channels. They were designed to absorb the kinetic energy of a dropped fuel stringer by means of plastic deformation of a preformed 'crumple' tube. Fortunately no such accident occurred during the 18 years of operation of the reactor. The mechanisms (Figs 6 and 7) are 1.9 metres long in their existing state. To fit them into the rad-waste containers it is necessary to reduce their length by at least 0.5 metres. This is done by utilising the crumple tube design in a controlled crushing operation. A flask has been modified by the addition of a winch and grab and a hydraulic ram to do this job (Fig 8).

The flask is ventilated from the top via a cyclone filter mounted on the flask body. This reduces the risk of airborne contamination from the mechanism.

The operation first is to use the Refuelling Machine to 'break out' the arrestor mechanisms from their holding clips and stack them in reactor channels. They are then lifted into the flask by the hand operated ball grab. A 10 tonne capacity ram crushes them to an overall length of 1.2m, the ram is retracted and the mechanism put back into its original channel. The flask is itself positioned over the relevant channel by the pile cap crane. The grab, winch wire and ram head are cleaned after every operation.

As with the NSPs the mechanisms are being stored in reactor channels to be picked up by the Remote Dismantling Machine for packaging into waste containers. At the time of writing 100 of the 250 mechanisms have been lifted, crushed and returned to the reactor for storage.

DISCUSSION

Radiation levels on the upper plug stringers were generally low, but a small number had substantial levels of beta contamination which tended to concentrate at the gag port section. Localised shielding in the form of sheet lead was used to protect operatives when unusually high (~ 30 $mSvh^{-1}$ $\beta\gamma$) readings were recorded. The great majority of the pieces (of unshielded contact dose rate of < 7.5 $mSvh^{-1}$) were sent for disposal at the low level waste site. The pieces with higher radiation were sent for decontamination and then to the low level waste site.

The radiation levels on the NSPs are about 1 Svh^{-1} at the bottom end and a few microseiverts at the top (Fig 9). It would be attractive from the disposal point of view if the top half of the NSP came within the LLW category. Careful monitoring is being done to assess this possibility. So far the two NSPs cut during the active commissioning trials have just exceeded the LLW limit.

CONCLUSION

The processing of WAGR operational waste is an activity which is being carried out early in the project programme in parallel with the planning and design of remote facilities to deal with the reactor itself.

The programme is to complete all operational waste processing by the middle of 1988 and current progress indicates that this date will be met.

It can be seen from the paper that every effort has been made where possible to use redundant reactor facilities suitably modified by the application of simple engineering techniques.

The small 20 man operations and maintenance team employed to carry out dismantling tasks has incurred a collective radiation dose of less than 100 man - mSv per year on this work - a figure well within the working target limit of 10 mSv per man per year.

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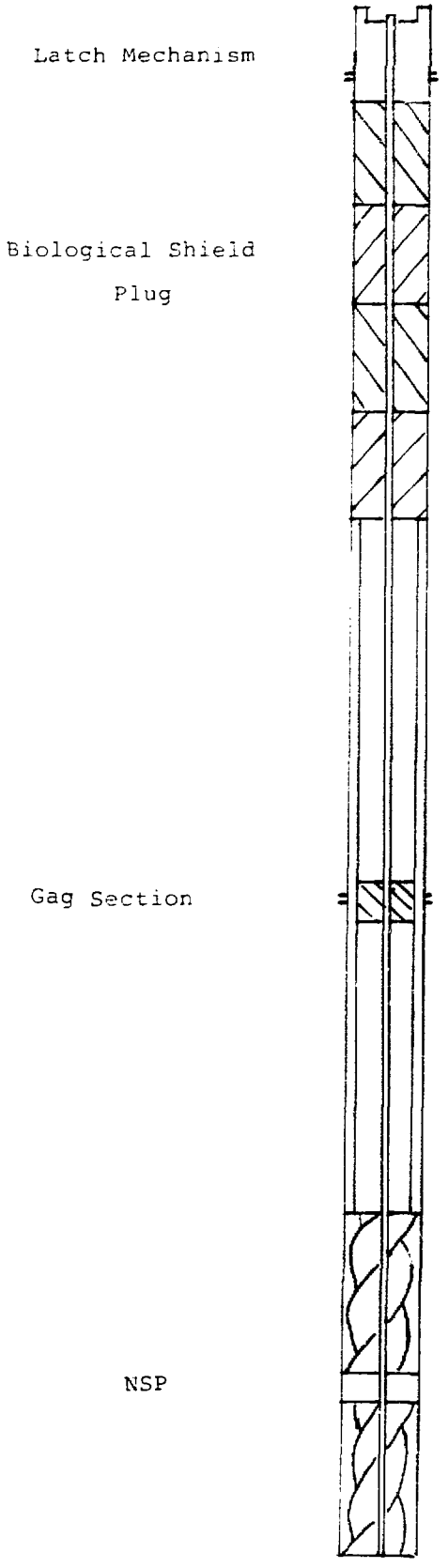


Fig 1 PLUG STRINGER

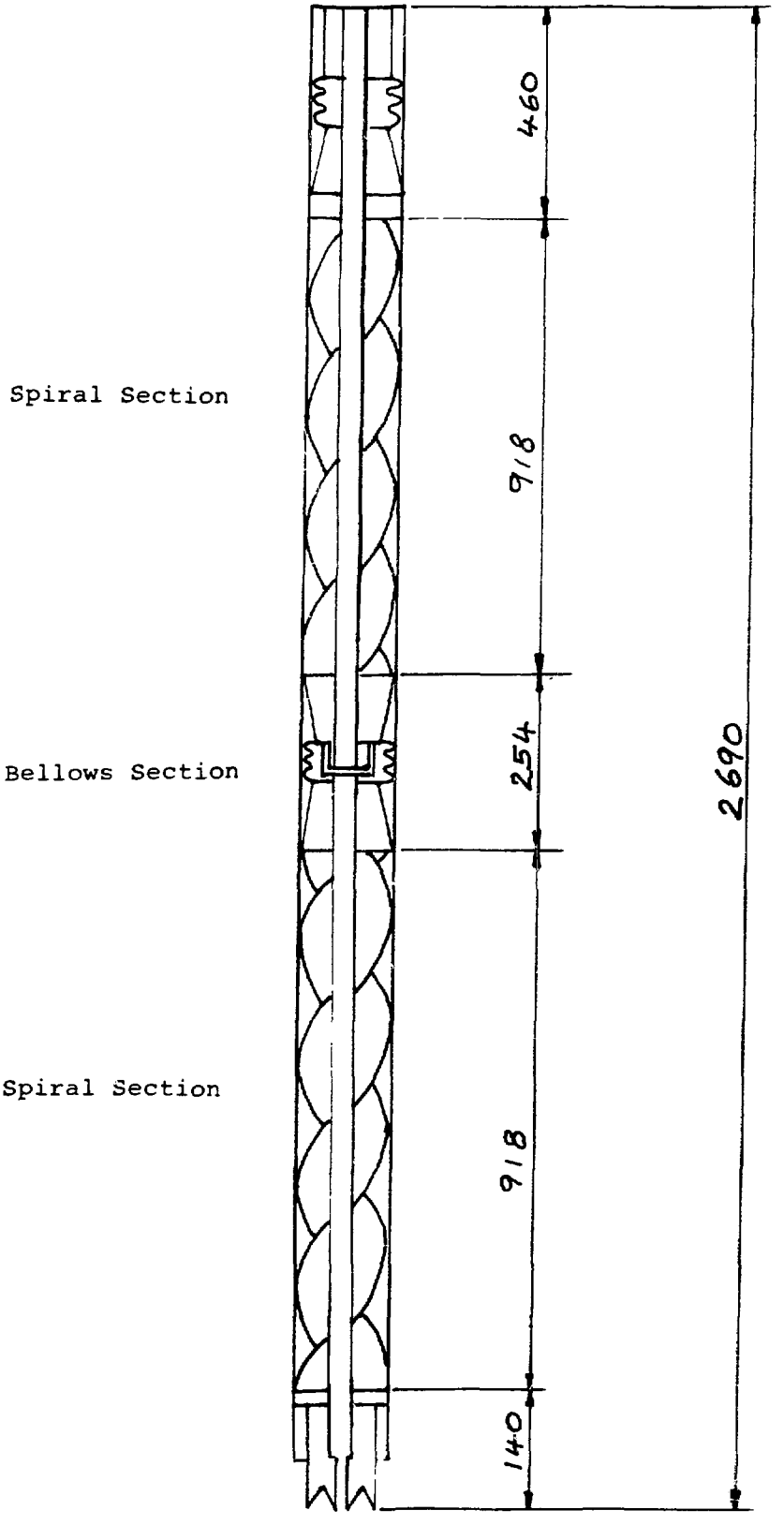


Fig 2 NEUTRON SHIELD PLUG (NSP)

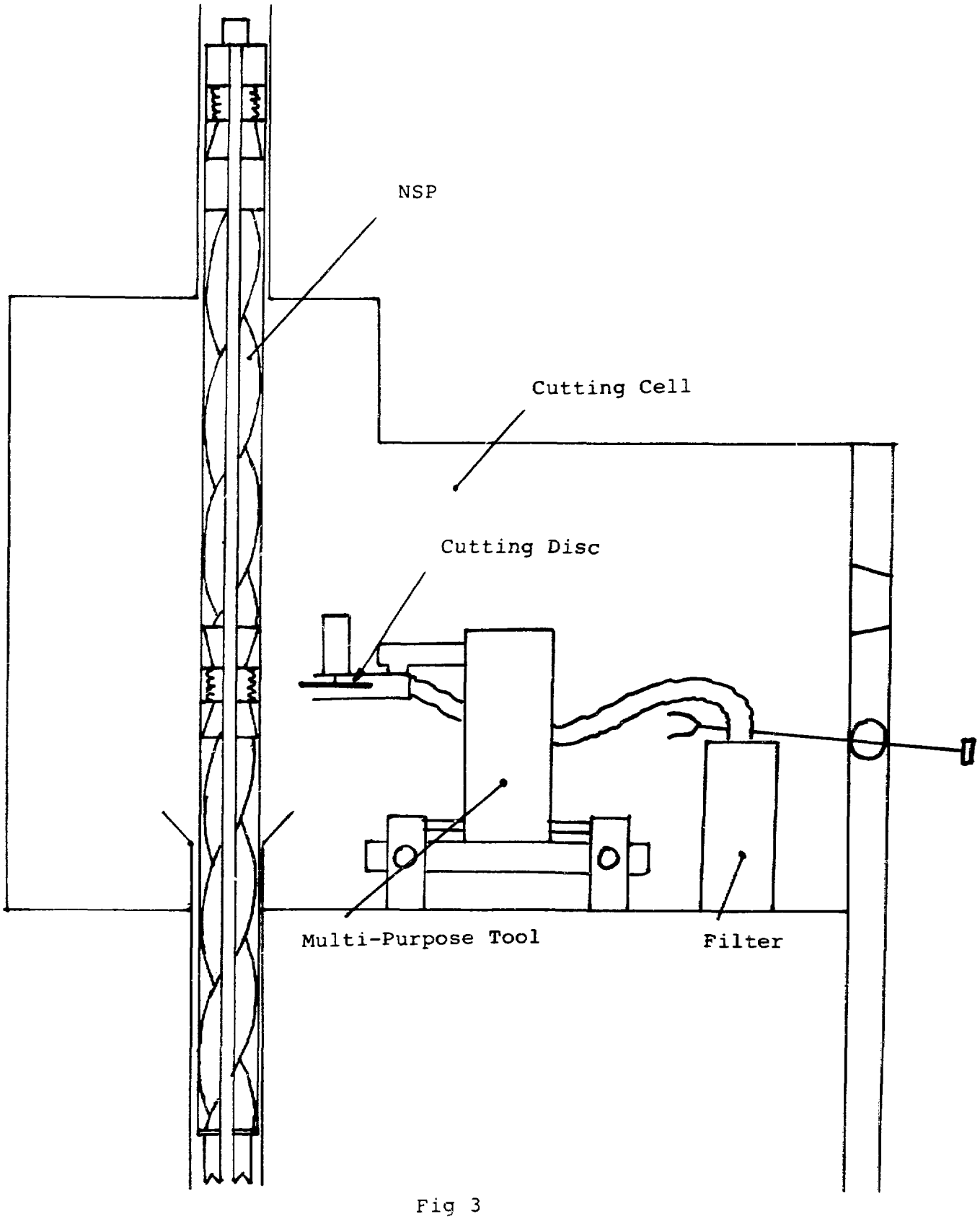


Fig 3

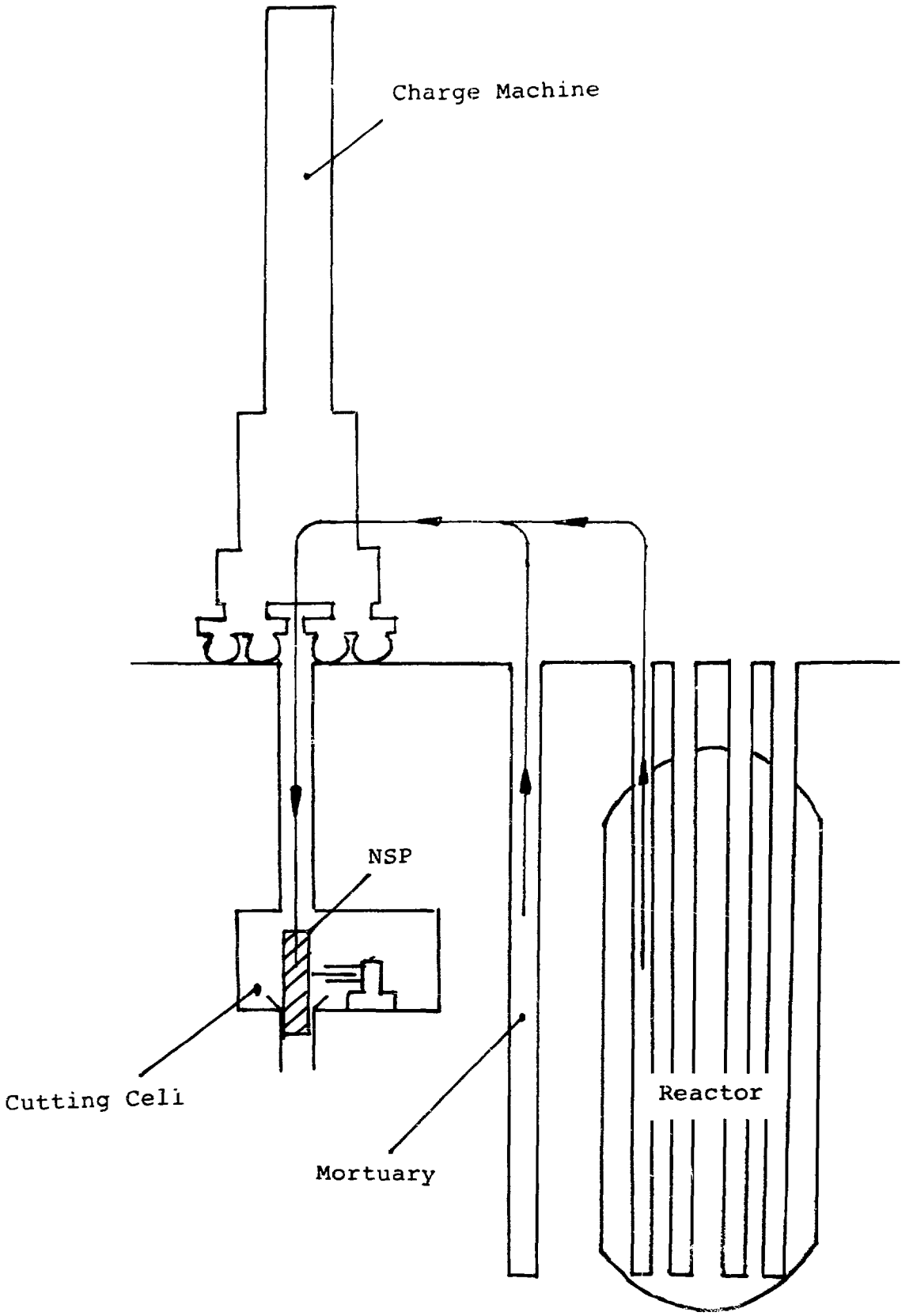


Fig 4 TRANSFERRING NSP TO CUTTING CELL

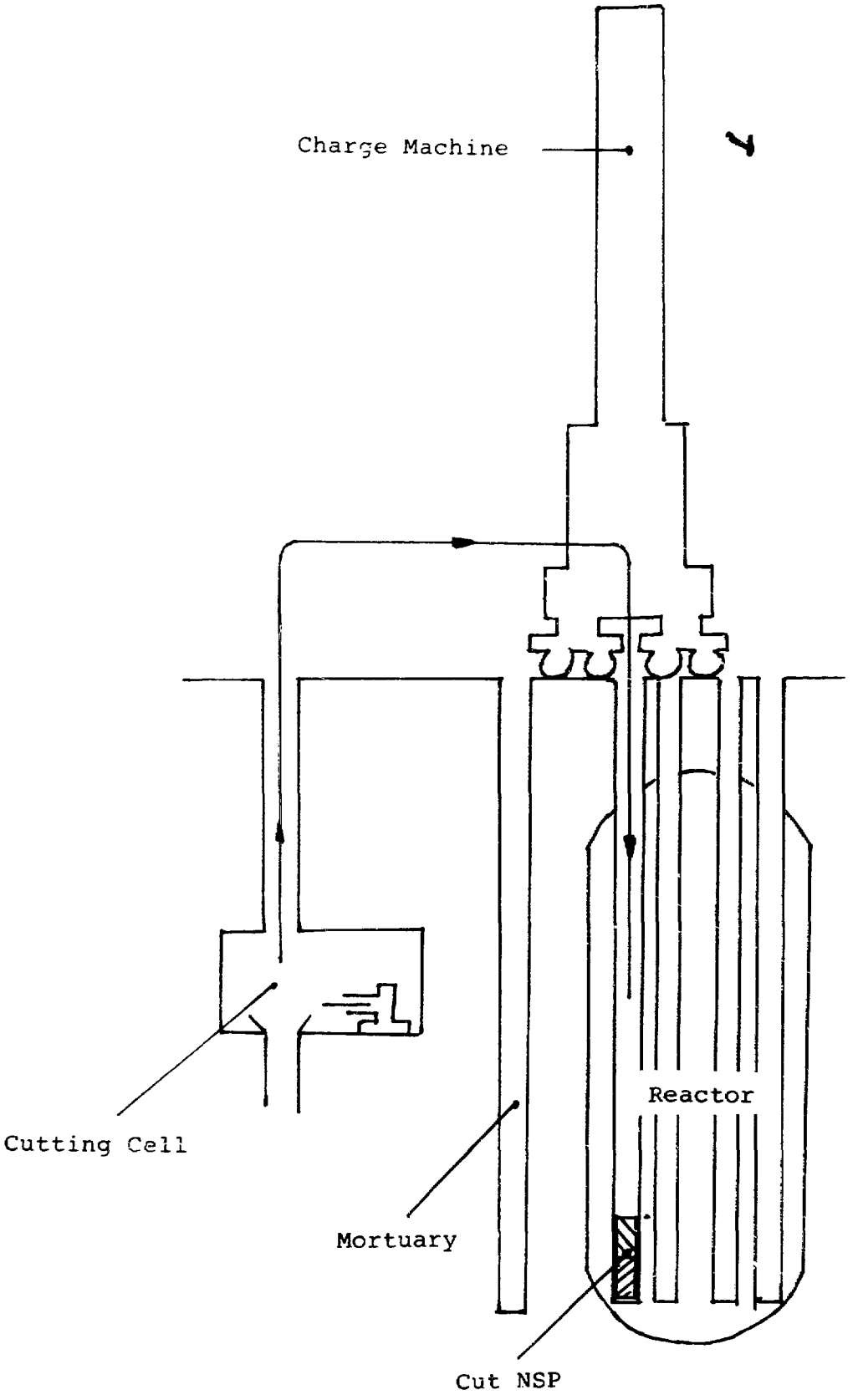


Fig 5 TRANSFERRING CUT NSP TO REACTOR

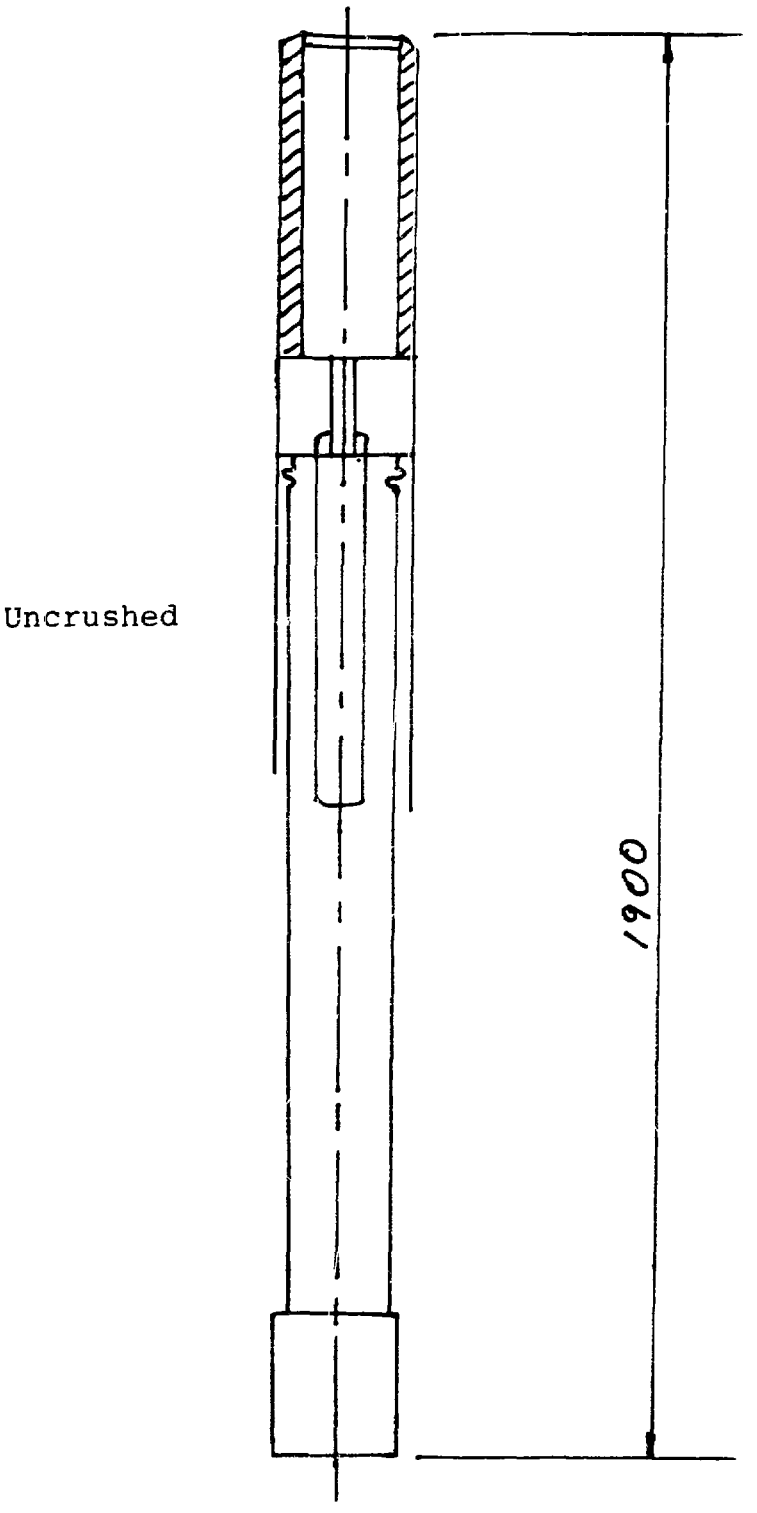


Fig 6 ARRESTOR MECHANISM

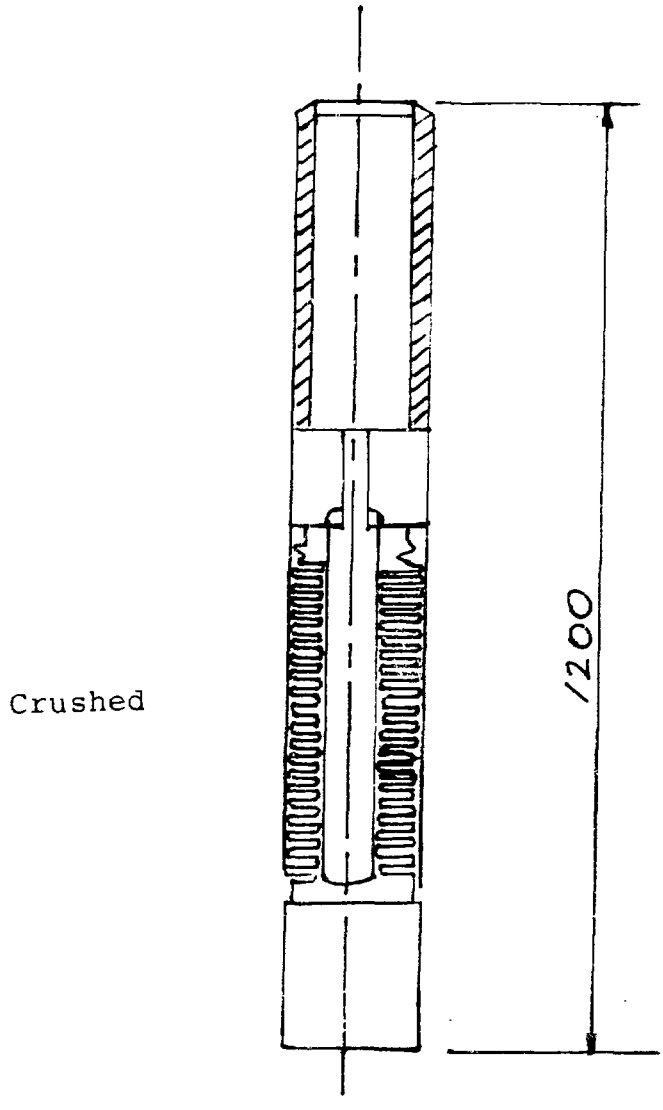


Fig 7 ARRESTOR MECHANISM

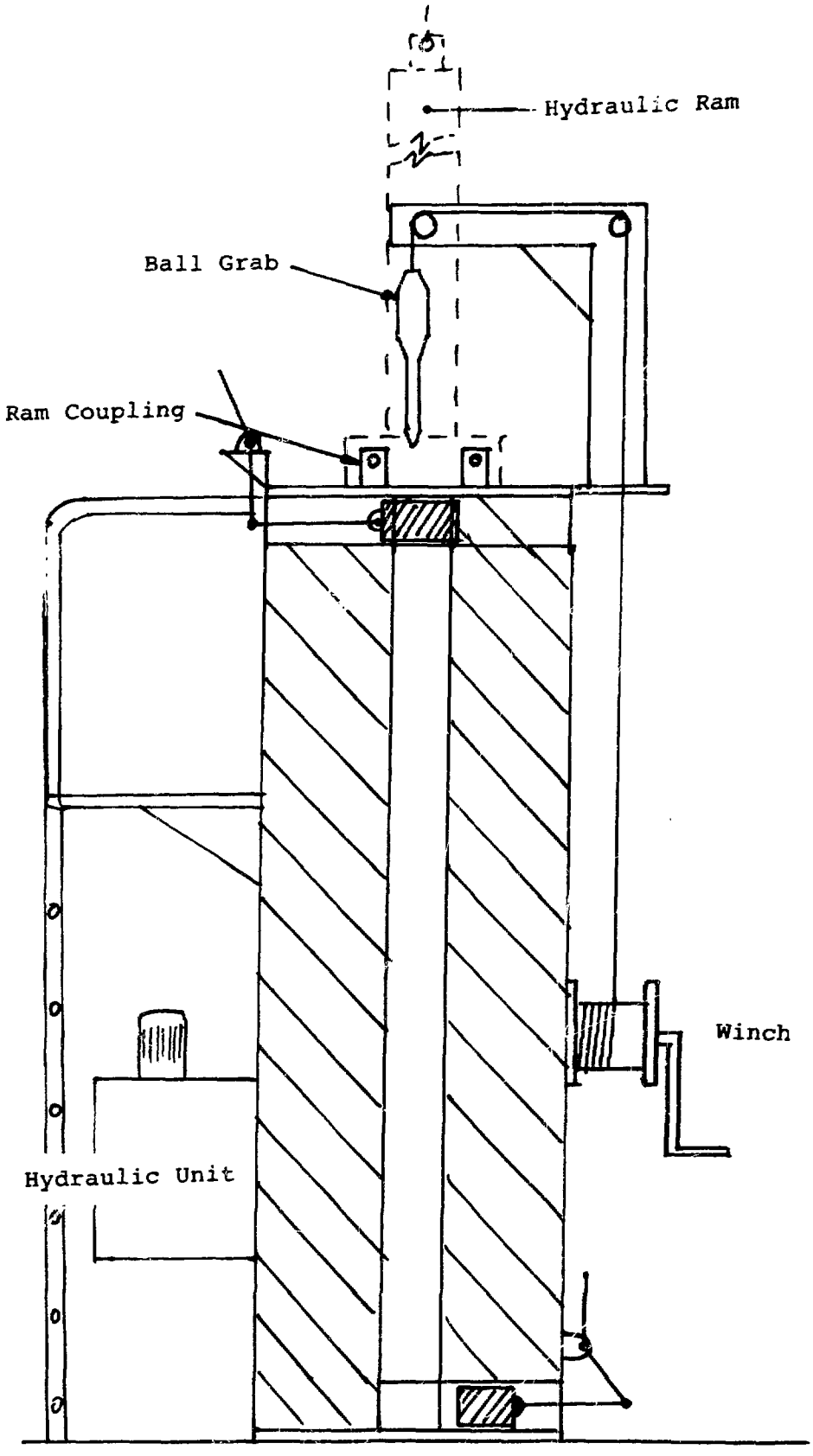
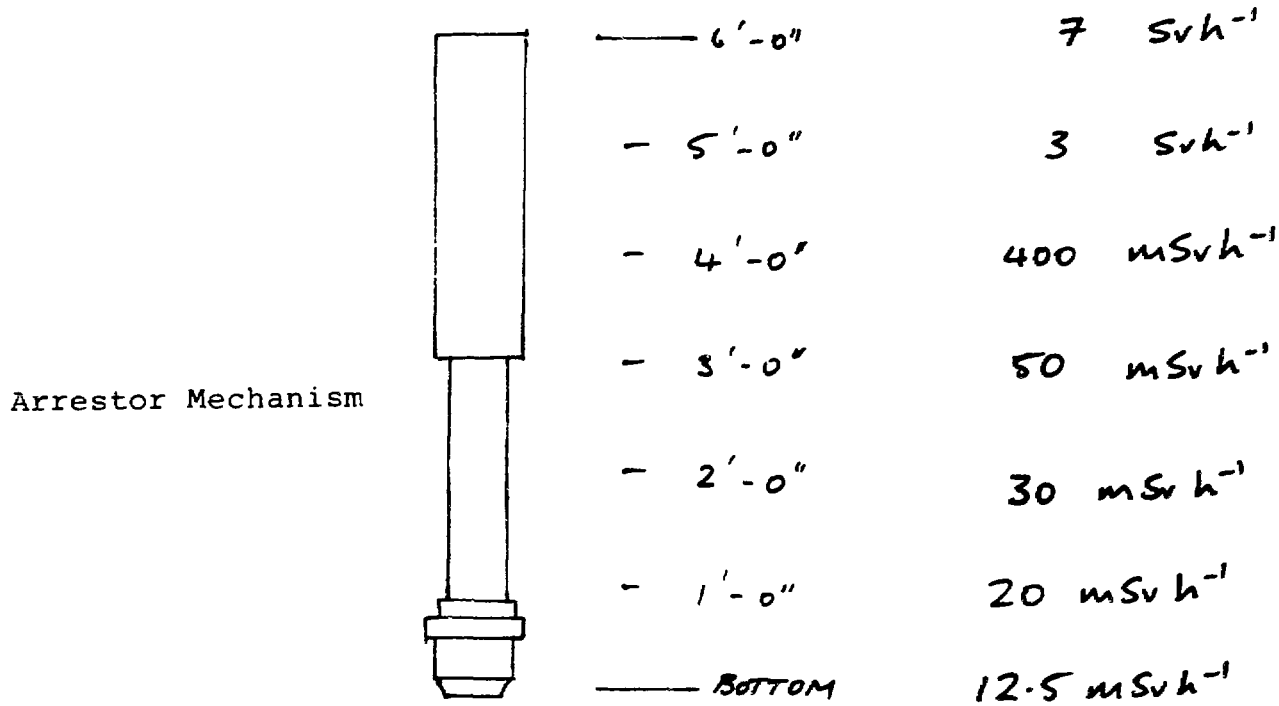


Fig 8 ARRESTOR MECHANISM CRUSHER



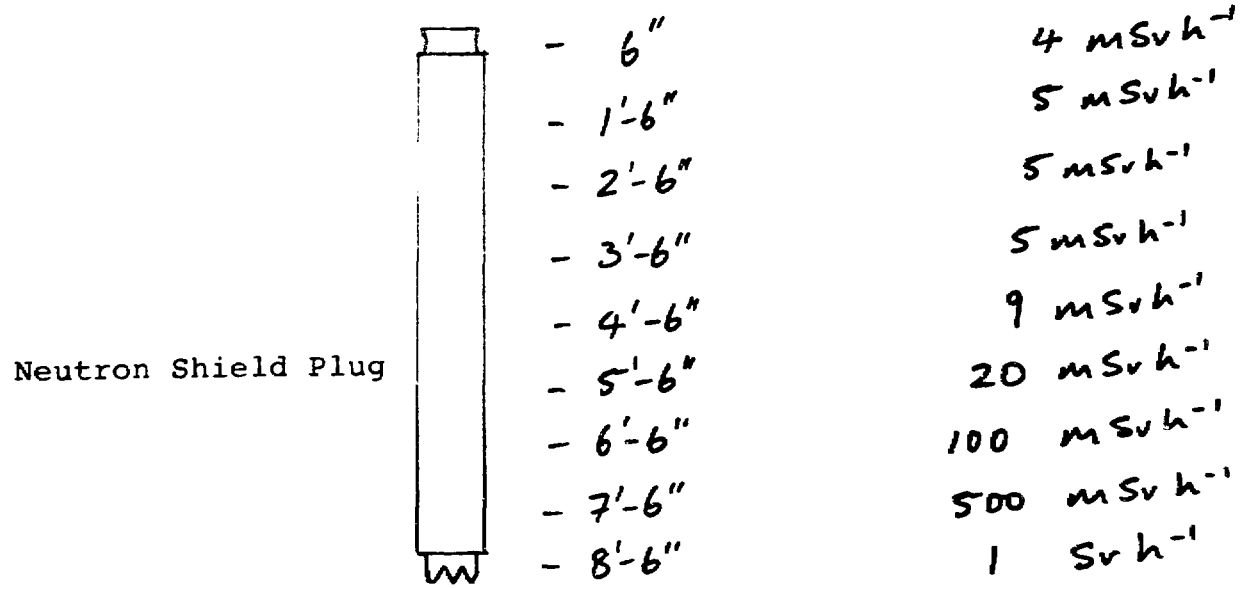


Fig 9 RADIATION PROFILES

PROPOSED DECOMMISSIONING OF RADIOACTIVELY CONTAMINATED NaK*

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ABSTRACT

This paper deals with a proposed method for stabilizing radioactively contaminated eutectic sodium/potassium (NaK) liquid metal. Approximately 680 liters (180 gal) of contaminated liquid NaK were generated in 1955 during testing with the Experimental Breeder Reactor (EBR-I) at the Idaho National Engineering Laboratory (INEL). Reaction of the NaK with chlorine gas to produce solid salts of sodium and potassium is proposed as a means to stabilize this waste. Preliminary testing was initiated to determine the reaction conditions required for this process. It was found that reaction with chlorine is feasible for safely treating the liquid metal.

INTRODUCTION

One of the more complex decommissioning projects at the Idaho National Engineering Laboratory (INEL), a U.S. Department of Energy (DOE) laboratory, involves the decommissioning of approximately 680 liters (180 gal) of eutectic sodium/potassium (NaK) liquid metal (78 weight percent potassium). This NaK is contaminated with transuranic (TRU) radioactive materials. The contaminated waste was generated in 1955 as a result of an incident at the Experimental Breeder Reactor (EBR-I), one of the world's first nuclear power reactors. The NaK was used as the primary coolant for the EBR-I.

The contaminated NaK was drained from the core region and placed into two 210 liter (55 gal) Mine Safety Appliance (MSA) stainless steel containers and two containers fabricated from carbon steel pipe sections. The contaminated NaK was stored at the EBR-I site until 1974, at which time it was moved to its present location, an underground bunker located near the center of the INEL.

*Work supported by U.S. Department of Energy, Surplus Facilities Management Program, under DOE Contract No. DE-AC07-76ID01570.

The radioactive contamination in the NaK is believed to include approximately 15 grams fission product radionuclides. The NaK may also contain a 10.5 gram foil of plutonium, which was inside the reactor but not accounted for when the core was decommissioned. There is probably also some nonradioactive zirconium fuel rod cladding material in the NaK. Due to the fission product contamination of the NaK, radiation levels are high. Measurements made during the 1974 movement of the NaK showed contact radiation levels of from 500 mR/hr to 7 R/hr.

In the past, liquid metal coolant materials have been decommissioned by reacting the liquid metal with steam under a nitrogen atmosphere. Although this procedure has proven effective, safety concerns exist over hydrogen gas evolution. The required filtration (HEPA) would pose significant radiological hazards should water vapor condense on the filters thereby rendering them ineffective. Because of these safety concerns, a study was undertaken to investigate other possible chemical methods to decommission the NaK. This study investigated several possible chemical schemes and assessed these possibilities using the following evaluation criteria:

1. Safety
2. Environmental impact
3. Waste form stability
4. Waste volume
5. Project costs
6. Service requirements
7. Processing rate
8. Simplicity of process scheme

The chemical process schemes evaluated included steam/nitrogen, oxygen, alcohols, sulfur, acids, and halogens (chlorine and bromine). An analysis of the merits of the various process options was performed. The results of this analysis indicated that the reaction of NaK with chlorine to produce NaCl and KCl was the optimum process option to be pursued under this project. The major reasons for this selection were improved safety, no by-product formation, and the potential for waste volume reduction.

A literature survey was undertaken to determine if previous empirical work had been performed on this reaction scheme, but the literature search failed to uncover any previous work on this process. A thermodynamic model was developed to bound possible reaction rates and chemical reactor size. Analysis of the modeling results indicated that information on salt density, heat capacity, and thermal conductivity were required before the reactor design could proceed.

PROCEDURE

The chemical reaction between NaK and chlorine yields two salts, sodium chloride (NaCl) and potassium chloride (KCl). Eutectic NaK is 78% potassium by weight. This ratio between the two metals translates into a molecular formula for eutectic NaK of approximately NaK_2. Selected

physical properties for the major species involved in this reaction are seen in Table I.

TABLE I. SELECTED PHYSICAL PROPERTIES

	Melting Point °C	ΔH rxn with Cl_2 (Kcal/mol)	Density at 20°C (gm/cm³)
Na	97.8	98.32	0.968
K	63.2	104.3	0.855
Eutectic NaK	-12.6	307.0	0.867
NaCl	801	-	2.164
KCl	368	-	1.988

Three ways of combining the chlorine and NaK were identified: gaseous chlorine into pooled NaK, NaK sprayed into gaseous chlorine, and NaK sprayed into liquid chlorine. Initial experimentation was performed for gaseous chlorine contacting pooled NaK, as this method was the least complicated and could alleviate processing complications on the full-scale system if successful in the laboratory.

The experimental reactor is constructed of stainless steel and has an interior volume of 8.0 liters. For preliminary experiments with pooled NaK, a 250 ml stainless steel beaker was placed inside the reactor to contain the NaK.

Instrumentation includes three thermocouples in a sheath installed axially in the reactor. Temperature measurements are obtained at 1, 12, and 24 cm from the reaction zone. Pressure is determined by means of a pressure transducer and a gage. A load cell on the NaK feed reservoir is used to measure NaK input to the reactor.

It must be mentioned that the design of the experimental facility was intended to provide only qualitative results concerning the feasibility of the process. The reactor was designed to withstand severe pressure and temperature loads anticipated during this and subsequent experiments. Rates of reaction reported herein are based on the consumption rate of chlorine and the cross-sectional surface area of the metal beaker. The rate of reaction was difficult to estimate when the reaction occurred at subatmospheric pressures, as the provision had not been made to read vacuum pressure. In such cases, reaction rate was determined from temperature rise time and initial NaK quantity.

EXPERIMENTAL RESULTS AND DISCUSSION

Results of experiments conducted with chlorine into pooled NaK are displayed in Table II. A discussion of each experiment is included below.

TABLE II. SUMMARY OF EXPERIMENTS

Experiment	Date	Conditions*	Results	Comments
1	4/15-4/25	~70 gm NaK $[Cl_2]$=0.05-0.52 T=21 - 35°C	Slow conversion to KCl, reaction rate from 44 to 7 mg NaK/m^2s	Insufficient time allotted to detect reaction, more NaK added until visual inspection revealed ~2.5 cm NaK in beaker. Metal (solid, liquid) remaining in beaker probably non-eutectic NaK.
2a	4/30-5/4	~10 gm NaK $[Cl_2]$=1.00 T~43°C	Some air in purge line. Slow conversion, reaction rate of 10 mg NaK/m^2s.	Yellowish deposit observed on visual inspection. Reassembled reactor for cleanup of remaining NaK. Heating coils used for remainder of experiment.
2b	5/4-5/5	Same	Very rapid temp. rise to full scale (540°C) pressure drop to subatmospheric. Reaction rate of 18×10^3 mg NaK/m^2s.	Reaction occurred 15 minutes after heating coilsturned on. Intent had been to dispose of NaK for next experiment. Apparent condensate (frozen) on metal beaker, purple in color. Purple coating on reactor walls, very acidic when H_2O added (to wash). Bottom TC melted off. SS beaker 114-->93 gm.
3a	5/7-5/8	~10 gm NaK $[Cl_2]$=1.00 Tm~51°C	Slow pressure drop (leak).	Experiment halted due to Cl_2 leak through damaged TC thermowell. Repaired leak. Greenish/yellow deposit on inside reactor walls.

TABLE II. Continued

Experiment	Date	Conditions*	Results	Comments
3b	5/11-5/12	Same	Slow reaction, 22 mg NaK/m^2s.	Exposed 0.09 gmol air to partially reacted NaK from Exp. 3a, slight temp. rise (17^oC).
4a	5/12-5/13	Unknown NaK $[Cl_2]$=1.00 $\bar{T}m$-49^oC	Slow but accelerating consumption of Cl_2 rate of 100 mg NaK/ m^2s.	$8.6x10^{-5}$ gmol air added, no effect.
4b	5/13	Same	Rapid temp. rise, P decrease, reaction rate of $75x10^3$ mg NaK/m^2s.	More air added (0.0014 gmol). Cl_2 added after evacuation of reactor. Reaction occurred immediately. Additional Cl_2 reacted with SS beaker (only bottom remained).
5a	5/18	Unknown NaK $[Cl_2]$=0.50 T=124^oC	Slow P decrease, reaction rate of 130 mg NaK/m^2s.	First run with heating tape, relatively slow reaction perhaps due to lower chlorine concentration.
5b	5/19	8 cm NaK added $[Cl_2]$=1.00 T = 46^oC	Rapid P rise, P decrease when T ~46^oC T_t peak=227^oC rate of reaction = $17x10^3$ mg NaK/m^2s	Additional NaK introduced without breaking reactor. Reaction could be due to 100% Cl_2 or 'seed'of already precipitated material.

TABLE II. Continued.

Experiment	Date	Conditions*	Results	Comments
6	5/20	15 gm NaK $[Cl_2]$=1.00 T = 53°C	21°C T rise on addition of Cl_2, reaction rate of 330 mg NaK/m^2s	Heating tapes not functional. Slow pressure decrease.
7	5/29	11 gm NaK $[Cl_2]$=1.00 T = 56°C	36°C T rise on addition of Cl_2, reaction rate of $40x10^3$ mg NaK/m^2s	Repeat of Exp. 6, new heating tapes and TC assembly. White deposit (salt)with some NaK included.

*T_b,T_m,T_t - Temperature measured by bottom, middle, and top thermocouples, approximately 1, 12, 24 cm from reaction zone, respectively.

Experiment 1

Approximatel' 70 gm of NaK was introduced to the reactor for this experiment, although it was intended to react a much smaller amount. Uncertainty as to the quantity actually reaching the reaction vessel through initially empty feed lines contributed to the reactor containing this quantity of NaK. The reactor was filled with argon gas at atmospheric pressure. Chlorine gas was introduced to the reactor and was consumed at a decreasing rate corresponding to a reaction rate of from 44 to 7 mg NaK/m^2s (0.032-0.005 lb/ft^2hr). The occurrence of a reaction was evidenced by a slow decrease in reactor pressure accompanied by an initial temperature rise (6°C). During this phase of the experiment, coils around the bottom of the reactor, originally intended for cooling, were used for heating the reactor in an attempt to increase the rate of reaction. This heating, to approximately 46°C, had no noticeable effect on the rate of reaction. Increasing chlorine pressure increased the reaction rate slightly, suggesting a process involving diffusion of gas through a reacted solid layer. When the addition of chlorine had no further effect on the reactor temperature or pressure, the reactor was opened for visual inspection. About 40 percent of the stoichiometric amount of chlorine to react the 70 gm of NaK had been consumed at this point.

Examination of the contents of the beaker revealed a layer of pea-sized brown and purple solid. The solid appeared very porous and contained some inclusions of unreacted metal. This was evidenced by the vigorous reaction that occurred when the solid was dissolved in water. Analysis of this solid by X-ray diffraction detected only KCl. Beneath the surface of the solid in the beaker was a mass of partially reacted metal. A small amount of liquid metal was also observed, but the beaker contained mostly solid. This solid material seemed to react with the residual oxygen in the glove box used to provide a blanket of argon gas) to form a white/gray solid which was also reactive with water. It was thought that potassium had reacted preferentially with the chlorine, which left an unreacted sodium-rich alloy in the bottom of the beaker. The solid alloy prevented the more dense solid product from sinking to the bottom of the beaker. It was thought that diffusion of chlorine through the solid salt layer had caused the slow reaction rate.

Experiment 2a

This experiment was temporarily halted due to plugged feed lines. The lines were cleared, and a purge line installed, which necessitated introduced approximately 0.002 gmol of air into the feed system which contacted a small quantity of NaK contained in the beaker. A reaction occurred with the NaK and this air purge (or more specifically, oxygen), indicated by a white smoke which was vented from the reactor. After first evacuating the argon gas, chlorine was fed into the reactor. A very slow decrease in chlorine pressure indicated a slow reaction rate (10 mg NaK/m^2s). The reactor was opened and the beaker contents examined,

revealing a deposit of purple, white and yellow solid in addition to liquid metal. These compounds could possibly be ferric chloride, salts of sodium and potassium, and potassium superoxide, respectively.

Experiment 2b

The purpose of this exercise was to completely react the NaK remaining in the beaker in preparation for another experiment. Heating coils were initially turned off for this experiment, as they had been shown to have little effect in Experiment 2a. After several hours, however, it was decided to use the heating coils in an attempt to increase the reaction rate.

About 15 minutes after the heat had been applied, a temperature rise to over 540°C was observed on the instruments, accompanied by a sudden loss of reactor pressure to below atmospheric pressure. The NaK reacted at a rate of approximately 18×10^3 mg/m^{2}s, or about three orders of magnitude faster than during the previous experiment. Inspection of the metal beaker revealed a condensed (frozen) deposit on the beaker exterior. The beaker also contained a solid mass which was essentially 100 percent converted, and which contained some traces of iron. The deposit was also less porous than obtained during the previous experiment. A loss of approximately 21 gm of the stainless steel beaker was detected, although the reactor vessel itself was not visibly changed.

It was thought from this experiment that oxygen could serve to promote the reaction being considered. It is possible that inspection of the beaker contents (Experiment 2a) had brought metal oxide into contact with unreacted metal, allowing a reaction such as the following to occur:

$$2M + O_2 \rightarrow M_2O_2; \quad M_2O_2 + Cl_2 \rightarrow 2\,MCl + O_2$$

$$M + O_2 \rightarrow MO_2; \quad MO_2 + 1/2Cl_2 \rightarrow MCl + O_2$$

(M being either Na or K metal)

A number of other mechanisms could be postulated, each involving oxygen in a chain reaction.

It is also possible that during the inspection of the beaker contents in Experiment 2a, sufficient unreacted NaK was exposed to the surface for reaction to occur. The heat evolved from this reaction could have melted the solid alloy, allowing the salt to sink to the bottom, exposing more NaK for reaction with chlorine.

Experiment 3a

This experiment was halted due to a chlorine gas leak which developed in the thermocouple sheath.

Experiment 3b

This experiment was intended to fully convert the partially reacted NaK from Experiment 3a. Approximately 0.09 gmol of air was introduced to the reactor, followed by evacuation of the vessel and introduction of chlorine. No detectable reaction occurred upon air addition. However, a 17^{o}C temperature rise was detected while adding chlorine to the reactor (thermocouple 12 cm above reaction zone). The reaction rate was about 22 mg NaK/m^{2}s, or about the same as the average rate during Experiment 1. Inspection of the interior of the reactor vessel revealed the same purple-colored solid material produced in previous experiments. The product was only slightly reactive with water.

Experiment 4a

The purpose of this experiment was to expose air to unreacted NaK. After the reactor which had been evacuated, approximately 8.6x10^{-5} gmol of air was introduced followed by 0.45 gmol of chlorine. A slow reaction (100 mg NaK/m^{2}s) was detected, evidenced by a gradual decrease in reactor pressure.

Experiment 4b

For this experiment, the reactor was purged with argon gas, and 0.0014 gmol of air was introduced to the reaction vessel. The reactor was then evacuated and 0.42 gmol of chlorine was introduced to the reactor. The reactor pressure dropped immediately to subatmospheric and temperature rose and peaked at 620 and 440 ^{o}C on thermocouples 12 and 24 cm from the reaction zone, respectively. The reaction rate was approximately 75 x10^{3} mg NaK/m^{2}s. Chlorine was fed continuously into the reactor until the temperature peaked and subsequently declined for several minutes. Upon inspecting the inside of the reactor, it was found that the stainless steel beaker containing the NaK had itself almost completely reacted with the chlorine. The excess chlorine had apparently consumed the stainless steel as well as the NaK, resulting in a deposit of brownish powder (perhaps $FeCl_3$). Again, nearly 100 percent conversion to salt was achieved.

Experiment 5a

This experiment involved the use of electrical heating tape which enabled heating to much higher temperatures than the hot water heating coils (120^{o}C as opposed to 46^{o}C). It was thought that the reaction would proceed at a faster rate, and chlorine concentration was held at 50 percent. A steady pressure decrease was observed, indicating a reaction rate of approximately 130 mg NaK/m^{2}s.

Experiment 5b

This experiment involved the addition of 10 gm of NaK to the quantity of NaK already in the reactor followed by the introduction of chlorine. The heating tape was turned on after the chlorine was added, and a rapid temperature rise and pressure decrease was noted as the upper thermocouple (24 cm above the NaK) approached 46°C. The upper thermocouple peaked at 225°C, while the lower thermocouple (24 cm above the reaction zone) exceeded 1090°C. The reaction rate was approximately 17 $x10^3$ mg NaK/m^2s, quite comparable to Experiment 2b. It is believed that the reaction might have been facilitated by the precipitate already in the reaction beaker, and not due entirely to the heating tape and pure chlorine in the reactor. Examination of the contents of the beaker revealed a completely reacted purple/brown solid. Approximately 13 gm of stainless steel beaker was lost to reaction with chlorine.

Experiment 6

This experiment was deemed unsuccessful due to malfunctioning heating tapes. Upon chlorine addition, a temperature rise of 21°C and slow pressure decrease were noted, yielding a reaction rate of approximately 330 mg NaK/m^2s.

Experiment 7

During this experiment, an immediate temperature rise of 36°C was noticed upon first introduction of chlorine to the reactor. The reactor had been evacuated after the introduction of 11 gm NaK. The rate of reaction was about 40 $x10^3$ mg NaK/m^2s. Introduction of additional chlorine had no further effect on temperature, and it was decided that the reaction was complete. The reactor was opened and a white deposit was found inside the beaker. Some inclusions of NaK were evidenced by the reactivity in water, and the faint bluish/gray color of some of the deposit, which had been shown to be reactive in previous experiments.

CONCLUSIONS AND RECOMMENDATIONS

These initial experiments with NaK and chlorine have demonstrated that exposing gaseous chlorine to pooled NaK can be an effective means of liquid metal stabilization. The reaction proceeds at relatively slow rates (approximately 50-100 mg NaK/m^2s) at room temperature in 100 percent chlorine. The reaction appears to form a solid top layer which slows further reaction by inhibiting gaseous diffusion. Increasing pressure can overcome this resistance only slightly. Potassium was found to react with chlorine preferentially to the reaction of sodium with chlorine. This initial extraction of potassium alters the Na/K ratio such that the unreacted metal partially solidifies, preventing the solid product from sinking to the bottom of the reaction vessel. This slows the reaction because of gaseous diffusion through the salt layer which must

take place. This results in not only a slow process, but significant inclusion of unreacted NaK in the resulting product.

It was found that heat applied to the reactor vessel increased the reaction rate to 17-75 $x10^3$ mg NaK/m^2s. Reaction at higher rates appeared to more fully convert the NaK to salt. This effect could be explained in two ways. (1) The exothermic reaction under investigation might require heat input to overcome the activation energy necessary for spontaneous reaction. (2) The increased reaction rate with heat addition is because the NaK alloy, depleted of some potassium, partially solidifies at room temperature and retains a passivating layer. This layer melts upon heat addition and allows a much faster reaction to occur as the product presumably sinks to the bottom and unreacted NaK is exposed to the chlorine.

Oxygen introduced to the reaction system appeared to promote the reaction to higher rates than by simply heating the vessel. This is possibly due to the formation of metal oxides which have a higher reactivity with chlorine and may participate in a chain reaction such as

$$M + O_2 \rightarrow MO_2 + 1/2\ Cl_2 \rightarrow MCl + O_2.$$

Further experimentation is necessary to determine the mechanism of salt formation prior to design scale-up. These experiments should include the following:

1. Reaction of larger quantities of NaK (50-70 gm) at temperatures in excess of 98^0C (m.p. Na) in order to substantiate the theory of solid diffusion limited reaction.

2. Replication of Experiment 1, with temperature being raised to above 98^0C after some reaction has occurred.

3. Introduction of metal oxide to determine the extent of oxygen participation in a chain reaction.

SECTION V

PROJECT PLANNING, MANAGEMENT, AND COST ESTIMATING

ESTIMATING VOLUME OF CONTAMINATED SOIL FOR EXCAVATION; A QUICK AND SIMPLE METHOD

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ABSTRACT

Estimates of the volume of radionuclide contaminated soil which must be excavated require data on the depth distribution of the contamination. Using a novel uncomplicated approach, vertical activity distribution data were obtained on a 1400 m^2 site. The volume of contaminated soil which had to be excavated was estimated to be between 227 and 425 m^3. When the site cleanup was done, 312 m^3 of contaminated soil had to be removed to satisfy the cleanup criteria.

The basic approach was to drill 193 holes over the entire site and to make measurements in each using a detector with a side-looking collimator. Soil samples from the holes were analyzed only to the extent necessary to obtain sufficient data for demonstrating a correlation between count rate and activity concentration in the soil. The site was then conceptually sliced into vertical strips of contamination concern. The product of the contamination depths which exceeded cleanup criteria and the areas of the slices gave the volume of excavation required.

INTRODUCTION

Reliable and cost effective estimates of the volume of radionuclide contaminated soil which must be excavated to achieve a given radiological criteria is often a major concern in site D&D projects. Unfortunately, accurate estimates of contaminated soil volume are usually difficult, time consuming and expensive. A major problem has been the inability to economically obtain sufficient data on the vertical (depth) distribution of the contaminant in soil. There is currently no simple or reliable in situ method to infer subsurface vertical distribution of radioactive contamination in soil from surface measurements only. In situ gamma analysis systems can accurately measure radionuclide concentrations in soil if the depth distribution is already known.[1,2] Also, most radionuclides which are covered by more than about 25 cm of soil are essentially hidden from efficient detection at the surface.[3] Consequently, attempts to predict contaminated soil volume from surface radiation measurements only and/or from in situ measurements in a "few" vertical holes can differ by orders of magnitude from what is actually present. The methodology described here for obtaining vertical distribution data and subsequent volume estimates are different from traditional procedures primarily in the approach to the problem and in the way that time and resources are used to obtain data.

The advantages of the alternative approach becomes evident when contrasted to more traditional approaches. Traditional approaches to obtain vertical distribution data over large areas usually involve the following steps:

1. Make exposure rate and/or gross count rate, and/or in situ spectral measurements on grid points (grid intersection) over the entire site with radiation detectors held above ground surface.

2. Take additional above surface measurements on smaller grid in suspect areas.

3. Dig shallow trenches, spread spoils between trenches and make in situ measurements above soil piles.

4. Take soil samples from walls of trenches. Perform laboratory analyses on all samples collected to determine radionuclide composition and concentration depth distribution to depth of trench in sampled area. In addition to trenching, or as an alternative, take a few core samples in suspect areas and perform laboratory analyses on entire length of soil core (in segments).

5. Gamma log the core hole. That is, a detector is placed down the core hole and radiation measurements are taken at depth increments. The detector is usually uncollimated or has a down-looking collimator (coaxial with the vertical hole).

By contrast, the alternate approach described here has the following steps:

A. The entire surface of the site is scanned with radiation detectors to identify and document locations of "hot spots". Discrete measurements are also taken at grid points. The detectors are held above the ground surface.

B. Soil corings are not taken. Rather, augured 10 cm diameter holes are drilled over the entire area at equally spaced grid points and at all hot spots. The efficiency of the drilling crew is optimized because it is not impeded by sample collection and decontamination procedures.

C. Take count rate readings in all holes with a detector which has a side-looking (horizontal) rather than a down-looking collimator. At each depth increment, measurements are taken with collimator pointing at each of the points of the compass.

D. Take soil samples from the walls of the holes rather than from the center core. The number of samples collected and analyzed are dictated by the number of data points required to demonstrate a correlation between count rate in holes and activity concentration in samples from the same location. In some cases, the number of analyses required can be as few as 20 to 50.

TEST SITE

This alternate approach to obtaining vertical activity concentration distribution data for soil was developed and applied to the site of a

former radium dial painting factory in Ottawa, Illinois, U.S.A. The size of the site was approximately 15,000 ft^2 (1400 m^2). The D&D objective was to reduce the average ^{226}Ra contamination at all depths to less than 15 pCi/g (0.56 Bq/g) and to assure that the exposure rate everywhere on the site would be no more than 20 μR/h at 1 meter above the surface.

SURFACE RADIATION MEASUREMENTS

Three different types of radiation measuring instruments were used to make the surface measurements. Surface measurements include both contact measurements and measurements made at distances up to 1 meter above the ground surface level.

One measurement was made with a 2 mm thick by 50 mm diameter shielded sodium iodide detector connected to a single channel analyzer-rate meter (Eberline PG-2 probe with PRM-5-3 analyzer rate meter). An example of this instrument is shown on the left in Figure 1. The analyzer was operated in the gross mode (window completely open) with the threshhold set at 60 keV (using gamma-rays from ^{241}Am). This detector and analyzer setting combination were selected because they gave the best signal to background ratio. Surface count rate measurements made with this instrument give only a relative indication of radium in the soil and cannot be directly equated to subsurface radium concentration distribution. Typical background count rate in noncontaminated areas was 500 counts per minute. Measurements were taken at 2 ft intervals (0.6 m x 0.6 m grid) over the entire site with the detector at the soil surface (1 to 5 cm).

A second set of measurements was made of exposure rate at about 1 meter above the soil surface. The instrument (Eberline PRM-7 μR Meter) used for the measurements had a 25 mm x 25 mm sodium iodide detector. The center object in Figure 1 is an example of this instrument. Typical background in noncontaminated areas gave readings of 8 to 11 μR/h (micro roentgen per hour). As with the other instrument, measurements were taken at 2 ft (0.6 m) intervals over the entire site. The highest noted radiation exposure rate on the site was 215 μR/h. A total of less than 5% of the site indicated this radiation level.

A third instrument was used to measure alpha contamination on the soil surface. The instrument used (Eberline PAC-4G-3) had a 100 cm^2 gas-flow proportional detector with a 0.85 mg/cm^2 window thickness. An example of this instrument is shown on the right in Figure 1. The detector was held as close to the surface as practical. Measurements were, again, taken at each 2 ft (0.6 m) interval. The effective efficiency of the detector used in this mode is really unknown. The effective efficiency (count rate/disintegration rate) could be as low as 1% and probably never greater than 20%. Therefore, the alpha count rate data is not a true quantification of the alpha activity on the surface. However, it did indicate, that in some areas of the site, there was radioactivity directly on the surface. Under certain conditions, this surface activity could become windborne.

A total of over 20,000 surface radiation measurements were taken on the 1400 m^2 site.

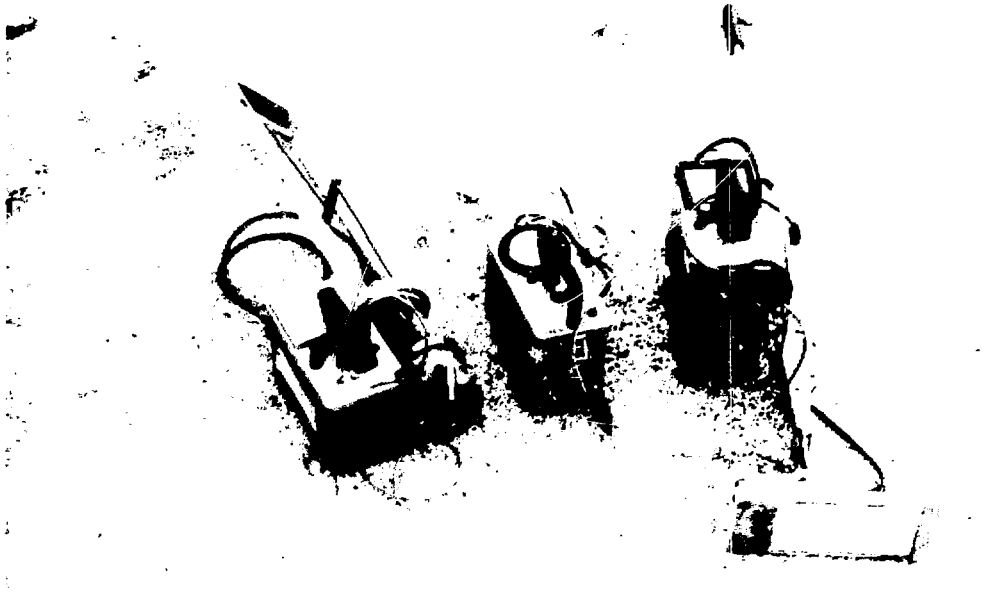

Figure 1 Instruments Used for Surface Radiation Measurements

SUBSURFACE IN SITU RADIATION MEASUREMENTS

Data Collection

As indicated in the introduction, a side-looking, rather than a down-looking collimator, was used with a 25 mm x 25 mm sodium iodide detector for taking count rate measurements at depth increments in holes drilled in the soil (downhole logging). Though the advantages of a side-looking collimator were seemingly intuitive, a test was done to measure the response of a detector to the same activity distribution when the collimator was down-looking and when the collimator was side-looking. Soil was simulated with concrete blocks and uniform activity distribution was simulated with a ^{137}Cs rod. The test was done by taking measurements parallel to a 30 cm long rod containing 7.4×10^7 Bq ^{137}Cs which was shielded from the detector by concrete blocks. The results are shown for two different shield thicknesses in Figure 2 for the down-looking collimator and in Figure 3 for the side-looking collimator. The results show (as expected) that with a down-looking collimator, a maximum count rate is seen before the detector reaches the edge of the upper end of the rod and then decreases at all depths. With a side-looking collimator, a maximum is seen at about the center of the rod. The change in count rate per depth increment is greater for the side-looking collimator than with the down-looking collimator. Also, the side-looking collimator has the additional advantage of having a lateral directional response. In general, data obtained using a side-looking collimator was considered easier to interpret.

A profile of the activity distribution in contaminated soil was obtained by making radiation measurements at different depths in holes

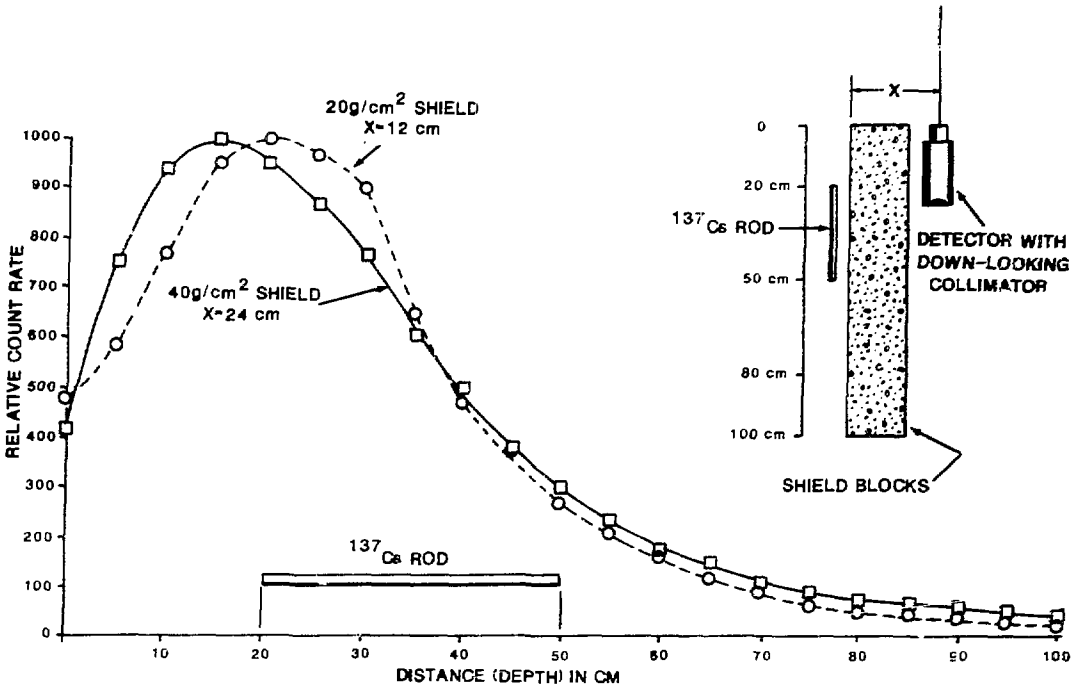

Figure 2 Response of 2.5 cm x 2.5 cm NaI(Tℓ) Detector With Down-looking Collimator When Moved Parallel to a Uniform ^{137}Cs Rod.

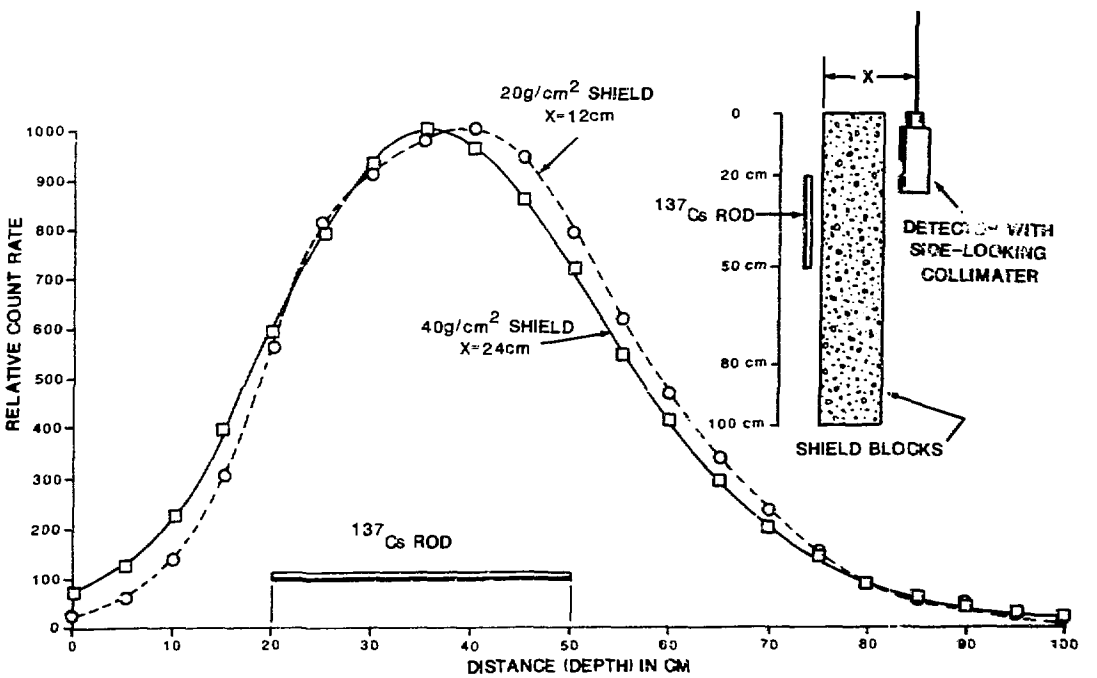

Figure 3 Response of a 2.5 cm x 2.5 cm NaI(Tℓ) Detector With Side-looking Collimator When Moved Parallel to a Uniform ^{137}Cs Rod.

drilled in the soil (downhole gamma logging). The instrument used to make downhole gamma measurements was a 25 mm diameter x 25 mm long sodium iodide detector connected to a single channel analyzer/rate meter (Eberline PRM-5-3). The analyzer was used in the gross mode (window completely open) with the threshold set at 60 keV. A sketch of the detector in its side-looking collimator is shown in Figure 4 and a photograph of the detector with its analyzer/rate meter is shown in Figure 5. The collimator was made of lead with a 1 mm thick steel outer sheath. The lead was 1.5 cm thick on the wall and 2.5 cm on the bottom. The aperture was a slit 2 cm wide by 3 cm long. The collimated detector was lowered into the 10 cm diameter holes, and measurements were taken at 15 cm increments. At each depth increment, a separate measurement was taken with the collimator pointing at each of the points on the compass; north, east, south and west. Directional differences in count rate give an indication that the hole may be near a sharp change in activity concentration or that it may be near a contaminated object such as a sewer line. The typical background count rate at the soil surface in known noncontaminated areas was 500 counts per minute. Samples of the downhole logging data are given in Table 1.

The depth profile of radium in the soil exhibited two different modes. In 26% of the holes logged, the count rate increased with depth, reached a maximum and then decreased. This mode of contamination is referred to here as a type 1 contamination profile (see Figure 6). The remaining 74% of the holes showed contamination profile where the count rate was maximum at the surface and decreased with depth (see Figure 7). The latter is referred to here as a type 2 profile.

For the case of the type 1 profile, one must infer that in some cases, contaminated subsurface soil may be covered with "clean" soil and that there was a relatively narrow band of uniform contamination centered at the depth corresponding to the peak count rate. This, of course, makes interpretation of any surface gamma measurements difficult. It also complicates the decontamination process.

One hundred and ninety-three holes of 10 cm diameter were drilled on the site for downhole logging. The holes were drilled at hot spots and more or less uniformly over the entire site. Drilling was done using a small drilling rig (Giddings Company Soil Explorer) mounted on a two wheel trailer. Figure 7 shows the drilling rig in use. While the trailer is usually pulled by a vehicle, it can be maneuvered into tight spaces by two or three people. An experienced operator was able to drill five to six holes per hour. Holes were drilled down to at least the sandstone strata, which varied from 60 cm to 120 cm in depth.

Correlation Between In-Hole Count Rate and Radium Concentration

To use the in-hole gamma count rate profiles to estimate excavation volumes, it was necessary to establish a correlation (at least a trend) between count rate at given depths and activity concentration in samples

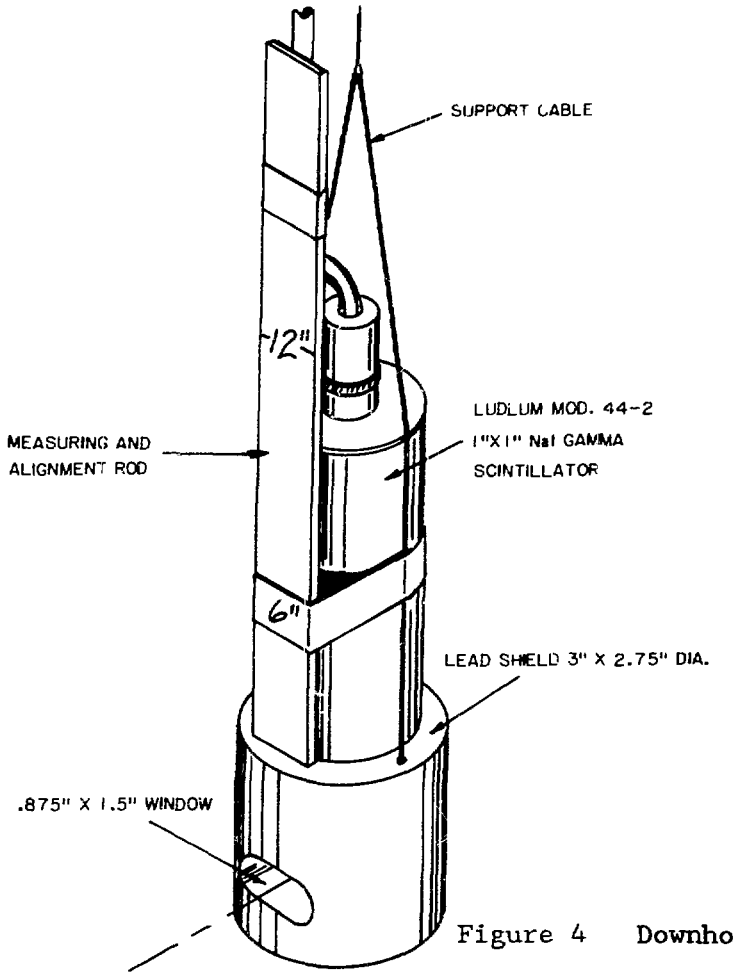

Figure 4 Downhole Logging Detector.

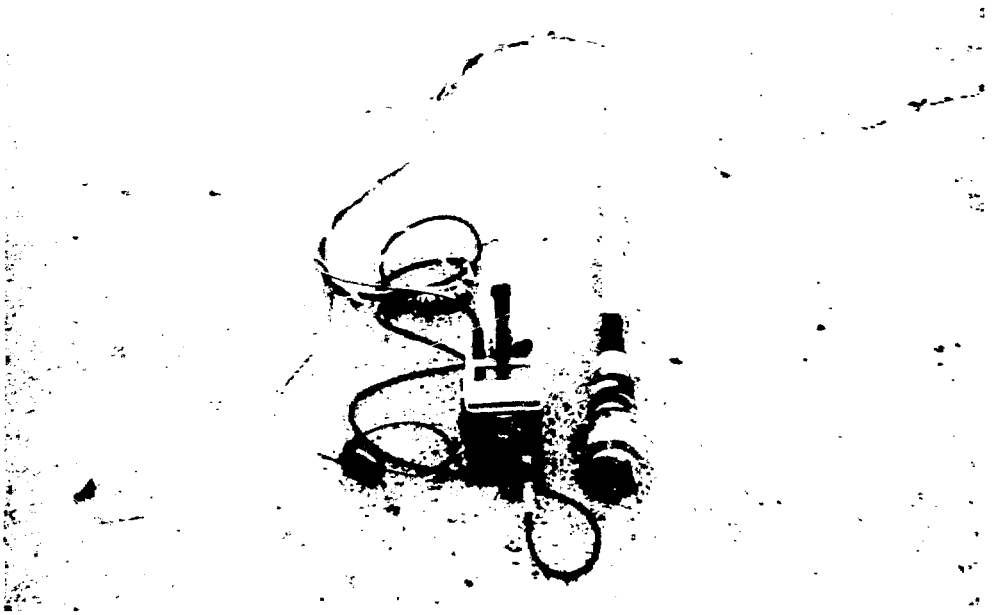

Figure 5 Downhole Logging Detector With Analyzer/Rate Meter.

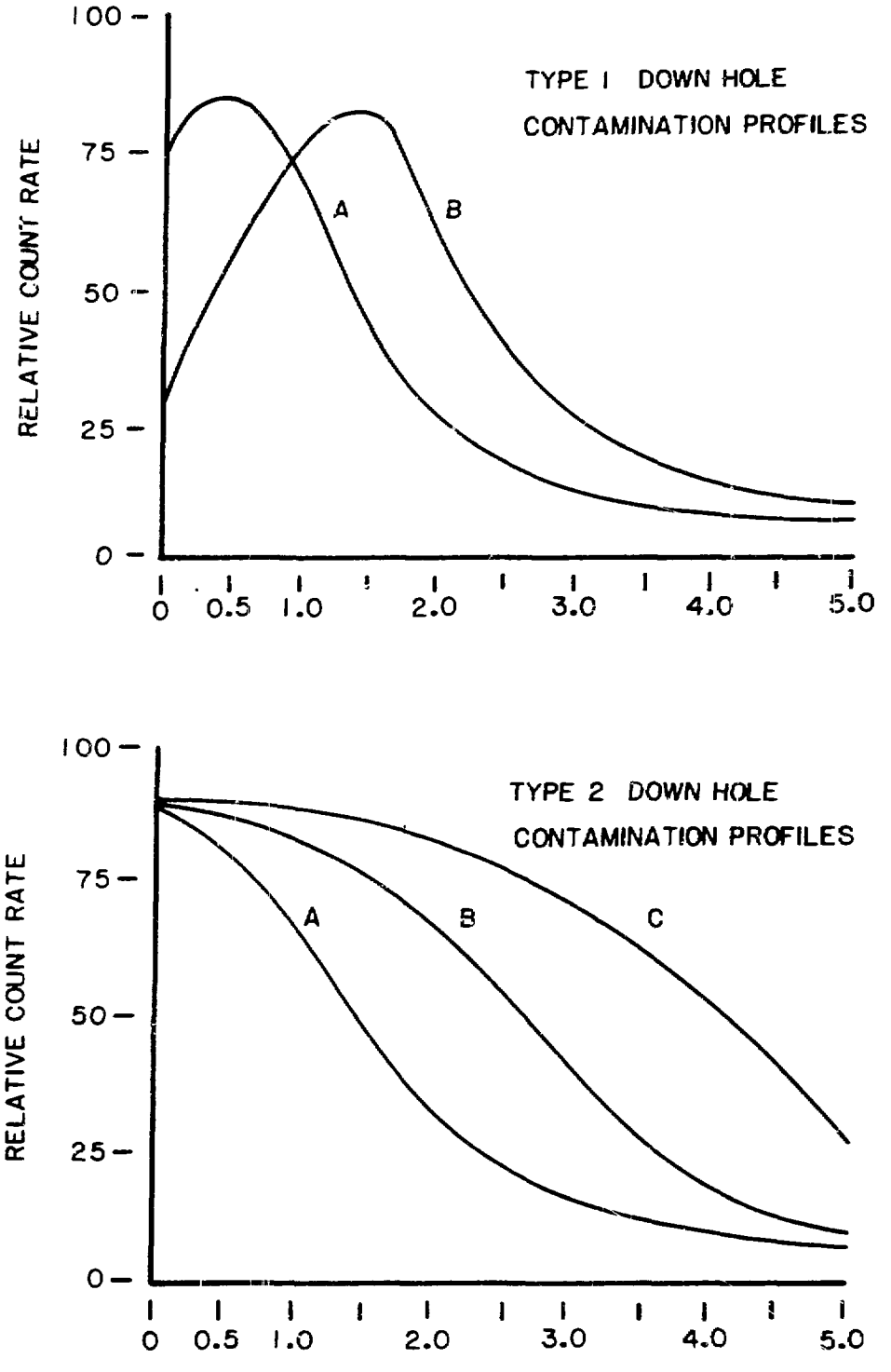

Figure 6 Types of Contamination Profiles Found on Site. (Axes are Relative Count Rate Versus Depth in Feet)

Figure 7 Drilling Rig

from the same location. No samples were collected until the count rate logging of all holes was completed. The in-hole count rate data were used to select sample locations which would give an adequate range of concentration values. Samples were then collected only from the walls of the hole. This approach offered some advantages:

- There is no ambiguity about the true depth interval of a sample.
- The volume of a sample is not limited by the diameter of a coring tube.
- Multiple samples can be collected from the same depth interval. If there are questions about the possibility of a sample being cross contaminated, that sample can be discarded and another one collected from the same depth interval.
- The rate of drilling is not impaired by the time required for sample collection and documentation.

Samples were collected from the walls of holes using a simple cup and scoop method. A cup was lowered into the hole and soil was scraped from the sides over a 15 cm depth increment and allowed to fall into the cup (see Figure 8).

The average count rate between depth increments was used for comparing the downhole count rate and the laboratory measured radium concentration.

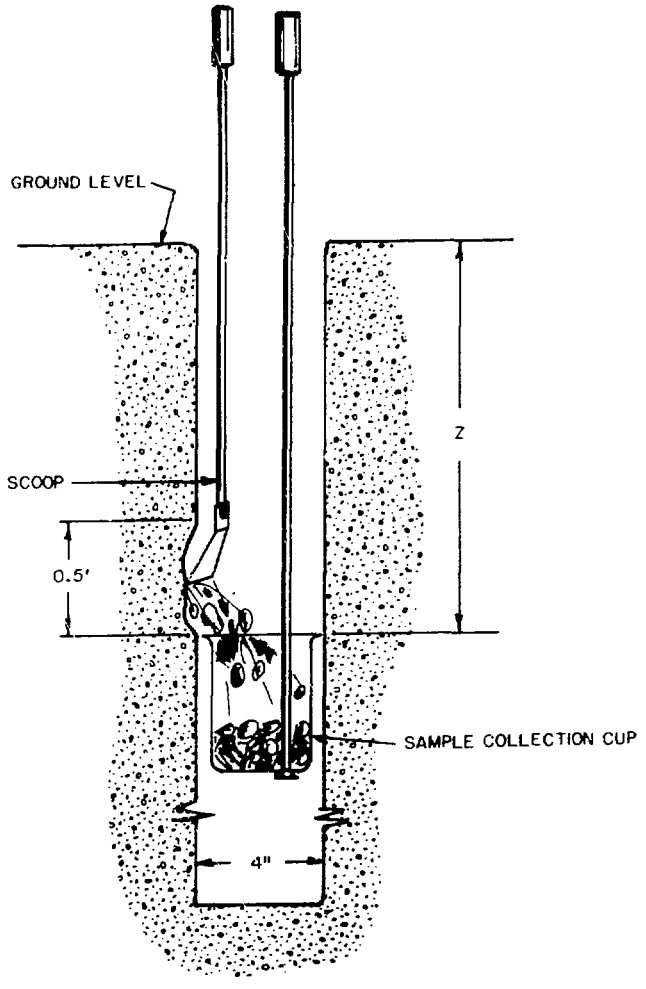

Figure 8 Method of Sample Collection From Logging Holes.

To keep sample collection cost and related effort to a minimum, an iterative process was used for sample selection and analysis, i.e., the results of each laboratory analysis were used as a basis for selecting the next sample to collect and analyze. This process continued until sufficient data were obtained to indicate a trend between downhole count rate and radium concentration in soil. Radium concentration in soil was measured with a HpGe detector and multichannel analyzer in an on site mobile laboratory. Measured activity was based on the count rate from the 186.211 keV gamma-rays from ^{226}Ra. The soil samples were placed on the detector in 500 cm^3 Marinelli beakers.

The results of downhole count rates and corresponding radium concentrations are given in Table 2. Some data in the Table were ignored because the data were obviously invalid. Those data are, however, shown to illustrate a point. For example, in Table 2, the concentration measured in sample 28-1.0 (soil sample #28 at 30 cm depth) was much too low. The fact that the count rate in the north direction at 30 cm depth

for this sample (see first entry in Table 2) was 5000 cpm more than the south and west directions, was a clue that something may be amiss. With the aid of a metal detector and minor excavation, a highly contaminated sewer line was found within 60 cm of the sampled hole. The count rate seen in the hole was due to the sewer line, and not to contaminated soil.

A plot of the valid data is shown in Figure 9. Based on the trend of the data, 1500 to 2000 cpm corresponds to approximately 5 pCi ^{226}Ra/g (0.19 Bq/g) and 3000 to 4000 cpm corresponds to approximately 15 pCi ^{226}Ra/g (0.56 Bq/g).

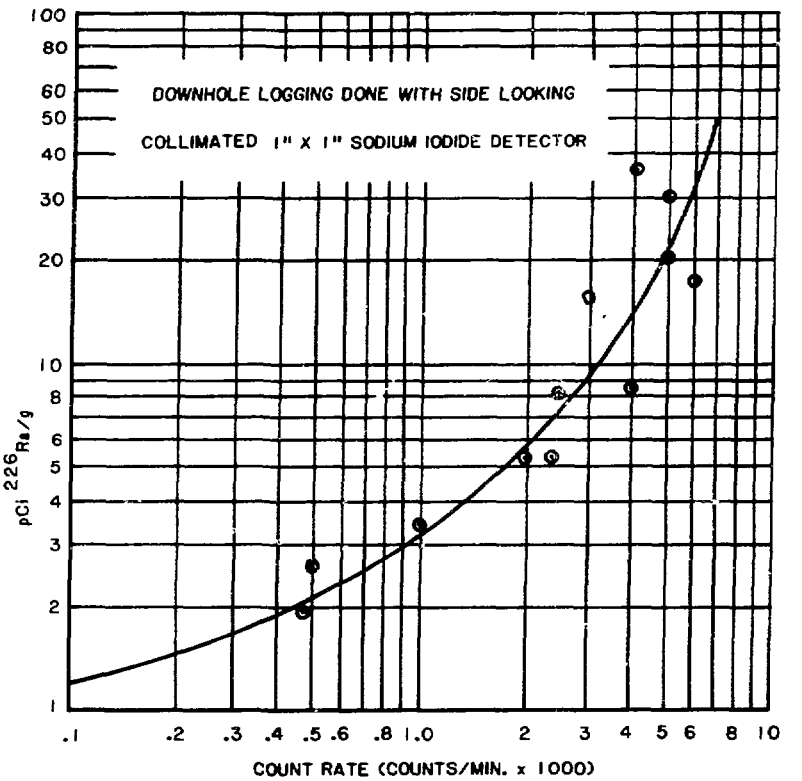

Figure 9 Effective Concentration in LPI Soil Versus Downhole Count Rate.

METHOD OF ESTIMATING CONTAMINATED SOIL VOLUME

The basic approach used in estimating the volume of soil which should be excavated to satisfy the cleanup criterion was to conceptually divide the site into cross sectional slices (see Figure 10) and to look for depths at which downhole count rates were no more than 3000 to 4000 cpm (this corresponded to approximately 15 pCi ^{226}Ra/g).

Each cross-sectional slice represented a 10-ft thick section and all of the downhole data as well as the surface radiation data within each

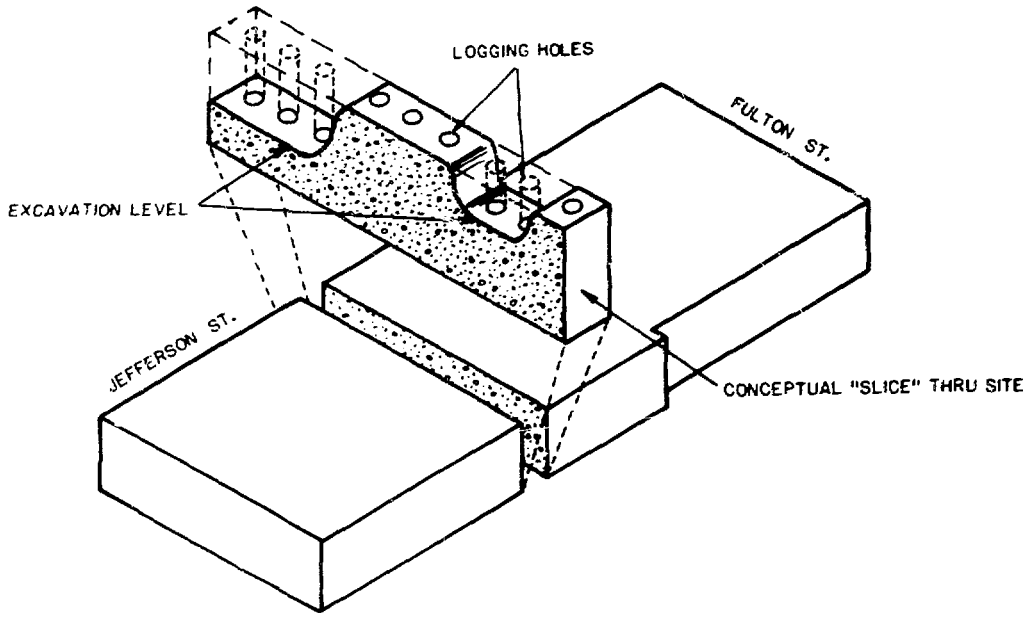

Figure 10 Conceptual Slicing of the Site to Estimate Required Excavation.

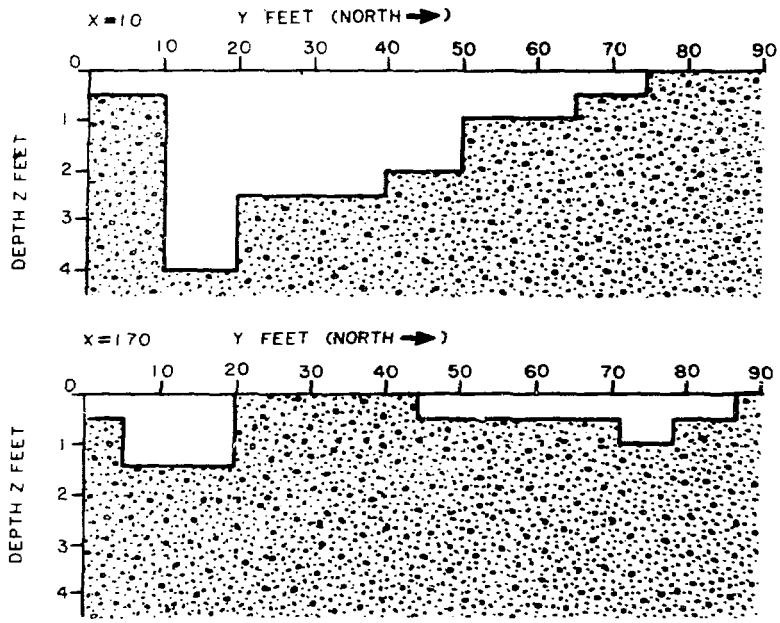

Figure 11 Two Sample "Slices" Through Site (Looking West) Showing Depths of Required Excavation at X Coordinates of 10 Feet and 170 Feet.

slice were examined. It is interesting that (because of type 1 contamination profiles) the highest surface measurements were not always in the areas requiring deepest excavation. The maximum depth to 15 pCi/g was plotted on a work sheet for each slice. Samples of two work sheets for slices taken at the X coordinates of 10 feet and of 170 feet are shown in Figure 11. The excavation depths within each 10-ft thick slice were transferred to the site plan at the corresponding X, Y coordinates for use by the site D&D crew.

The sum of all of the products of areas and depths gave the total volume of soil which must be excavated to reduce the ^{226}Ra concentration to 15 pCi/g. This volume of soil (which must be excavated to reduce the ^{226}Ra concentration to 15 pCi/g) was estimated to be a minimum of 8,000 ft^3 (227 m^3) and a maximum of 15,000 ft^3 (425 m^3).

The mechanics of computing the volume of contaminated soil is in principle the same as the borrow-pit method used by land surveyors to compute excavated soil volume.[4]

To prevent contaminating clean holes by waterborne contaminated soil (during rain) and by fill-in during future excavation work, all holes were backfilled with clean mason's sand and capped with about 3 inches of Portland cement.

COMPARISON OF ESTIMATED AND ACTUAL VOLUME

When the site cleanup was done (six months after estimate). 9,845 ft^3 (279 m^3) of contaminated soil had to be excavated to satisfy the cleanup criteria. The packaged burial volume was 11,830 ft^3 (335 m^3). The estimated volume was a minimum of 8,000 ft^3 (227 m^3) or a maximum of 15,000 ft^3 (425 m^3).

DISCUSSION AND CONCLUSION

The accuracy of estimating the volume of contaminated soil which must be excavated to achieve a specified cleanup criteria is often a mute point because the excavation contractor simply excavates exactly the amount of the estimate. In the cleanup of the site described here, every step of the excavation was guided by radiation measurements and on site laboratory analyses. The volume of soil actually removed was therefore independent of the initial estimate.

Concerns raised about the method described here are (1) that the method is practical only for small sites, (2) that it requires "too many" holes, and (3) that the simplicity of the method disappears if there are different criteria for each radionuclide in a mixture whose relative concentration is spatially variable. There is no technical reason as to why the method could not be applied to any size site. The accuracy of estimates for large sites may actually improve simply because there would be a greater number of volume slices to average. However, for a very large site it would probably be more efficient to divide the site into a number of areas and deal with each as a small site. The intuitive feeling

about "too many" holes is understandable, but it also has no technical or economic bases. The concern about dealing with a mixture of radionuclides whose relative concentration is spatially variable certainly complicates the problem of finding a correlation between in-hole count rate measurements and activity concentration. The problem is compounded further if the nuclides in the mixture have different cleanup criteria. In either case, it would be necessary to make gamma spectral measurements in the holes and generate a correlation curve for each radionuclide of concern. A spatially variable mixture of radionuclides adds to the complexity of any method of estimating contaminated soil volume. In the final analysis, it is a question of objectives and priorities. If it is important to obtain an estimate which is as accurate as possible, then the data on vertical distribution of activity concentration must be obtained as efficiently as practical. The method described here is therefore believed to be a viable alternative to traditional methods.

TABLE I

SAMPLE OF DOWNHOLE LOGGING DATA

SAMPLE[a] NO.	LOCATION X, Y, Z (FEET)		CPM X 1000[b] IN LISTED DIRECTION N	E	S	W
	146, 37,	0	14.0	10.0	5.0	5.0
		0.5	15.0	13.0	10.0	10.0
28-1.0- - - - - - - - - -		1.0	10.0	10.0	8.0	8.0
		1.5	3.5	3.5	3.5	3.0
		2.0	1.5	1.0	1.5	1.5
		2.5	0.5	0.7	0.7	0.7
		4.0	0.5	0.5	0.5	0.5
	10, 38,	0	4.0	4.0	4.0	4.0
		0.5	12.0	26.0	24.0	26.0
		1.0	16.0	17.0	16.0	15.0
		1.5	7.5	7.5	7.5	10.0
		2.0	4.5	4.2	4.2	4.6
		4.0	0.9	0.9	0.9	0.9
	74, 8,	0	10.0	10.0	10.0	10.0
		0.5	8.0	8.0	8.0	8.0
		1.0	3.9	4.0	4.0	4.0
		1.5	2.1	2.2	2.2	2.3
13-2.0- - - - - - - - - -		2.0	1.9	1.7	1.7	1.7
		3.0	1.4	1.3	1.2	1.3
	173, 6,	0	2.5	2.5	2.5	2.5
		0.5	6.0	6.0	6.0	6.0
		1.0	8.0	8.0	8.0	8.0
17-1.5- - - - - - - - - -		1.5	9.0	9.0	9.0	9.0
		2.0	7.0	7.0	7.0	9.0
		2.5	5.0	5.0	7.0	7.0
		3.0	5.0	5.0	5.0	5.0
	88, 25.5	0	0.45	0.45	0.45	0.45
		0.5	0.45	0.45	0.45	0.45
		1.0	0.45	0.45	0.45	0.45
		1.5	0.5	0.5	0.5	0.5
		2.0	0.5	0.5	0.5	0.5
		3.0	0.3	0.3	0.3	0.3
	146 87,	0	2.0	2.0	2.0	2.0
		0.5	10.0	12.0	12.0	12.0
		1.0	10.0	11.0	10.0	10.0
		1.5	5.0	7.0	7.0	7.0
25-2.0- - - - - - - - - -		2.0	5.0	5.0	6.0	6.0
		2.5	4.5	4.5	5.0	5.0

[a] Samples at the indicated location were taken after all holes were logged.

[b] At each depth, measurements were taken with the collimator opening pointing north (N), east (E), south (S), and west (W).

TABLE II

SUBSURFACE ACTIVITY CONCENTRATION AND DOWNHOLE COUNT RATE

SAMPLE NO.	LOCATION X, Y, Z (FEET)	DOWNHOLE GROSS COUNT RATE (cts/min x 1000)	^{226}Ra pCi/g	COMMENTS
1-0.5	190, 24, 0.5	0.45	2.6	
3-2.9	165, 24, 2.9	0.48	1.9	
14-1.5	92, 15, 1.5	0.5	2.6	
19-1.5	72, 26, 1.5	0.95	3.5	
15-2.0	131, 7, 2.0	2.0	8.4	
13-2.0	74, 8, 2.0	2.0	11.1	
18-1.0	180, 16, 1.0	2.4	5.3	
22-1.5	136, 85, 1.5	2.5	8.4	
5-1.0	173, 58, 1.0	2.7	15.7	
26-1.0	137, 38, 1.0	3.0	5.4	Near Hot Pipe
4-2.0	86, 7, 2.0	3.2	26.1	
8-1.0	28, 28, 1.0	4.0	8.4	
16-1.0	164, 17, 1.0	4.0	37.8	?
4-1.5	86, 7, 2.0	4.5	35.0	
7-1.5	27, 36, 1.5	5.0	21.6	
23-1.0	153, 107, 1.0	5.0	27.9	
6-2.5	75, 14, 2.5	5.5	22.9	
25-2.0	146, 87, 2.0	6.0	17.7	
11-1.5	18, 8, 1.5	7.5	46.1	
4-1.0	86, 7, 1.0	8.0	25.0	
17-1.5	176, 6, 1.5	8.6	95.5	
24-1.0	146, 82, 1.0	10.0	28.0	
28-1.0	137, 37, 1.0	10.0	2.1	Near Hot Pipe
10A-1.0	21, 33, 1.0	11.0	8.3	?
10B-1.0	21, 33, 1.0	11.0	10.5	?
4-0.5	86, 7, 0.5	19.5	243	
27-0.5	44, 45, 0.5	35.0	1180	Basement

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DECOMMISSIONING COST ESTIMATING AND CONTINGENCY APPLICATION

T.S. LaGuardia, PE

ABSTRACT

The funding of nuclear power plant decommissioning has matured into an integral part of utility planning. State public utility commission regulators and the U.S. Nuclear Regulatory Commission have recognized the need to assure the availability of funds to safely decommission these facilities at the end of their useful lives. The cost estimates for decommissioning need to reflect the changes in labor and material costs due to inflation, changes in waste disposal costs for packaging, transporting and burying radioactive materials, and the site-specific factors for each unit that account for differences in plant design and construction. Decommissioning activities involve remote tooling to segment the reactor vessel and internals, decontamination of contaminated systems to reduce occupational exposure, controlled blasting to demolish concrete structures, and removal and disposal of radioactive wastes by controlled burial. The unforeseeable problems encountered in performing these activities result in additional costs that are accounted for through contingency.

This paper summarizes the recent progress in nuclear power plant decommissioning cost estimation and contingency application. The important factors to be included in planning for the establishment of a decommissioning fund are identified, and typical results of recent estimates are provided. The nuclear industry is probably one of the first industries to plan for the eventual retirement of its facilities, and the public needs to be aware of these efforts.

INTRODUCTION

The decommissioning of power reactors has drawn much attention in the last few years. While the forecasted dates of when these large power plants will be taken off the line is decades into the future, the problems we will face in decommissioning exist or are being created now. Unless we provide the necessary funds to properly decommission these facilities at the end of their useful lives (whenever that occurs), we run the risk of shortfalls in funding. Such funding shortages could easily lead to decommissioning program delays and inefficiencies.

Most of us have traveled through older industrial cities where factories or process plants have been abandoned, leaving behind unsightly, deteriorating structures potentially contaminated with hazardous wastes. The federal government is now attempting to remedy the worst of the lot through its Superfund program - a program likely to go on for decades. Yet this is a program funded through taxed wage earners who may or may not have received any direct benefit from the facility when operational!

The individual taxpayer is not the only affected party. States are now being required to host the disposal of hazardous materials that never originated in their territories and, therefore, never contributed any tax income. We have learned from these early experiences to clean up after ourselves or, more importantly, to prevent spills or prohibit unrestricted dumping.

Recent decommissioning regulatory activity has necessarily focused on assuring that adequate funds will be available to safely retire power reactors and other facilities. Effective preplanning is needed to provide adequate funds to assure the work will be done safely. The U.S. Nuclear Regulatory Commission (NRC) has been developing regulations to assure funds for the safe removal of residual radioactivity. Public utility commissions in Maine and New Hampshire, for example, have similar laws to assure their utilities have provided adequate funds for each facility and thus avoid the state's bearing any unfair financial burden.

The estimated costs for the actual decommissioning of large power reactors has risen sharply in the last few years. This increase is essentially due to the present lack of burial space for radioactive and hazardous materials. Regulatory and financial pressure has been applied to force individual states to act upon the formation of regional compacts for waste disposal. In the meantime, the Low-Level Radioactive Waste Policy Amendments Act of 1985[1] has imposed a surcharge at Barnwell, SC, Beatty, NV, and Richland, WA on all radioactive waste generators from states that have not yet formed a regional compact. Additional surcharges will be imposed if regional waste disposal sites are not operational by the year 1992.

Another reason for the rise in decommissioning costs is increased attention to controls for protecting the worker and general environment. Extending current applications of ALARA principles at operating reactors into the projected activities for decommissioning requires additional crew training, more extensive use of remote tools (costly and slower), the

installation of temporary shielding and a more extensive application of decontamination technology. Recent cost estimates include these factors.

Costs for waste packaging and transportation have also escalated. Public pressure and federal and state regulatory agencies are demanding greater assurance against spills during transport. Burial regulations mandate the isolation of deposited waste from the environment. The resulting costs of waste packaging and transportation to meet these stringent requirements have increased the costs for decommissioning.

To identify the depth of the decommissioning funding issue, we need to review recent federal regulatory requirements, public utility commission and Federal Energy Regulatory Commission involvement. The US NRC has issued Proposed Rules[2] on decommissioning funding, and has emphasized the importance of ALARA considerations on potential decommissioning exposures. The US Department of Transportation (DOT) regulations on packaging and transport of radioactive materials[3] result in costly shipments - an important factor driving the minimization of wastes generated. Title 10 CFR Part 61[4] requires the evaluation of waste form and packaging to assure radioactive wastes will be isolated from the environment. Each of these regulatory factors have a strong influence on decommissioning costs and need to be addressed by utilities in preplanning for decommissioning. These factors will be discussed further in the next section. Utilities have begun to cope with this situation and have taken several positive steps towards preplanning and informing the regulators and the public on the utility decommissioning issue. The next sections will describe the current regulatory issues and what is being done to anticipate and pre-plan for future decommissioning.

REGULATORY REQUIREMENTS

U.S. NRC Proposed Rule

The U.S. NRC began a review of its policies and rules for decommissioning in 1978. Up to this time the only guidance provided by the NRC was its Regulatory Guide 1.86[5], "Termination of Operating Licenses for Nuclear Reactors," last revised in June of 1974. The Regulatory Guide identified four methods of decommissioning acceptable to the Commission: Mothballing, In-Place Entombment, Removal of Radioactive Components and Dismantling, and Conversion to a new nuclear system or a fossil fuel system. These decommissioning alternatives are briefly described as follows.

Mothballing consists of putting the facility in a state of protective storage. In general, the facility may be left intact except that all fuel assemblies and the radioactive fluids and waste should be removed from the site. Adequate radiation monitoring, environmental surveillance, and appropriate security procedures should be established under a possession-only license to ensure that the health and safety of the public is not endangered.

Entombment consists of sealing all the remaining highly radioactive or contaminated components (e.g., the pressure vessel and internals) within a structure integral with the biological shield after having all fuel assemblies, radioactive fluids and wastes, and certain selected components shipped off-site. The structure should provide integrity over the period of time during which significant quantities of radioactivity remain with the material in the entombment. An appropriate and continuing surveillance program should be established under a possession-only license.

Removal/Dismantling consists of removing all fuel assemblies, radioactive fluids and waste, and other materials having activities above accepted unrestricted activity levels from the site. The facility owner may then have unrestricted use of the site with no requirement for a license. If the owner desires, the remainder of the facility may be dismantled and all vestiges removed for disposal.

Conversion consists of separating the original nuclear steam supply system from the generating system and disposing of the nuclear system in accordance with one of the previous decommissioning alternatives. The existing turbine-generator system may be powered by a new steam supply system.

The Commission held state workshops in 1978 and 1980 to gather public comments on existing regulations and to formulate NRC policy for decommissioning. Concurrently, the NRC sponsored several decommissioning studies of feasibility, cost, and environmental impacts for each type of reactor and process facility, and each decommissioning alternative. During this period, the Three Mile Island Unit 2 accident raised serious question to the ability of power utilities to pay for recovery from such accidents and to subsequently decommission these large reactors. This incident was the primary driving force behind the NRC's current policy to assure that adequate funds will be available to safely decommission nuclear facilities of all types.

The culmination of the NRC-sponsored studies was the publication of a Draft Generic Environmental Impact Statement

(DGEIS) on Decommissioning[6] (NUREG-0586). This report summarized estimated decommissioning costs for each type of facility and reported the estimated environmental effects, primarily the exposure to workers and the radioactive waste volumes generated. Of particular importance was the conclusion that indefinite mothballing or entombment was not recommended, and that decommissioning ultimately should include removal of all residual radioactivity from the site. The report further concluded that conversion was not really another decommissioning alternative but instead an alternative use for potentially salvageable turbine-generator equipment.

With these studies and the DGEIS in hand, the NRC drafted Proposed Rules on Decommissioning and published them for comment in the Federal Register on February 11, 1985[2]. These Proposed Rules defined decommissioning as the removal of "nuclear facilities to a level that permits release of the property for unrestricted use and termination of license."

The NRC proposed two methods to establish the cost of decommissioning for power reactors. The first is to prepare a site-specific cost estimate for each facility and to provide for adjusting funding levels over the life of the facility. The second method is to certify that at least $100 million (1984 dollars) would be available adjusted for inflation at the rate of twice the Consumer Price Index (CPI).

The Rules emphasized the need to assure the availability of funds for decommissioning. Four methods were proposed to accumulate funds, including:

1. Prepayment: Cash or other liquid assets deposited prior to startup in an account segregated from licensee assets.

2. External Sinking Fund: A fund into which periodic payments would be made and whose interest earnings, together with the deposits, would be sufficient to pay for decommissioning. The fund would be segregated from licensee access.

3. Internal Reserve: A negative net salvage depreciation method that allows the utility to invest the funds in utility assets that may later be used to float bonds to raise the capital to pay for decommissioning.

4. Insurance: An insurance policy may be purchased to cover ultimate decommissioning costs.

The $100 million certification amount in the proposed Rule was based on studies[7,8] prepared by Battelle Pacific Northwest Laboratories (PNL) during the period 1978-1980, and in an overview study by Professor Siegel at the Wharton School[9]. These were generic studies that did not reflect the site-specific factors that have a major cost effect on decommissioning. Moreover, the early studies did not sufficiently reflect the ALARA considerations for decommissioning workers. Such worker exposure considerations are routinely included in site-specific cost estimates.

The NRC has received 143 letters of comment on the Proposed Rule. Of those who commented on the $100 million certification amount, more than 80% of nuclear proponents and opponents were opposed to it. Both groups believed the $100 million was too low for large plants and did not reflect site-specific factors. It is important to note that the $100 million is only to cover the costs of removal of radioactivity and does not address the dismantling of remaining structures. Although not specifically stated in the Proposed Rule, it is believed the $100 million includes 25% contingency - a justifiable amount for this type of work. However, it then casts further doubt as to whether decommissioning can be accomplished for $80 million without contingency. Based on discussions with the NRC staff reviewing the public comments on the Proposed Rule, the NRC will likely delete or significantly increase the $100 million certification amount.

Recent decommissioning cost estimates for 1100 MWe reactors prepared by utilities and consultants are in the range of $170 million for PWRs and $220 million for BWRs. Demolition of nonradioactive structures and site restoration adds about $30 to $50 million in 1987 dollars. These estimates include 25% contingency.

ALARA Considerations

The NRC intentionally excluded from the Proposed Rule the issue dealing with acceptable levels of residual radioactivity for release of property for unrestricted use. The nuclear industry has long sought definitive guidelines from the NRC on this topic. Current guidelines are essentially based on precedent. The extent of contaminated/activated equipment and structure removal required for controlled burial could be increased if the NRC imposes more restrictive residual radioactivity levels. This could increase the cost of decommissioning to remove, package, ship and bury the radioactive materials. While the increase may not be great, the uncertainty created from the lack of federal guidance will require future updates to current decommissioning estimates. The NRC

plans to address this residual radioactivity issue. The Environmental Protection Agency (EPA) is currently developing a standard for the NRC to adopt for the regulation.

Just as the principle of keeping exposures "as low as reasonably achievable" (ALARA) is important for operating reactors, it is obviously equally as important for decommissioning. The activities involving exposure include decontamination, component removal, and container loading. The primary areas of occupational exposure, and therefore the areas where greatest emphasis should be placed, include:

Cutting and removal of piping and components
Removal of steam generators and pressurizers
Cutting and removal of the reactor vessel and internals
Cutting and removal of the reactor coolant pumps
Demolition of the activated concrete biological shield
Removal of components without segmentation.

However, some exposures are undoubtedly going to be unavoidable. The increased use of remotely operated tools (robots) can reduce some exposures (at perhaps higher cost), but current robots are not capable of climbing into a pipe chase and cutting pipe selectively so that other piping systems might remain operational while supporting decommissioning activities. While aggressive chemical decontamination of certain high radiation level systems may be necessary, it is generally not cost effective for lower radiation levels. Accordingly, some worker exposure may be unavoidable.

DOT Packaging Requirements for Transport

The U.S. DOT has established rules for the safe transport of radioactive materials in Title 49 of CFR, Parts 173-178.[3] These rules set limits on the quantities of radioactivity (curies), external radiation exposure rates and the number of packages that may be transported in various types of vehicles. They further specify rigorous tests each container must pass to be allowed for transport of radioactive materials. These federal requirements, coupled with individual state requirements to limit allowable payloads over the roadways during spring thaws, highlight the importance of efficient packaging techniques.

Several methods have been proposed to improve packaging efficiency and to reduce the generation of waste from decommissioning. Some of these methods are:

Incineration of combustible wastes
Waste compaction

Chemical decontamination to prevent contamination
Coating concrete to prevent contamination
Administrative controls to segregate wastes.

LLW Burial Requirements

Recent industry estimates of the decommissioning radioactive waste volume for a single 1100 MWe reactor is approximately 35,000 cubic yards of packaged waste. If the decommissioning of a large reactor were to commence today, the monthly operations volumetric allotments afforded each utility at any of the three existing commercial burial grounds (approximately 1500 cubic feet per month, per utility) would be exceeded by the third day of decommissioning operations! However, if all operating power reactors were decommissioned today, the existing burial grounds could accomodate this waste volume.

The limited commercial burial space available has sparked regulators to force states to form regional waste compacts, and to select sites for waste burial on a schedule aimed at meeting the needs of the industry. More recently, the passage of Public Law 9924[1], "Low-Level Waste Policy Act Amendment of 1985," permits current burial ground host states to impose a surcharge on waste generators from states who have not yet formed a regional compact.

Decommissioning waste disposal is currently estimated to cost approximately $35 million for an 1100 MWe reactor in 1987 dollars. If these surcharges are imposed, the burial costs should escalate to approximately $54 million in 1988. Allowing that decommissioning for most operating reactors will not occur for 10 to 30 years, it is unlikely that new burial grounds can be constructed and operated at rates lower than the existing commercial grounds. The surcharges themselves may disappear in the future but will probably be replaced by base burial charges that exceed these rates. On a side note, it should be evident that these surcharges are forcing the escalation of total decommissioning costs well in excess of the NRC's proposed inflation adjustment factor of twice the CPI. This is a critical issue when a utility tries to establish a decommissioning fund that must earn interest at a rate to meet future projected costs.

Radioactive waste packaging and burial requirements have become more restrictive in recent years. Title 10 of the CFR, Part 61[4], established specific classifications of radioactive waste and rules for packaging and disposal to assure the waste will be safely buried and isolated from the environment to protect man. In general, these rules require more expensive containers than were formerly permitted for waste disposal.

Some utilities have begun volume reduction programs for their operating reactors so that the procedures will be in place for decommissioning. Volume reduction equipment such as super-compaction and incineration are being purchased to reduce waste volumes. It is important to maintain this equipment and have it operational for use during decommissioning.

UTILITY RATE REGULATION

Public Utility Commission Involvement

The federal regulatory requirement to establish a decommissioning fund is only part of the total picture. An equitable, assured method must be provided to collect the monies from the utility consumer. All privately owned utilities selling retail electricity are regulated by Public Utility Commissions (PUCs). The commissioners are usually political appointees from all backgrounds, including engineers, accountants, attorneys, and homemakers. Some view themselves as anti-utility consumer advocates while others try to seek a balanced outlook, although there seems to be more of the former than of the latter. All generally acknowledge the need for decommissioning funds and try to negotiate an equitable funding mechanism. The PUCs generally have a support staff of engineers, accountants, attorneys, and economists to advise the commissioners on specific issues. This support staff has also been influenced by precedent to keep utility retail rates as low as possible for consumers.

Some PUCs have adopted the decommissioning cost estimates based on outdated generic studies and have applied simplistic CPI escalating factors to attempt to update the estimates to current cost levels. However, the CPI is a poor indicator for the escalation of such costs as radioactive waste burial. In 1978, the burial cost for the lowest radioactive level container was approximately $2.65 per cubic foot. By 1985, before the LLW Amendment surcharges, the cost was approximately $24.89 per cubic foot. This is an increase of approximately 840% in a period of seven years. The CPI only increased 68% during the same period.

The generic studies do not reflect the site-specific factors that add significantly to the decommissioning cost. These factors include:

1. Site labor costs - management and crew
2. Shipping distances and routing difficulties
3. Regional compact burial costs
4. Site facilities available - rail siding, barge docks, or truck roadway restrictions

5. Site factors - seismicity, hydrology, site restoration requirements
6. Site structures - cooling towers, ocean, river or lake cooling, intake/discharge structures and stacks
7. Plant size - power rating
8. Degree of contamination in secondary systems - turbines, condensers, and feedwater heaters
9. Architect/engineer design differences
10. Plant modifications and backfits
11. Errors in calculated generic vessel and internals radioactive inventory - curies and weights
12. Allowable exposures to workers
13. Two-shift vs. one-shift decommissioning operations
14. Utility and Decommissioning Operations Contractor (DOC) staffing levels
15. Removal of nonradioactive components/structures to gain access to radioactive structures
16. Sorting and segregation of radioactive wastes
17. Waste volume reduction equipment available on-site
18. On-site temporary storage facilities for radioactive wastes and laydown areas for scrap/salvage materials.

Some PUCs have recently adopted the NRC's Proposed Rule certification amount of $100 million as a ceiling. This is in direct conflict with the NRC's stated intent to assure that sufficient funds will be available to safely decommission these power plants. These PUCs have disallowed the utilities' site-specific estimates which, in most cases, were prepared by reputable architects/engineers and consultants who have had direct "hands-on" decommissioning experience. In so doing, the PUCs have virtually disallowed the recovery of costs to dismantle the nonradioactive structures and restore the site.

By favoring a low estimate today, the amount to be collected from current ratepayers is obviously less than if a greater amount were adopted. Such short-sightedness unfairly favors current ratepayers by not charging them their share of the cos of decommissioning for the electricity they consumed. It defers the real costs to later ratepayers who will then have to play "catch-up" to have sufficient funds available to safely retire the facility.

Not to leave this issue on so sour a note, it is encouraging to report that several PUCs, namely those of Maine and New Hampshire, have an aggressive state law to assure sufficient funds will be available for decommissioning so that the state does not find itself having to assist the utility to complete the job at the taxpayer's expense.

FERC Involvement in Decommissioning Ratemaking

The Federal Energy Regulatory Commission (FERC) is the governing agency that regulates wholesale rates for electricity that is sold to other utilities, municipalities, or across state boundaries. Its role is very similar to PUCs in so far as it has the responsibility to provide equitable electricity rates for the consumer.

FERC, like the PUCs, has opted for the lowest estimated decommissioning cost, irrespective of evidence of higher costs in the form of detailed site-specific studies prepared by reputable consultants. It has based its estimates on generic studies and applied CPI escalation factors to adjust for inflation and bring the estimates to 1987 dollars. FERC has not made its own independent estimates to challenge the more reasonable site-specific estimates.

IRS Involvement in Decommissioning Funding

Most utilities have prepared site-specific cost estimates for decommissioning their power plants, and have developed funding methods to accumulate sufficient funds for ultimate plant retirement. Some have used external sinking funds with investments in commercial or government bonds. Others are using net negative salvage depreciation. The Internal Revenue Service allows tax free treatment of the monies collected for decommissioning (removal of radioactivity) if these monies are placed in an external fund, subject to certain restrictions on the type of investments. Otherwise, such collections are treated as ordinary income and taxed at the utility's normal tax rate.

GUIDELINES FOR COST ESTIMATING

The Atomic Industrial Forum National Environmental Studies Project (AIF/NESP) is a utility and power industry group dedicated to promoting nuclear power. It recently funded a study to TLG Engineering, Inc. to prepare a report entitled "Guidelines for Producing Nuclear Power Plant Decommissioning Cost Estimates" (AIF/NESP-036)[10]. The guidelines are intended to identify all the appropriate costs of decommissioning in a logical format based on current cost estimating methodology. The guidelines provide a methodology to estimate costs on a consistent basis, and to lend credibility to the cost estimates to segment reactor vessels and internals, steam generators, contaminated turbines and condensers, contaminated piping, valves, pumps, tanks, heat exchangers and electrical equipment. The document includes costing factors for the demolition of heavily reinforced concrete containment

buildings, cooling towers, and structural steel buildings. These calculations are not normally within the experience of most estimators. The guidelines have been reviewed by the industry, decommissioning experts, architect/engineers, consultants, and both NRC and FERC staff representatives. While the regulators may be reluctant to embrace the guidelines as a national standard, they generally agree it provides the necessary consistency needed for cost estimation and comparison.

Methodology

The basic elements of all cost estimates include labor, materials, equipment, energy and services (shipping and burial). The guidelines place the types of costs into three categories: (1) activity-dependent costs, (2) period-dependent costs, and (3) collateral or special item costs. In addition, the guidelines also address the impact of salvage and scrap values and the incorporation of contingencies. The guideline document should be used in conjunction with the user's own cost estimating data base to identify and calculate these decommissioning cost elements.

Cost Categories. Activity-dependent costs are those that are directly related to discrete activities, e.g., decontamination, removal, packaging, shipping, and disposal. They include all labor, materials, equipment and services (shipping and disposal) associated with the "hands-on" activities.

Activity-dependent costs may be calculated by means of unit cost factors ($ per unit of output - cubic yards, tons, linear feet, etc.). For each type of component, these unit factors include the estimated labor hours, crew size and composition, worker base rate plus fringe benefits, consumable materials, special equipment and subcontractor overhead and profit. The labor hours include work difficulty factors associated with work on scaffolding, in protective clothing or in respirators, and the appropriate time for crew ALARA planning and routine work breaks. With this approach, unit cost factors can be developed for various sizes of piping, valves, pumps, heat exchangers, tanks, and for structural concrete and steel. The estimator may then apply these unit cost factors to the detailed inventory of components and structures to develop the removal cost.

In a similar manner, costs for various sizes of radioactive waste shipping containers may be estimated from typical manufacturers, and the number of such containers estimated from the inventory volume of waste. Transportation of these containers to a waste burial facility may be estimated from

published truck, rail, or barge shipping companies. The number of shipments may be determined from the number of containers that fit on the carrier (truck, railcar, etc.) and the total weight to meet state and federal roadweight limits.

The cost for radioactive waste burial may be estimated from published burial facility rates at operating burial grounds. These rates include (1) a basic per cubic foot charge, and (2) surcharges for curies, radiation levels, weight and other special handling. As noted earlier, regional compact burial facilities (when developed) may likely impose higher charges than currently imposed, but such effects can be adjusted during future cost estimating updates.

Period-dependent costs are those associated with project management, administration, routine maintenance, radiological, environmental, and industrial safety and security. They are not directly assignable to any one activity, but are essentially period-dependent; that is, these costs continue for the duration of the decommissioning program or period.

Period-dependent costs may be calculated based on the projected time-phased staffing level of personnel for project management, administration, etc., for both utility employees and a Decommissioning Operations Contractor (DOC), if used. Typically, the utility staff will be greater during the first year of decommissioning and decrease thereafter to a monitoring function. The DOC staff will start small, grow as the decommissioning operations peak, then decrease as the remaining facility is removed. The personnel costs can be estimated from typical utility salaries for the appropriate job positions for the duration of the period; hence, the costs are period-dependent.

Collateral Costs. Collateral (or special item) costs are not directly attributable to one or more specific decomm'ssioning activities. Some of these costs may be period-dependent. This category includes such items as heavy equipment purchase, health and safety supplies, energy costs (heating, diesel fuel, gasoline, etc.) taxes and permits/licenses.

Collateral costs for equipment purchase or rental may be estimated from handbooks or similar estimator experience. Energy cost estimating guides for lighting, heating, or air conditioning are provided in the guidelines, or may be estimated from plant experience during extended shutdowns. Real and property taxes may be obtained from utility records, or from the local tax assessor. The guidelines also provide estimates for permits and licenses.

Salvage and Scrap

The guidelines provide suggestions for an assessment of the suitability of equipment for salvage and scrap, and the effect on net cost. However, the current salvage market for used equipment is virtually nonexistent as a result of recent nuclear plant cancellations and the availability of new salvaged components at $0.10 - $0.15 on the dollar. The extra care required to preserve the value of these components during decommissioning to requalify them for use as nuclear or non-nuclear grade equipment would drive the removal costs up significantly, without any guarantee of a market. Furthermore, complete renovation of this 40-year old equipment would be quite costly to requalify for nuclear applications.

The scrap market has been similarly depressed in recent years and the sale of scrap is expected to barely offset the costs to haul the material from a loading dock to the scrap dealer's yard. The scrap steel and nonferrous markets are currently severely depressed ($65/gross ton for steel, $0.45 per pound for copper). The costs to load it on the trucks, haul it to a scrap yard large enough to stockpile it (perhaps 50-100 miles), cut it to steel furnace charging box size (18"x60"), store it, and load it onto barges or rail cars are barely covered by the scrap value.

Contingency

Virtually every cost estimator of large construction or demolition projects includes some contingency. Contingency is a specific provision for unforeseen elements of cost within the defined project scope - particularly important where previous experience-related estimates and the eventual (actual) costs have shown that unforeseen events and the resulting increase in costs are likely to occur.

The guidelines grouped the decommissioning activities into several categories because they share similar characteristics and potential for contingency events. Fourteen categories were identified in the guidelines, as summarized in Table 1.

As noted earlier, contingency is a highly contested issue in utility rate cases because of the apparent misunderstanding of the basis for contingency and its application. When contingency elements are broken down (as shown in Table 1) and applied on a consistent basis from prior experience, it becomes clear that contingency is a real cost of decommissioning, and is fully expected to be spent.

Typical Decommissioning Cost Estimates

Recent utility planning for decommissioning funding has included site-specific studies for various alternatives. The prompt removal/dismantling alternative appears to have become the standard for comparison of decommissioning costs among different plants. Without attempting to account for site-specific differences, typical 1100 MWe PWR and BWR estimates for the three major alternatives are shown in Table II. The breakdown of costs for the Prompt Removal alternative into their respective major elements is shown in Figure 1.

CONCLUSIONS

The utility industry is probably one of the first industries to ever plan for the eventual retirement of its facilities in a planned program of commitment and funding. Regulators and consumer advocates alike should rally support to utilities for their efforts to be responsible participants in the community, taking care of their wastes and not deferring these problems to future generations. The regulatory processes currently underway are a major step to achieving this self-sufficiency, and the public needs to be made aware of these efforts.

TABLE I: CONTINGENCY ESTIMATES

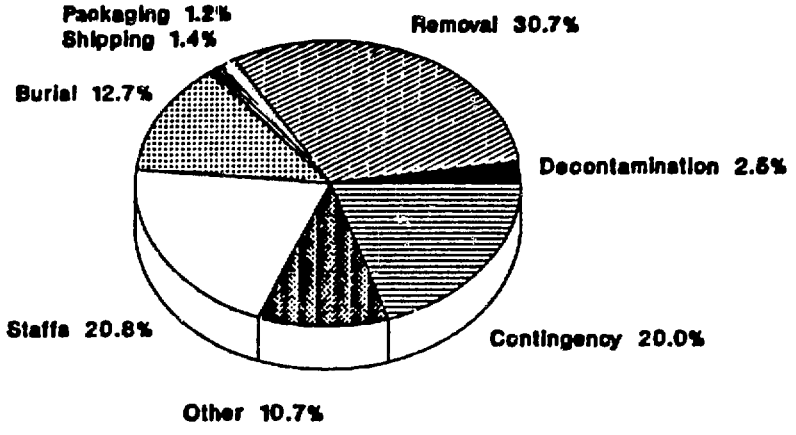
Category	Contingency
Engineering, Project Management, Demolition Management	15%
Utility and DOC Staff Costs	15%
Decontamination	50%
Contaminated Component and Concrete Removal	25%
PWR Steam Generator, Reactor Coolant Pumps and Piping Removals	25%
BWR Recirculation System Pumps and Piping Removals	25%
Reactor Vessel and Internals Removal	75%
Reactor Packaging	25%
Reactor Shipping	25%
Reactor Burial	25%
LSA Packaging	10%
LSA Shipping	15%
LSA Burial	25%
Clean Component and Concrete Removals, Clean Waste Disposal	15%
Supplies and Consumables	25%

TABLE II: TYPICAL COST ESTIMATES (1100 MWe)

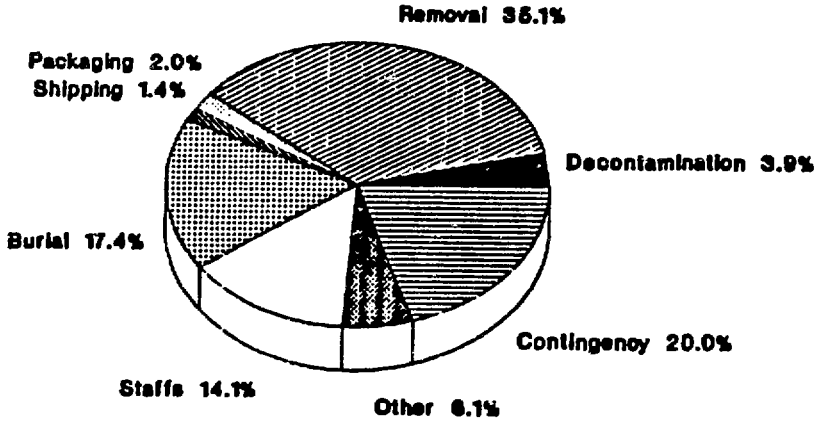
Alternative	$, Millions PWR	BWR
Prompt Removal/Dismantling	200	260
Entombment with 30 Years Dormancy and Delayed Dismantling	230	280
Mothballing with 30 Years Dormancy and Delayed Dismantling	235	285

FIGURE 1: PROMPT REMOVAL COST ELEMENTS

1100 MWe PWR



1100 MWe BWR



Note: "Other" includes engineering, energy costs, insurance, and staff relocation expenses.

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MANAGEMENT OF G2 DECOMMISSIONING (UNGG REACTOR)

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ABSTRACT.

Technicatome, together with the French CEA, has developped engineering methodology for nuclear plant decommissioning management. Illustration of this methodology is provided through its applications to G2 reactor decommissioning.

1. GENERAL.

The purpose of this presentation is to describe G2 Reactor decommissioning management, as an illustration of methodology worked out from available experience.

At definitive shut down time, nuclear facility responsible - owner and/or operator - has to face the problem of choosing a decommissioning solution, which can be expressed as a stage according to IAEA definition. Afterwards, when decision on decommissioning stage has been taken, an appropriate scenario has to be set up, to define the operations leading to the planned stage, taking into account the goals, the R and D program if any, and the facility specific constraints.

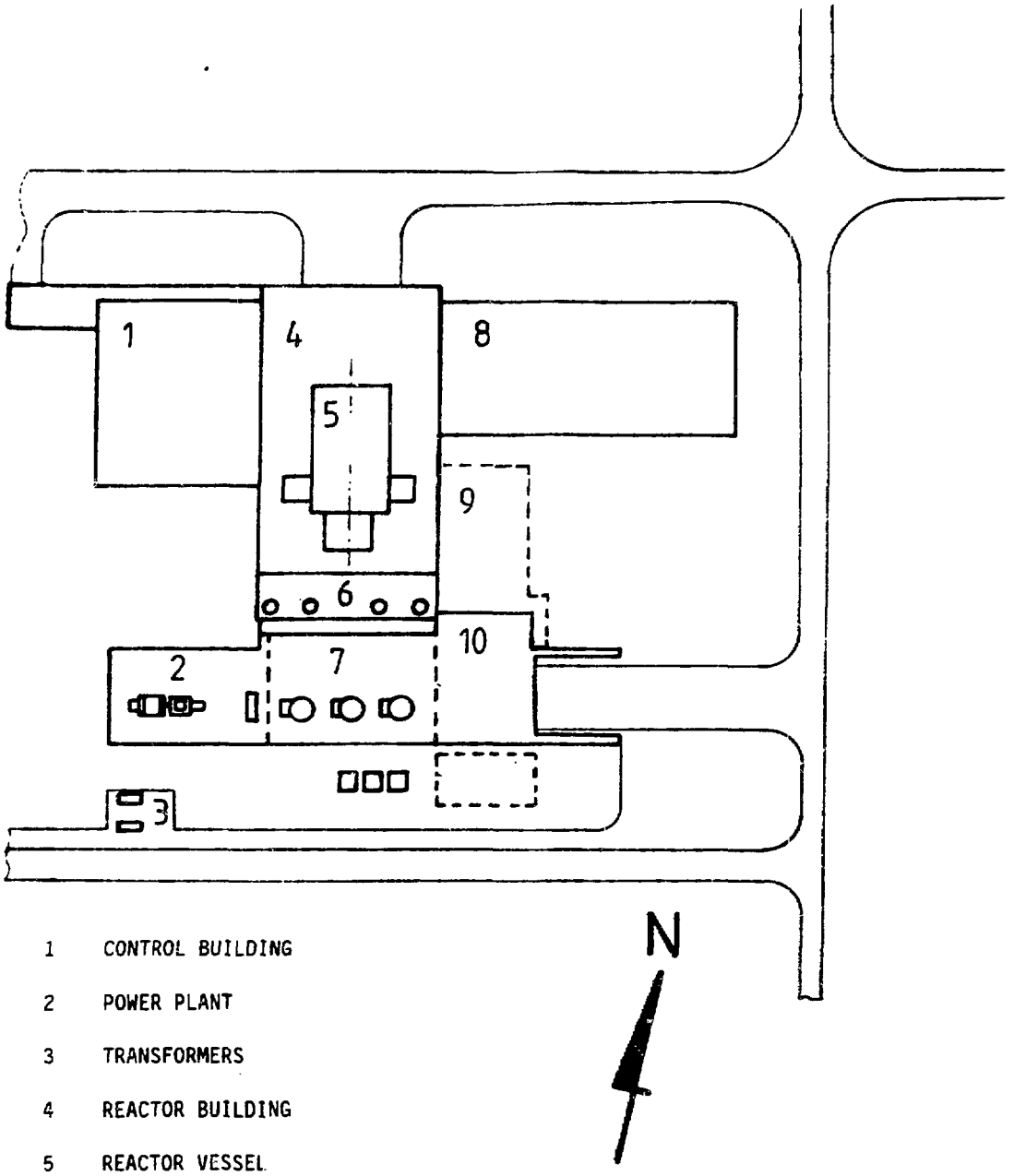
After operating for more than 20 years, French CEA G2 reactor has been definitively shut down in June1980. The specialized department UDIN of the CEA/IPSN is responsible for its decommissioning, TECHNICATOME being in charge of the engineering work, including site work management.

The reactor is a graphite moderated CO_2 cooled at a pressure of 15 bar. Its nominal output was 260 MWTH (40 MWe).

2. G2 MAIN TECHNICAL CHARACTERISTICS IN DEFINITIVE SHUT DOWN SITUATION.

2.1. Description.

In this facility, cooling circuits and steam generators are located outside the reactor vessel ; the reactor vessel and the cooling circuits are housed in metal frame buildings, whereas the steam generators are installed outdoors (fig 1).



1 CONTROL BUILDING

2 POWER PLANT

3 TRANSFORMERS

4 REACTOR BUILDING

5 REACTOR VESSEL

6 STEAM GENERATORS

7 TURBO BLOWERS

8 WORKSHOP - STORES

9 CO_2 GAS STORAGE

10 MOTOR BLOWERS-COOLERS

FIG. 1

G2 GENERAL LAYOUT

2.1.1. Reactor Core (fig 2). The core is composed of a pile of 15,000 graphite bars, its section is quite octogonal of 9.5 m of axis and 9 m long. This pile is crossed by 1200 horizontal channels for the fuel elements and 51 vertical channels for the control rods. This active part is surrounded by a graphite reflector of about 0.8 m thick, entirely shielded by 12 cm thick steel plates and by an insulation. This block is enclosed in a prestressed concrete vessel in form of an horizontal cylinder of 14 m internal diameter and 18 m long closed at each end by an hemispherical dome. The concrete thickness of the cylinder and of the domes is 3 m. In order to withstand the 15 bars in service pressure and the 30 bars test pressure the vessel is prestressed by 161 cables tightened each one at 1200 tons. Its inner face is equipped with a 30 mm thick steel liner.

2.1.2. CO_2 Circuits (fig 3). They are made of carbon steel, and, - not comprising the steam generators - they involve 1700 m of pipes, 1000 m of which are over 1 m in diameter (up to 1.6 m), with thickness ranging from 7 to 25 mm. Cooling circuits include number of specific components such as blowers, coolers, valves, expansion bellows .... Large pipes are equipped with internal fittings, such as deflector grids, temperature sensor thimbles.

Overall steel weight is in the range of 4000 tons, among which 2500 tons contaminated, including the steam generators, which are described below. Piping over 0.8 m in diameter represents 700 tons, piping below 0.8 m, 83 tons.

2.1.3. Steam Generators (fig 4). G2 is equipped with four steam generators, 3.5 m in diameter and 32 m long, arranged vertically outside the building. Each one includes several sections, composed of finned tubes and weights 300 tons.

2.2. Radiological Status.

2.2.1. Reactor Vessel. The major part of the radioactivity is concentrated into the core and is due to activation of materials and structures. Activated material weights are approximately the following :

graphite : 1300 tons
steel : 1300 tons

part of the concrete bio shield : estimated at 4000 tons.

Total radioactivity inventory in the graphite is about 4000 Ci (1.5×10^8 MBq) essentially due to :

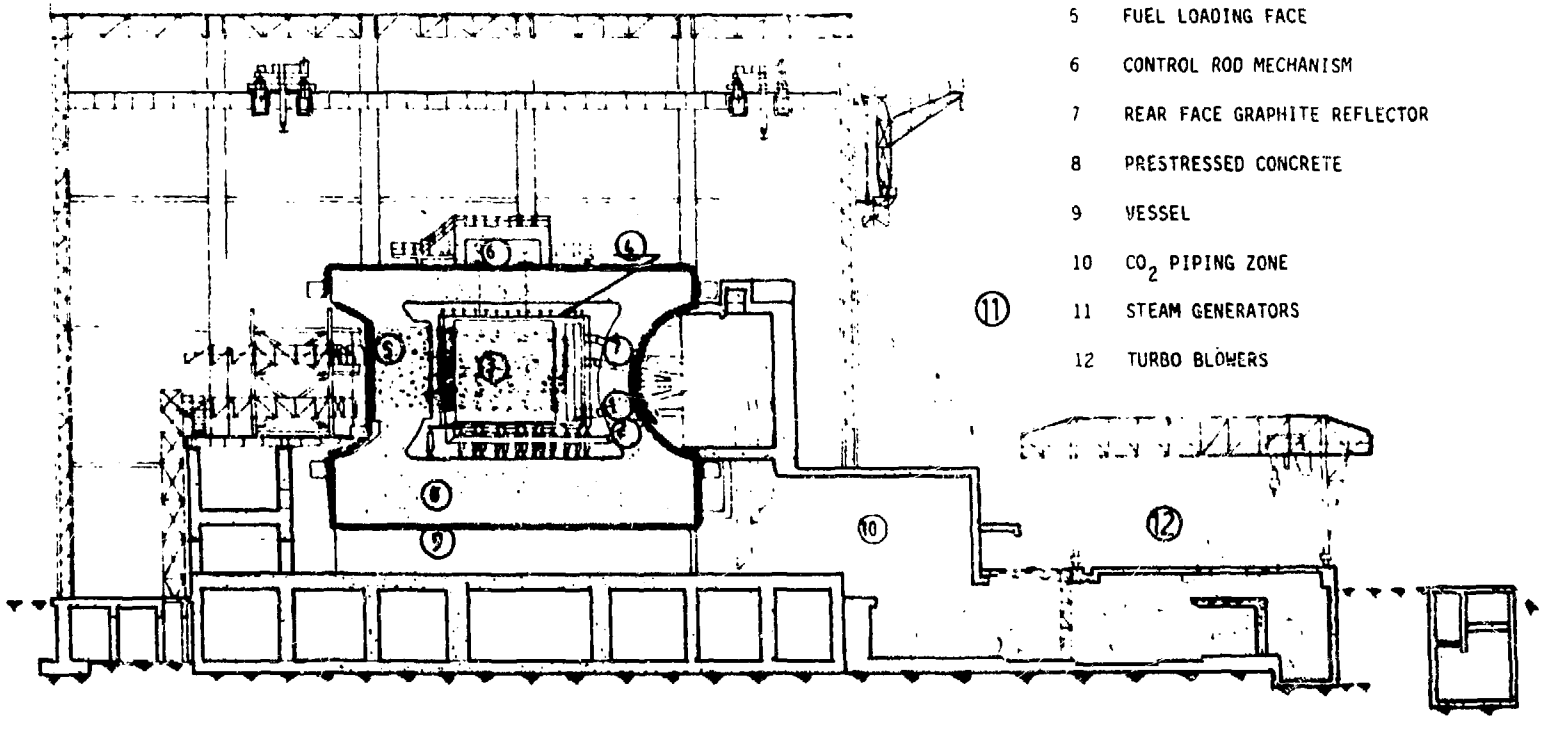
C14 : 500 to 1000 Ci (20 to 40 x 10^6 MBq)
half life : 5600 years.
H3 : 2000 to 3000 Ci (80 to 120 x 10^5 MBq)
half life : 12 years.
Co 60 : approx. 500 Ci (20 x 10^6 MBq)

Average dose rate inside the reactor vessel is about 30 to 50 Rad/h, (0.3 to 0.5 Gy/h).
(Measurement performed on G2 in May 1981).

FIG. 2

G2 REACTOR

GENERAL SECTIONAL VIEW



1 CO_2 INLET
2 CO_2 OUTLET
3 GRAPHITE BLOCK
4 THERMAL SHIELD
5 FUEL LOADING FACE
6 CONTROL ROD MECHANISM
7 REAR FACE GRAPHITE REFLECTOR
8 PRESTRESSED CONCRETE
9 VESSEL
10 CO_2 PIPING ZONE
11 STEAM GENERATORS
12 TURBO BLOWERS

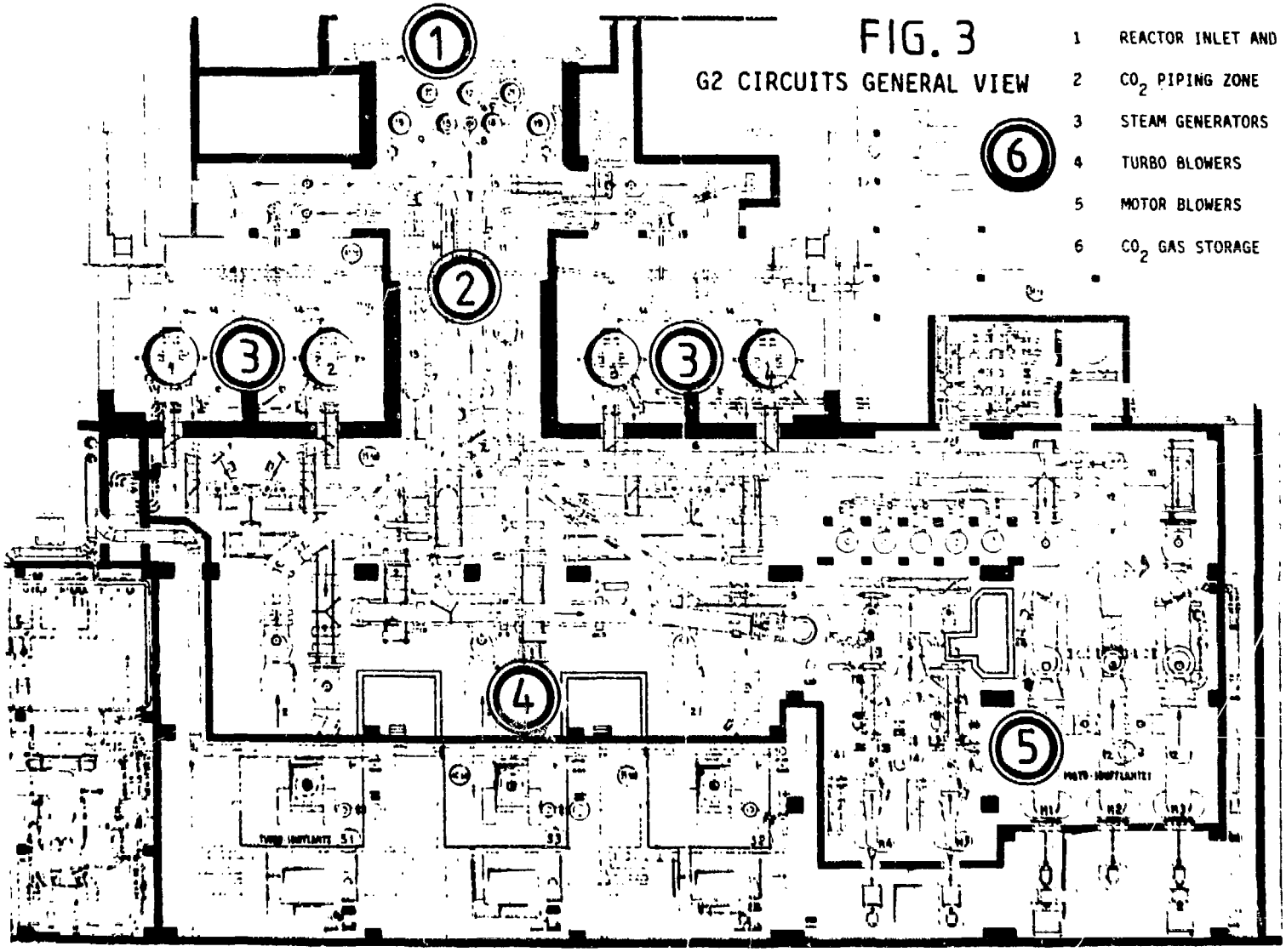


FIG. 3

G2 CIRCUITS GENERAL VIEW

1 REACTOR INLET AND OUTLET

2 CO_2 PIPING ZONE

3 STEAM GENERATORS

4 TURBO BLOWERS

5 MOTOR BLOWERS

6 CO_2 GAS STORAGE

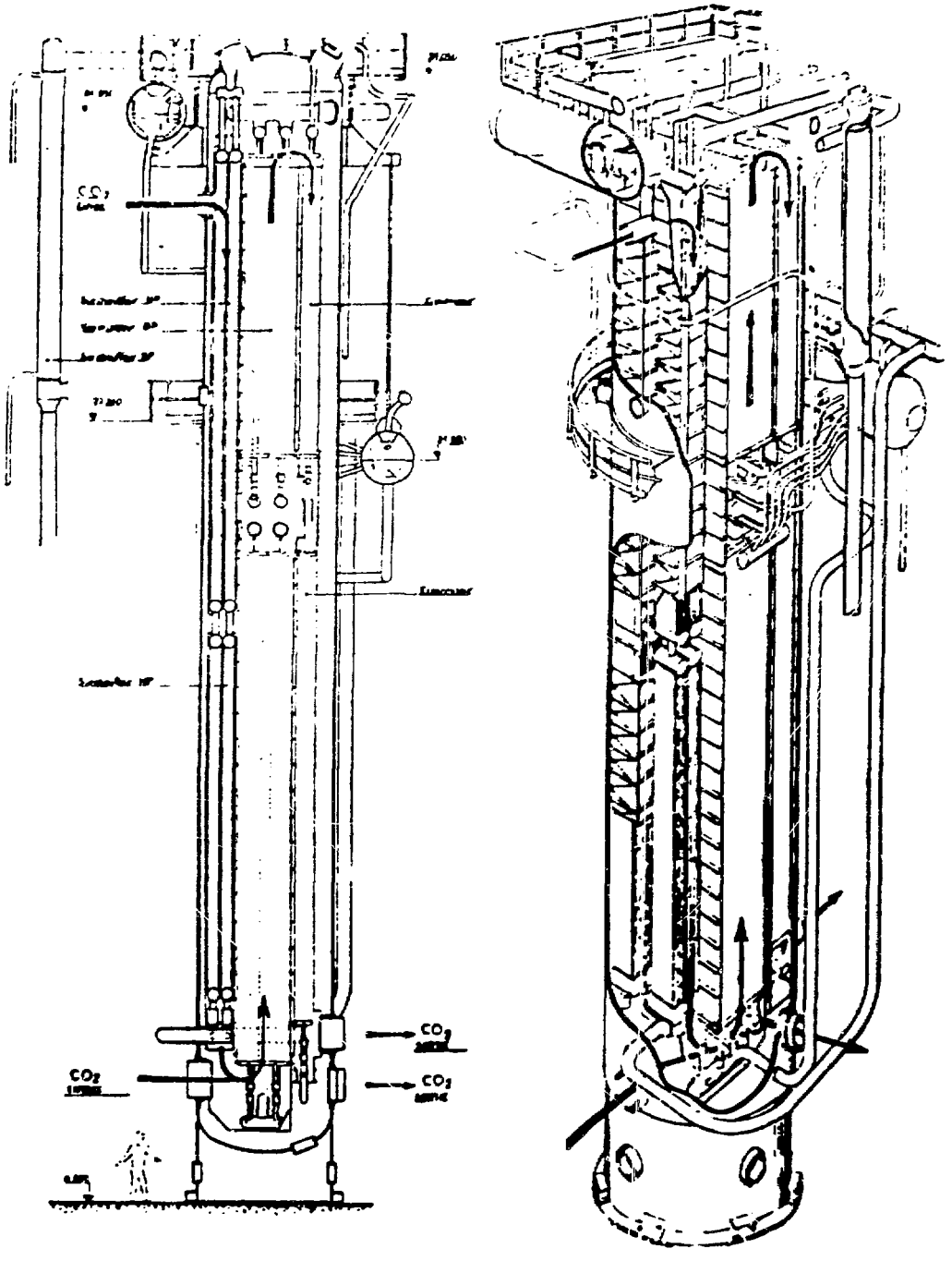


FIG. 4 G2 STEAM GENERATORS SECTIONAL VIEW

2.2.2. Circuits, Steam Generators, Ancillary Equipment.
Average contamination level is 5 x 10^{-4} µCi/cm² (20 Bq/cm²), essentially due to Co 60, half life 5 years. (Measurement performed on G2 in May 1980).

Corresponding overall radioactivity inventory is about 3 to 5 Ci (1 to 2 x 10^5 MBq).

Dose rates in circuit areas do not exceed a few mrad/h (10^{-5} Gy/h).

In order to avoid contamination spreading risk, reactor vessel and contaminated circuits are kept at a small negative pressure of air.

3. DECOMMISSIONING STAGE CHOICE.

To make the decision among the three considered stages, the different studies point out the following main factors as being relevant :

- National decommissioning policy, including government responsibility and public relations aspects ;

- The status of the installation, after shutdown and before decommissioning as regards equipment, radiological and overall control aspects ;

- The safety and radiation protection constraints contained in the body of national regulations and international recommendations ;

- The problems connected with waste management (storage, disposal, etc) ;

- The possible re-use of equipment, premises, buildings and land, and the recovery of materials for unrestricted use ;

- The availability of the means to decommission to a specific stage : techniques, processes, equipment and workforce ;

- The estimates of the cost, the availability of funds and their management within the context of the financial policy of the enterprise as well as the national authorities ;

- The likely social and environmental consequences.

Today in France, no requirement from the regulatory bodies for a definite decommissioning stage :

- First of all, keep the installation at the same safety level as during operation.

- Then make the choice on the basis of the other criteria.

- Gain experience so that a policy can be set up for the future.

Such an analysis was applied to G2 reactor as follows :

a) National policy in matter of decommissioning.

From a French point of view, G2 decommissioning appears to be a pilot operation, as well as for the reactor type (graphite moderated - CO_2 cooled) as for its industrial scale.

Targets considered in matter of decommissioning have been the following :

- Dismantling the whole reactor, i.e. the assembly consisting of reactor core, cooling circuits and fuel loading-unloading equipment.

- Conducting the dismantling operation in good cost conditions.

- Meeting ALARA principle, by minimizing both hazards for environment and workers exposure.

- Completing the works within a reasonable time.

b) Reactor status.

b1. Quality of containment.

Core radioactivity is contained in the steel line prestressed concrete pressure vessel, which is no more at service pressure. A study of mechanical behaviour has been performed, showing that no special short term measures had to be taken to ensure containment integrity : prestressing is maintained, cable tightening and vessel displacement controls are performed periodically.

CO_2 circuits represent a huge contaminated volume, which has to be kept under negative pressure to avoid contamination spreading risk. Corrosion of piping and supports has to be controlled, with appropriate maintenance, especially for outdoor structures.

b2. Radiological status involvements.

As to the core, direct access work can not be envisaged : dose rate is too high, and radioactivity, essentially due to activated materials, can not be eliminated.

Moreover, better knowledge of radioactivity in graphite and concrete was needed to correctly handle waste generation and disposal aspects.

As to the circuits and steam generators, direct access work was considered possible, as it was for inspection and maintenance during reactor operation.

c) Safety, protection, industrial security.

Solutions envisaged were analyzed in accordance with ALARA principle and within the frame of national regulations, with respect to radioactivity, containment barriers, dismantling work, waste disposal and by-products.

c1. Considering core dismantling, safety, protection and industrial security involve :

- appropriate concept of necessary remote operated equipment, both in service and maintenance conditions,

- bio shield and containment restoration at necessary penetrations,

- verification of core structure correct mechanical behaviour during dismantling work.

c2. Considering cooling system dismantling, same aspects involve :

- possible on site decontamination in order to reduce contamination spreading and workers exposure during cutting, using processes with minimum waste generation,

- avoiding transfer of chemical products - or other - used or generated during work, towards unaccessible parts of the reactor, namely the core cavity,

- providing for security of handling operations and correct behaviour of the piping support system all along dismantling work progress.

In case of deferred dismantling, maintaining containment quality has to be provided for, with respect to ageing.

d) Waste management and disposal.

d1. Reactor core.

No experience was available concerning graphite disposal. Two ways could be considered : disposal as a solid waste, or incineration. Five disposal processes were identified and analyzed :

- Three in the sea : sea bed burial, deep sea dumping, off shore geological burial.

Those three processes are not so far usual practice : their acceptance, in the frame of an international consensus, is also depending on national policy.

- Deep geological land repository projects do exist, especially in France, but decision of construction has not yet been taken.

- In the present state of technics, C14 embedding possibilities do not allow shallow land burial : further progress could make it possible.

Considering graphite incineration, studies relating to process acceptance showed it necessary to improve graphite radioactivity knowledge, which also applies to burial solutions.
Appropriate R and D program is in progress (cumulated neutron flux, impurities, leaching ...).

d2. Cooling system.

Considering the low contamination, disposal is possible in shallow land repository. But the large amount of material was reasonable incentive to avoid such a solution and to have in view maximum reuse or unrestricted release.

The minimum figures taken into account in the analysis were those admitted on a case by case basis by French regulatory bodies :

10^{-6} µCi/cm² (3.7×10^{-2} Bq/cm²) for α emitters

10^{-4} µCi/cm² (3.7 Bq/cm²) for β γ emitters

e) Existing means, methods, process and tools.

e1. Reactor core.

As already shown, there was no immediately available technical means for reactor vessels dismantling and graphite disposal, but already existing tools, methods, and processes could be used, subject to some adjustments.

e2. Cooling circuits.

Considering low existing contamination, and subsequently moderate required decontamination factor ($DF \leq 10$), it was decided to use chemical or mechanical (sand blasting ...) decontamination in a two step process :

- on site application,
- application, after cutting, in a decontamination workshop.

Obviously, generated waste processing should be taken into consideration in decontamination process choice.

Cutting was considered feasible by oxyacetylene or plasma, together with usual mechanical processes, subject to appropriate implementation (generated aerosol collection and processing).

f) Cost elements

Surveillance and maintenance costs : they are of course "cancelled" in the case of reactor complete dismantling ; they can be reduced in case of circuits dismantling and reactor mothballing : less health physics controls and maintenance operations, possible suppression of under pressure requirement after appropriate reactor vessel containment reinforcing.

Due to the amount of material involved in the case of G2 and G3, potentially industrial scale work is a positive factor in dismantling costs. Moreover, vicinity of twin reactor G3, definitively shut down in 1984, allows considering that not only studies, R and D, processes implementation and necessary equipment design performed for G2 will be a priori applicable to G3 - unless regulations changes, or new processes or technology developments - but that part of equipment made and used for G2 dismantling will possibly be reused for G3, which is also true for staff.

g) Local site considerations.

G2 is not a separate plant, but is located in a nuclear center : surveillance can more easily be envisaged if necessary and economically acceptable. A waste treatment station is operating in this nuclear center, including waste disposal procedures : benefit can be taken from processing generated decontamination wastes in the station.

Conclusion.
On the basis of the foregoing factors, decommissioning option studies have demonstrated the technical feasibility of stage 2 decommissioning, which is presently under way.

On another hand, from policy considerations, G2 position as first French production reactor and our desire to carry out the complete decommissioning of a full size industrial nuclear plant led us to decide stage 3 decommissioning.

Summarized option was :
- First, dismantling whole cooling and auxiliary circuits.
- Then, dismantling reactors core (G2 and G3 operations being carried out in a common non stop program).

4. WORKING OUT THE SCENARIO.

4.1. Circuits - Steam Generators - Auxiliary Equipment.

a) Technical options.

Selected scenario (stage 2) is the following :
- On site decontamination, using self acting devices, then cutting into as big as possible pieces.
- Conveying the pieces in a specialized workshop for final decontamination.
- As for pieces which cannot be reused or released, second step cutting for packaging and disposal to radwaste storage facility.

This scenario has the following objects :

As already mentioned, main option consists in achieving steel maximum unrestricted release or reuse. Target considered is the following : unrestricted release of 80 % of initially contaminated steel (50 % if steam generator internals can not be decontaminated). Parts slightly activated (vessel inlet and outlet) are not concerned by the above target and are included in radioactive wastes.

Implementation of this guideline leads, in the case of G2, to the following technical options (fig 5) :

- two steps decontamination system : first on the spot whenever possible, second, if necessary and achievable, in an auxiliary facility,

- two steps cutting system : first on the spot making big pieces, second for further treatment (decontamination, control before disposal) in an auxiliary cutting workshop,

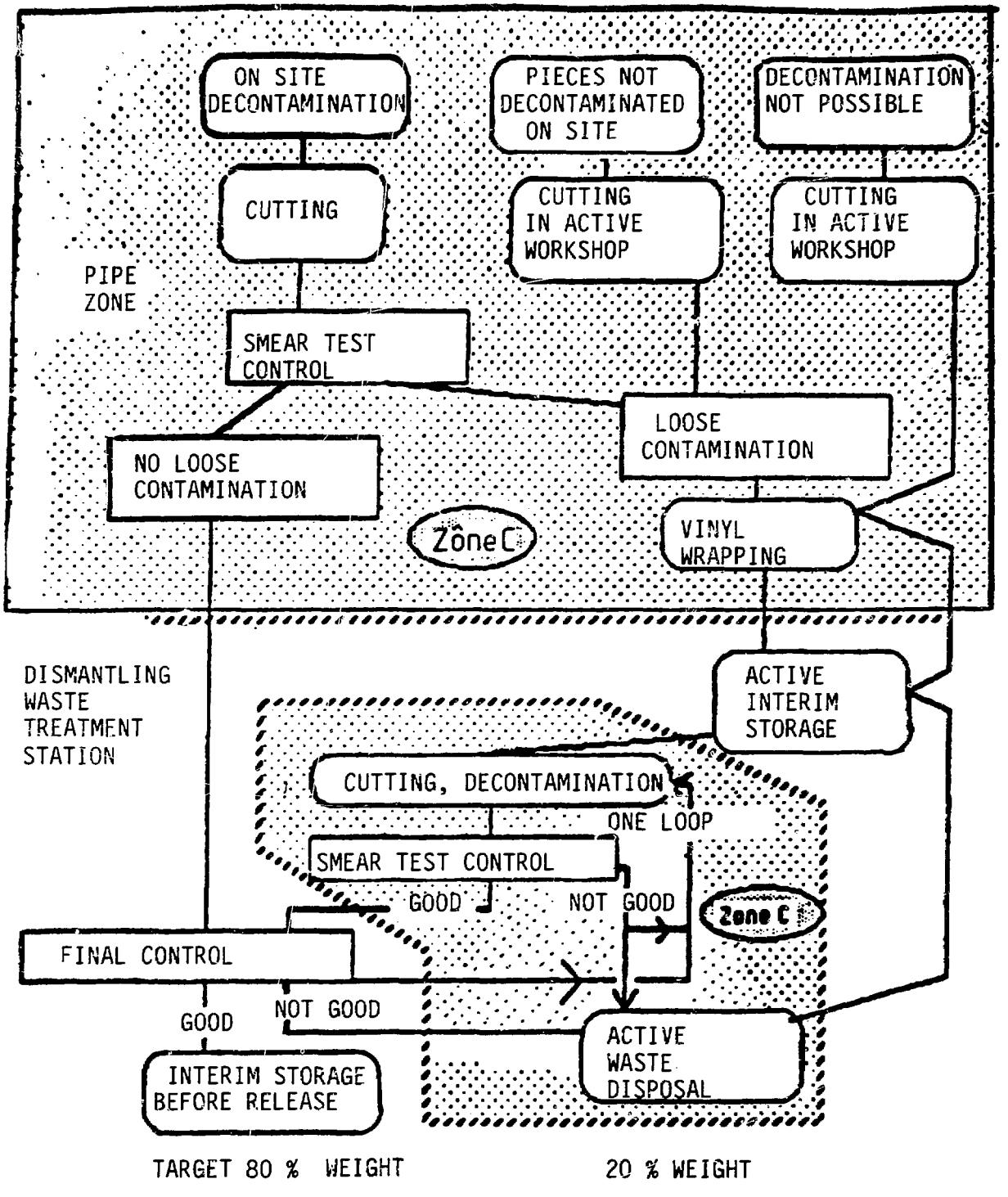


FIG. 5 DISMANTLING WASTE TREATMENT SCENARIO

- considering the amount of work to be done in second step processing, design of corresponding dismantling waste treatment station as a production plant, located next to G2 circuits, with anticipated reuse for G3. Layout of this facility is shown on figure 6,

- considering the large quantity of material involved, techniques are to be applied on an industrial scale : before working out the detailed dismantling procedures, a full scale test of processes, procedures and specific tools and equipments is performed on a pilot section of the circuit, representative of the whole (hot and cold lines, various diameter and wall thickness pipes, various components).
Such a preliminary operation allows to select most appropriate handling methods as well as working area ventilation and containment system. Pilot section (see figures 7 and 8) involves about 290 tons of metal, corresponding to a contaminated surface of 3600 m^2, and a radioactivity of approximately 100 mCi (3.7×10^9 Bq).

b) Associated developments.

Processes and techniques selected to meet the dismantling options are tested and operated on a significant section of G2 circuits. Works are conducted like or through a pilot procedure. Are concerned in particular :

b1. on site

- piping decontamination by spraying a chemical gel, with three targets :
  - first step maximum DF, making easier second step decontamination, if needed,
  - reducing, if not cancelling, radioactive hazards during cutting operation,
  - reducing liquid waste generated by decontamination, and preprocessing in order to meet the specifications of Marcoule nuclear center waste treatment station.

- automatic cutting for large diameter pipes, using a plasma torch carried by a self propelling trolley,

Both processes can be remote operated, in order to reduce operator exposure.

b2. in the waste treatment station

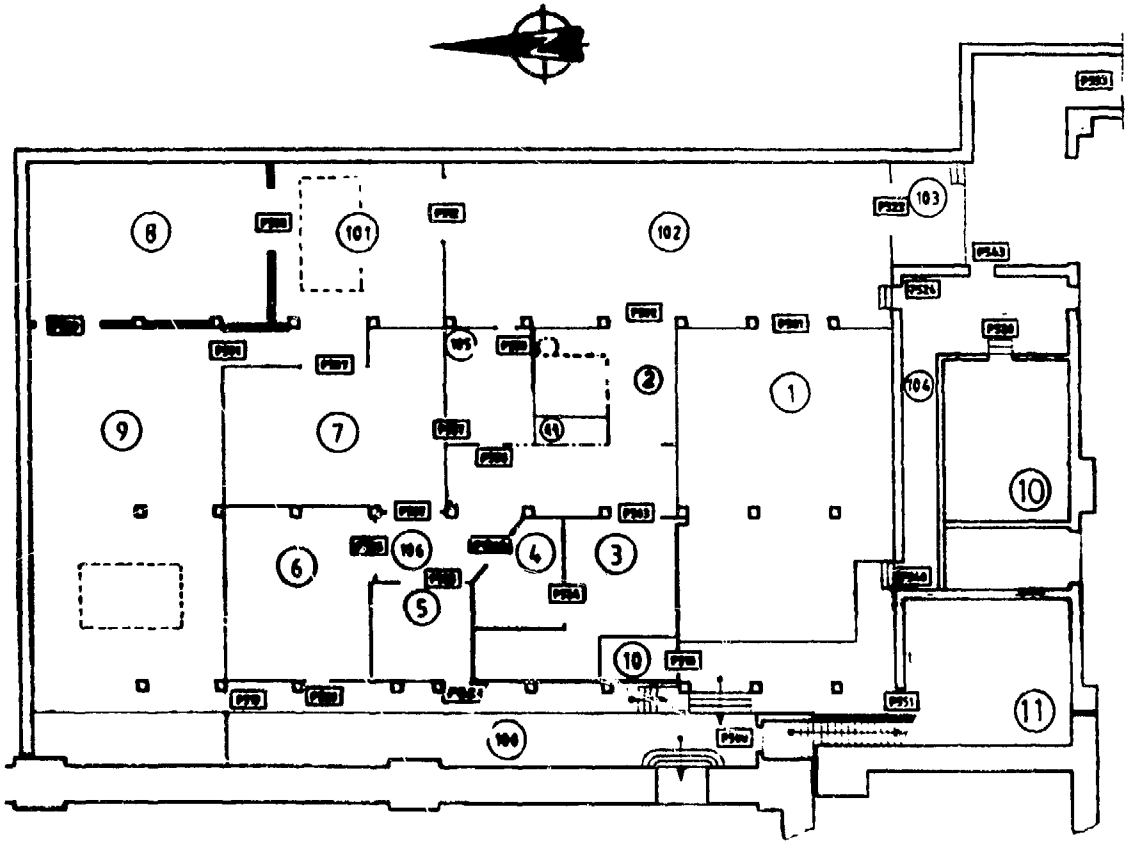
- semi-automatic system for high sensibility and high speed measurement of residual contamination of decontaminated and cut pieces, allowing on line identification of pieces meeting reuse or unrestricted release criteria (surface contamination and mass contamination).

In the same field, melting tests have been planned on low contamination steel pieces, in order to determine the efficiency of homogenization and dilution with respect to residual contamination traces.

c) Resulting dismantling scenario.

Corresponding schedule, taking into account the scenario main phases, is shown on figure 9.

FIG. 6 DISMANTLING WASTE TREATMENT STATION



1 ACTIVE INTERIM STORAGE

2 BUFFER STORAGE

3 AUTOMATIC DECONTAMINATION CELL

4 ROBOT MAINTENANCE

5 MANUAL DECONTAMINATION

6 SECOND STEP CUTTING

7 WASTE PACKAGING

8 FINAL CONTROL

9 INACTIVE INTERIM STORAGE

101, 103, 104 AIR LOCKS

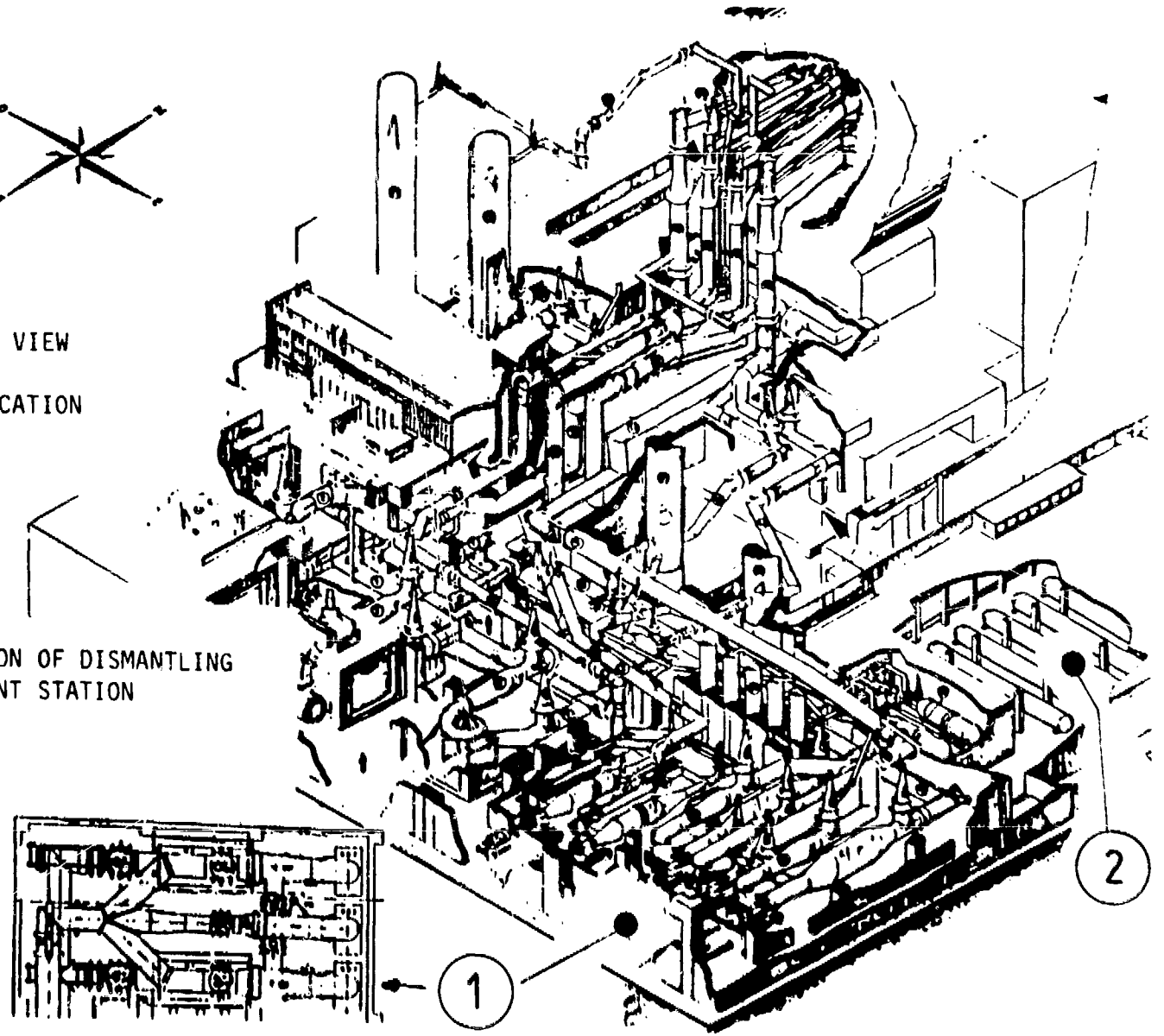
10 LIQUID WASTES

11 VENTILATION

FIG. 7

G2 PIPING ISOMETRIC VIEW

PILOT SECTION LOCATION



(1) PILOT SECTION

(2) FUTURE LOCATION OF DISMANTLING WASTE TREATMENT STATION

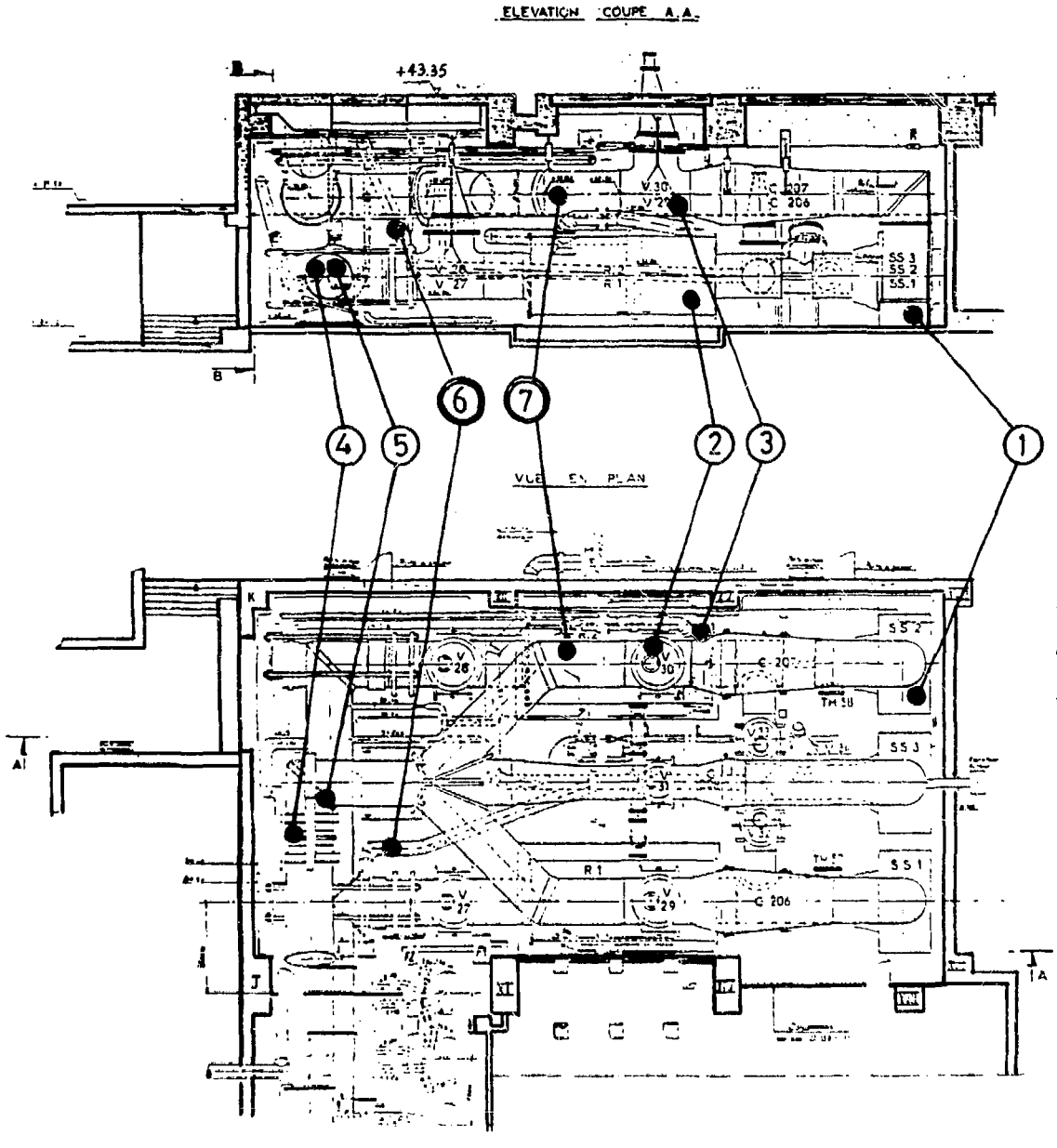


FIG.8

G2 PILOT SECTION

1 BLOWER

2 COOLER

3 VALVE

4 EXPANSION BELLOWS

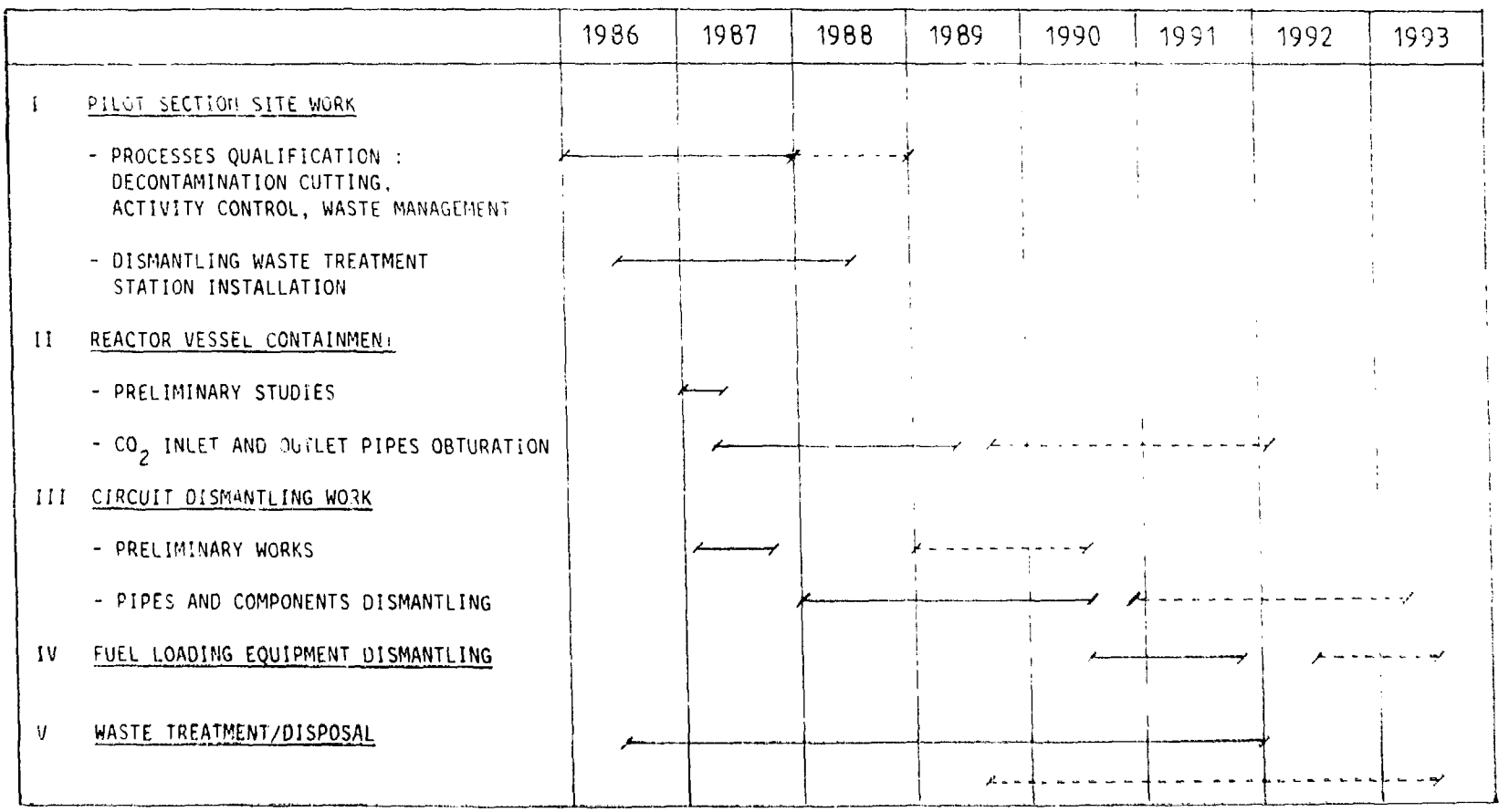
5 ELBOW (WITH DEFLECTING GRIDS)

6 ⩽ Ø 400 PIPES

7 Ø 1600 PIPES

FIG. 9 PLANNED SCHEDULE OF G2 -G3 REACTORS CIRCUITS DISMANTLING

G2 ——————
G3 - - - - - - - -



4.2. Reactor Core.

Selected scenario is based on wanting to limit the extent of operations inside the reactor vessel (taking out of the vessel as big as possible pieces) and to carry out maximum of cutting and decontamination operations in a specialized workshop, which must allow, by using self-acting devices, better performance and efficiency.

Such a strategy necessitates access means, involving heavy machines and shielding equipments. Two solutions have been envisaged :
- vertical access through the top,
- horizontal access through the fuel loading face (front face).

Difficulties to be handled were the following :

- access through the top needed cutting a large opening on the vessel section where prestressing cables axial layers cross circular layers (this solution has the advantage of using more common and easy design means for dismantling and handling).
- front access perfectly matches prestressing cables structure. On the other hand, handling and radiation shielding equipment design call for more complex and less experienced solutions.

Furthermore, R and D work had to be started in order to complete knowledge of reactor structures residual radioactivity, as well as means for concrete dismantling and disposal. In the same way, work had also to be done concerning tools and methods for graphite removal and disposal.

In such reactors, graphite volume is so important that it has a very significant impact of the scenario : dismantling and handling means are linked to graphite future.

Option for graphite future is not yet decided : it is linked to options relating to UNGG reactors family. So, program has been started to measure its radioactivity porosity and leachability.

5. EXPERIENCE ALREADY GAINED IN CIRCUIT DISMANTLING.

Pilot section dismantling work started in 1986 second half year.

a) Preliminary operations.

According to scenario guidelines, a specific ventilation system has to be designed and installed to serve the working zone and prevent contaminated air transfer to other sections and to reactor vessel.

Performing decontamination involved reagent and generated waste storage area.

Second step operations involved dismantled pieces interim storage and waste treatment station installation.

Recovering room for these auxiliary facilities needed previous dismantling of ancillary circuits most of which considered inactive in France : it was the case for about 270 tons. Only 18 tons were disposed off as radioactive, because surpassing permitted limits and consisting in small size components, efficient contamination of which was not anticipated.

So as to give access to the main piping and components of the pilot section, 35 tons of 400 mm diameter circuit equipment has to be dismantled and put in interim storage to be later decontaminated in the dismantling waste treatment station.

b) First pilot subsection dismantling campaign.

It was performed in representative containment and ventilation conditions, with respect to future work : pilot section was made separate from the other circuits, by cutting the main pipes at the pilot section limits - with no prior decontamination - and welding caps at both ends of the cut. It was then possible to ventilate the pilot section independently of the other sections.

Main target was testing gel decontamination and automatic plasma cutting of 1600 mm diameter pipes.

Concerning full size decontamination tests, gel spraying and rinsing were performed by a decontamination specialist entering the pipe in protective suit. Target was to gain data and experience in order to prepare remote operated decontamination procedures.

Dismantling elementary sequence was the following :

- fixing a mobile containment air lock at the pipe inlet, negative pressure being held in the pipe by the auxiliary ventilation system,

- decontaminating about three meters of pipe in a single operation,

- controlling surface contamination (smear test),

- removing the mobile containment,

- cutting the decontaminated section into approximately 1 ton pieces (up to 1.7 m long) using automatic plasma cutting machine,

- keeping pieces considered decontaminated for release in an inactive storage area, to be later processed through the final control station,

- keeping pieces either not decontaminated or still contaminated above release limit in active interim storage, to be later processed through second step operations.

An auxiliary second step cutting facility is used if necessary, to reduce pieces size before storage.

Weight balance of dismantled pieces handled from the beginning up to the end of first subsection work is summarized in table 1.

Table 1

G2 DISMANTLING WASTE
(First pilot subsection, including preliminary operations)

Part dismantled	Inactive	Active		
	Weight (tons)	Weight (tons)	Contamination level	Processing
Preliminary operations. Auxiliary facilities.	272	18.5	3.7 to 37 Bq/cm²	Rad waste disposal.
Pilot section : up to ϕ 400 piping.		35	3.7 to 37 Bq/cm²	Interim active storage at G2. To be decontaminated.
First pilot subsection. SEPI expansion bellows.	10.5	7.6	37 Bq/cm²	Interim active storage at G2. To be decontaminated.
		2.9	37 Bq/cm²	On site decontaminated. Inactive storage before final control.
ϕ 1600 piping.		3.5	37 Bq/cm²	Interim active storage at G2. To be decontaminated.
		3.5	37 Bq/cm²	On site decontamination Inactive storage before final control.
TOTAL	282.5	71		

Comments are the following :

Concerning pipes plane surfaces, on site smear test controls, and γ spectrometry measurements on samples show that for initial contamination up to 2.5 x 10^{-2} μCi/cm² (900 Bq/cm²), residual contamination is below 10^{-5} μCi/cm² (0.37 Bq/cm²).

Concerning internals, such as deflecting grids, expansion bellows, with initial contamination around 10^{-3} μCi/cm² (37 Bq/cm²), DF reaches a few units, but is not sufficient to allow unrestricted release. It appears necessary to remove the internals before performing pipe decontamination in order to reach release limit ; a preliminary decontamination before removing the internals limits contamination fixing during cutting operations.

It has been verified that due to the low contamination level, fixing during cutting was not an obstacle to sufficient second step decontamination.

Concerning liquid waste, generation from decontamination process is about 10 litres for 1 m².

c) Subsequent operations.

Second subsection dismantling started in spring 1987. This section, and the following comprise not only pipes, but circuit components such as valves, blowers, coolers.

The good results obtained in the first subsection (homogenous decontamination along the pipe, knowledge of operating times) has allowed doubling the pipe length to be decontaminated in a single operation. For example, decontamination of a 6.2 m long section - 27 m² - was performed within three hours and generated 270 litres of liquid wastes.

Technique for circuits components decontamination with minimum waste generation and minimum preliminary cuttings is being developped : tests with chemical foam are under way and process is planned to be available in the second half of 1987. In the meantime, components, after cutting connecting pipes, are kept in active interim storage.

d) Dismantling waste treatment station.

Final control and second step decontamination facilities are planned to be in operation in the second half of 1987.

6. CONCLUSION.

Already performed studies and experimentations resulted in the following statements :

a) Circuits.

- Decontamination : to have a good process is not enough, appropriate means to put it into operation are also necessary.

- Residual radioactivity measurement.

A stationary, high sensitivity, counting station has been developped : it would be appropriate to have also available a mobile on site counting device, after decontamination.

Important comment :

- When facing so low radioactivity, decontamination processes must be able to allow unrestricted release of a great part of dismantled metal pieces. If not, decontamination should not be attempted : wastes are to be packaged for disposal and final storage in active repository.

b) Reactor core.

- So far, only preliminary strategy studies are performed.

- It is necessary to first correctly appraise constraints in connection with penetration opening : among these, the big volume of outer cell and access air lock.

Therefore, feasibility is investigated for basically different solutions.

- A set of important technical aspects depends on selected solution regarding waste future.

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DECOMMISSIONING PROJECT READINESS REVIEWS AT THE DEPARTMENT OF ENERGY'S HANFORD, WASHINGTON, SITE *

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ABSTRACT

Two Hanford Site contractors independently formulated readiness review methods to prepare for decontamination and decommissioning (D&D) projects. One readiness review method provided an independent management review process. The other method provided a review by personnel directly involved in the project and concise documentation procedures. A unified system is now used at Hanford which combines the best aspects of both readiness review methods. The unified method assigns category levels based on certain job characteristics. The category assigned to the project then indicates the required level of management review prior to proceeding with the D&D project. In addition, the concise documentation procedures are now used for all category levels.

INTRODUCTION

The initiation of decontamination and decommissioning of major nuclear facilities should always be preceded by a logical systematic readiness review prior to beginning work. Standard methods have been developed from similar readiness reviews performed for many years in the nuclear industry during the startup of new facilities. Examples of such methods are the Occupancy Use Readiness Manual, ERDA-76-45-1, SSDC-1, published by the Energy Research and Development Administration in 1976 and the Work Process Control Guide, DOE-76-45/15, SSDC-15, published by the U.S. Department of Energy. Two separate contractors at the Hanford Site have developed readiness review systems and outlines for decontamination and decommissioning projects. These systems were developed independently and were based upon experience and guidance found in generic readiness review outlines. This paper summarizes the two systems, referred to below as System One and System Two, and explains a refinement for Hanford decommissioning which combines the best parts of both into one Unified Readiness Review System.

* This work was performed for the U.S. Department of Energy under Contract DE-AC06-77RL01030.

SYSTEM ONE READINESS REVIEW PROCESS

The System One readiness review process is initiated with the selection of a Startup Team and a Review Board. The Startup Team is made up of persons knowledgeable in the Decontamination and Decommissioning (D&D) task to be undertaken. The team chairman is usually the representative from the organization performing the actual decommissioning (referred to as Operations). Other team members are selected from Quality Assurance, Research and Engineering, Safety and Environment, and Quality Information. The Startup Team evaluates all aspects of readiness and provides written documentation of each individual aspect.

The Review Board is made up of senior persons from the same organizations; however, they are selected by the top-level managers (Directors) of the organizations and are operationally removed from any direct connection with the performance of the decommissioning. The Review Board judges the adequacy of documentation of readiness and, when satisfied that all aspects are ready, recommends to the Operations Director to approve startup. The Operations Director in turn directs Operations to perform the work.

The Startup Team reviews each individual item in the Readiness Review Checklist for appropriateness to the decommissioning to be done. Items which are judged to be not applicable are documented as such and are not reviewed further. Items which are judged to be incomplete are supplemented with additional items. The Team then assigns each individual item to one or more Team members for completion. The documentation of readiness for each item is reviewed by the Team and, if complete, approved and submitted to the Review Board for review and approval. In this way each individual item is reviewed and approved (closed out). In some cases there will be items which cannot be closed out prior to the start of decommissioning, such as cutting and welding permits. These can only be granted within 24 hours of the actual cutting or welding. In this case, the item is carried forward as an anomaly and is closed out formally after the work has started. In other cases there will be items which can be closed prior to the start of decommissioning but, for the convenience of operations, are selected for closure after approval to start is granted by the Operations Director. These items are carried on an Open Item list and approval is granted contingent upon closing each of the items prior to actual startup. This process is graphically represented in figure 1.

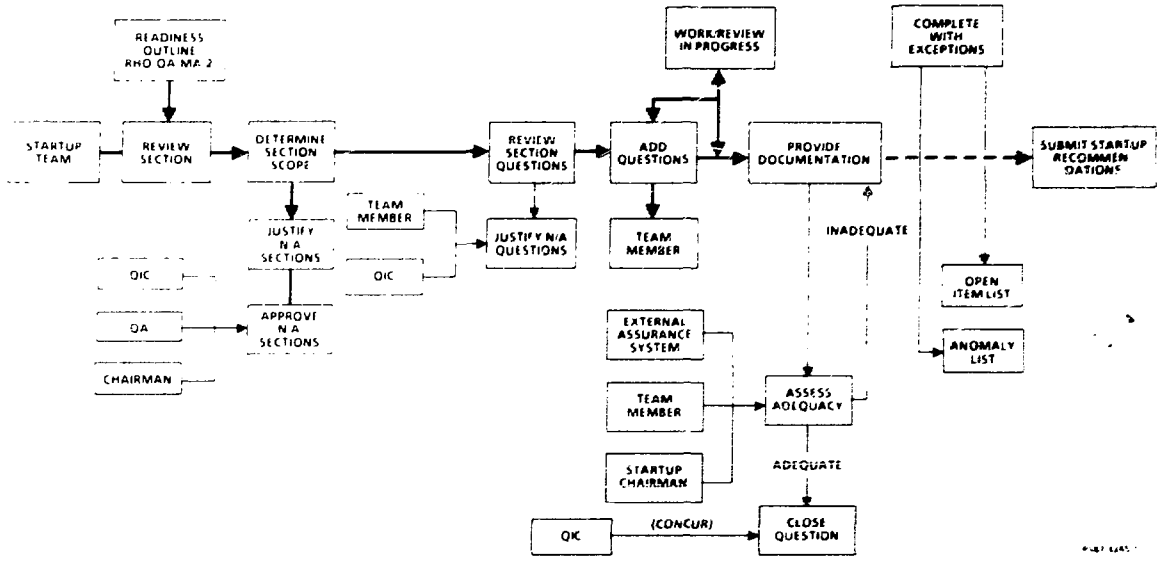

Figure 1. Startup Team Review Process Flow

System One Readiness Review Outline

The System One Readiness Review Outline is organized into three major areas: Facilities and Related Equipment, Procedure and Managerial Control Requirements, and Personnel. Each of these major areas is broken down into sub-areas, as noted in figure 2. The sub-areas are further broken down; an example of which is shown in Attachment 1, Industrial Safety and Hygiene. This outline is intended to serve as a catalyst to the thought process in evaluating readiness to start a project. It is not intended to be a rigid road map which does not allow deviation.

System One Readiness Review Outline (Cont'd)

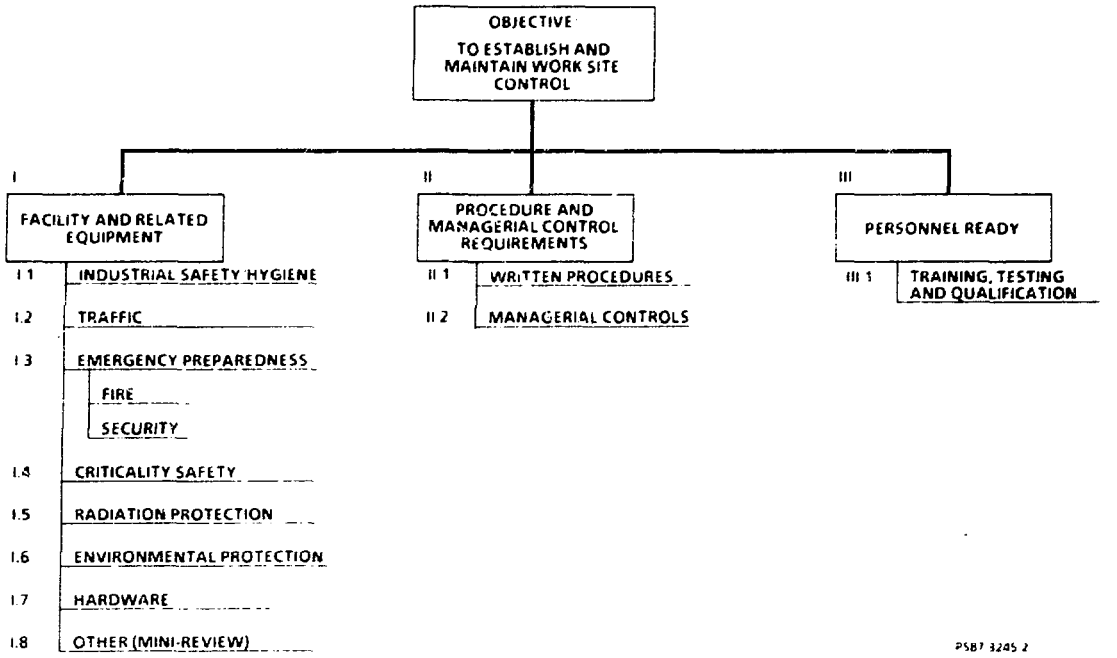

Figure 2. System One Decontamination and Decommissioning Readiness Review Outline

SYSTEM TWO READINESS REVIEW PROCESS

The System Two readiness review method is a two-step process. The first step is the completion of readiness review checklists, which are compiled into a Readiness Review Report. The second step is the Readiness Review Meeting, where participants have a final opportunity to review the readiness checklists prior to project authorization.

The overall compiling of the readiness review checklists is the responsibility of the Decontamination and Decommissioning (D&D) engineer assigned to the project. The checklists cover Engineering, Health Physics, Operations, and Safety requirements which must be addressed prior to the start of decommissioning activities.

SYSTEM TWO READINESS REVIEW PROCESS (Cont'd)

The engineering portion of the checklists is completed by the assigned D&D engineer and covers the following topics: project scope definition, photo documentation, design information review, on-site inspections verifying drawings and identifying security matters, work task identification, detailed work procedures and job safety analysis, environmental and hazard assessment documentation, site preparation, and cost estimates.

The health physics portion of the checklists is completed by the D&D health physicist assigned to the project and covers the following topics: project scope review, radiological data review and radiological surveys, drawings and operational history, sample requirements, type and extent of surveys, release and survey criteria, preliminary and final Allowable Residual Contamination Levels (ARCL) calculations, As Low As Reasonably Achievable (ALARA) considerations, radiation work procedure review, environmental controls, and special instrumentation.

The operational portion of the checklists is completed by D&D Operations personnel and covers the following topics: documentation and posting requirements, training, equipment and materials needs and inspection, security matters, and industrial and radiological safety.

When all aspects of the project have been systematically reviewed and the checklists have been completed, the D&D engineer assures that all checklists are signed off and compiled in a Readiness Review Report with all appropriate supporting documentation. The next step in the readiness review process is the formal Readiness Review Meeting.

Participants in the Readiness Review Meeting have direct and indirect involvement in the project. In addition to the assigned engineer, health physicist, planner, and operations supervisor, the managers of D&D Engineering, Health Physics, Planning, and Operations have direct involvement. The personnel indirectly involved are the D&D Department manager and the Department of Energy representative.

At the formal meeting, copies of the Readiness Review Report are distributed for review. The material is discussed and questions are resolved. Questions which are not adequately addressed in the report are resolved subsequent to final project authorization. Once the participants are satisfied that all relevant aspects of the project have been adequately addressed, the D&D manager gives a formal approval to begin the project as outlined in the detailed Decommissioning Work Procedures (DWP) for the project.

System Two Readiness Review Outline

The System Two Readiness Review Outline is organized into Safety, Operations, and Engineering requirements. Each of these major headings is detailed further as shown in figure 3. Many of the subheadings shown in figure 3 are further broken down as shown in Attachment 2. This outline is also designed to serve as a catalyst for a readiness thought process and is not intended to be a cookbook type of review outline.

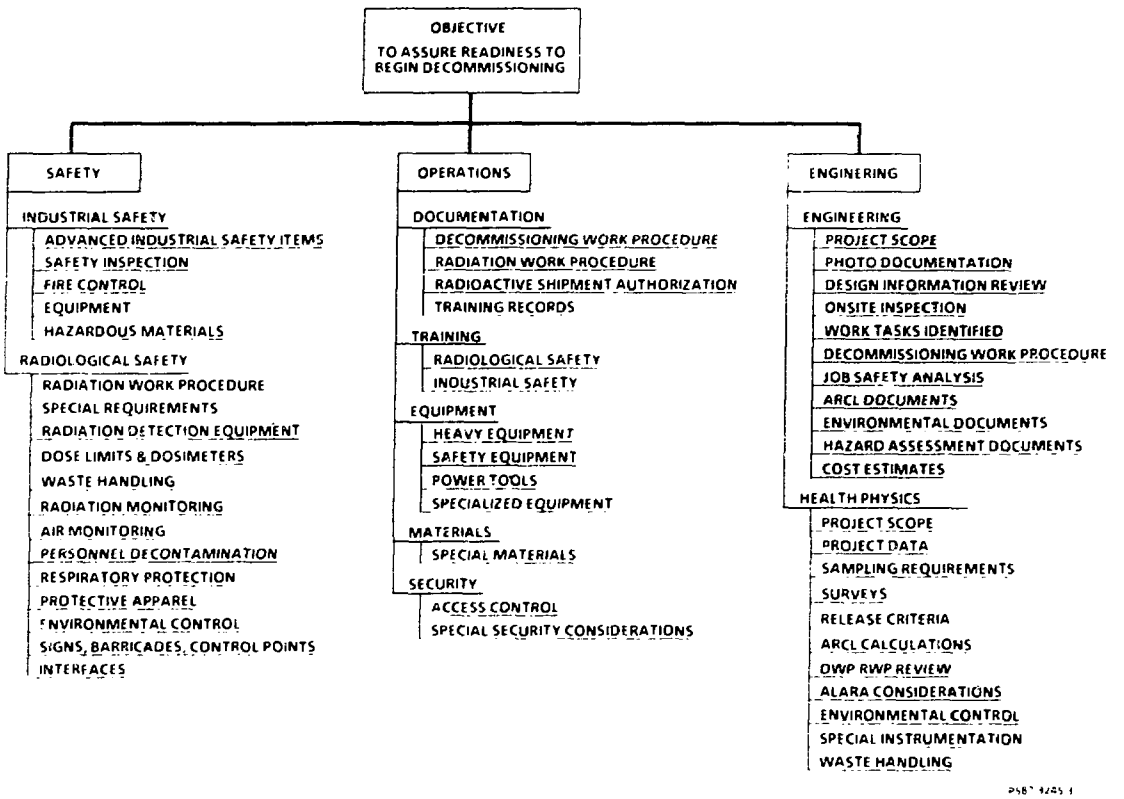

Figure 3. System Two Decontamination and Decommissioning Readiness Review Outline

EVALUATION OF SYSTEM ONE AND SYSTEM TWO

Both systems have good and bad qualities. System One is highly formalized with a system of checks and balances to assure top management of a detailed review. At the same time, System One does not incorporate a standard, concise format for documenting readiness for each individual review item. System Two lacks the formal one-over-one type of review of each readiness item; instead, it relies heavily on the reviews of individuals. System Two does utilize a concise format for gathering together all of the major documentation related to a decommissioning task such as review checklists, environmental documentation, safety analysis, etc.

Unified Readiness Review System

The Unified Readiness Review System combines the best features of System One and System Two and incorporates a new method to assign review categories. The highly formalized System One, with a system of checks and balances, is used to assure top management review when needed as described below in Category 3. The System Two Readiness Review, using the less formal Readiness Review meeting, is used when needed as described in Category 2 below. A new level of review, described below in Category 1, is also now being used. The concisely formatted Readiness Review Report from system two is also incorporated into all categories of the Unified System.

In the Unified Readiness Review, the project scope is assessed before the project begins to compare the project's relationship to other previously performed projects based on the tasks involved, environmental and occupational safety, and quality assurances. Based on this assessment, the managers of the D&D Engineering and D&D Operations put the project into one of three categories, described below as Category 1, Category 2, or Category 3.

Category 1. The tasks identified in the project scope are comparable to previously performed tasks and all environmental and occupational safety and quality assurance considerations are also comparable. Therefore, the tasks may be performed via a letter from the Manager, D&D Department, authorizing work to commence based on the reviews and recommendations of the Managers of Decommissioning Operations and Decommissioning Engineering. Documents used for comparative review are referenced to file.

Category 2. The tasks identified in the project scope are comparable to previously performed tasks and all environmental and occupational safety and quality assurance considerations are also comparable, but some additional safety and/or quality assurance concerns may exist. An "Informal" Readiness Review Meeting is held. All items that are different from the comparison documents are identified and all additional safety and/or quality assurance requirements are included on "Informal"

Readiness Review Checklists. An informal Readiness Review Meeting must be held (as described in System Two) to assure that everything is ready prior to commencing work activities.

As a minimum, the Informal Readiness Review (IRR) Committee members will consist of representatives from D&D Operations, D&D Engineering, Safety, and Quality Assurance. The work scope tasks may be authorized via a letter from the Head Manager of Defense Waste Management and Decontamination and Decommissioning, based on the review and recommendation of the Informal Readiness Review Committee.

Category 3. The tasks identified in the project scope, when compared to previously performed tasks, have major differences, are new or unique, and/or are of major significance to warrant a full, formal, readiness review as described in System One.

CONCLUSION

Combining System One and System Two readiness review processes is resulting in efficient, uniform methods to prepare for decommissioning projects and providing uniform concise documentation for all Hanford Site projects.

The project readiness reviews are more efficient because work which is similar to previous projects does not require extensive top-level management review, thus allowing "routine" work to progress must faster from readiness review to project commencement. Other projects which are more complicated receive stricter reviews by different levels of management, depending upon job complexity.

These different review levels or categories also allow for more flexibility in arranging projects during the year because routine projects can be slipped into the schedule and implemented ahead of schedule if needed.

Lastly, the concise gathering of readiness review documentation into a Readiness Review Report benefits all projects because information for each particular project is uniformly laid out and readily accessible for third-party review and historical record.

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1. Decontamination and Decommissioning Readiness Review Outline, RHO-QA-MA-2, Rockwell Hanford Operations, Richland, WA, 1984.

Attachment 1. Industrial Hygiene and Safety Review Items
System Two Readiness Review Outline

1. Are systems controlled by lock and tag? Identified and implemented? Assessed for adequacy?
2. Are physical safety systems operational and functioning? Can any be inadvertently by-passed?
3. Are required utilities in place? Operational?
4. Ventilation system balanced and functioning? Difference of Pressure tested?
5. Approved storage for flammable material available?
6. Restrooms, change rooms, eating facilities available? Field office space available?
7. Routine services provided, i.e., bus service, snow and trash removal?
8. Recorders with time indicators set properly?
9. Are containments available for sampling/maintenance, i.e., plastic wrap, pre-fabricated tents?
10. Are special handling devices required? Identify.
11. Identify unique or special cleaning materials, equipment or waste disposal requirements.
12. Warning signs in place, i.e., Restricted Area, Keep Out, On-Mask Only, Radiation, High Voltage?
13. Sufficient lighting available?
14. Noise control/protection required and available?
15. Are all fixtures and equipment requiring dismantling tagged or identified?
16. Evacuation routes designated and posted?
17. Personnel protective clothing provided:
    Head - Hard hats or other head covering.
    Eye - Safety glasses or goggles.
    Hand - Gloves
    Foot - Safety shoes, shoe covering.
    Hearing - Hearing protectors, ear plugs.
    Respiratory - SCBA
    Skin - Coveralls
    Special Work Procedure Clothing
18. Occupational hazards reviewed:
    Electrical
    Mechanical
    Chemical
    Explosive (gases)
    Combustible
    Flammable
    Pyrophoric
    Corrosive
    Toxic: Poison
    Pathogenic
    Carcinogenics
    Pressure
    Nuclear Radiological

Acoustical
Thermal
Kinetics
Mass, gravity, height
Biological agents (snakes, spiders, etc.)

19. Will any confined space entry be required? Is it adequately controlled?
20. Are adequate barriers established to separate D&D from operations?
21. Is demolition of a facility involved? Have all requirements of Industrial Health and Safety been met?

Attachment 2. Advanced Industrial Safety Review Items
System Two Readiness Review Outline

a. Are access routes safe for vehicles required?

b. Are loading and unloading areas adequate?

c. Are access doors operable?

d. Are services to the site functional?

e. Are working surfaces adequate for personnel and equipment?

f. Are heavy lifting needs identified and equipment in serviceable condition?

g. Are safety rails and barricade needs identified?

h. Are electrical systems able to be energized?

i. Specific safety concerns. (List)

j. Have piping systems/conduit been identified in accordance with Special Procedures?

k. Have air samples been taken for oxygen content?

DEVELOPMENT AND VERIFICATION OF CODE SYSTEM FOR MANAGEMENT OF REACTOR DECOMMISSIONING (COSMARD) FOR THE JPDR DECOMMISSIONING

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ABSTRACT

A code System for management of reactor decommissioning (COSMARD) has been developed to estimate management data such as manpower, worker exposure, cost, related to the decommissioning of the Japan Power Demonstration Reactor (JPDR). It is also possible to make schedules of the dismantling activities by PERT calculations using the data in the code system. We have used the COSMARD to evaluate and plan the dismantling activities in the JPDR decommissioning program. During this study, the COSMARD proved to be a useful tool for the management of the decommissioning activities.

INTRODUCTION

In planning the economical and safe decommissioning of nuclear power plant, management data such as manpower, worker exposure, cost needs to be estimated. However, it is not easy to estimate these data because they vary with decommissioning procedure and inherent plant characteristics. Therefore a code system for managing and planning reactor decommissioning is required to easily estimate these data according to the various decommissioning scenarios. Japan Atomic Energy Research Institute (JAERI) has developed COSMARD as one of the developmental techniques* of the JPDR decommissioning program.[1),2)] The COSMARD was used to evaluate JPDR decommissioning management data. The preliminary planning was made by PERT calculations contained in the COSMARD using the management data estimated under realistic JPDR decommissioning conditions. Dismantlement of JPDR started in December, 1986. A wide variety of decommissioning data has already been collected through the program. These data will be used to verify and if necessary improve the COSMARD. The code system will be applied to planning and managing of commercial power reactor decommissioning in the future.

This paper describes the results of the evaluation of the JPDR decommissioning plan by the COSMARD and compares dismantling activity calculated and collected data.

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* This work was performed by the Japan Atomic Energy Research Institute under contract from the Science and Technology Agency of Japan.

CONCEPT OF THE COSMARD

Methodology

Work breakdown structure. Before developing the COSMARD, dismantling activities were analyzed by Work Breakdown Structure (WBS) in a technical design study for the JPDR decommissioning. Figure 1 shows an example of activity analyzed by the WBS. Furthermore, each of the tasks in the dismantling activities were placed into one of the five categories, preparation, dismantling, packaging, shipping, and cleaning up, in order to simplify the dismantling activities. The calculation sequence reflects the analyzed dismantling procedure in the COSMARD. Therefore, the management data calculated by the COSMARD are expected to correspond closely with those of the actual JPDR decommissioning.

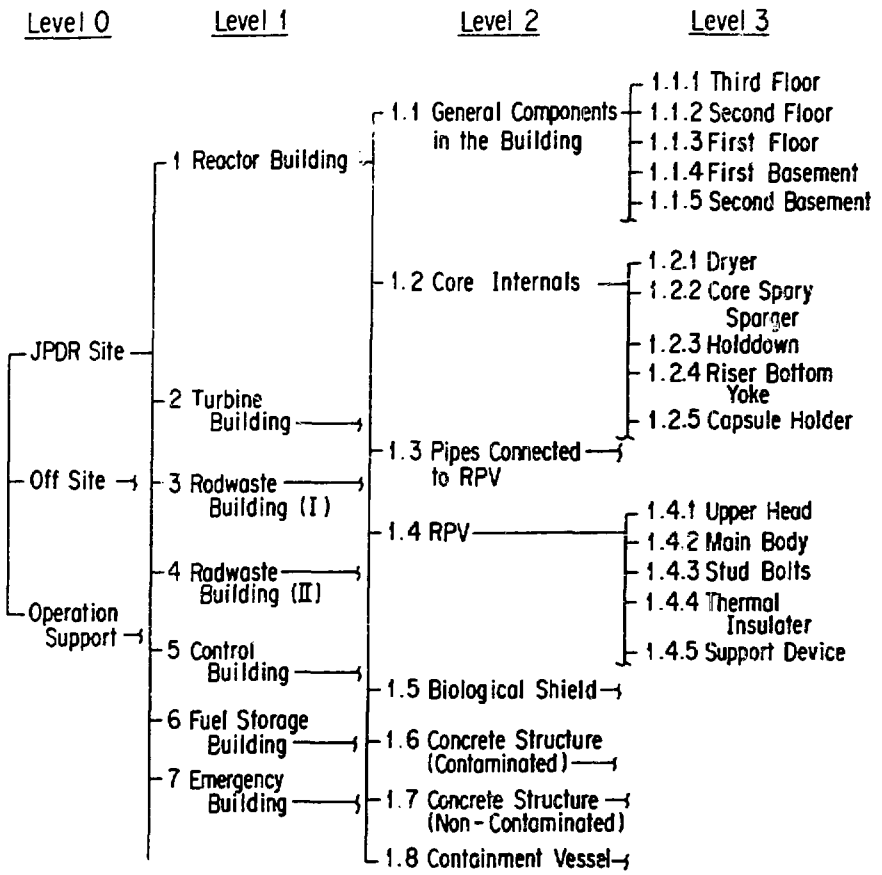

Fig.1 WORK BREAKDOWN STRUCTURE FOR THE JPDR DECOMMISSIONING

Unit manpower factors. Having stressed the importance of appropriate planning for the decommissioning activities, efforts were made to standardize manpower factors for various dismantling activities by using the WBS. A manpower expenditure for each activity is calculated based on the unit manpower factor which is contained in a data base of the COSMARD. The unit manpower factors are characterized according to the activities. The difficulty of each dismantling activity in a high radiation environment and/or confined space was accounted for applying a suitable multiplication factor.

Dismantling techniques. Various cutting techniques with remote handling will be applied to the dismantling activities for highly activated and contaminated components in the reactor building. The dismantling sequence and related cutting techniques developed in the program are shown in Fig. 2. The COSMARD has the capability to evaluate the management data regarding activities using techniques developed. Characteristics of the cutting devices such as cutting speed and utility consumption are contained in the data base to evaluate the management data when applying these techniques.

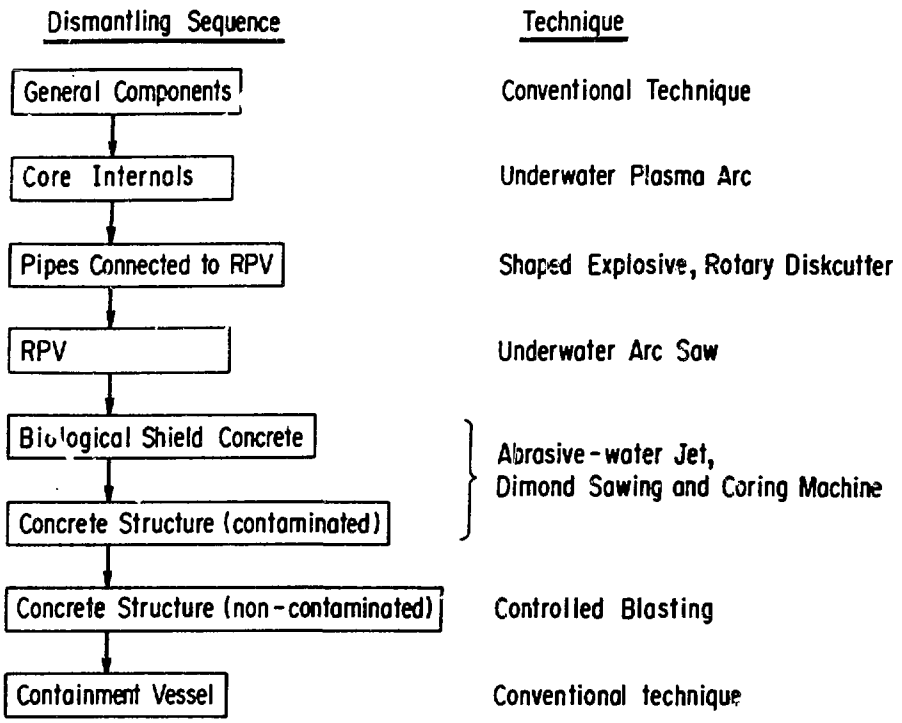

Fig.2 SEQUENCE AND TECHNIQUE FOR DISMANTLING THE REACTOR BUILDING

Cost estimation. Decommissioning cost is evaluated according to activity dependent and period dependent items, as normally used for the cost estimation.[3] The activity dependent items are related to manpower, number of containers, utility consumption, etc. The period dependent items are related to contingency, routine maintenance, rental fee of temporary buildings etc. The total decommissioning cost is estimated for each year by adding the activity dependent costs expended in the year to the period dependent costs. A few percent inflation rate is also included in the cost estimation.

Structure Of The COSMARD

Figure 3 shows structure of the COSMARD. As shown in the figure, the COSMARD consists of three parts; a management data calculation code, PERT (Program Evaluation and Review Technique) calculation code and a data base of the JPDR decommissioning. These parts contain the following elements.

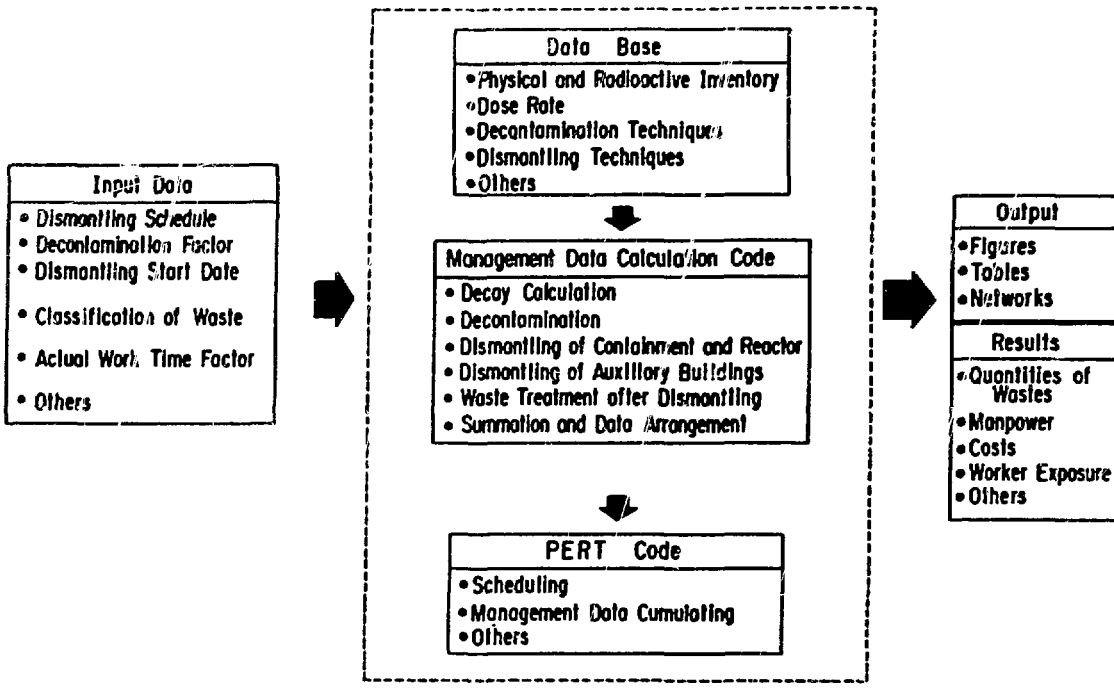

Fig. 3 DESCRIPTION OF THE COSMARD

Management data calculation code. This code has subroutines to calculate: (1) radioactive inventory, (2) management data from decontamination activity before the dismantling, (3) management data from dismantling of

reactor building and components, (4) management data from dismantling of auxiliary building and components, (5) management data from activities after completion of the dismantling, and (6) data processing.

Radioactivities of components are calculated considering the decay time in subroutine(1). Dismantling wastes are categorized into five radioactive levels. The total waste and radioactivity of each level are also calculated by this subroutine. Management data from dismantling are derived in subroutine (2) through (5). Calculated results are summed in subroutine (6) to output and transfer for scheduling.

Scheduling. The calculated results of the management data calculation code are transferred to the PERT code. They are also summed to derive a schedule. The management data are obtained as a function of days together with main scheduling parameters such as critical path and network relation between the activities.

Data base. The data base has six subfiles which include physical and radioactive inventories, dismantling procedures and machine performance used for the dismantling. The physical inventory file contains plant component information such as location, weight, volume and characteristics. The radioactive inventory file has becquerel numbers of activated nuclides contained in the components. These are Co-60, Ni-59, Mn-54, Eu-152, C-14, Fe-55, etc. The other files contain information to use for the calculation of manpower expenditures and container numbers, etc.

RESULTS AND DISCUSSION

An estimate of the amount of radioactive waste from the dismantling is essential to the decommissioning planning work. Therefore we calculated the amounts of radioactive waste according to the activity levels and materials using the COSMARD. Then, the JPDR decommissioning plan was analyzed to clear the features based on the estimated data.

Estimate Of Waste Arising From The JPDR Decommissioning

The total amount of waste arising from the JPDR decommissioning is estimated by the COSMARD to be 27,800 tons. This includes 3,300 tons of metals, 24,420 tons of concrete and 80 tons of the other materials such as glass and asbestos. In Fig. 4, the estimated weights of the radioactive waste categorized by the activity levels as a function of years of reposal are shown. Since actual JPDR dismantling has started after more than 10 years of reposal, the amount of the radioactive waste is expected to be 420 tons of metals and 790 tons of concrete. By waiting for 10 years, the amount of radioactive metals and concretes decreased to 80% and 50% the initial weights, respectively. Japan has not yet selected radioactivity levels associated with various methods of waste disposal. However, weights of radioactive waste are estimated based on expected radioactivity levels. The results are shown in Fig. 5. Since the concrete waste is distributed mostly among the activity level between 10^{-7} and 10^{-6}Ci/ton, the amount of radioactive concrete waste increases when decreasing the exemption level below 10^{-6} Ci/ton.

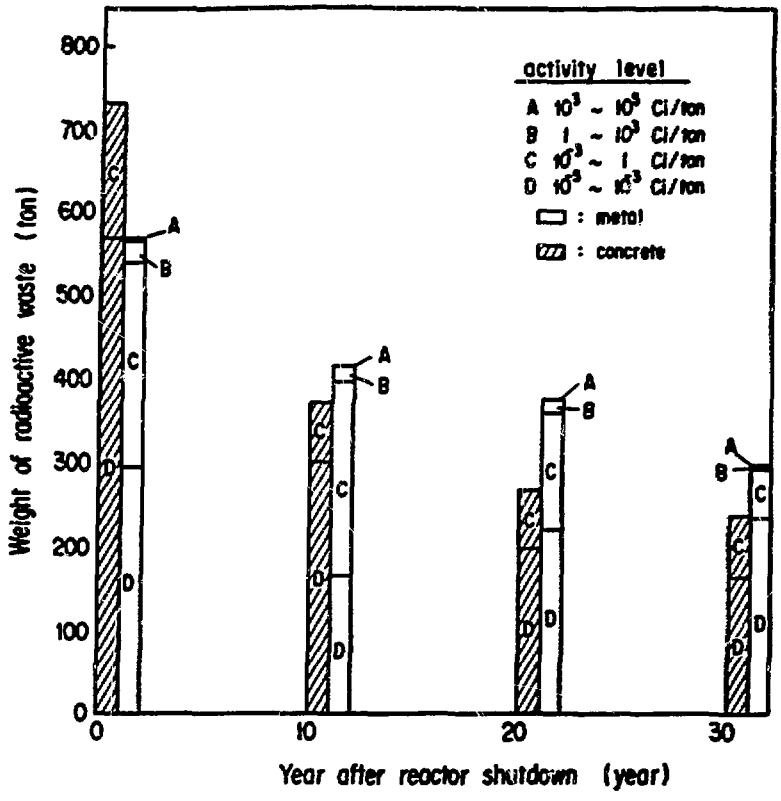

Fig.4 ESTIMATED WASTES AS A FUNCTION OF REACTOR SHUTDOWN DURATION

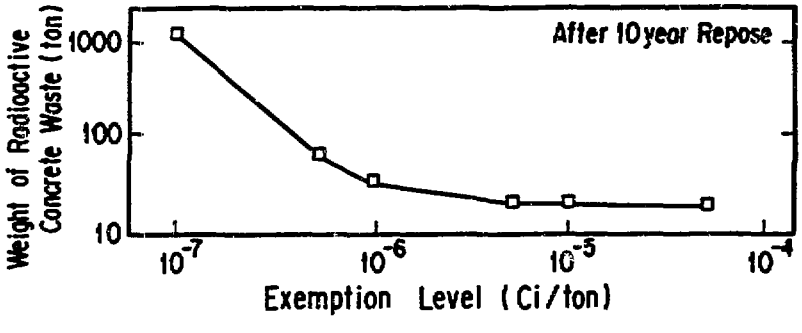

Fig.5 WEIGHT OF RADIOACTIVE CONCRETE WASTE AS A FUNCTION OF EXEMPTION LEVEL

Evaluation Of JPDR Decommissioning Plan

We evaluated the JPDR decommissioning plan under actual decommissioning conditions. The calculation parameters are shown in Table I. The results are as follows.

Table I. PARAMETERS FOR ESTIMATION BY THE COSMARD

- Dismantling Start Date	12/1986
- Dismantling Schedule	6 year
- Dismantling Region	A*
- Expected Decontamination Factors	
· Pre-dismantling	
Chemical Cleaning	10.0
System Flashing	1.2
Tank Flashing	2.0
Water Jet	5.0
· Post-disassembled Components	
Electropolishing	500.0
Immersion Decon.	15.0
- Activity level applying for decontamination of concrete surfaces	$\geq 10^{-5}$ Ci/ton
- Actual Work Time Factor	6 hour/day
- Classification of Waste	
High	$\geq 10^{3}$ Ci/ton
Intermediate	$\geq 10^{0}$
Low	$\geq 10^{-3}$
Extremely Low	$\geq 10^{-5}$
Nonradioactive	$< 10^{-5}$

Reactor building : except for basemat, Other buildings ; except for the basement below 1m from the ground

Management data of JPDR decommissioning. Table II shows the estimated manpower expenditures, worker exposures and costs associated with dismantling activities for each building during the JPDR decommissioning. For reference, the amounts of wastes arising from the dismantling activities in each building are also shown in the table. The dismantling activity in each building consists of dismantling of components, decontamination of concrete surfaces, demolition of the building and restoration of the site by landscaping. As shown in the table, the total manpower expenditure is estimated to be 83,240 man-days with a worker exposure of 82.5 man-rem. The total cost is estimated to be approximately 11 billion yen excluding period-dependent items. The reactor building will require approximately 40% of the manpower to be expended in the dismantling activities and will contribute about 80% of the total worker exposure.

Management data of reactor building dismantling. The dismantling activities in the reactor building are divided into seven components to be dismantled. Therefore the manpower expenditure and worker exposure estimated above are further classified into seven items as shown in Fig. 6.

The reactor internals and the reactor pressure vessel (RPV) are highly activated. Therefore manpower expenditures will be high for the dismantling activities of those components due to the use of remote cutting techniques.

Table Ⅱ. MANAGEMENT DATA CALCULATION RESULTS

Level 1. Activity	Waste (ton)	Manpower (man-day)	Worker Exposure (man-rem)	Cost (%)
Pre-dismantling Decontamination	——	240	0.8	0.1
Reactor Building	6,100	33,100	65.0	47.2
Turbine Building	5,900	12,900	4.5	12.7
Radwaste Building	3,500	6,100	6.0	5.5
Control Building	3,600	3,400	0.2	3.2
Other Building	7,800	15,700	3.3	18.4
Outside Facilities	900	2,000	1.1	2.0
Waste Treatment after Dismantling	——	9,800	1.6	10.9
Total	27,800	83,240	82.5	100.0

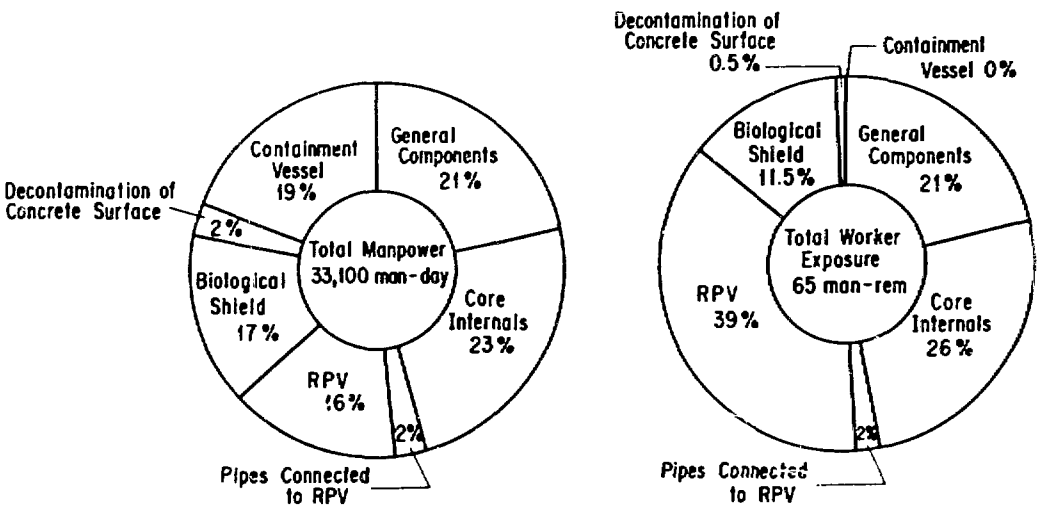

Fig. 6 ESTIMATED MANPOWER AND WORKER EXPOSURE IN REACTOR BUILDING DISMANTLEMENT

The worker exposure during these dismantling activities is estimated to be approximately 65% of the total reactor building dismantling. Figure 7 shows the worker exposure and manpower expenditure arranged in five categories regarding the RPV main body dismantling activity. Since these components are dismantled under water using the newly developed cutting techinque and remote handling, worker exposure of the dismantling activity is expected to be small. However, a tank will be installed in the cavity between the biological shield and the pressure vessel for the under water cutting. Since activities of preparation and cleaning up include installing and disassembling the tank, respectively, the worker exposure and manpower expenditure are estimated to become larger in the preparation and cleaning up activities compared with the others.

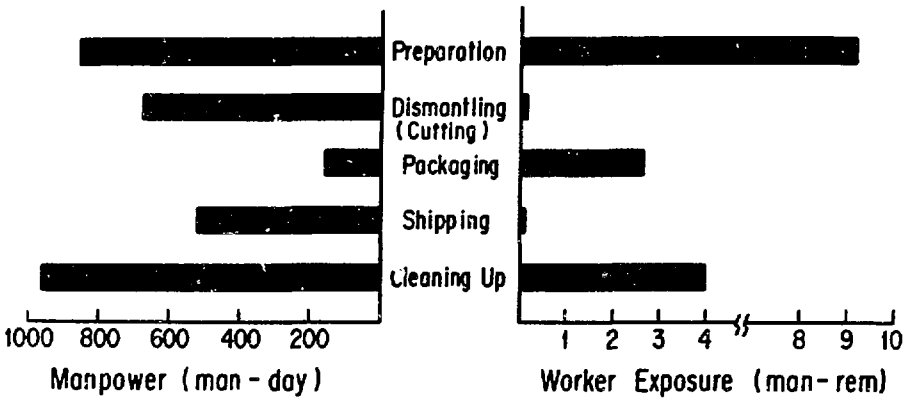

Fig.7 DETAILED MANPOWER AND WORKER EXPOSURE IN RPV MAIN BODY DISMANTLEMENT

Scheduling. To insure safe and economical dismantling activities, it is of great importance to make an accurate schedule. Based on the estimated management data, preliminary scheduling was made for the dismantling activity of each building. Figure 8 is an example of a schedule and histogram of manpower expenditure predicted by the PERT code in the COSMARD. The main input parameters are activity period, network relation between the activities and starting date of the task. The schedules made by the code were evaluated in accordance with the management data derived by various input conditions to evaluate the effectiveness of the planning. Figure 9 shows the histogram of manpower expenditure and the cumulative worker exposure during the whole decommissioning period based on the JPDR decommissioning plan evaluated by the parameter study. A large amount of manpower is expected to be expended at the project beginning and after 1991. In the beginning, general components with no or low radioactivity will be dismantled in the buildings to provide working spaces. All remaining components and buildings will be dismantled simultaneously after 1991. These are the reasons for the increase in the manpower expenditure at the beginning and end portions of the dismantling project.

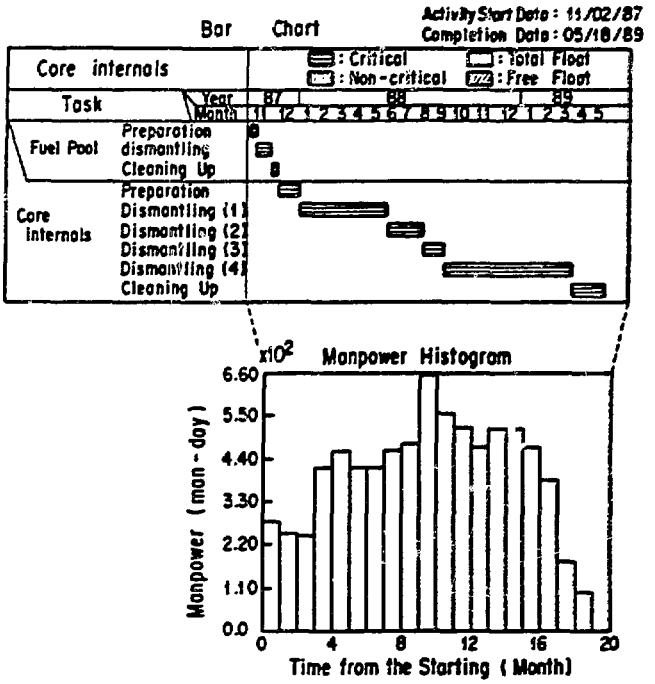

Fig.8 BAR CHART AND MANPOWER HISTOGRAM OF CORE INTERNALS DISMANTLEMENT

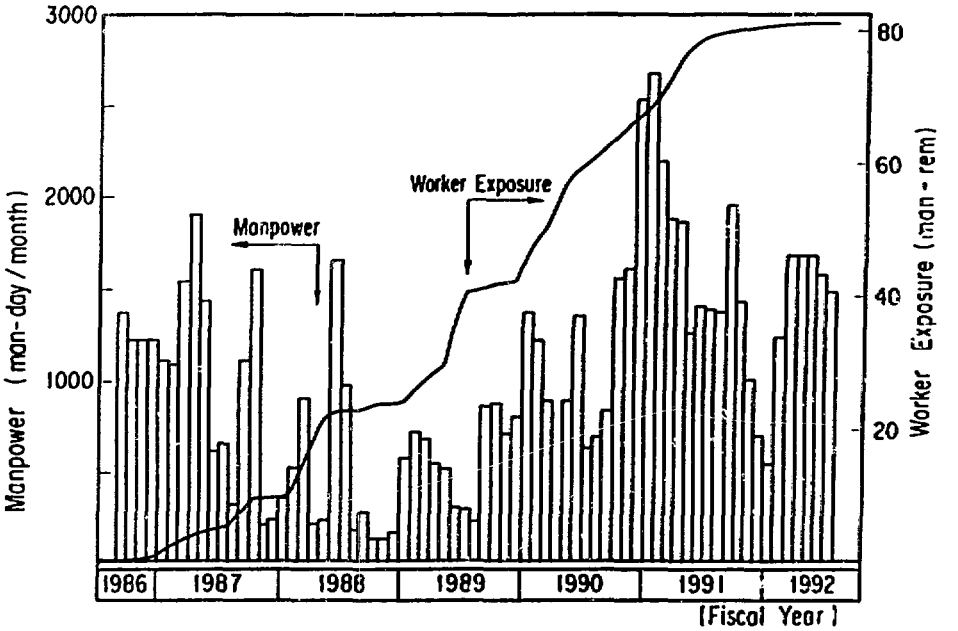

Fig.9 MANPOWER AND CUMULATIVE WORKER EXPOSURE OF JPDR DISMANTLEMENT DERIVED BY THE COSMARD

Verification

Data collection. A wide variety of useful data can be expected from the JPDR decommissioning. These data will be collected and processed with a data collection and retrieval system developed by JAERI. The collected data are to be used for verification of the COSMARD as well as management of the JPDR decommissioning. By March, 1987, the upper head of the reactor pressure vessel and components in the reactor building and others had been dismantled. The management data collected in the dismantling activities were stored in the data collection system.

Comparison with collected data. The management data calculated by the COSMARD were compared with the data collected during the dismantling activities. Figure 10 shows both the estimated and collected cumulative manpower expenditures as a function of activity period. The actual manpower expenditures have also been compared with the estimated data concerning the dismantling activity of the upper head of the reactor pressure vessel. The calculated and collected manpower expenditures are almost identical.

A detailed study for the verification of the COSMARD is in progress to prove the applicability to future commercial power reactor decommissioning.

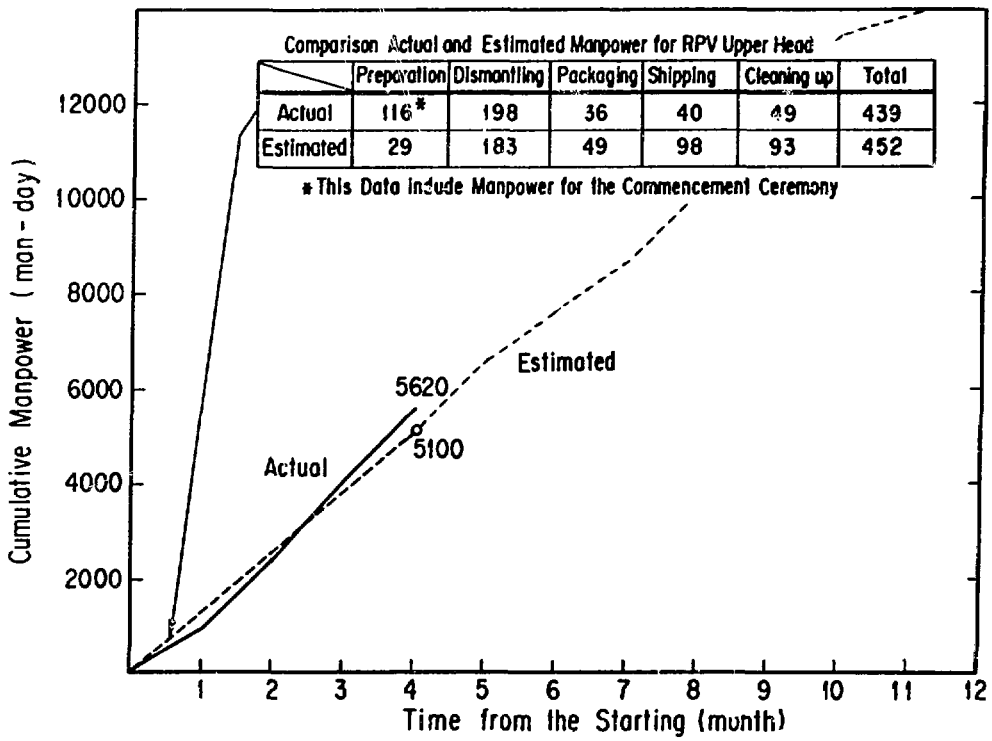

Fig.10 CUMULATIVE MANPOWER FOR JPDR ESTIMATED AND ACTUAL

CONCLUSIONS

In order to estimate the suitable management data for safe and efficient decommissioning of the JPDR, a code system for management of reactor decommissioning (COSMARD) has been developed. The code can estimate detailed management data based on unit manpower factors for standardizing each dismantling activity and has the capability to make schedules using the estimated management data. The COSMARD was applied to estimate management data for the planning of the JPDR decommissioning. Many features and detailes of the JPDR decommissioning were included in a parameter study by the COSMARD. Furthermore, a comparison between calculations and actual data showed that the calculated manpower expenditures were almost identical with the data collected during the dismantling activities. The COSMARD will be improved, if necessary, through verification. Then it will be applied to planning and management of future commercial power reactor decommissioning.

ACKNOWLEDGMENT

The authors wish to express our appreciation to Messrs. K. Kamike, H. Ishikawa and S. Ohkawa for their helpful advice and valuable discussions.

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COMPUTER PROGRAM FOR ESTIMATING DECOMMISSIONING COSTS OF NUCLEAR POWER PLANTS

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ABSTRACT

Reliable cost estimates are essential for the planning of a decommissioning program for a nuclear facility. The preparation of such cost estimates is a complex task because there are a large number of components, equipment, and piping systems to be considered, many different levels of radioactive contamination and activation levels to be examined, corresponding waste volumes for disposal to be determined and multiple decommissioning scenarios to be evaluated.

While conducting the initial decommissioning studies on the Gentilly-1 Nuclear Station a computer program called DECOM was developed by AECL to address this requirement. It utilizes close to 150 models or Unit Cost Factors (an approach rapidly becoming the accepted industry estimation methodology) to describe the various decommissioning activities and, in addition to providing decommissioning cost estimates, it can compute predicted radiation exposure to workers and waste volumes for nearly any decommissioning scenarios.

The DECOM computer program was initially developed in 1983 for use on an IBM main frame computer. In 1985, in order to provide more flexibility and ease of operation for the users of the DECOM computer program, the DECOM Code was converted so as to be useable on an IBM-PC type microcomputer.

The DECOM program has been used in the past few years to do decommissioning estimates for both CANDU (Canada Deuterium Uranium) and PWR (Pressurized Water Reactor) type reactors. The estimates have been found to be within the range reported in the OECD (Organization for Economic Cooperation and Development) decommissioning cost surveys. It can be adapted to other types of reactors and may be extended to non-nuclear facilities with suitable modifications of the cost codes.

In order to test the validity of the cost estimates prepared using DECOM, a sample of actual cost and manhour data from the Gentilly-1 Decommissioning operation have been processed through the DECOM code and it was observed that the total cost figures were accurate within a 20% range, though costs for individual activities in some instances differed.

The AECL DECOM computer program, based on the widely accepted unit cost factor approach, is a versatile tool for applications in decommissioning studies. Current incorporation of the DECOM code into a format suitable for use on a microcomputer provides even greater flexibility and wider accessibility.

INTRODUCTION

As many nuclear power plants around the world approach the end of their expected lives, decommissioning has taken on more than an academic interest. Worldwide, a number of plants have been offered for various stages of decommissioning in recent years. These include Gentilly-1 in Canada, the power demonstration reactor (JPDR) in Japan, Windscale (WAGR) in the United Kingdom, Shippingport in the USA, and several others within the European Community.

Currently, there are several decommissioning alternatives that are technically, socially, and politically acceptable. A reliable cost estimate is essential to assist in the planning and selection of the most suitable decommissioning programme from among the options available, and to establish a practical funding mechanism for it.

The preparation of such cost estimates is a complex task because there are a large variety of plant inventories, radioactivity levels and, waste volumes and decommissioning options. In addition, a number of "what-if" questions regarding cost-benefit analysis must be answered before a decision is made as to which stage a plant will be decommissioned.

AECL faced this situation in 1983 during the initial decommissioning studies on the Gentilly-1 nuclear station and came to the conclusion that a computerized cost model was essential to permit the analysis of numerous decommissioning scenarios and for optimization purposes. To address this requirement, a computer program called DECOM was developed for use on an IBM main frame computer using a utility program called ADRS (A Departmental Reporting System).

COST ESTIMATING METHODOLOGY

Preparation of cost estimates be a straight forward task if all cost components are known and the estimating methodology is well defined.

The components that make up the total cost for decommissioning a nuclear plant can be grouped into four categories, each of which need to be handled in a slightly different fashion. These are:

a) activity dependent costs
b) period dependent costs
c) special activity costs.
d) dormancy period costs (if delayed dismantling is envisaged)

The flow diagram in Figure 1 describes the methodology developed below.

Activity dependent costs are those associated with tasks that are discrete, measurable and of a repetitive nature and can thus be analysed by developing typical Unit Cost Factors (UCF)* which can be applied to the category of equipment that they represent (e.g. cutting pipe, removing pumps, dismantling structural steel). The type and category of equipment which are to be dealt with dictates the type and number of cost factor models that need to be developed for a given nuclear facility.

Period dependent costs are those associated with the durations of different phases of the project such as engineering, project and construction management, licensing, quality assurance and security.

The duration of the total program is established by the dismantling experts taking into consideration the most efficient methods of removing all materials. Once the schedule is established, the critical path is then identified by processing the schedule through a scheduling computer program. The project schedule is then used as a base to determine the period-dependent cost.

Special activity costs are split into two categories: special items that are non-repetitive such as the reactor vessel removal and miscellaneous items such as the operation and maintenance cost, cost of energy and the like.

The activity dependent, period dependent and special item costs are amalgamated to develop the total decommissioning cost.

In the case of delayed dismantling, costs have to be calculated for a delay or dormancy period which may vary from over between 40 years to 100 years (typically).

Once the total costs are calculated, cash flow requirements can be obtained based on the schedule. Project costs can be updated periodically as needed, taking into consideration the appropriate escalation factors, technology developments, waste disposal/transport costs, and changes in the Regulatory requirements etc.

* Defined under DECOM

COMPUTER PROGRAM DEVELOPMENT

DECOM

The DECOM computer program mentioned earlier, makes use of an on line, interactive data base and was developed for estimating decommissioning activity dependent costs, waste volumes and worker radiation exposure. It is based on the Unit Cost Factor approach. The users key in the plant inventory with the proper equipment code, site location, quantity, radioactivity level and schedule activity. The computer program calculates the costs and the radiation exposure for the various operations based on the inventory and the built-in (user defined) Unit Cost Factors. The result of calculations can be sorted in a multitude of ways and at many levels, such as by buildings, rooms, schedule activities, radioactivity levels and equipment categories. The program is very flexible and allows for easy addition, deletion and rearrangements data. Furthermore, the data base spread sheet can accept up to hundreds of fields and as many inventory items as the user may desire. This flexibility represents one of the many attractive features of the program, since it permits a user to carry out optimization studies for a large variety of decommissioning scenarios.

Each entry in the data base (a record listed in a row) is a collection of fields (listed in columns) chosen to contain useful information such as physical characteristics, radioactivity levels, unit cost data (cost factors) and so forth, for each equipment and structure for a given nuclear plant.

The data base is divided into several subfiles (i.e. technical data of an equipment is in one subfile, cost data in another subfile, etc).

The main body of the Data Base contains information on:

1. Unit Cost Factors
2. Physical inventory of the plant.
3. Radiological inventory.

These are described in more detail below.

Unit Cost Factors

For the purposes of the DECOM computer program, AECL has developed a total of approximately 150 Unit Cost Factors which represent up-to-date experience with the various techniques applied to the decommissioning of nuclear power stations.

Unit Cost Factors are essentially models which take into consideration all the typical activities associated with, for example, the exercise of dismantling piping. They consider manpower requirements, duration of various tasks and special equipment involved. By factoring in labour costs, equipment rental or maintenance costs and so forth, the model then converts the whole range of activities into a cost, expressed in dollars per unit ($/Ftrun in the case of piping). The cost factor is thus defined as the estimated amount of money required to remove one unit of a component. Unit Cost Factors have been developed for dismantling, packaging, transportation, and disposal for a variety of categories of equipment.

The Data Base maintains all the necessary Unit Cost Factors in a special file and uses it to calculate cost. The Unit Cost Factors can be updated as required to reflect changes in technology, actual hands-on experience and so on. They can also be adapted to local conditions by factoring in local labour rates and productivity. Furthermore, new Unit Cost Factors can be developed as required for different types of nuclear plants. This is a powerful feature which allows the program to be kept up to date with a minimum amount of effort.

Physical Inventory of the Plant

The DECOM program requires the input of information on the total physical inventory (equipment, materials, structures) of the plant. The physical properties (size, volume, weight, type of material, etc.) of each piece of plant component are entered into the Data Base utilizing specially designed input sheets. The information on component properties can either be obtained from the physical inventory records of each piece of equipment or by extracting this from relevant system drawings. It is preferable, if at all possible, to obtain this information from the physical inventory records.

All plant components are classified by major types of equipment such as pumps, tanks, heat exchangers, valves and piping. These are further subdivided, (for example, pumps and pumpmotors are classified into six subgroups, to allow for the development of more representative Unit Cost Factors. The selection of these groupings depends largely on the type and size of crew required for component removal. Each item is then given a cost code number consistent with the Work Breakdown Structure (WBS) of the project. The allocation of this WBS cost code number is essential for cost roll ups.

Once the basic physical characteristics of all the major plant components have been entered into the Data Base, the program then calculates the total quantities for each type of component and retains this information in a separate subfile for later use.

Radiological Inventory

Radiological information is required by the program to calculate predicted radiation exposure to workers, and to determine the quantities of radioactive wastes by category. AECL has developed procedures for generating radiological inventories by field surveys of all plant components and structures. The radiological data is entered into the Data Base preferably at the same time as the physical inventory data is entered.

Radiation fields (milli-rem per hour) at contact, one meter from the component and the background radiation level in the room are measured. The program can then calculate the total predicted radiation exposure to personnel in each room or area, and provides the cumulative value for the entire plant (project).

DECOM II

The DECOM Computer Program in its first version was mainly used as a database for physical and radiological inventory, and as a tool to estimate the activity dependent costs and radiological doses to the workers. The other components of costs (period dependent, special items, and dormancy period costs) had to be added to the activity dependent costs through another database oriented program. The total decommissioning costs were analysed manually, or by way of some other commercially available computer program.

In view of ongoing and anticipated work in decommissioning studies, a strong need was recognized for improving the DECOM code to make it versatile, powerfull and comprehensive enough to consider all the cost components simultaneously.

In 1985, in order to provide all these features, more flexibility and ease of operation for the users of the DECOM computer program, the DECOM Code was transferred from the ADRS Mainframe program to the dBase III*microcomputer based program. Even though PCs have limitations in the areas of storage space** and speed of operations, they are far less expensive to buy and maintain. This transfer allows greater access of the code to potential users, since an expensive mainframe computer is no longer required.

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* dBase III program is proprietary of Ashton Tate, Culver City, California, USA.

** DECOM runs on an IBM PC-XT or PC with external hard disk drive.

The DECOM II program, as it is called now, is fully menu driven, user friendly, and can be used to prepare a range of cost estimates for any decommissioning project. Each of the components of costs can now be estimated by inputting the relevant data into the DECOM II program through submenus. Operation has been simplified even further by the preparation of a manual that walks the user through the program step by step.

APPLICATION OF THE COMPUTER PROGRAM

Gentilly-1 Plant

During the studies associated with the G-1 plant, decommissioning cost estimates for stages 1, 2 and 3 were prepared, using the DECOM computer program (ADRS Version).

This section describes the step by step approach which was pursued to prepare the G-1 stage 3 decommissioning cost estimate and cash flow. This approach can be applied to any nuclear power plant. Major steps were as follows:

- Survey of equipment inventory
- Application of a computer code
- Survey of radioactive inventory
- Radiological exposure to workers (man-rem)
- Development of unit cost factors
- PERT/CPM network to determine critical path
- Manpower requirements
- Integration of cost and schedule
- Summary of costs
- Financial analysis and cash flow

The physical inventory of all the plant components (equipment, structures, etc.) was obtained from a room by room survey. All the components were grouped first into major equipment categories such as pumps, tanks, heat exchangers. Each major equipment group was further divided into subgroups which could represent a component for any type of plant.

All component items were then entered into the database of the DECOM computer program.

In order to estimate radiological doses to the workers, a survey of radioactive inventory was done for each component in every room of the plant. For each component, two dose readings were taken: at contact (1 cm away) and at one meter distance. The background radiation in the centre of the room was also measured. These data were entered in the DECOM to calculate man-rem exposure to the workers based on the number of workers and the duration they handled the components.

A detailed CPM was developed from a master schedule and was computerized through a CPM/PERT program. Each activity on the schedule was given an identificaiton number (1, 2, 3, etc.). All cost items in the data base associated with one scheduled activity carried the same number, as a link between the CPM program and the cost estimate code. For example, all cost items associated with activity no. 3 - "clear feeders and steam drums", were grouped together and summarized to facilitate accurate cash flow computation. The activity dependent costs associated with the schedule were then added to the period dependent and special item costs to obtain total costs. Several alternative decommissioning scenarios were then studied.

The experience from the G-1 study suggests that although total cost is an important factor in choosing a preferred decommissioning alternative, it may not necessarily be the most dominant one.

Other considerations such as the annual cash availability, future use of the site, availability of a radioactive waste disposal facility play important roles in selecting a decommissioning alternative. Each plant should be treated individually.

It must be emphasized however that the estimates helped in optimizing the decommissioning decisions given the financial and other constraints imposed on the project.

Other Applications

Apart from use on the the Gentilly-1 project, the DECOM program has been used in the past few years in the preparation of decommissioning cost estimates for both CANDU (Canada Deuterium Uranium) and PWR (Pressurized Water Reactor) type reactors. Specific Applications are for SAN ONOFRE Units 1, 2, 3 in the USA, Point Lepreau-2 (hypothetical case) and NPD (Nuclear Power Demonstration Station) in Canada.

The estimates have been found to be within the range reported in the OECD (Organization for Economic Cooperation and Development) decommissioning cost surveys. It can be adapted to other types of reactors and may be extended to non-nuclear facilities with suitable modifications of the cost codes.

VALIDATION OF THE CODE WITH SITE DATA

In order to test the validity of the cost estimates prepared using DECOM, a sample of actual cost and manhour data from the Gentilly-1 Decommissioning operation were processed through the DECOM code and it was observed that the total cost figures were accurate within a 20% range, though costs for individual activities in some instances differed significantly due to changes in the project technical concepts, reduction in the production work day because of unanticipated clothing changes, showers or breaks, increased radiation

protection coverage provided, and special features such as asbestos removal. However, sufficient confidence has been developed in the capabilities of the DECOM Code through this experience. UCFs built into the code have been updated based on this bench marking exercise. The accuracy of the estimates can be further improved by constantly reviewing the UCF's in the DECOM code and adjusting them as required.

CONCLUSION

For effective pre-planning for decommissioning, the need for a credible cost estiamte cannot be over-emphasized. A logical, standardized and consistant estimating method will assist greatly in the decision making process leading up to the selection of a decommissioning alternative and/or in establishing a decommissioning estimate for ratebase setting purposes.

The large amounts of data to be considered, the issue of radiation exposure to workers, the waste volumes generated and the large number of decommissioning scenarios that can be analyzed make this a time consuming and tedious process if done manually. Computerization using a code, like DECOM II, represents a feasible and and attractive alternative to preparing such cost estimates manually.

The AECL DECOM computer program based on the widely accepted unit cost factor approach is a versatile tool for applications in decommissioning studies.

It can manipulate large amounts of data and make use of a large number of models that have been developed expressedly for this purpose. Results can be generated for various decommissioning scenarios and these can be reported in numerous ways which suit the specific requirements of the user.

The computer program has already been used successfully to estimate costs for two CANDU type reactors and three PWRs. It can be adapted to any type of nuclear reactor.

The present DECOM II Computer Program which is a vastly improved version of the DECOM, provides even greater flexibility and acceptability since it is menu driven, user friendly and in a dBase III format that can be run on any IBM PC-XT or equivalent compatible micro computer.

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SECTION V

HEALTH AND SAFETY ASPECTS OF DECOMMISSIONING

ESTIMATION OF THE RISK FROM RESIDUAL RADIOACTIVITY*

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ABSTRACT

A new approach for estimating the potential radiation hazard from residual radioactivity at decontaminated sites and facilities has recently been developed to aid the USEPA's selection of decommissioning criteria for regulatory purposes. The approach is intended to generate conservatively realistic estimates of radiation doses to onsite residents from radioactivity both in the environment and in buildings. It does so using a comprehensive yet relatively simple set of physically-based, state-of-the-art risk-level environmental transport and exposure pathway models. The transport models have been interfaced to permit time-dependent, mechanistic consideration of radioactive transfers within and among the various environmental media. Radioactive decay and ingrowth are explicitly accounted for. Doses in contaminated multiroom buildings are estimated using a set of unique transport and exposure models. The overall approach has been encoded in a computer code REUSEIT which runs on a personal computer.

INTRODUCTION

To develop justifiable criteria for permissable radioactivity levels following cleanup of contaminated lands and buildings, it is necessary to quantitatively relate the amounts and types of residual radioactivity to the potential hazard. An earlier review of the various approaches available for estimating the hazard to humans from residual radioactivity concluded for a variety of reasons that no existing approaches were suitable to aid the U. S. Environmental Protection Agency's (USEPA's) selection of decommissioning criteria for regulatory purposes.[1] Thus the USEPA decided that a new approach for estimating potential radiation doses from residual radioactivity should be developed. Among the goals the USEPA decided upon for the new approach were that it should be applicable to a broad range of decontaminated sites and facilities; that it should be applicable to contamination both in the environment and in buildings; that it should predict not only any potential hazard at the time of release of a site or facility for unrestricted use but also any potential increased hazard at future times (due to factors such as movement of radioactivity within the environment and radioactive ingrowth); that (where reasonable) it should generate estimates which were conservatively realistic (rather than needlessly conservative); that it should be a risk-level approach (rather than a detailed research-level approach); that it should be able to take into account significant site-specific differences; and that it

*This work was initiated while the author worked as a consultant to SC&A, Inc. under contract to PEI in turn under contract to the Office of Radiation Programs at the USEPA. It was continued while the author worked as a consultant for the USEPA.

should be implementable on a personal computer. The resulting approach for generating estimates of potential doses to onsite residents[2] is reviewed in this paper.

MODELS AND ASSUMPTIONS

The new approach utilizes a comprehensive yet relatively simple set of physically-based, state-of-the-art risk-level environmental transport and exposure pathway models. Those models include certain models long in use, some of whose data bases have been updated; scaled-down versions of some recent, more sophisticated models (the crop exposure model,[3,4] the surface soil transport model,[5-7] and the external ground exposure model[5-7]); and new models developed for this approach (the building transport and exposure models). The models have been interfaced to permit "mechanistic", time-dependent consideration of interactions within and among the various environmental media. Radioactive decay and ingrowth are explicitly taken into account.

The basic flow of information for estimating individual doses is indicated in Figure 1. The approach considers residual radioactivity initially present in surface soil, subsurface soil and/or in buildings. The envionmental transport models and the building transport models predict

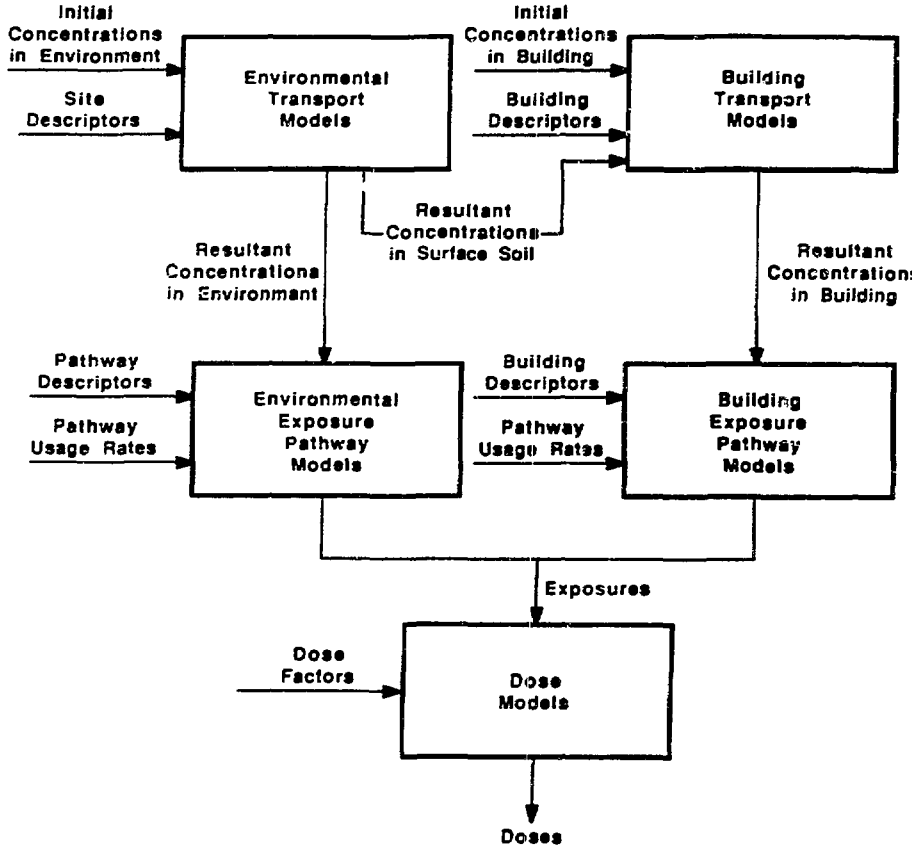

Figure 1. Schematic representation of dose estimation procedure.

the concentration of each long-lived radionuclide in each key environmental medium, e.g., in the atmosphere, and associated with each major building component, e.g., in the ventilation system, respectively. The environmental pathway models and the building pathway models predict both the radionuclide concentrations in each exposure pathway, e.g., in crops for the plant consumption pathway and on floors for the building external exposure pathway, and the resultant radiation exposures for each pathway. The dose models predict the annual individual doses associated with each of the potential exposures. The foregoing procedure is repeated for each of a series of representative times between the time of release of the site and 10,000 years later.

Environmental Transport Models

Movement of radioactivity within the environment is assumed to occur via the processes depicted schematically in Figure 2. For the atmosphere, (re)suspension of surface soil and subsequent deposition are taken into account. For surface soil, addition of radioactivity by irrigation and by percolation downward from higher surface soil (if any), as well as removal by leaching accompanied by downward movement (to lower surface soil), are modeled. In addition, (re)suspension and subsequent transport of surface soil contamination into any buildings

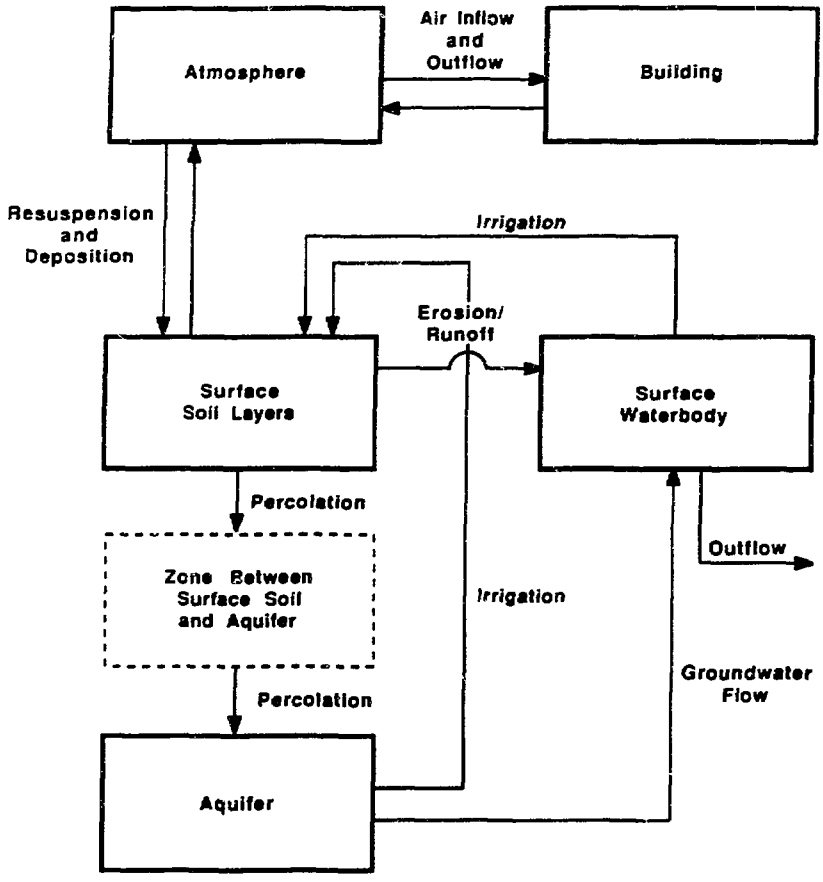

Figure 2. Schematic diagram of the environmental media and the transport mechanisms in the approach.

is accounted for. For subsoil, additions by percolation from surface soil and losses by removal of groundwater for irrigation are considered. For surface water, contamination both by erosion and/or runoff of surface soil and by subterranean flow of subsoil contamination are modeled.

More specifically, airborne radioactivity, taken to be present only as a result of resuspension, is modeled using an assumption of uniform mass loading of the air by surface soil. The fraction of the suspended particle load derived from the contaminated site is taken to be a function of the site size.

Surface soil radioactivity, assumed present only as a result of initial surface soil contamination or as a result of irrigation with contaminated well water or surface water, is modeled using a series of surface soil compartments. In particular, the top meter of soil is divided into 5 horizontal layers of increasing thickness with depth (0-1 cm, 1-5 cm, 5-15 cm, 15-30 cm and 30-100 cm).[5] Downward transport of radioactivity by rainfall and other infiltrating water is modeled using the leaching model of Baes and Sharp.[8] Addition of radioactivity by irrigation is modeled assuming the concentration in groundwater is related to the concentration in the subsoil by the appropriate distribution coefficient for the element. For irrigation water contamination derived from surface soil contamination, additional dilution effects are considered.

Only one subsurface waterbody, an aquifer, is considered. Subsoil radioactivity, taken to be present only as a result of initial subsoil contamination or as a result of subsequent contamination by downward movement of surface soil radioactivity, is considered by assuming that a well protrudes into the aquifer at the location with the highest concentrations of radioactivity. As noted, the concentrations in the well water and in the subsoil are assumed related via distribution coefficients and where appropriate by additional dilution considerations. The region between the surface soil layers and the aquifer is not explicitly modeled, although account is taken of the delay resulting from passage of surface soil contamination through that unmodeled layer to the aquifer.

Only one surface waterbody, a uniformly-mixed reservoir, is considered. That waterbody is assumed contaminated by erosion or runoff of contaminated surface soil or by underground flow of contaminated aquifer water or by both. All the radioactivity which runs off the site or which by subterranean flow travels a distance sufficient to reach the nearest shore is assumed to enter the reservoir. Radioactivity leaves the reservoir only as a result of hydrologic turnover.

Environmental Exposure Pathway Models

The estimated environmental exposures include external exposures from contaminated ground, air, and water; and internal exposures from inhalation of contaminated air, as well as from ingestion of contaminated water, crops, animal products, and aquatic foods. Most of the environmental exposure pathway models are adaptations, in two cases involving extensive modifications, of the traditional USNRC Regulatory Guide 1.109[9] models often used for analyses of potential radiation exposures. The two substantially modified models are the external ground exposure model[5-7] and the agricultural products model.[3-4] Another significant departure from the RG 1.109 models involves the inclusion of area dependencies for some pathways.[10,11]

Direct external exposure from contaminated ground is calculated using the depth-dependent exposure model of Kocher et al.[5-7] for the five surface soil layers previously described. Radionuclide-dependent (energy-dependent) attenuation by the soil is built into those models. An adjustment for the finite size of any contaminated area (in contrast to the infinite area assumed in the dose conversion factors for those models) is included.[10,11] External exposure from "immersion" in contaminated air and water is estimated using dose conversion factors for semi-infinite clouds and waterbodies, respectively.

Inhalation exposure is estimated by considering intake of contaminated particles (dust) only and not intake of radioactive gases (although all radioactive decay products of the radionuclides in the dust-both gases and nongases-are taken into consideration in determining the total inhalation dose). The relatively lower airborne concentrations associated with smaller areas of surface soil contamination are taken into account.

Drinking water exposure is estimated assuming water is taken from the more contaminated water source, the worst onsite well or the nearest surface waterbody. The effects of drinking water treatment can be taken into account, with those effects depending on the character of the water.[12]

Exposures from consumption of contaminated crops and animal products are estimated using a scaled-down version of the TERRA models and data bases of Baes et al.[3,4] Five categories of crops are considered: leafy vegetables; exposed produce (e.g., tomatoes); protected produce (e.g., potatoes and oranges); grains; and hay, silage and pasture. Concentrations in crops are estimated using an updated version of the RG 1.109 expression for concentrations in plants which includes uptake both from the plow layer and from soil below the plow layer,[10] as well as direct contributions from aerial deposition and irrigation. An area factor for crops takes into account that only a limited harvest can be taken from a small contaminated area.[10] Similarly, concentrations in animal products are estimated using a RG 1.109-level expression for two products, milk and beef. An area factor for animal products takes into account that only a limited fraction of an animal's diet can be grown on a small area.[10] (The area factors in this approach are different than those in the referenced report.)

Exposures from consumption of contaminated aquatic foods are estimated for up to three types of organisms (finfish, crustaceans and molluscs) for either salt water or fresh water. Bioaccumulation factors are used to relate the radionuclide concentrations in the edible parts of the organisms to the concentrations in the water in the reservoir.[13,14]

Building Transport and Exposure Pathway Models

The estimated indoor exposures include external exposure from contaminated interior building surfaces (walls, ceilings, and floors); ventilation system ducts, and abandoned equipment; and inhalation from contaminated dust, both from inside the buildings and from outside. The models used to estimate exposures in contaminated buildings were developed as part of the formulation of the overall approach described in this paper. Their levels of sophistication and complexity of use are comparable to the corresponding levels of the USNRC RG 1.109 models traditionally employed for estimating exposures in the environment.

The building models assume a maximum of 27 rooms, with all the rooms in the building being the same size and with the contaminated rooms being contiguous. Resuspension of radioactivity within ventilation system ducts and subsequent movement of some of that radioactivity into rooms is considered, as, was already noted, movement of radioactivity into buildings via (re)suspension of contaminated surface soil. The basic mechanisms are indicated in Figure 3.

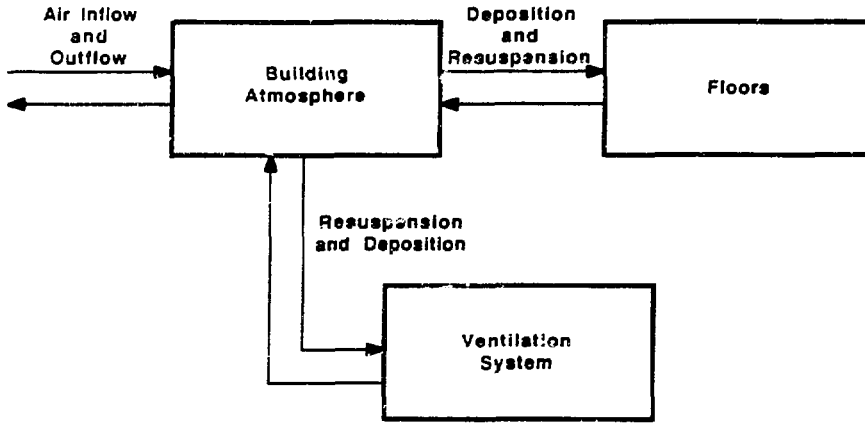

Figure 3. Schematic diagram of the transport mechanisms in the building models in this approach.

External exposures are estimated separately for the following:

1. Rooms with all surfaces initially contaminated;
2. Rooms with just floors contaminated (e.g., contaminated by deposited dust);
3. Ventilation system contamination (still in the ducts);
4. Localized "hot spots" of radioactivity; and
5. Contaminated equipment.

Attenuation by building walls is estimated on a radionuclide-dependent (energy-dependent) basis.

As for the environment, indoor inhalation exposure is estimated assuming intake of dust only (including all the decay products of the radionuclides in the dust). As noted, the contaminated dust can be from outside the building and/or from the ventilation system.

Dose Models

The individual dose to any organ o for any exposure pathway p from any radionuclide i and all its radioactive progeny is written in the general form

$$dose^{o}_{i,p} = U_p \sum_{m=1}^{m'(p)} \sum_{j=0}^{n} W_{i+j,m,p}\, C_{i+j,m}\, D^{o}_{i+j,p} \quad , \qquad (1)$$

where

U_p = annual rate of usage of pathway p, e.g., annual consumption rate;

$W_{i+j,m,p}$ = weighting factor for radionuclide (i+j) for environmental medium or building component m and pathway p;

$C_{i+j,m}$ = activity (concentration) of radionuclide (i+j) in environmental medium or building component m; and

$D^o_{i+j,m}$ = dose factor for radionuclide (i+j) in pathway p for organ o.

The summation over m is over all m'(p) environmental media and building components which contribute directly to pathway p. The summation over j is over the entire radioactive decay chain of radionuclide i, that is, over i and its n progeny.

Pathway usage rates are not built into the approach but are specified by the user. The weighting factors for the initial contaminants are calculated by the exposure pathway models and the environmental media and building component concentrations for those contaminants are calculated by the transport models as described in the previous three subsections of this paper. The weighting factors and corresponding concentrations of the radioactive decay products of those initial contaminants are estimated as described in the next subsection. Organ-dependent internal dose factors are read from resident files.[15-17] Organ-dependent external dose factors are calculated from dose factors to air[5-7] using radionuclide-dependent factors for estimating doses to organs from a semi-infinite cloud.[15-17]

Radionuclide Decay and Ingrowth Models

Radionuclide decay and ingrowth occuring after intake (ingestion or inhalation) are accounted for, as is typical, in the internal dose factors.[15-17] Radioactive processes occuring in the environment are accounted for as described below.

All the radioactive decay products of any parent radionuclide are assumed to move through the environment at the same rate and to behave in any exposure pathway in the same manner as the parent. Thus the concentrations of the various members of any decay chain in any environmental medium or building component are related only by decay considerations and the pathway weighting factors in Equation 1 are the same for all members of a radioactive decay chain. Consequently, in this approach, the dose for any chain with n progeny can generally be written as

$$dose^o_{i,p} = U_p \sum_{m=1}^{m'(p)} W_{i,m,p}\, C_{i,m} \sum_{j=0}^{n} (C_{i+j,m}/C_{i,m})\, D^o_{i,p} \quad . \qquad (2)$$

As a result, to include the effects of environmental ingrowth in this approach, the dose factors for any parent radionuclide are replaced in Equation 1 for the parent only by modified or effective dose factors which represent composite, activity-weighted dose factors for the entire

decay chain. That is, the following modified dose factors

$$"D^{o}_{i,p}" = \sum_{j=0}^{n} (C_{i+j,m}/C_{i,m})\ D^{o}_{i+j,m,p} \qquad (3)$$

are used in

$$dose^{o}_{i,p} = U_p \sum_{m=1}^{m'(p)} W_{i,m,p}\ C_{i,m}\ "D^{o}_{i,p}" \qquad (4)$$

Unlike regular dose factors, the modified dose factors are functions of the activities of all the decay chain members and therefore are functions of the time.

SOME OUTSTANDING FEATURES OF THE APPROACH

The new approach has some notable features which make it especially useful for generating estimates of the potential radiation doses from residual radioactivity in a variety of situations. Several of these features are considered in this section.

Environmental Transport and Exposure Pathway Models

The direct, physically-based interfacing of the transport models with each other and with the exposure pathway models provides consistent predictions of concentrations of radioactivity both throughout the major environmental media and in the critical portions of the pathways. Such consistency in not found in many other risk-level approaches for residual radioactivity.[1] In some of those approaches, the user must perform extensive calculations "on the side" to ensure consistency. In still others, non-physical models and nonexistent or non-physical interfaces preclude such consistency altogether.

For example, the USNRC Regulatory Guide 1.109 crop model and many derivative models assume (nonmechanistic) buildup of irrigation contamination in surface soil for 50 years for all radionuclides. Those models ignore that some radionuclides could reach substantially higher surface soil concentrations by buildup for periods much longer than 50 years while some other radionuclides could attain at most only much lower maximum concentrations and would attain them in much shorter periods. (The anticipated variation of buildup of surface soil contamination by irrigation is illustrated in Figure 4 for a very long-lived (stable) radionuclide and a much shorter-lived radionuclide (with a half-life of 30 years) for two values of the distribution coefficient, a parameter which describes the interactions of a radionuclide with the soil.) As a result of the incorrect predicted surface soil concentrations in most RG 1.109-derived models, the crop dose estimates for such models can be much too low for some radionuclides and much too high for other radionuclides. And the temporal behavior of the doses is inadequately described by such models for almost all radionuclides. That the buildup period and the maximum concentrations would in reality be very element dependent is ignored in those models. In contrast, that behavior is automatically accounted for in the new approach.

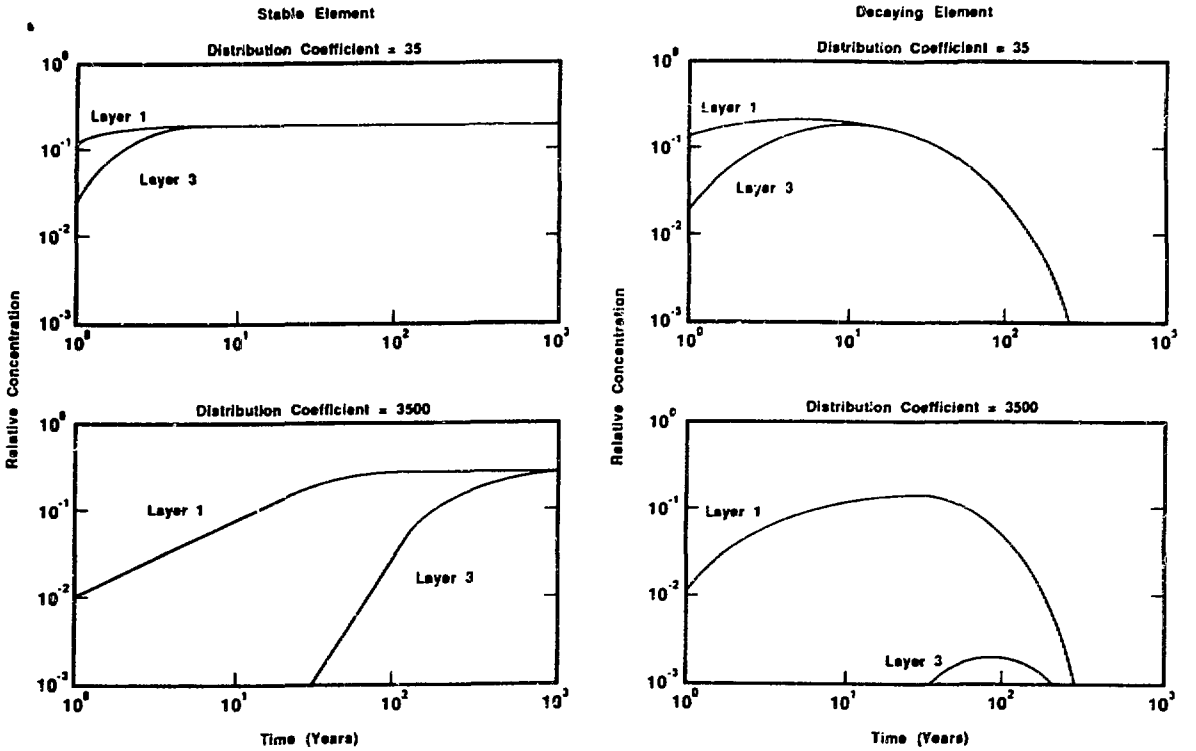

Figure 4. Buildup of radioactivity in surface soil by irrigation with contaminated water.

Another example of the improved predictions resulting from the "mechanistic" treatment of radioactive transport can be seen in the ground dose model. The surface soil transport model with its five surface soil layers and its leaching assumption provide a reasonable approximation of the physical distribution and movement of radioactivity within surface soil. The consideration of energy-dependent attenuation by the soil further improves the estimation of the external dose from contaminated soil. Combining all these factors, the doses estimated for some radionuclides, especially for low-energy γ-emitters, are substantially lower than those estimated with the typical, very conservative models that assume all surface soil contamination resides on the ground surface (and therefore experiences no attenuation by the ground). The doses estimated by the new approach are also "more realstic" than those estimated with approaches which consider an initial distribution of radioactivity within surface soil but do not adequately follow its movement with time. The doses estimated by the new approach will in some cases be larger and in other cases be smaller than those estimated by approaches using non-physical and unphysical assumptions.

Building Models

The building models in this approach go far beyond any other models for estimating doses inside contaminated structures. All other approaches include at most a 1-room "building" model. Unfortunately, one room, no matter how large, is not conservative for some radionuclides,

in particular, for those with significant high-energy γ-emissions which can pass through walls with little attenuation. For such radionuclides, radioactivity in rooms an individual never enters can contribute significantly to his (her) external dose.

Some estimated relative individual doses for two radionuclides, a high-energy γ-emitter and a low-energy γ-emitter, are depicted in Figure 5 for 3 buildings with 1 room, 5 rooms and 27 rooms contaminated, respectively, for a range of floor sizes. As expected, for both radionuclides, the dose from a contaminated floor approaches the dose from an infinite plane (approximately the dose from an area of 10,000 m^2) as the floor size increases. As can been seen, for both radionuclides the dose from a contaminated room whose height is fixed approaches the dose from 2 infinite planes as the room size increases (and the side walls recede). As can also be seen, for the high-energy γ-emitter, whose radiation is not substantially attenuated by the walls, the individual external dose in a building with 27 contaminated rooms is approximately 5 to 6 times the individual dose in a building with only 1 room contaminated. (The dose can be even higher if more than 27 rooms are

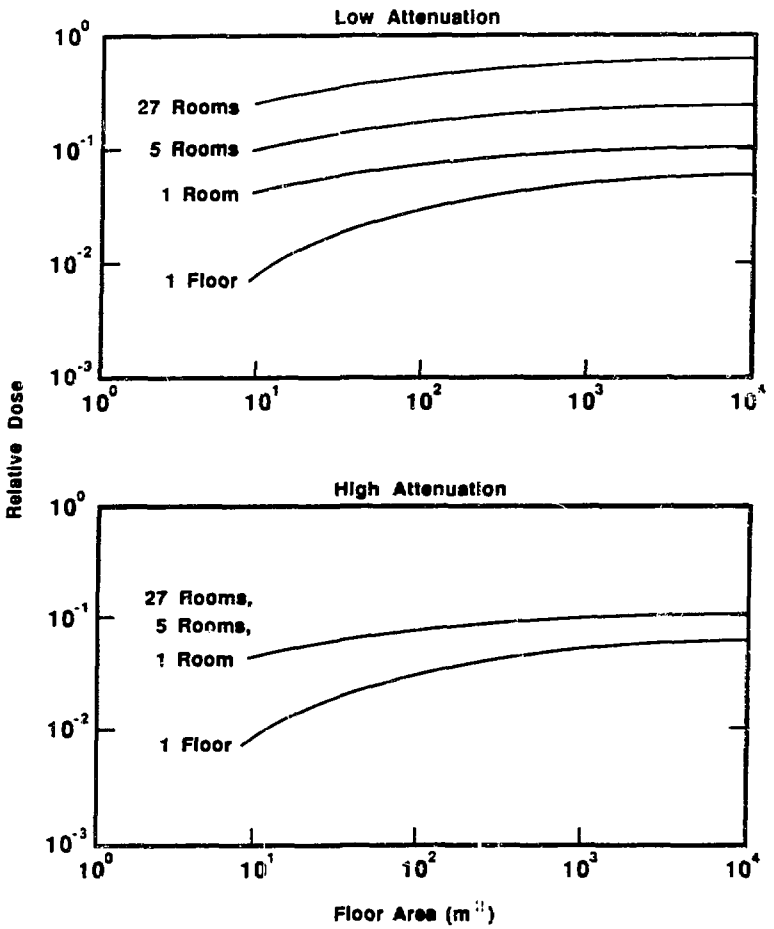

Figure 5. External individual doses from two radionuclides, a high-energy γ-emitter (for which attenuation by walls is relatively low) and a low-energy γ-emitter (for which attenuation is substantial).

contaminated.) For the low-energy γ-emitter, whose radiation is strongly attenuated by even relatively thin walls between rooms, the entire external dose comes from the room the individual occupies and so the dose is independent of the number of contaminated rooms.

Radioactive Ingrowth

Radioactive decay typically tends to lower the potential radiation dose with time. However, decay which results in significant ingrowth of more hazardous radioactive progeny can substantially increase the potential hazard with time. The effects of environmental ingrowth, especially for lengthy actinide chains, are often neglected in risk-level approaches for residual radioactivity (and for low-level waste).

The dependence of the potential doses on radioactive decay and ingrowth is depicted schematically in Figure 6 for two radionuclides for which long-term ingrowth is not important (Ru-106) and for one radionuclide for which such ingrowth can be significant (Th-230). As can be seen in the Th-230 example, failure to consider ingrowth and to consider

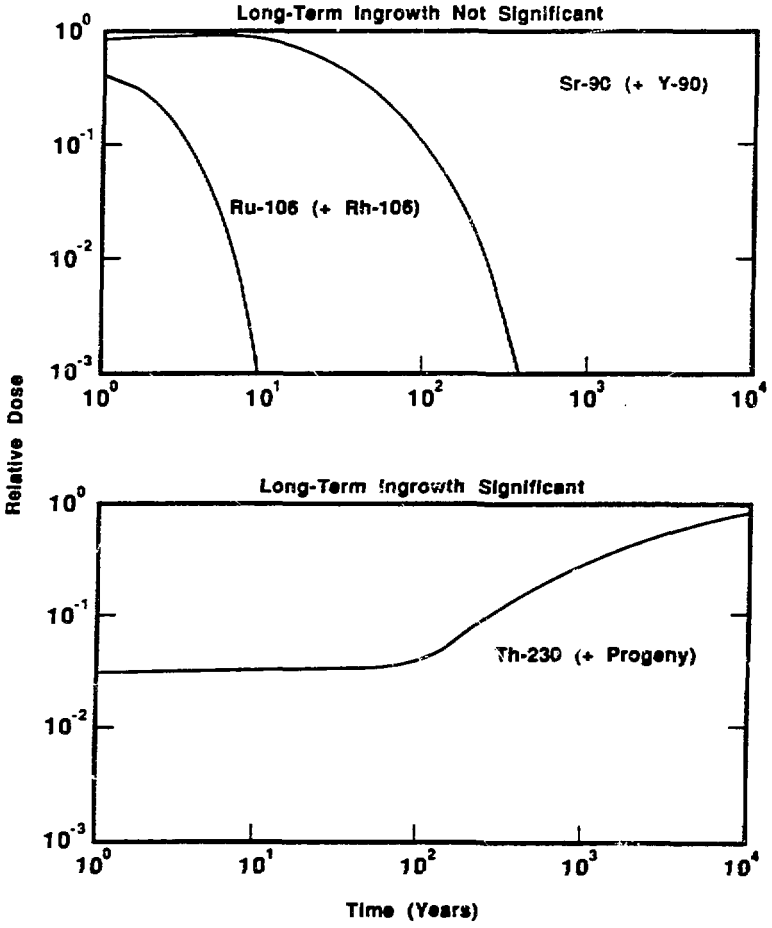

Figure 6. Relative potential individual doses as a function of time for 2 radionuclides for which long-term ingrowth is not significant and for one radionuclide for which such ingrowth can be important.

times long after the date of decommissioning can potentially result in underestimation of certain doses by over an order of magnitude.

Conservative Realism

Unlike many risk assessment approaches which have been designed to be very conservative and which may thus significantly overestimate the doses for some exposure pathways, the new approach has been designed to be "conservatively realistic", that it, to be somewhat more "realistic" (in a modeling sense) and to default on the conservative side when what is realistic is not known or is not readily amenable to risk-level modeling. As a result, in the new approach the doses estimated for some pathways and some radionuclides are substantially lower than those estimated by most other risk approaches. For example, as noted, the much improved accounting of the distribution of surface soil radioactivity and the consideration of energy-dependent attenuation by soil (instead of energy-independent attenuation or no attenuation as in many models) result in external dose estimates for some radionuclides which are much smaller (an order of magnitude or more) than the comparable dose estimates for other approaches.

However, the increased "realism" in some models has resulted in higher dose estimates for certain situations. For example, as already noted, the inclusion of multiple rooms in the building models results in dose estimates for some radionuclides which are much higher (even an order of magnitude higher) than estimates of the 1-room models in other approaches. As another example, also already noted, the consideration of long-term ingrowth can result in substantially higher dose estimates at later times than approaches neglecting such ingrowth.

Overall, the use of more "realistic" models has resulted in a state-of-the-art risk-level approach which provides more consistent dose estimates than other approaches. However, it should be noted that the estimates provided by the approach are critically dependent on the data used. As the data used are not necessarily conservative (especially the data in the element-dependent and radionuclide-dependent data bases), the dose estimates generated by the approach are not necessarily conservative. (The same can be said for the more conservative approaches which relie on some of the same nonconservative data bases.)

Solution Techniques and Implementation

Although the approach is comprehensive and incorporates some very desireable features not typically included in residual radioactivity approaches, e.g., physically-based modeling of movement of radioactivity within the environment and inclusion of radioactive ingrowth, it is still a very practical approach. The adoption of extremely efficient solution techniques, e.g., the use of analytical solutions throughout and the restriction of detailed dose calculations to a set of judiciously-chosen times, has resulted in an approach whose models can be economically implemented. Thus consideration of a large variety of situations is practical with the new approach.

The approach has been implemented in a computer code which runs on a personal computer.[18] Thus the approach is potentially accessible to a wide variety of interested users.

SUMMARY

The new approach constitutes a significant first step toward a comprehensive, versatile set of relationships appropriate for providing radiation dose estimates for residual radioactivity at a variety of sites and facilities. Thus it can potentially be used to help choose decommissioning criteria for residual radioactivity. As such it serves to fill a significant gap in the spectrum of previously available approaches.

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RADIATION CONTROL TECHNOLOGY APPLICABLE TO THE JPDR DECOMMISSIONING

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ABSTRACT

From the view of radiation control, the main features of the reactor dismantling are listed as follows;

1. The works under high level radiation and high level radioactive air contamination area will be expected.
2. The new dismantling techniques which have not been experienced in the controlled area before will be adopted.
3. A great amount of materials, tools, radioactive waste and so on will be taken out from the controlled area.

Considering these features, we have developed the following five instruments to adapt for the JPDR decommissioning;

- Remote high dose rate measuring instruments (underwater and in the air)
- Contamination inspection monitor
- Respirable dust monitor
- Extremely low level waste γ-scanner
- Waste package contamination and dose rate monitor

INTRODUCTION

In the reactor dismantlement, the methods of radiation control during the reactor operation may be used. However, further effort should be made to reduce exposure to the dismantling workers and to increase the efficiency of dismantling operations. This paper discusses from the above point of view, the features of the monitoring instruments developed for radiation control of the JPDR decommissioning, considering various circumstances encountered during the reactor dismantling.

DEVELOPED INSTRUMENTS AND MONITORS

Remote High Dose Rate Measuring Instruments
(for underwater and in the air)

These instruments were developed for the purpose of remotely measuring dose rate of reactor internals and pressure vessel underwater and their segments in the air in case of lifting them from the water. They contribute to planning the appropriate demolishing programs and the relevant radiation works.

Photo 1 shows the remote underwater instrument. The detector is a water proof-type ionization chamber (the ionization chamber is contained in a stainless steel container whose submersible limit is 20m). The effective

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This work was performed by the Japan Atomic Energy Research Institute under contract from the Science and Technology Agency of Japan.

volume of the ionization chamber is 100cm^3 and the measuring range is 10 to 10^5R/h. The measured value is displayed on a LCD (Liquid Crystal Display) and recorded on an X-T recorder. The detector can be moved vertically and horizontally. The detector is suspended by submersible coaxial signal cable and the vertical movement is controlled by drawing out and in the cable. The vertical position of the detector is displayed on a detector control panel. Horizontal movement is controlled by expansion (maximum 4m) and swing (45° right and left each) of an extendible arm. The operation is done by the control panel or a wired remote controller. Fig.1 shows surface dose rate of JPDR core shroud measured underwater by this instrument.

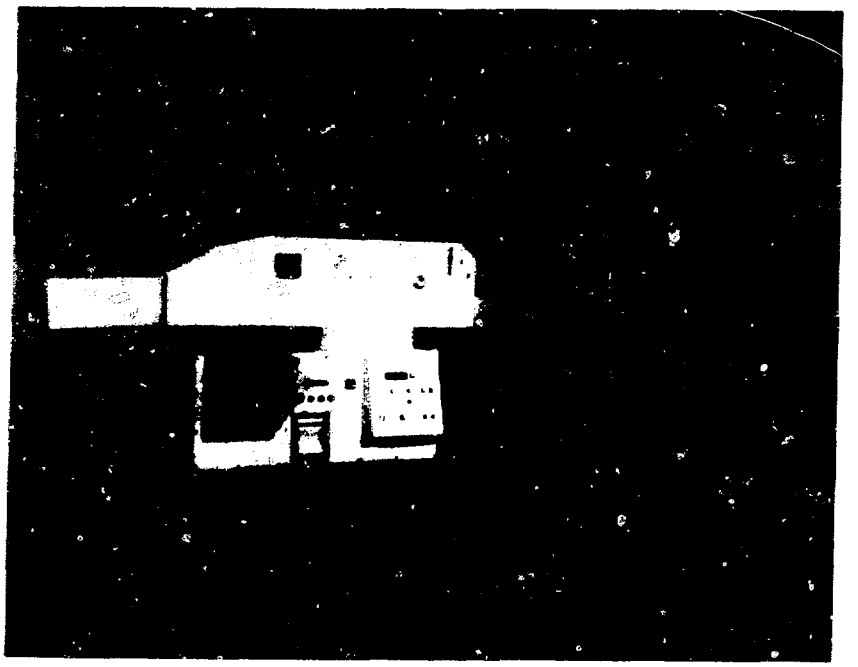

Photo 1 Remote high dose rate measuring instrument underwater

The remote instrument in the air consists of a measuring subsystem and a control subsystem. Photo 2 shows the measuring subsystem of the instrument. The control subsystem can be located apart 20m from the measuring subsystem. The detector is the ionization chamber (effective volume is 28cm^3) and its measuring range is 0.1 to 10^3R/h. The measured values are displayed on a panel meter in the measuring subsystem and also recorded on an X-T recorder. The detector is mounted on an extendible arm. The arm is extendible to the maximum length 3.5m and retractable to the minimum 0.6m. When the arm was retracted, it can be wounded onto a spool. It can swing within elevation angle 60°, depletion angle 15°, and right and left angles 45° each. The extension and swing of the arm are done by watching a TV monitor of the control subsystem. In order to avoid touching the measuring object during operation, non-contact light reflection type of distance sensors are equipped (to extension direction and right and left directions).

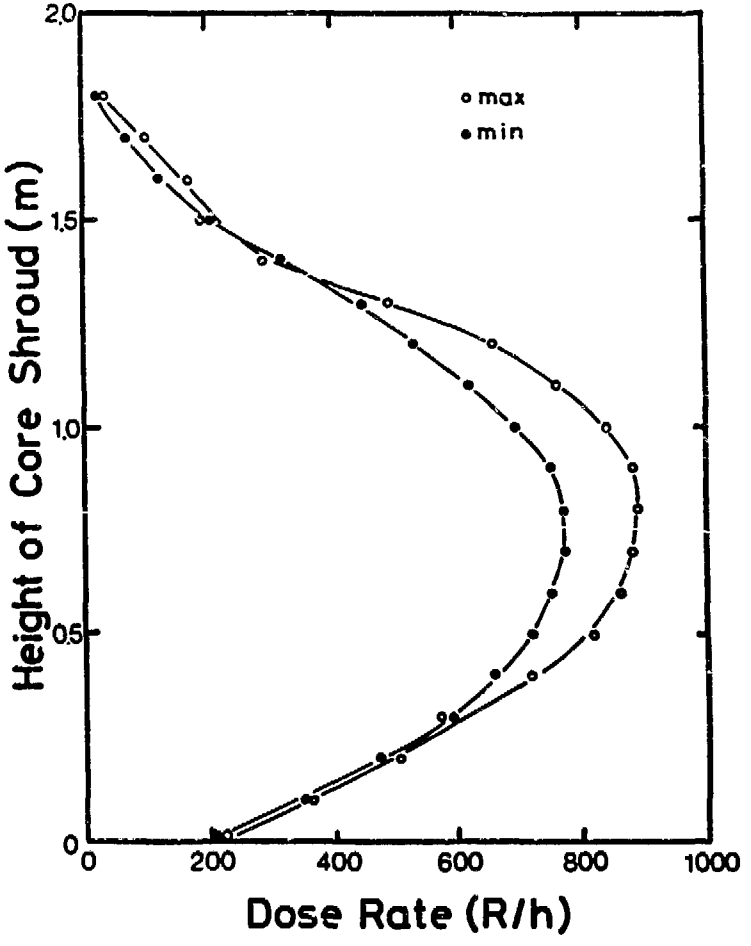

Fig.1 Underwater dose rate of JPDR core shroud

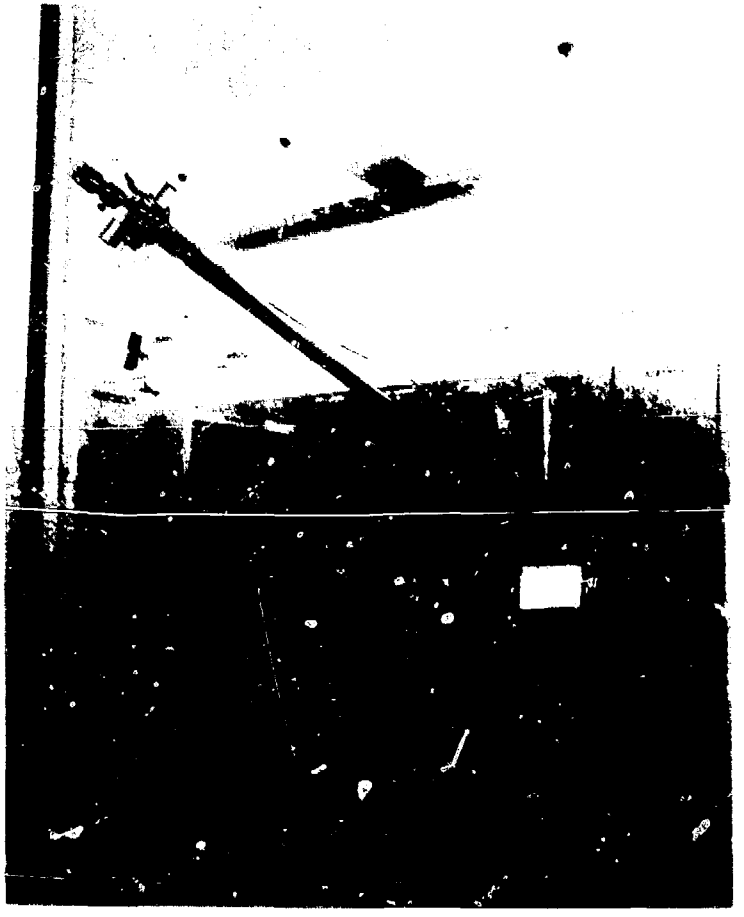

Photo 2 Remote high dose rate measuring instrument in the air

Control of the workers is extremely important in high radiation working areas. Exposure of measuring personnel will be largely reduced by introduction of these instruments compared with that of commercial telescopic type surveymeters.

Contamination Inspection Monitor

This monitor was developed for the purpose of inspecting the surface contamination of taken out articles from the controlled areas, for example, hand tools, drawings and electric measuring instruments, and reducing the labour of contamination inspection by radiation control personnel.

The monitor is composed of a main body, a control subsystem and an article conveying subsystem. Photo 3 shows the monitor. The main body contains two types of detectors and a detector driving gear. Gas-flow counters are used for the outer surface contamination detection (for β-rays), while a plastic scintillation counter is used for the inner surface contamination detection (for γ-rays). The measured article can be surrounded by the gas-flow counters (6 sides). The upper side detector descends automatically according to the height of measured article. For the upper and lower sides, each gas-flow counter which has effective incidence area 640×400mm is used. The plastic scintillation counter mounts the scintillator with a volume 440×200×100mm and two photoelectron multiplier tubes (2-inch ϕ). Signals of each detector are inputted to each scaler and their counts are transmitted to a 16-bit micro-computer via parallel interface. Their BG counts are subtracted from gross counts and converted to the surface concentration of radioactivity. If the surface concentration

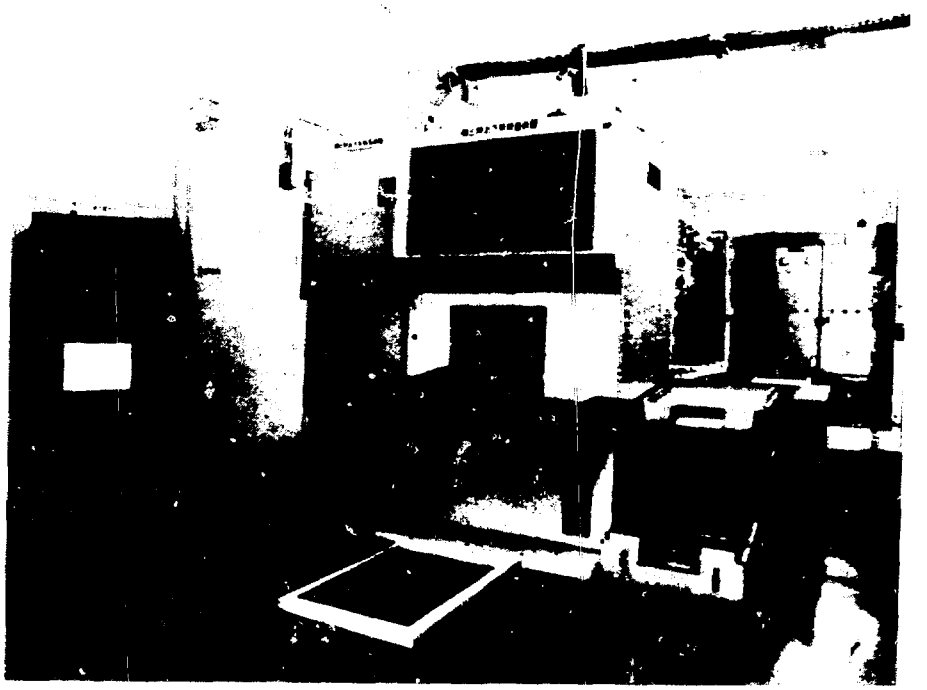

Photo 3 Contamination inspection monitor

of radioactivity is less than the administrative level, then the article will be conveyed out automatically to the non-controlled area by the article conveying subsystem. If the surface concentration of radioactivity is beyond the administrative level, then the articles will be returned to the controlled area side and the alarm will be sounded at the radiation control office. The maximum size of the measured articles is 444×470×300mm and the maximum weight is 20kg. The lower limit of detection is less than 0.37Bq/cm^2 (1×10^{-5}μCi/cm^2) at the measuring time 30 sec. One operation cycle of the monitor involving the articles set and the conveying time is less than 1 minute.

Respirable Dust Monitor

This monitor was developed for the purpose of monitoring continuously the radioactive respirable dust concentration under high level radioactive air contamination generated by cutting of reactor internals, pressure vessel and biological shielding concrete, separating respirable dust aerodynamically from the gross to prevent the clogging of a sampling filter paper. The schematic diagram of the monitor is shown in Fig. 2. The monitor is composed of a dust sampling subsystem and a measurement and a control subsystem.

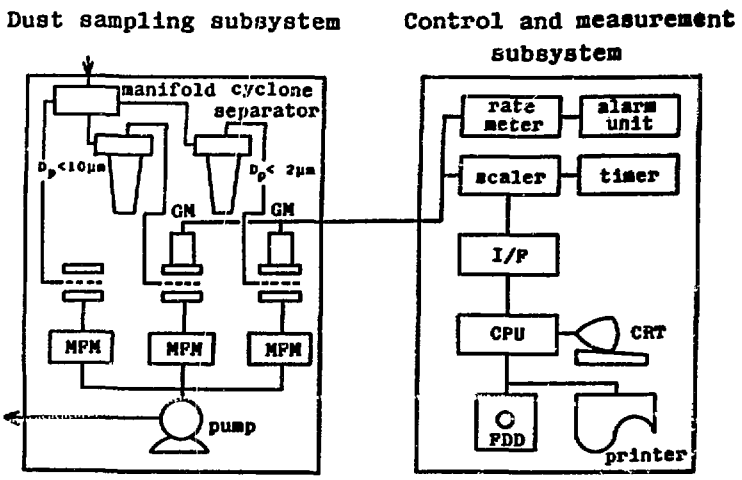

Fig.2 Schematic diagram of the respirable dust monitor

The dust sampling subsystem can collect respirable dust on filter papers by separating dust aerodynamically and measure their radioactivities. The separation is made by two cyclone separators under 2μm and 10μm from sampling dust. Radioactivities of the separated dust are measured by GM-counters continuously. The sampling flow rates are measured by mass flow meters (MFMs). Card type filter papers (a 60mm ϕ filter paper is mounted on 80×100×t1mm card) are used, and if any sampling flow rate of three lines lowers than the set value

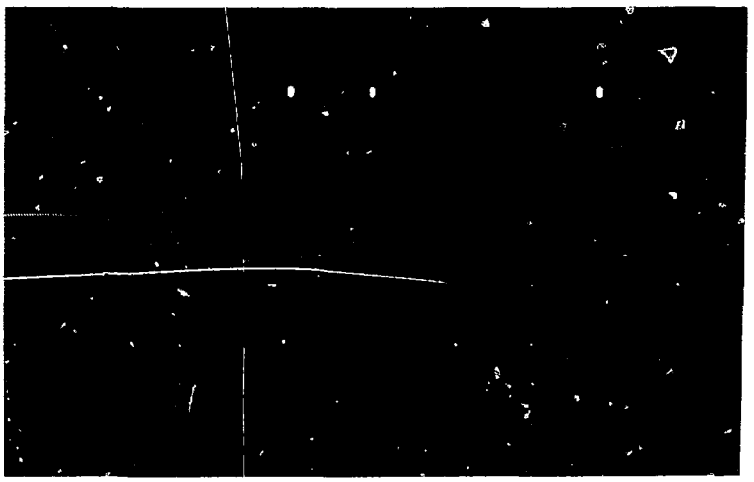

Photo 4 CRT during measuring

due to the filter paper clogging, then the card type filters are exchanged automatically. Signals of GM-counters and MFMs transmitted from dust sampling subsystem are inputted to an 8-bit micro-computer of the control and measuring subsystem. The BG counts are subtracted from the gross counts of GM-counters and the net counting rates are displayed on a CRT and plotted on a printer. If any counting rate exceeds the set value, an alarm will be sounded. The average radioactive respirable dust concentration is displayed on the CRT. Photo 4 shows the data displayed on the CRT during measurement. All these values measured are stored in floppy disks and will be able to display later as necessary.

Extremely low level waste γ-scanner

In the reactor dismantling, it is important that a large amount of radioactive solid waste arisings should be reasonably treated and disposed according to the radioactive level of them.

This γ-scanner was developed for the purpose of classifying and confirming the radioactive solid waste less than the extremely low level or the exemption level from a large amount of the biological shielding concrete and the other waste. The γ-scanner can automatically quantify the radioactivities for each nuclide in it. The schematic diagram of the γ-scanner is shown in Fig.3 and the appearance is shown in Photo 5. The γ-scanner consists of a driving mechanism, a detector subsystem and a data analyzing subsystem. The driving mechanism is capable of elevating, rotating and weighing the waste package to measure accurately the radioactivity in it. The detector subsystem consists of two high-purity

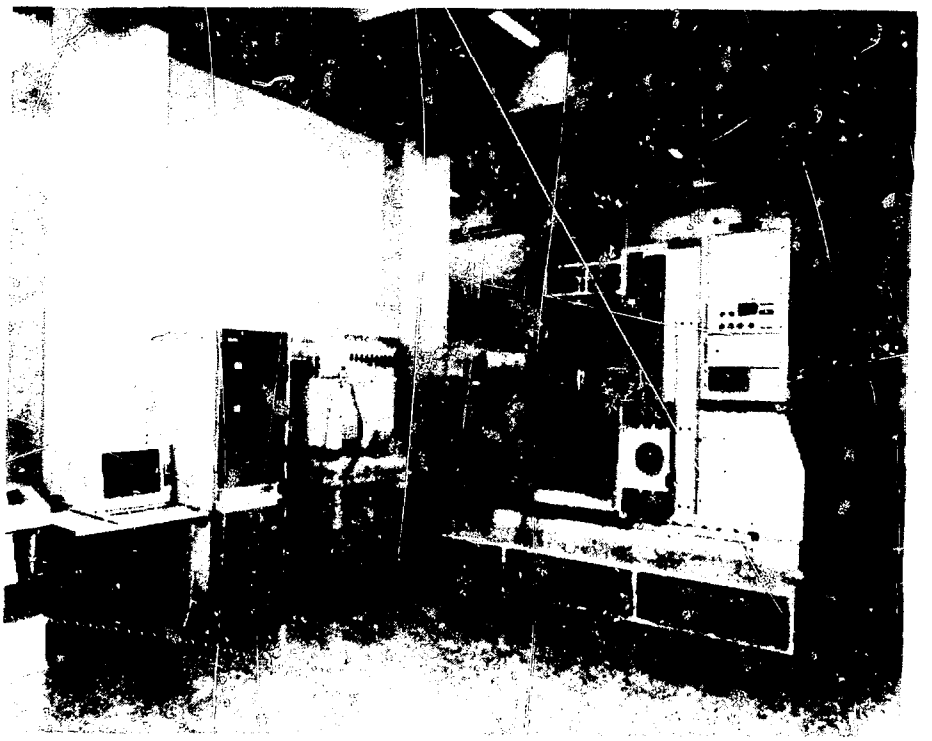

Photo 5 Extremely low level waste γ-scanner

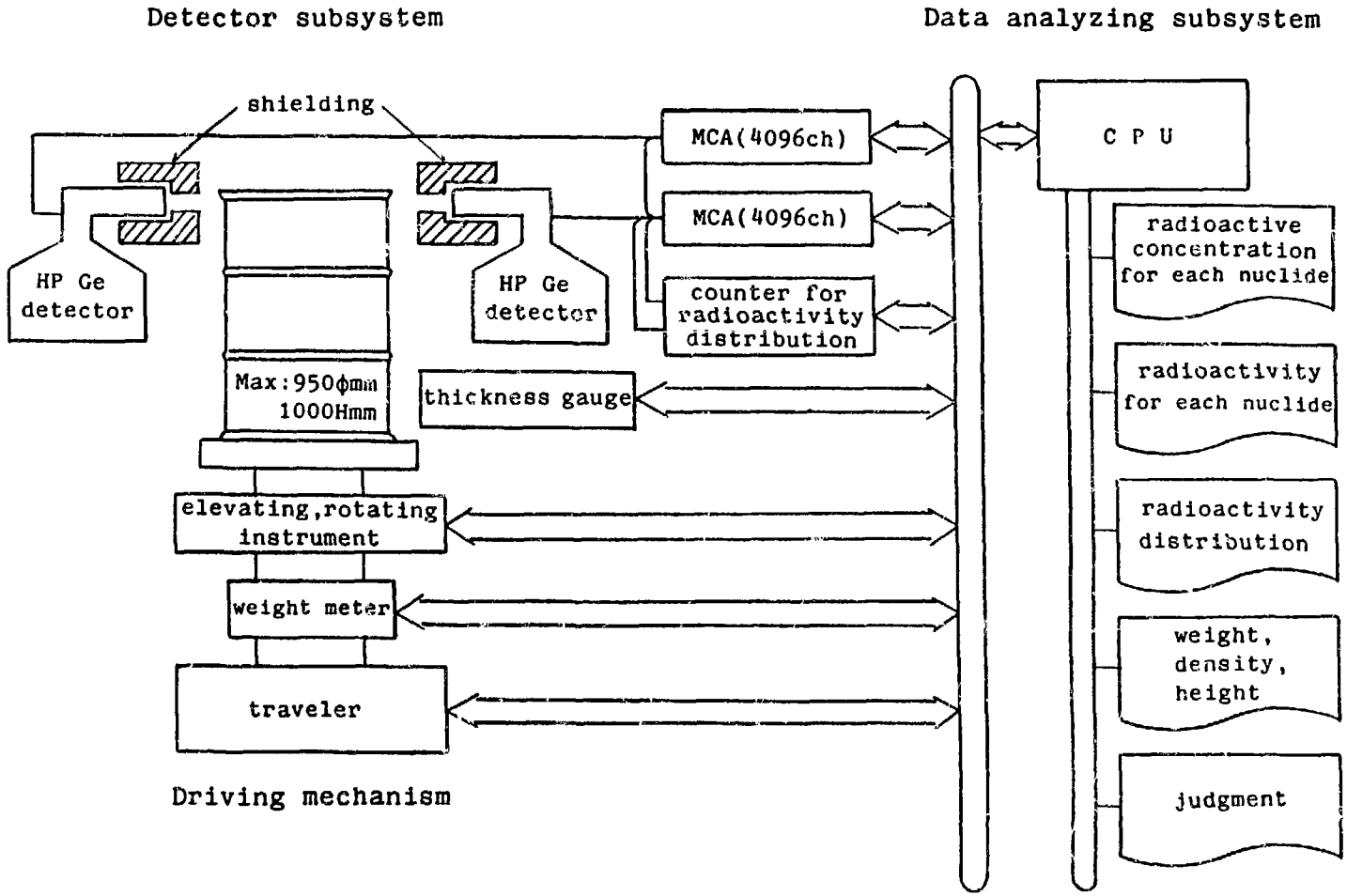

Fig.3 The schematic diagram of the extremely low level waste γ-scanner

germanium (HP Ge) detectors faced each other, two multi-channel analyzers (MCAs) and a thickness gauge. The numbers of channel of the MCA are 4096. The data analyzing subsystem with a 16-bit mini-computer can control the driving mechanism and analyze the γ-ray spectrum, simultaneously. The γ-scanner is capable of dealing with the waste package whose maximum size and maximum weight are 950mmφ×1000mmH and 500kg. respectively. However, the drum of 200ℓ is usually used as the standard waste container. The γ-scanner can measure about 30 waste packages a day. The lower limit of detection of γ-scanner is about 1.1×10^{1}Bq/kg (3×10^{-7}μCi/g) for ^{60}Co and ^{137}Cs in the measuring time of 10 minutes regardless of their density.

This γ-scanner can confirm efficiently with a certain accuracy that the level of the radioactive waste sorted in the dismantling place is less than the extremely low level or the exemption level.

Waste Package Contamination and Dose Rate Monitor

This monitor was developed for the purpose of inspecting, in compliance with the transport standard, the surface concentration of radioactivity and the dose rate of the definite waste package which contains low level radioactive solid waste.

The monitor consists of a waste package conveying mechanism, a driving mechanism, a sticker labeling mechanism, a monitor panel and a 16-bit micro-computer. The schematic diagram of the monitor is shown in Fig.4 and the appearance is shown in Photo 6. A hydraulically controlled servo-arm robot is used for the waste package conveying mechanism. The griper of the robot can move the maximum distance 2100mm vertically and 1500mm horizontally, and

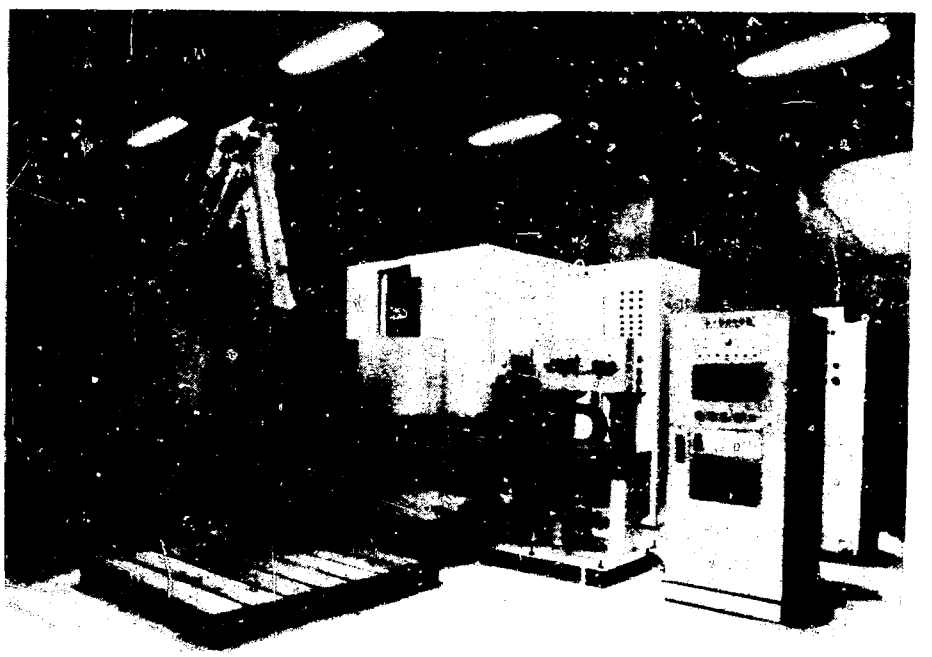

Photo 6 Waste package contamination and dose rate monitor

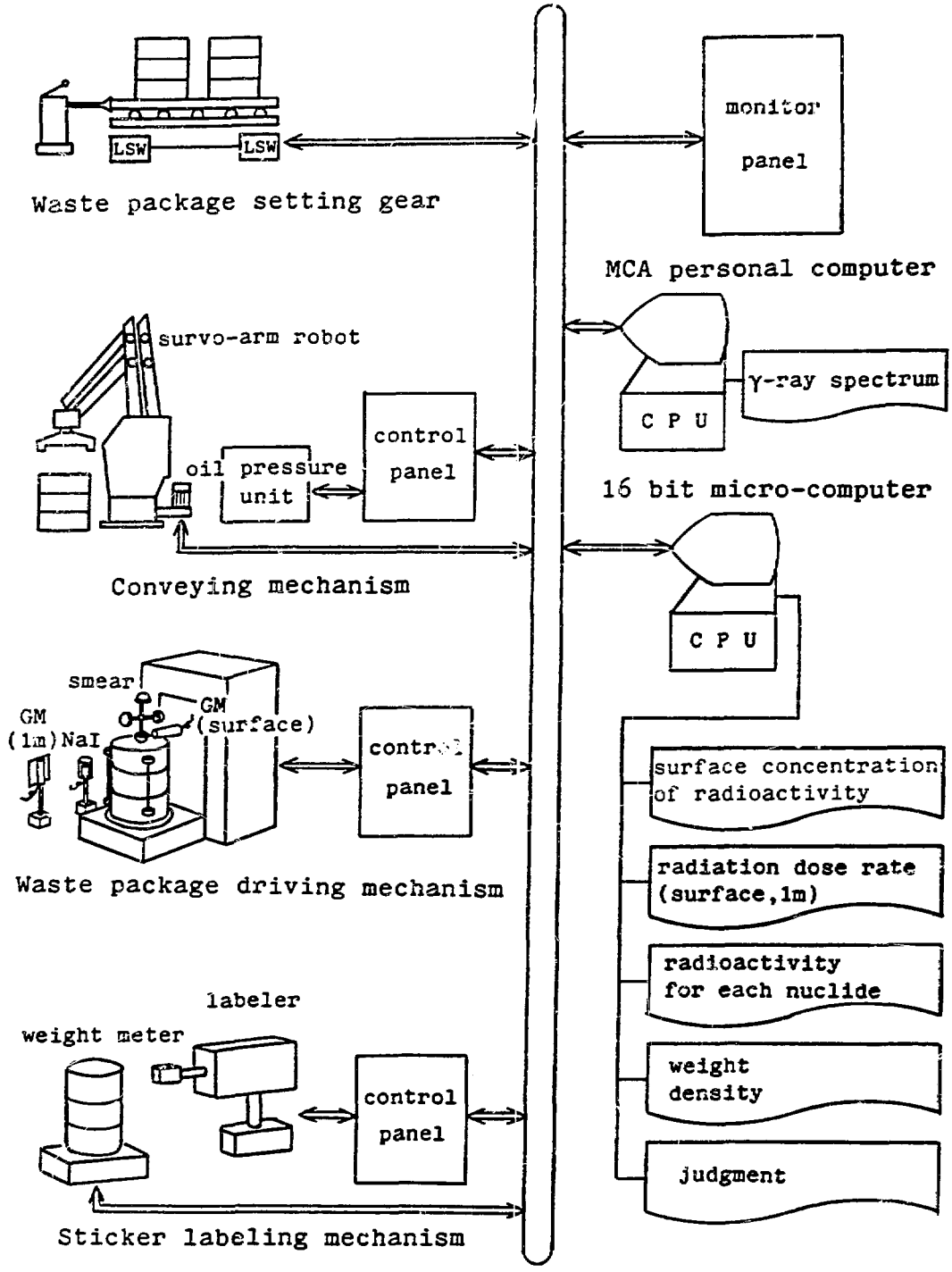

Fig.4 The schematic diagram of the waste package contamination and dose rate monitor

the arm can swing 270° horizontally. The robot is capable of conveying the drum to the set measuring position according to the micro-computer program. The smear sampling submechanism and the eight GM detectors for measuring the surface concentration of radioactivity and the surface radiation dose rate are assembled into the waste package driving mechanism. This mechanism takes and measures smear samples and also measures the surface radiation dose rate, by rotating the drum. The radiation dose rate (GM detector) at 1 meter distance from the drum and the contained radioactivity (NaI(Tℓ) scintillation detector) in it can be also measured simultaneously. The measured data are transmitted to the monitor panel and are processed to obtain the results. The sticker labeling mechanism has the capability of automatically labeling the sticker on which the measured data are printed, to the drum. The 16-bit micro-computer controls all the actions of the mechanisms, and processes the data and files the results. The measured waste package is 200ℓ drum (600mmϕ×890mmH) in the maximum weight 500kg. The lower detectable limit of the surface concentration of radioactivity is 3.7×10^{-2}Bq/cm^2 (1×10^{-6}μCi/cm^2). The measuring range of radiation dose rate is 0.1mR/h to 1R/h. This monitor can deal with a drum in about 15 minutes and 30 packages a day.

The monitor can obtain the measured results with a definite accuracy and reduce the labour and the exposure of the measuring personnel.

TECHNOLOGY OF AIR CONTAMINATION MEASUREMENTS

The size distribution, concentration, the dispersion rate and the resuspension factor of the dust particles generated by dismantling operations within the contamination control envelopes in the working environment are needed for evaluation of the internal exposure to workers and environmental impact. For obtaining these data, a laser spectrometer and two cascade impactors are used for determining the concentration of smaller size and larger size particles, respectively.

The efficiency of these radiation control technology will be confirmed in actual decommissioning of JPDR. It is expected that the data and experiences which will be obtained through the use of the technology in the JPDR dismantling will prove to be useful for decommissioning of commercial power reactors in the future.

HEALTH AND SAFETY ASPECTS OF DECOMMISSIONING RELEVANT TO LICENSING

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ABSTRACT

After a brief introduction on the use of nuclear power and decommissioned nuclear installations in the Federal Republic of Germany, the safety goals are introduced the decommissioning plan has to comply with.

The safety goals have consequences concerning certain health and safety issues. Some of them which are regarded to be of special importance in the licensing procedure are addressed specifically: Recycling and reuse of steel scrap, potential accidents during decommissioning, occupational exposure and decommissioning options.

1. INTRODUCTION

The share of nuclear power of the overall electric power production in the Federal Republic of Germany has increased to 36 %. Reactor types, status and power capacity are summarized in Tab. I.

Tab. I: Number, types and status of nuclear power plants in the Federal Republic of Germany (as of April, 1987[1])

Status	Number of Units /Power capacity (MWe)							
	PWR		BWR		Others		Total	
In Operation	11	12411	7	7207	3	343	21	19961
Under Construction	3	3985	–	–	1	327	4	4312
Planned	8	10606	–	–	1	1500	9	12106
Total	22	27002	7	7207	5	2170	34	36379
Decommissioned	1	58	4	557	1	100	6	715

The nuclear power plants and research reactors which have been taken out of service are listed in Tab. II. Most of the decommissioning projects have been presented in publications or during international conferences [2,3,4,5,6,7,8,9,10,11]. Some of them will be addressed specifically during this conference.

The legal aspects of decommissioning in the Federal Republic of Germany have been discussed in some detail in a previous paper[4]. The objective of the present paper is to discuss some health and safety aspects which are important issues within the licensing procedure.

Tab. II: Permanently shut down nuclear power plants and research reactors in the Federal Republic of Germany (April, 1987)

RESEARCH REACTORS

Name	Location (Operator)	Type	Power rating (MW_{th})	Operational phase	Decommissioning
FRN	Neuherberg (GSF)	pool type, TRIGA Mark II	1 (stationary)	1972 – 1982	stage 2, reactor hall decontaminated for unrestricted release, safe storage of activated material in the thermal column and the dry irradiation room
FRJ–1 (MERLIN)	Jülich (KFA)	medium energy research light water moderated industrial nuclear reactor	10	1963 – 1985	planned (1986): stage2, reuse of reactor hall, partial dismantling, restricted controlled area for safe storage
FR–2	Karlsruhe (KfK)	tank–type heavy–water reactor	44	1961 – 1981	stage2 will be reached after in situ decontamination and removal of cooling system, stage 3 is deferred for about 30 years

POWER PLANTS

Name	Location (Operator)	Type	Power rating (MWe)	Operational phase	Decommissioning
MZFR	Karlsruhe (KfK)	heavy–water–moderated and cooled PWR	58	1965 – 1984	stage2: dismantling of peripheral systems active components will be securely enclosed in the reactor building
HDR	Großwelzheim (Karlstein)	BWR	23	1969 – 1970	stage 3, converted into a non–fest–facility
KKN	Niederaichbach	HWGCR	100	1972 – 1974	stage 3 licensed
KRB–A	Gundremmingen	BWR	250	1966 – 1977	Dismantling and decontamination of turbine building
KWL	Lingen	BWR	258	1968 – 1977	stage 1 for 25 years
VAK	Kahl	BWR	16	1962 – 1985	License expected

2. HEALTH AND SAFETY ASPECTS OF DECOMMISSIONING

2.1 Safety Goals

The purpose of the licensing procedure is to establish the safety goals listed in Tab. III.

Tab. III: List of Safety Goals

1. To protect life, health, and property against the hazards of nuclear energy and the harmful effects of ionizing radiation
2. To provide compensation for the damage caused by nuclear energy or ionizing radiation
3. To prevent any unnecessary radiation exposure
4. To keep all types of radiation exposure as low as practicable, even where the values are below the specified limiting values
5. 30/90 MREM/A-concept
6. Planning value for incidents
7. Arrangements for mitigating the consequences of accidents or incidents
8. Utilization of radioactive residues without harmful effects and disposal of radioactive waste

The licensee has to prove that his decommissioning plan meets these central goals and thus the workers, the public and the environment are adequately protected.

2.2 Recycling And Reuse Of Steel Scrap From Decommissioned Installations

2.2.1 The Dimension Of The Problem Only a small percentage of the steel in a nuclear power plant is activated or contaminated to a degree that the material will be a hazard to people handling the material or being exposed to its radiation. The amount of steel scrap with potential for recycling ranges from 5000 - 10000 t for a PWR and from 10000 - 15000 t for a BWR. The actual decision whether to decontaminate for recycling or not can only be taken on the basis of prevailing boundary conditions including technical and economical data and release criteria.

At present, licensing for release is done on an ad hoc basis. A general rulemaking by means of a guideline is under way and some essential features of the corresponding thoughts shall be discussed in the following.

2.2.2 Exposure Scenarios Release criteria are based on radiological assessments of the recycling option. Numerous studies exist in this field[12,13,14,15,16]. One of the common results is that in most cases the external irradiation by γ-emitting radionuclides dominates the doses to people exposed to these materials. Tab.IV contains a number of exposure scenarios which might be used to calculate maximal individual doses per year. They obviously differ in their degree of conservatism.

But even the most conservative scenario in Tab.IV is not a worst-case-consideration, as, at least in principle, the specific activity in the final product could be higher than in the released material. This could occur by mechanical separation of scrap with a non-uniform activity distribution, for instance one might think of chips from contaminated surfaces. On the other hand, a case like scenario 1 with increased specific activity can certainly be regarded as extremely unlikely.

It seemed desirable however, to have a method available to assess the actual doses to the general public. Based on this information, one can define exposure scenarios with an adequate degree of conservatism to derive release criteria from them.

Tab.IV: Exposure scenarios for external irradiation

Released at: 1 Bq/g
Nuclide: Co-60

Scenario	Thickness d (mm)	Steel making dilution factor V	Exposure time (h/a)	Dose H (µSv/a)
1	100	1	8000	10000
2	100	1/10	8000	1000
3	100	1/10	2000	250
4	2	1/10	2000	16

2.2.3 Results Of A Stochastic Simulation Of The Release Process The release of radioactive metals from the controlled area for unrestricted use and the processes of recycling have basically a stochastic character. The path ways of these materials can not be predicted deterministically and the quantities which are relevant for the radiological assessment are fluctuating.

A stochastic simulation of the complete process of scrap release, scrap processing, steelmaking, product manufacture and the use of products including exposure scenarios on each of these stages presents an adequate approach. Such a probabilistic model has been developed[3]. The results of the model are individual dose distributions.

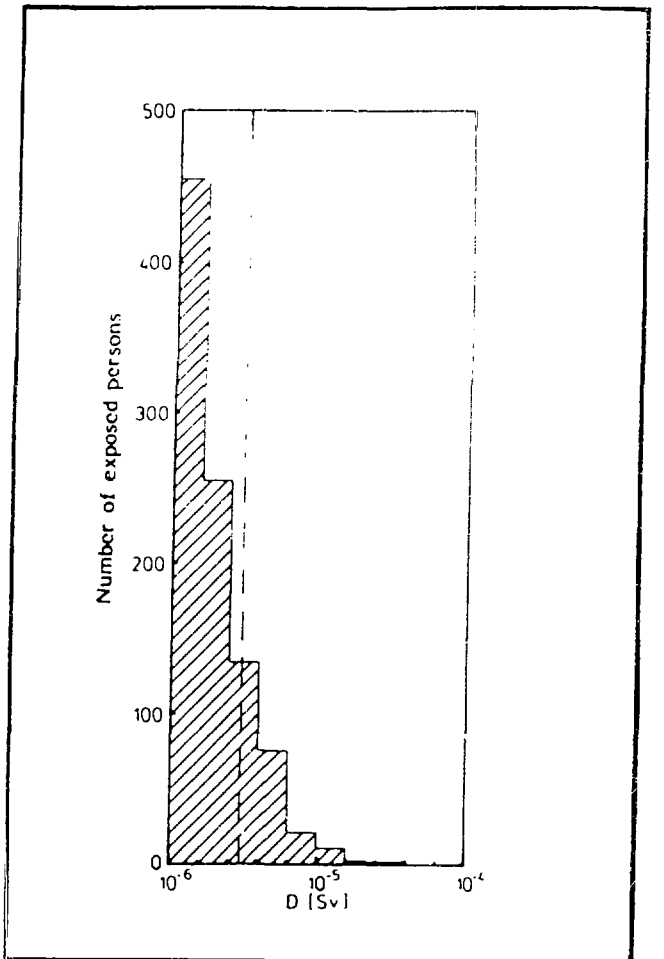
The following results are valid for the base scenario. Some of the important input parameters are listed in Tab. V

Tab. V: Important parameters of the basic scenario

Amount of scrap released	1000 t
Contamination release level	0.37 Bq/cm^2
Release level for "specific total activity" ((surface + bulk activity)/mass))	0.37 Bq/g
Nuclide	Co-60
Average fraction of activated scrap	0.1

The result of each simulation is a distribution of individual doses.

Fig. 1 shows an example which is representative for the average distribution which is obtained by multiple repetition of the simulation.



Typical result of simulation of the distribution of individual doses in the first year after release (base scenario)
(⋮ : 3 μ Sv)

Fig. 1

The results shown indicate that a release based on the criteria in Tab. VI does seem acceptable from the radiological point of view.

2.2.4 Proposed Rulemaking The Commission on Radiological Protection, an advisory body to the Federal Minister for the Environment, Nature Conservation and Nuclear Safety, has formulated the basic principle that the recycling within the nuclear industry (waste casks, shielding plates etc.) should be given priority to. Only where this option is technically not feasible or economically not justifiable material might be released as scrap.

A working party of the European Community has proposed clearance levels for the release of steel scrap of 1 Bq/g for the specific activity and 0.4 Bq/cm² for surface contamination (0.04 Bq/cm² for α-emitters). In the Federal Republic of Germany, this proposal is regarded to be acceptable in its outlines.

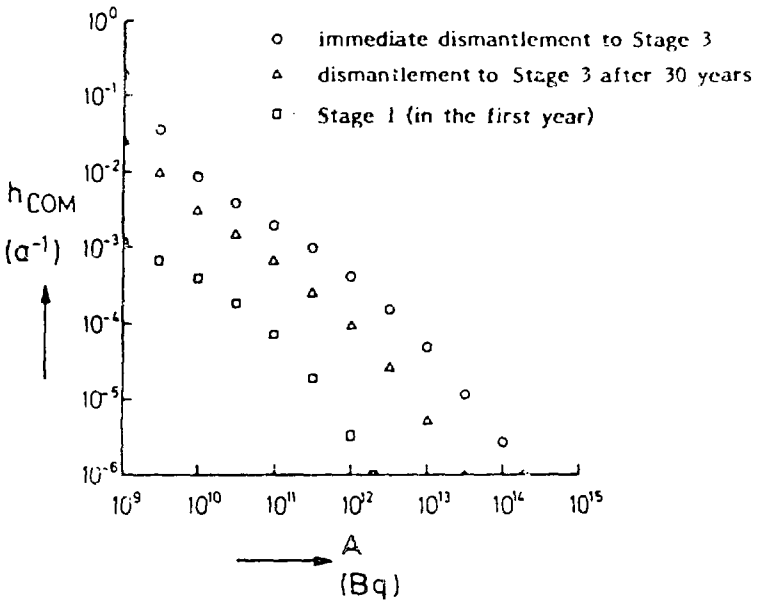
As these levels have been justified by dilution arguments in the radiological assessment, they should not be applied to direct reuse or to the release of products fabricated from active steel under a license. A lower clearance level of around 0.1 Bq/g should be used in these cases.

2.3 Risk Estimate

Before decommissioning starts it is assumed that the nuclear fuel, active plant media and operational wastes are removed under the operating license. Criticality is no longer possible. In the case of an LWR, the remaining radioactive inventory is roughly $^{1}/_{1000}$ of that during operation.

Though the damage potential has been lowered drastically it must not be concluded that the remaining risk is negligible. Due to external (lightning, storm, earth quake, explosion waves, plane crash) or internal causes, the plant might be damaged and activity released to the environment.

In a study[3], a rough estimate of the risk of airborne activity emissions has been given. The results are shown in Fig. 2. In the case of immediate dismantlement to Stage 3 the frequency of activity releases significantly above the licensed values is of the order of 10^{-3}/a.



Complementary frequency distribution h_{COM} for accidental airborne emissions during decommissioning of a 1300 MWe PWR (= average number of events per year with airborne emissions $A_{em} \geq A$)

Fig. 2

Such considerations are intended to support the realization of safety goal 2 (provision of compensation for damage).

2.4 Occupational Exposure And Decommissioning Variants

The experience available so far and planning results indicate that the annual exposure during decommissioning is lower or at most equal when compared to operation. In Tab. VI some data on occupational exposure for decommissioning projects in the Federal Republic of Germany are displayed. Note that the study results[17] for the Biblis and Brunsbüttel plants are conducted for the future decommissioning after the end of their useful plant life of approximately 30 years. There are no actual plans to take these plants out of service.

Tab. VI: Occupational exposure data during decommissioning including planning results (PL) and experiences (EX)

	Project	Project Characterization	Occupational Exposure (manSievert)
PL	FR2	HWR dismantlement to stage 3	1
	KKN	HWGCR dismantlement to stage 3	3
	Biblis A	PWR immediate dismantlement to stage 3	10
	Brunsbüttel	BWR immediate dismantlement to stage 3	14.5
EX	NS"OTTO HAHN"	Nuclear ship with PWR, removal of RPV in one piece, complete decontamination	0.25
	KRB-A	BWR, decontamination and dismantling of the turbine building	0.7

There is an ongoing debate on the optimal timing of decommissioning. International workshops have been dedicated to this subject[18]. The radiological benefits from delayed dismantlement are frequently discussed in terms of decay factors and it is sometimes neglected that the major part of the dose reduction due to the decay is achieved during the first 10 - 15 years, as long as Co-60 is the dominating radionuclide. Decisions for a more extended enclosure or entombment period before dismantlement in most cases are primarily based on financial considerations or unavailability of a waste repository.

From this point of view, the dismantling of a nuclear power plant with a limited work force which is familiar with the plant over a period of 10 - 15 years proceeding from the slightly contaminated areas towards components with higher activities appears to be an interesting alternative to the usually discussed options (immediate dismantlement to stage 3, implementation of stage 1 or 2 for 30 years followed by dismantlement to stage 3).

From the point of view of licensing, however, all decommissioning variants are acceptable provided that they meet the safety goals mentioned above.

2.5 Final Repository

Licensing of major decommissioning activities will pose problems as long as there is no waste repository available. The radioactive waste disposal strategy of the Federal Republic of Germany is outlined in[19]. There are two repositories for deep geological disposal in different stages of realization, the licensing for the former iron-ore mine "Konrad" is underway, while the salt dome at Gorleben is being explored (Tab. VII).

The decommissioning wastes of all nuclear installations in the Federal Republic of Germany have an estimated volume after conditioning and packaging of 150 000 m^3. Scenarios including the other waste flows show that the repository capacities are sufficient for the already existing wastes and those arising in the next decades.

Tab. VII: Repositories "Konrad" and "Gorleben"

Feature	Repository	
	Konrad	Gorleben
Geologic Formation	Oxford (marl, iron-ore)	Salt-dome
Current Status	Licensing Procedure afoot	Characterization afoot
Start up of Operation	1991	2000
Categories of Wastes	Radioactive Wastes with negligible Heat Generation	all Categories including HLW, HTR-Fuel Elements
Capacity	500.000 - 1.000.000 m^3	millions m^3 depending on Emplacement Strategy

3. CONCLUSIONS

Health and safety aspects are important issues in the licensing procedure for the decommissioning of nuclear installations. The available experience and planning results show that there are no obstacles to implement the safety goals during decommissioning.

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AUTOMATION OF THE RADIOLOGICAL SURVEY PROCESS: USRADS ULTRASONIC RANGING AND DATA SYSTEM*

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INTRODUCTION

The Radiological Survey Activities (RASA) program at Oak Ridge National Laboratory (ORNL) serves as the Inclusion Survey Contractor (ISC) in the Department of Energy's (DOE) Uranium Mill Tailings Remedial Action Project (UMTRAP). The ISC is required to identify properties in the vicinity of 24 inactive uranium mill sites where presence of uranium mill tailings is suspected. Tailings are ^{226}Ra-bearing by-product originating from the processing of uranium ore. Mobile gamma scanning is the primary method used to identify these properties.[1] Once identified, the ISC conducts an inclusion survey on the property. The objective of this survey is to perform sufficient radiological measurements to determine if uranium mill tailing contamination is present, and, if so, if it is in excess of relevant Environmental Protection Agency (EPA) criteria.[2] Radon gas emanating from ^{226}Ra is the primary component of exposure to human occupants at these sites. EPA criteria focus on controlling ^{226}Ra concentrations in soil. (^{226}Ra concentrations in soil shall not exceed background levels by 5 pCi/g in the top 15 cm of soil averaged over 100 m^2 area, or 15 pCi/g in any subsequent 15-cm depth averaged over a 100 m^2 area beneath the upper 15 cm.) The concentration of ^{226}Ra in soil can be measured directly by soil sampling and subsequent gamma spectrometric analysis of the sample, or can be estimated by direct measurement of the gamma exposure rate at the soil surface using portable instrumentation in the field. In both methods, the concentration of ^{226}Ra is inferred by examining the frequency of gamma emissions of $^{214}Bi/^{214}Pb$, radioactive decay products in the ^{238}U decay chain.

ORNL was requested to perform 8,000 vicinity property inclusion surveys in three years in the communities near 24 inactive mill sites. In an effort to conduct this radiological survey activity in the most cost-effective manner possible, ORNL has developed a technology to automate

Research was performed by members of the Environmental Assessments Group of the Health and Safety Research Division at Oak Ridge National Laboratory under U.S. DOE contract DE-AC05-84OR21400 with Martin Marietta Energy Systems, Inc.

much of the radiological survey process and provide tabular and graphical survey data output in the field or in the office for report generation. This technological development is called the UltraSonic Ranging and Data System (USRADS).

SYSTEM DESCRIPTION

System Application

The primary component of the inclusion radiological survey is the gamma scan across the entire surface of a vicinity property. The gamma scan consists of a field technician traversing a property in parallel transections while observing and noting the ambient gamma radiation level on an analog portable meter attached to a NaI(Tl) detector. Usually, every square meter of the accessible property is covered by this survey technique and gamma radiation levels are noted in field logbooks. If elevated levels of gamma radiation relative to background are observed, the areal extent of that anomaly is recorded and a soil sample is collected. The radiological survey information is transcribed into a microcomputer (PC) and later formatted into a survey report. The report recommends to DOE whether the property be included into or excluded from the UMTRA Project based on extant radiation levels as compared to EPA criteria.

The USRAD system was developed to automatically locate a surveyor's position on a property and transmit that information along with the instantaneous instrument measurement to a PC in the field. The system conserves technician time in the field by eliminating gridding of the property to locate measurements taken. USRADS conserves time in the office by automatically transcribing data into a PC and reduces report production time by preformatting information into report-ready output. The automation process reduces errors in instrument readout and errors of data transcription, and, because information from the USRAD system is transmitted every second, more information is gathered upon which to base assessment decisions.

System Hardware

The USRAD system consists of a surveyor's backpack (SB), fifteen stationary receivers (SR), a master receiver (MR), custom computer interface or counter timer module (CTM), Compaq Portable II personal computer (PC), and a small trailer to transport this equipment. The SB contains the interface circuitry to receive the signal from the portable gamma detector, an ultrasonic transmitter and radio frequency (rf) equipment to establish a bidirectional communication link with the PC mounted in the trailer. The ultrasonic transmitter is a lead-zirconate-titanate crystal that is in the form of a circular cylinder with a hollow core. The crystal dimensions are 2.2 in. in diameter and 1.445 in. in height. This crystalline material and its dimension result in a natural resonating frequency of 19.5 kHz. The crystal is pulsed for 10 msec each second as the

data from the portable survey instrument are transmitted to the PC via the rf telemetry link. If the PC detects any problems, either with the data or in determining the surveyor's location, a message is transmitted to the surveyor and displayed on the handheld terminal to alert the surveyor of the malfunction. The SB can be operated for a normal eight-hour day from the rechargeable gel-cell located in the SB.

The stationary receivers contain an ultrasonic receiver and a rf transmitter. The dimensions of the metal box that houses the ultrasonic receiver card, rf transmitter card, and rechargeable gel-cell battery pack are 10 x 10 x 15 cm. Each SR has a unique rf frequency so that the MR can identify which SRs heard a valid ultrasonic signal. The MR therefore contains 15 rf receivers, one for each SR and a receiver and transmitter for communication with the SB. Both the MR and PC are powered by a gasoline-operated generator also carried in the trailer.

System Operation

The hardware and software interface is crucial to the system operation. The system synchronization is obtained through the different rf links. The SB is the master timer for the system since the data are transmitted from the SB at the same instant that the ultrasonic crystal is pulsed. This data transmission occurs each second and instructs the MR to "start" the timers on the CTM card associated with each of the SR. As an SR hears a valid ultrasonic signal, it relays a "stop" signal to the MR via its unique rf channel. The "stop" signal from the SR instructs the MR to disable the timer on the CTM card associated with that SR, thus the time-of-flight of the ultrasonic signal from the SB to that SR has been recorded. With a simple calculation, the distance from the SB to that particular SR can be calculated (Fig. 1).

The conversion factor to determine the distance is the speed of sound which is determined during the setup phase of the system operation. During the system setup, the SRs are placed on the property so that the surveyor should be in view of at least three of the SRs from any location on the property. Once the SRs have been placed on the property, the speed of sound is determined and the locations of the SRs that were placed on the property are computed. A display of the computerized location is plotted on the CRT along with the property plat for the operator to view. When the operator has completed the system setup, which usually takes only about fifteen minutes, the surveyor begins the property survey.

As the surveyor begins to survey the property, the ultrasonic crystal is pulsed each second as the data from the survey instrument are transmitted to the PC. Each second, the PC reads the time-of-flight data from the fifteen SRs, determines which ones are valid, triangulates the surveyor's location, plots the surveyor's location on the CRT, and stores all raw data. By plotting the surveyor's location each second, the surveyor can view the surveyor coverage of the property at any time during the survey.

In addition to plotting the surveyor location, the CRT displays on the plot any data point that exceeds a threshold entered by the surveyor, so that any areas of concern are identified on the display, to ensure that sufficient data have been obtained to characterize that area.

System Software

A digitized schematic drawing of the property is stored in the PC prior to the survey using AutoCAD, a commercial computer-assisted drawing software package. The survey data are added to this information. The property schematic is displayed on the PC's monitor (CRT). As the surveyor traverses the property, his past and present position are displayed to denote the completeness of coverage by the surveyor. During the survey, the software checks incoming information and alerts the surveyor (via the backpack terminal) if errors are detected either in the survey data or position data. To ensure data integrity, all data are stored on the hard disk every 30 seconds.

Onsite data reduction is accomplished by several software packages. The USRAD system enables the surveyor to analyze the survey data to ensure sufficient data is obtained to characterize the property before leaving the site. The surveyor can view the data in a number of different graphical formats as well as obtain summary reports. The graphical formats supported by the USRADS are Replay, Block Statistics, Contour, and 3-D plots of the radiation data. The Replay program will generate the same display that the surveyor viewed when the survey of the property was completed. The data are replayed in the same order as the data were taken. The Block Statistics routine enables the operator to select a grid block size and have the data analyzed for each block. If the mean of the data for a particular grid block is greater than the operator-entered threshold, then that block is highlighted on the CRT, and the statistical information for that grid block are stored in the summary report. Raw data are converted to appropriate units and displayed or printed out in tabular or graphical format. By indicating preset thresholds, areas of contamination can be identified and vital statistics can be calculated (area size, number of measurements, measurement range, average and standard deviation). Graphical representations are made in two- and three-dimensional display (Fig. 2). The Contour routine generates a summary report and outlines the areas that exceeded the user input threshold. The 3-D plot generates two different views of the data and provides a means by which the surveyor can view the entire data obtained during the survey. Information can be displayed in the field and is output directly into a report-ready format without transcription.

BENEFITS OF THE SYSTEM

The USRAD system is a hardware/software ranging and data transmission system that provides real-time position data and combines it with other portable instrument measurements. Live display of position data, onsite

data reduction, presentation, and formatting for reports, and automatic transfer into databases are among the unusual attributes of USRADS.

Approximately 25% of any survey-to-survey report process is dedicated to data recording and formatting, which is eliminated by USRADS. Cost savings are realized by the elimination of manual transcription of instrument readout in the field and clerical formatting of data in the office. Increased data reliability is realized by ensuring complete survey coverage of an area in the field, by elimination of mathematical errors in conversion of instrument readout to unit concentration, and by elimination of errors associated with transcribing data from the field into report format. In one DOE-sponsored program, this represents a savings of almost $500/survey (over 9,000 surveys performed, e.g., $4,500,000 savings).

The USRAD system can be adapted to measure other types of pollutants or physical/chemical/geological/biological conditions in which portable instrumentation exists. Applications may include chronically contaminated areas; emergency response in mapping pollution areas where accidents have occurred; and, combined with the use of robotics, contaminated areas may be assessed which are too hazardous for human access (i.e., nuclear reactor cores, highly contaminated areas, etc.). Additionally, it may be used for biological or geological characterization of an area. As a surveyor traverses a region of interest, geological formation, magnetic fields, species of trees, and insects could be noted and transmitted.

The USRAD system fills a void in the automatic data collection/correlation/reduction system's arena, particularly for the outdoor suburban environment. In addition, USRADS fills this void in a very cost-effective and innovative manner that allows a surveyor to rapidly collect, spatially, high quality data with a minimum amount of training. System setup time is minimal and easy. Repeatability of the survey results from multiple surveys is excellent. The versatility inherent in USRADS allows a wide variety of sensors to be utilized and even multiple sensors simultaneously. Onsite verification of the completeness of the survey is automatic. While approximately half the savings of field data collection costs may be attributed to the field portion of the USRADS data collection, the other half of the savings has to be attributed to the data reduction, presentation, and integration into computer-assisted-drafting electronic files and into electronic databases where reports and plots are automated.

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HEALTH PHYSICS PROCEDURES IN THE REMOVAL OF CONTAMINATED EXTERNAL DUCTS IN A POPULATED AREA

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ABSTRACT

The Special Metallurgical (SM) Building at the U.S. Department of Energy facility located in Miamisburg, Ohio is currently in the decommissioning process. An early task in this project was the removal of external ductwork which was highly contaminated with plutonium-238. Approximately 450 linear feet of duct which averaged three feet in diameter were included in the project. One duct, approximately 150 feet in length, connected an external filter bank with a stack plenum and was moderately contaminated. This duct was removed first so that contamination control techniques could be developed for use later in the removal of the second duct which was highly contaminated. Because SM Building is located approximately 100 meters from the fenceline procedures were required which offered a high degree of assurance that airborne release of contamination would be strictly controlled. Special fixation techniques, decontamination methods, and other procedures were employed in the removal of the ductwork and subsequent removal and replacement of the roof.

INTRODUCTION

The Special Metallurgical (SM) Building was constructed in 1960 at the U.S. Department of Energy (DOE) Mound Facility in Miamisburg, Ohio. SM Building was used to: 1) process plutonium-238 into a final product form for encapsulation in isotopic heat sources; 2) recover plutonium-238 residues generated from isotopic fuel production, from R&D activities, and from analytical support activities; 3) encapsulate plutonium-238 fuel forms in the fabrication of isotopic heat sources; and 4) perform analytical activities required to support the plutonium-238 production and recovery processes (Figure 1). Construction of a new plutonium facility, PP Building, was completed in Dec, 1967.

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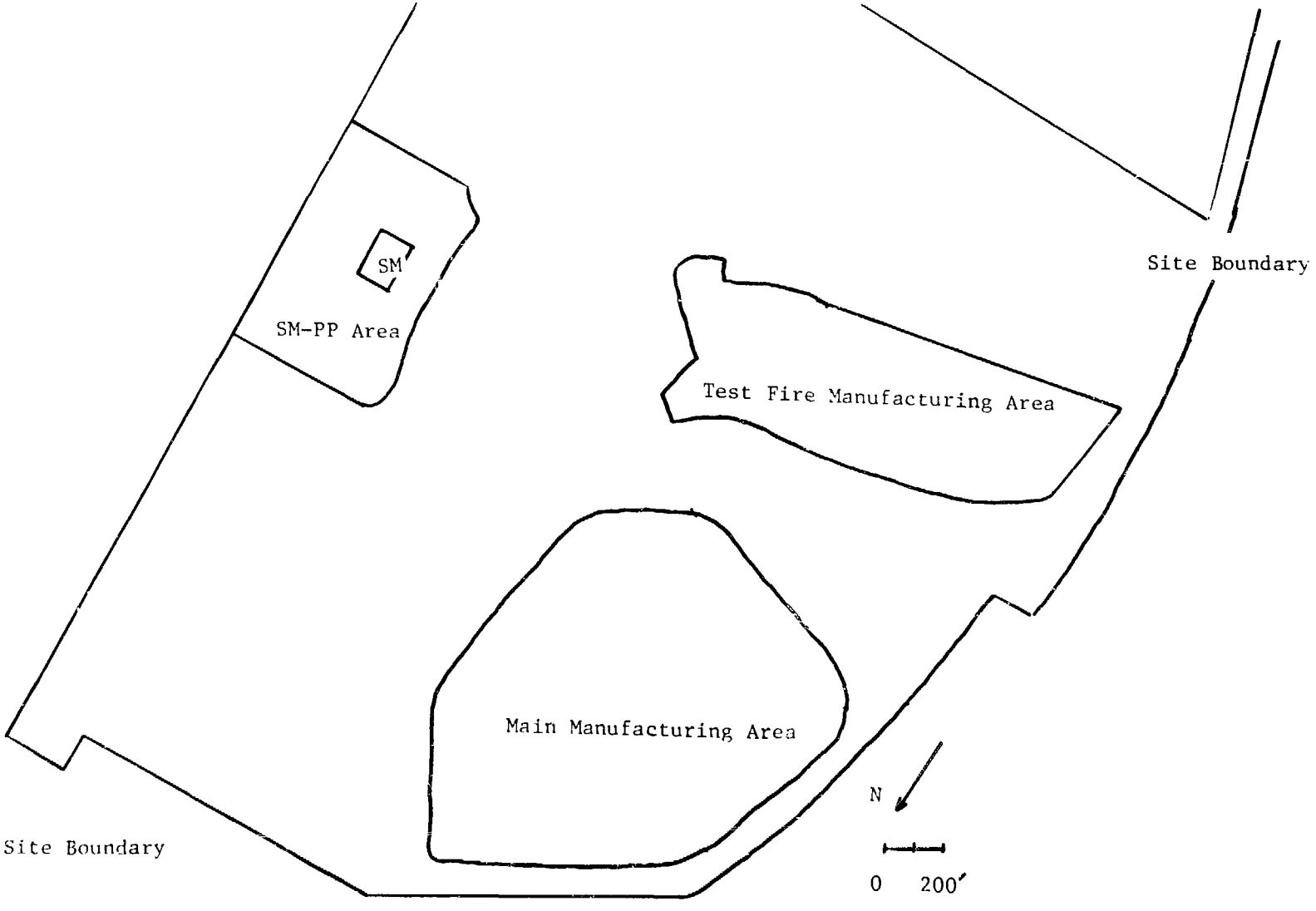

Figure 1. Location of the Special Metallurgical (SM) Building at Mound

The design of the new facility incorporated numerous safety-related features which the SM Building did not have. Plutonium operations were then transferred from the SM Building to the PP Building. Termination of the SM Building began in August, 1968, and continued through Aug, 1972. During this effort, approximately 6 kg of plutonium-238 were removed from the facility. Termination of 600 feet of gloveboxes, fumehoods, and enclosures was also accomplished during this period. All internal room walls, ceilings, floor coverings, piping, conduit, and miscellaneous services were removed. The termination effort ceased when it became progressively more difficult and costly to remove the small amount of plutonium remaining in the structure. When the termination effort ceased, the building was left with approximately 13,000 $ft.^2$ contaminated with plutonium-238. At this time, SM Building was placed into a stand-by mode awaiting decommissioning funding. Approximately 0.3 Ci of plutonium-238 remained in SM Building (Table I). Routine surveillance and maintenance was performed to maintain the building in a radiologically safe condition until decommissioning could be initiated. In FY 1982, a program of surveillance and major maintenance was initiated. Decommissioning was initiated in FY 1983.

Table I

Location of Plutonium-238 Inventory Above-Ground Level

Pu Inventory (Ci)	Location
0.094	Ball Room
0.010	Insulation (wall)
0.008	Roof (tar)
0.002	SMA Filter Bank Ducts
0.006	SMA Filter Bank
0.006	SM-2 Filter Bank Ducts
0.040	SM-2 Filter Bank
0.007	SM-3, 4 Filter Bank
0.007	Repackaging and Redrumming Facility (R&R)
0.001	Low Risk Waste Tanks
0.003	Stack and Plenum
0.031	Surface of Ground
0.012	Low Risk Waste Drains and Vents
0.003	Other
0.3 Ci Total	

One of the first major tasks associated with the SM Building decommissioning included removal of excess contaminated exhaust ducts from the roof and subsequent removal and replacement of the roof which had deteriorated during the previous twenty years. Contamination from various activities during the history of the building was present in the roofing material, primarily in the tar area. Although this contamination was not dispersible, it was determined that a fire could result in an environmental release. Also, the deteriorated condition of the roof decking made walking on the roof potentially dangerous. A significant snowfall might cause local collapse. A good, solid non-leaking roof was necessary before further removal of contaminated services inside the building could proceed. Health Physics was tasked to provide planning support, monitoring, and providing survey support to Engineering and Nuclear Operations.

The duct removal phase was begun in 1982. Detailed surveys were performed on the exhaust ducts that connected the inactive Special Metallurgical Addition (SMA) filter bank to the plenum, the glovebox exhausts, and the mist eliminator that fed the active SM EF-2 filter bank, and the Vault 37 duct (Figure 2). The plenum, stack, and roof were also surveyed (Table II). The survey showed that in most areas the contamination level was measurable but could be handled with relative ease using existing technology. Two areas, the glovebox exhaust and mist eliminator, contained significant contamination. At the levels of contamination found in these two areas, the total release of contamination from one square centimeter would exceed the USDOE air quality standards of 2000 cubic meters of air. Because the fenceline was located approximately one hundred meters east of the building, there was concern on the part of Operations, Engineering, and Health and Safety management.

Table II

Contamination Survey Results

	Pu-238 Contamination Average Removable, dpm/100 cm^2	Average Fixed dpm/100 cm^2
Duct- SMA Filter Bank to Plenum	2,000	50,000
Plenum Interior	400	100,000
Vault 37 Duct	100	100
Stack (at base)	<200	25,000
Glovebox Exhaust	400,000	>4,000,000
Mist Eliminator	400,000	>4,000,000
Roof (Tar material)	25	100

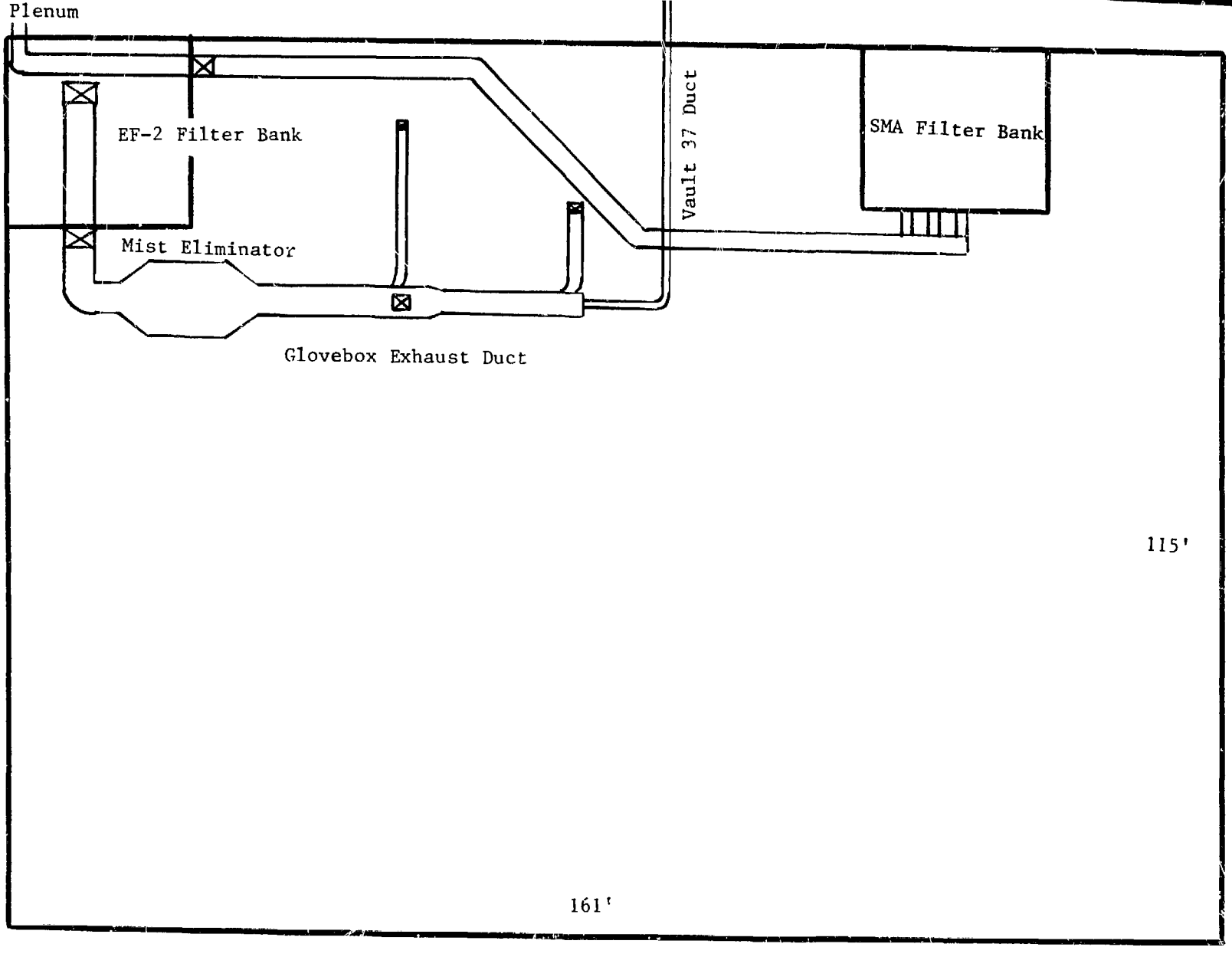

Figure 2. SM Building Roof and Ducts

Engineering decided to phase the project as follows:

1. Remove the Vault 37 duct
2. Remove the duct from SMA Filter Bank to the plenum
3. Remove the exhaust duct
4. Remove the mist eliminator
5. Remove and replace the roof

New methods of contamination control would be tried and evaluated on the less contaminated materials. Those that met with the greatest success would be applied to the removal of the more contaminated ducts.

Management guidelines decreed that contamination would be controlled to the maximum extent at the worksite, workers on the jobsite would be monitored while the work was progressing, and no contamination should leave plantsite. With these criteria in mind, the following contamination control methods were considered by Health Physics:

* Use of sheet metal enclosures where appropriate
* Use of portable plastic and sheet metal enclosures
* Use of portable exhausters with HEPA filtration
* Use of bagging techniques for separation of duct sections
* Strict zone control with step-off pads
* Use of fixatives to control wipeable contamination

Because much of the ductwork required size reduction prior to packaging for shipment to a burial site, a facility had to be constructed near the worksite for this operation. The abandoned filter bank area inside SMA penthouse was selected for this use.

Health Physics chose appropriate monitors for use inside the enclosures, at the zone control stations, and at the boundary of the worksite so that air levels, contamination control, and work conditions could be monitored. Weather considerations were important; it was agreed by all parties not to separate duct if rain was imminent or if the average windspeed exceeded 10 miles per hour (5 m sec^{-1}).

Because Vault 37 duct was lightly contaminated and located in a rather inaccessible area, it was removed using bagging techniques and portable enclosure where possible. Contamination was contained inside of the duct during the operation. Size reduction proceeded and the resulting waste package was lowered to the ground by crane.

The next step required removal of the duct which connected SMA Filter Bank with the stack plenum. The first task was to isolate the duct from the plenum. This was accomplished at a rubber boot near the plenum. During a stack shutdown, the boot was cut and two metal caps clamped onto the separated ends. The other end of the duct, in the SMA filter bank, was blanked off using a similar cap. A four-inch exhauster hose was connected through an adapter from the EF-2 Filter Bank to the static duct to provide continued negative airflow. Two-inch holes were cut in the duct at approximately one meter intervals and the duct interior painted with a non-flammable (water-based) household paint using an air sprayer. The holes were covered with sheet metal plates. After the paint dried, the portable tent enclosure was moved into position. A large plastic bag was taped onto the duct with tools inside. Hand ports with gloves were taped into slits in the bag. This arrangement was then used as a glovebox for separating the duct. After the duct was sectioned the parts were separated approximately one-half meter. The bag was tied off in two places about ten centimeters apart and the bag cut between. The tools were bagged out in the same manner and sheet metal end caps installed over the duct ends. With the metal caps in place, the duct section which weighed approximately fifty kilograms was transported to the size reduction facility. In some cases, it was not practical to separate a short section of duct. When this occurred a crane was used to handle the heavy piece of duct and lay the length on the roof for sectioning into a more appropriate length.

This duct was completely eliminated from the roof by removing one section at a time, working toward the end which had the exhaust connector so that negative pressure was always maintained. The use of the paint fixative, exhaust, bagging technique, and portable enclosure successfully precluded contamination from becoming airborne during this operation. The removal of this duct required approximately eight weeks. This effort provided the experience necessary to attempt the removal of the highly contaminated exhaust duct and mist eliminator.

Health Physics lessons learned included the following:

* Continuous high-volume air sampling (55cfm) provided at the roof perimeter provided adequate sampling volume for projected dose calculations. The immediate work area inside the tent enclosure was monitored with the standard alpha Constant Air Monitor with a background subtract feature to minimize radon interference.

* Air proportional alpha instruments are inadequate for outdoor use in this region of the country due to humidity problems. They are adequate for short durations of time. Alpha scintillation instruments are extremely light sensitive and may cause false high readings if not used carefully. This was noticed particularly at the hand/foot check stations established at zone control points.

* Painting with a standard paint such as a rust preventative provides excellent control of wipeable contamination. Although this paint may chip and peel, the resulting particle is sufficiently large that it does not create a dispersible airborne problem. Painting inside a duct with an exhaust requires a prefilter at the end of the duct to preclude clogging the HEPA filters.

* The bagging technique employed allowed contaminated tools to be reused often because they were not allowed to be exposed to the environment. They would be bagged into a section to be cut, then bagged out. This bag was then used in the subsequent section.

* Industrial safety considerations for working on rooftops presents considerable problems when the workers are wearing respirators, plastic shoecovers, gloves, and hardhats. Also, heat factors multiply when working in a protective suit with respirator on a black tar roof.

* Because of the weather constraints that were included in the work procedure no difficulties from wind were encountered. On two occasions a perfectly beautiful morning clouded up within two hours. On one occasion the work was completed before a storm hit. In the other case the storm passed south of the site. Coordination with a local weather station may have precluded these storms from surprising the work crew.

* Use of the crane allowed waste packages to be moved to and from the size reduction area. In some cases the duct was moved by crane to a more favorable location for continued work.

* Maintaining a negative pressure on the duct during the entire operation gave great assurance that no contamination inside the duct would escape during the removal procedure.

The glovebox exhaust duct provided additional challenges to the Engineering Group. This duct penetrated into the contaminated building in a number of drops (Figure 2). Also, a thick layer of contaminated dust had filled the bottom of the duct to the point that the paint layed on top rather than soaking through and fixing the contamination in place. This duct had rusted considerably and therefore was not nearly as strong. Handling created a problem. Because of the large duct size it was not possible to size reduce in the SMA Filter Bank area. This task would have to be accomplished in the SM Building in the contaminated area. This required that the duct be lowered from the roof to the first floor and guided through a set of double doors that entered the contaminated area of the building. This duct did, however, exhaust directly to the active EF-2 Filter Bank so that a negative flow was automatically maintained.

Before the job could begin, a method for removing the contamination was necessary. Eventually, a person entered the mist eliminator with a

HEPA filtered vacuum sweeper with a wand onto which additional sections could be added. He was able to reach most of the duct from the mist eliminator. The actual removal of duct sections was able to proceed as discussed except for those sections that penetrated the roof. In these cases, the portable enclosure was placed over the penetration and the roof was cut around the duct from inside the building. The open duct end was capped with a sheet metal blank. This duct end was then supported from inside the building with a lift truck. Contamination outside was cleaned from the area while the enclosure was in place, then the area was wrapped in plastic and held in place with sandbags. The portable enclosure was then moved into location to section the duct. The crane was attached so that the duct could be immediately removed from the roof. Bagging techniques were used to separate the duct, end caps placed on the sections, and the duct piece lowered to the first floor. This duct was removed in approximately six weeks.

The next task to be completed was removal of the mist eliminator. Because the mist eliminator was so large and was not located near an edge of the roof, it was decided to size reduce this piece in place and load it directly into a waste package. A sheet metal and plastic enclosure was constructed over the mist eliminator and attached to the EF-2 Filter Bank Penthouse. An airlock was constructed on this enclosure for waste staging and packaging. The mist eliminator was painted, cut with reciprocating saws, and wrapped in plastic. The pieces were passed into the airlock and placed into the waste package.

Two associated drains were removed and the roof penetrations sealed. The job required approximately four weeks to accomplish.

After the ducts had been removed the next task was removal and replacement of the contaminated roofing material and repair of the roof decking. Samples indicated that low levels of contamination were present in the roof insulation. Recognizing that the survey was not exhaustive in nature, it was decided that the roof should be removed inside an enclosure. This enclosure had to be large enough to support a sizeable work area, yet be light-weight enough to move manually. The enclosure was constructed and used for the first two roof sections. However, it was noted that the insulation was so wet that it was not dispersible at all. Monitoring inside the enclosure indicated that no contamination was being generated, so a field decision was made to abandon the use of the enclosure. A complication arose from the roofing insulation itself. It was so wet that it could not be packaged in its condition. Therefore a method of reducing the moisture content was required. A solar drying tent was constructed on the asphalt drive behind the building. A HEPA exhausted fan was installed on one end and a filtered air supply on the other end providing a sweep across the material which was placed on oven-type racks and stacked along the walls. This drying tent proved to be quite efficient and the typical residence time for roofing material was two days. The material was then packaged in waste containers. The roof removal required approximately thirteen weeks.

Once the roofing material and underlying insulation were removed the deck was cleaned and surveyed. No wipeable contamination in excess of 20 dpm per cm^2 was found. In some cases fixed alpha contamination exceeded 200 dpm per 100 cm^2. Because of this localized fixed contamination, the decking was painted as a fixative. Because the contamination had been so reduced a commercial contractor was permitted to install the new roof. This was completed in a few weeks. The new roof is functioning perfectly and has allowed further work to proceed inside the building.

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* Former Mound employee

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SECTION VI

DECOMMISSIONING TECHNIQUES AND TOOLING

METHODOLOGY FOR SELECTING TOOLING AND TELEMANIPULATED EQUIPMENTS FOR A DECOMMISSIONING PROJECT

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ABSTRACT

The paper in concentrating on the "end effector" aspect of a Remote Handling machine dedicated to dismantling nuclear facilities. End effector comprises the toolings and tool handling device (the Manipulator). This end effector is placed at work thanks to a transporter generally large and unaccurate machine. Desirable qualities of toolings (analysis is volontarily limited to metal cutting processes) are discussed with respect to the particular dismantling application. Comparisons are presented and discussed in order to help for the selection in front of a given project. Similar approach is presented concerning manipulators, quality parameters are proposed and discussed considering three major classes of machines : the power Manipulators - Servomanipulators and Industrial Robots.

An a priori and universal choice is not reasonable to consider particularly because of the basic duality of decomissionning activity : the project rounds around the waste identity which is the key product, and the "production" plant design has to comply with an existing facility and all the corresponding constraints. In conclusion possible development orientations are proposed considering a new field of investigations which addresses small size devices and work in parallel with multiple machines to reduce costs and down times. Decomissionning projects of the future may be looked at as flexible workshops.

I - INTRODUCTION

The paper is addressing essentially remotely controlled machinery for dismantling nuclear installation. Few projects have involved, up to now full remote handling in the field for several reasons :

- Low level of radioactivity allowing "hands-on" work.
- Limited number of facilities already been decommissionned.
- Low level of dismantling often selected.
- Cost effectiveness with respect to hands on workers.

This situation make it uneasy to extrapolate and to identify general trends. However future projects will involve highly activated and contaminated structures or components, (power plants, reprocessing facilities) mandating to develop important remote handling devices. An effort has to be made in order to prepare this future. Present projects involve rather heavy machines difficult or impossible to transport on different sites. Even when looking at subcomponents of the system (elements of transporters, cabling and transmissions, etc.) one can hardly identify "reusable" components except for two subsystems :

- The manipulators.
- The toolings.

Indeed a direction to work on to reduce costs for complex remote handling machines, is to look for a form of standardisation of the components. However dismantling machines are very site-specific mainly because very poor decommissioning provisions where taken in account during the design of the facilities. We could envisage some kind of "multi purpose" transporters or transfert system but these would certainly remain much more plant-specific than the head end of the machine wich receives the manipulator that handles the tool.

Our objective in the paper is consequently to concentrate on these two devices, attempting to set a selection methodology which could help to identify the correct candidates and to identify the "standards".

I.1 Definitions

The words manipulator-tooling-transporters may become ambiguous when applying to a given project. A tool (plasma torch for example) may include in its packaging some sort of degrees of freedom (fine positionning) that could functionnally be considered as part of the manipulator. Moreover a tool may be fixed directly at the tip of a transporter without any easy identification of the manipulator.

We shall precise what is our proper understanding :

. **The tool** is the active end-effector of the machine, it supports the working process (handling, cutting, decontaminating, etc.). The tool may include certain degrees of freedom : for fine position adjustments, at grasping interface with the manipulator or simply to actuate its main process (spining the grinding wheel).

. **The manipulator** is directly handling the tool. It has first, to place it at work in the proper positions and orientations when the tool is "autonomous" (shear machine or explosive ribbon). Then the manipulator will guide the tool continuously along a trajectory (positions and orientations) when the tool is not autonomous (plasma torch). Additionally the manipulator may have to support the weight of the tool (this is not always the case, the weight can be balanced thanks to auxiliary means, which is of particular interest when the tool is large and heavy). It has finally to counterbalance reaction forces generated by the process, and here also the forces may have to be finely regulated (force control, force feedback in case of grinding for example).

. **The transporter** is carrying the manipulator to place it grossly at work. The transporter is not able to perform fine motions (with respect to general requirements of toolings) its function is essentially to extend the reach of the manipulator and to provide a support base.

Naturally other subsystems of a dismantling machinery can be identified (waste transportation, power and signal transmission systems, control station, etc.) but they are out of our present scope.

I.2 Preliminary statements

A dismantling project can be compared to an industrial workshop in which processed product is the "waste". Whole scenario is conditionned by the characteristics of this key element. The "product" is characterized by following items :

1. **Radiological identity** (contamination and activation) which conditions its final processing and the bioshielding provisions in the installation. It will have also some influence on the size of the transported package in the plant and decontamination procedures.

2. **Final dimensions and packaging method** (naturally coupled with 1) will drive the segmentation process and complexity of additional conditionning zone.

3. **Material nature and initial shapes** influence the tooling selection and segmentation strategy.

4. **Initial environment** may preclude the utilisation of certain processes (explosive cutting - air or under water operations).

5. **Accessibility** has direct influence on remote handling machinery design.

6. **Overall production cost** including capital and operating costs which leads to difficult but necessary comparisons between alternatives.

The second fundamental specificity of dismantling, is that one have to comply with the actual capabilities and architecture of an existing facility. Classical industry is building the plant around the product. This duality has always to be present during the dismantling design process. One cannot define toolings and the way to operate them only looking at the tubings or vessels to cut-down.

II - TOOLING SELECTION

Main tasks to perform in a dismantling project are following :

1. **Inspection** : general survey and close examination in order to prepare the segmentation sessions.

2. **Cutting and removal of equipments** : comprising metallic parts (tubings, vessels, massive elements, shielding) as well as concrete and few miscellaneous (electric wires).

3. **Handling and transportation of wastes to a conditionning zone:** which may be bottlenecking if the appropriate throughput is not to provided (without dramatic modifications of the facility).

4. **Wastes processing, compacting, packaging and posting out** : which requires to build (or refurbish) a special conditionning zone in the facility. The cost implication of this zone is important in the project because it is non recurrent and plant specific.

5. **Decontamination** takes place before any dismantling in order to reduce dose rates, during dismantling process to clean the pieces (eventually at a level authorizing its reuse in the classical industry), and after all mechanical disassembly to render the facility available to other uses.

6. **General purpose tasks** : house keeping. Improvement and refurbishment of auxiliary equipments of the facility, modifications in order to allow installation and work of the remote handling machines.

Tooling will then be required at several locations within the facility, and operated in different ways : hand operated in low radiations areas, installed on fixed machinery in the waste packaging zone (high capacity cutting), handled by mobile remotely operated "Robots" to address the most active and unaccessible areas. As mentionned before we limit our present approach to this last category and more particularly to metal cutting operations (part of class 2 task description).

A selection methodology for toolings can be based on the cross coupling of the two following parameters fields :

The **intrinsic performance** parameters which concern the performances of a given tool with respect to a given criterion (weight, rapidity of cut, pollution generated, etc.). This evaluation is independant from any kind of application.

. The **specific constraints** coming from a given dismantling application.

II.1 Assessment of performance parameters for metal cutting tools

Classification of existing processes can be following :

1. **Mechanical**

   . abrasion, chips removal = x milling, nibbling, sawing, grinding
   x grit charged water jet

   . cutting = shearing

2. Thermal

   . fusion = oxygen and plasma torches, oxygen lance, laser, electrode
   . erosion = arc saw
   . cracking = welding cracking.

3. **Explosive**

We can divide the parameter field into three categories :

. The **internal** parameters addressing only the intrinsic performances of the tool (type and nature of components able to be cut, velocity of the cutting process, etc.).

. **The external** parameters characterise the ease to operate and service the tool (weight, required accuracy in position guiding, maintainability, cable management, etc.).

. **The safety** parameters which are separated in order to highlight this important aspect (pollution by airbone particles, fumes, etc.).

Desirable "qualities" of a tool with respect to decommissionning application will then be following :

II.1.1 Internal parameters

1. **Large cutting capacity** : elements generally encountered are tubings of various diameters and thickness (nominally Ø 10mm) metallic

plates (including large vessels or pipes) and massive structures (pressing - machine).

2. **High cutting velocity** not to slow down the overall process.

3. **Ability to cut any current material** in order to avoid multiple changes of tool end in front of multi metals structure.

4. **Reliability and robustness** to support without damages the rugged operating conditions unavoidably encountered during dismantling work.

5. **Easy to maintain** and quick return to work which may include the decontaminability requirement.

6. **Low capital cost**

7. **Low operating cost**

II.1.2 External parameters

1. **Weight** of the tool head is of prime influence it will condition the the manipulator choice.

2. **Size** of the tool package shall be as small as possible not to preclude accessibility to certain areas.

3. **Reaction forces** exerted on the tool carrier shall be small or better equal to zero. An important distinction has to be pointed out here because its influence on the Manipulator selection is central, two types of forces may be exerted at tool/Manipulator interface :

.. **parasite forces** asking only to the supporting device to comply with this forces without suffering any damages (ex. : jaming during shearing process - vibrations and shocks during grinding operations).

.. **forces that need to be controlled** in intensity and in directions in order to operate correctly the process. The simplest class is **permanent** reaction forces (water jet reaction) that need to be balanced, the most complex is **contact forces** adjustment on the trajectory (grinding) which asks for a Manipulator able to perform a hybrid position/force control automatically or with manual help (force feedback manipulators).

4. **Large tolerances in tool/piece relative positionning.** Tenth of a millimeter is very demanding for the manipulator controls and mechanics. 1 centimeter is a comfortable value.

5. **Large tolerances in speed control stability.** Very low and continuous advancing speeds are difficult to ensure with conventional

manipulators.

6. **Simple supply leads.** Large and complex supply lines (as plasma torch) will generate troubles in cable management and connections (secondary cooling lines...).

7. **No parasitic effects** that will disturb or make more difficult the direct control of the process (intense light or fumes emissions - high noise levels...).

II.1.3 Safety parameters

1. **Low production of chips or slags,** which pollute the environment and may spread some contamination.

2. **Low production of airborne particles** that will increase the levels of disperse contamination and will cause some troubles on the filtering systems.

3. **Low operationals risks** to damage the surroundings or the bioshields (explosive cutting may damage the filtering system. Thermal cutting may initiate fires).

II.2 Comparison table

. Referring to previous classification of metal cutting processes we propose an evaluation matrix on table 1. Process of similar performances have been grouped together. Others are standing alone in the columns because of their unique particularities. Qualification of the performances is given through 3 numbers, high number means good performances with respect to corresponding parameter. Water jetting, laser cutting and thermal cracking are mentionned because they are promising though not having realy been used in dismantling already.

II.3 Discussion

. With an equal ponderation of all the performance parameters, one establishs that shearing tool seems to be the best. Indeed shearing is clean and rather easy to operate but is is very slow and not applicable to a large variety of metallic structures. This is highlighting the fact that each parameter has to be ponderated according to the characteristics of the application. There is no best cutting process as a general rule.

. A ponderation column is present on table 1 to introduce these project-specific aspects. One can quote each performance parameter on a relative basis depending on its importance for the application.

Classes of processes / Performance parameters	MECHANICAL	ABRASION	GENERAL	WATER JET	SHEARING	THERMAL	FUSION	LASER	ARC SAW	THERMAL	CRACKING	EXPLOSIVE	
Internal													
1. Large capacity	1		3	1		3		2	2	1		2	
2. High velocity	1		3	1		3		3	3	1		2	
3. Multi metals	3		3	3		1-2		2	3	1		2	
4. Reliability	2		3	2		2		2 ?	2	3 ?		3	
5. Maintainability	2		1	2		3		1	2	1		3	
6. Low capital cost	3		1	3		1-2		1	1	1		3	
7. Low operating cost	3		2	3		1-2		1	3	3		2	
External													
1. Low weight	1		2	1		3		2	2	2		3	
2. Small size	2		3	1		3		2	2	2		3	
3. No parasite forces	1		2	1		2		3	1	3		1	
4. No controlled forces	1		2	3		2		3	2	3		2	
5. Positionning tolerance	3		3	3		2		1	1	1		1	
6. Speed tolerance	3		2	3		1-2		1	1	1		3	
7. Simple supply	3		1	2		1-2		1	1	1		3	
8. No parasites effects	3		1	3		1		1	2	2		1	
Safety													
1. No chips, slags	2		1	3		1		1	2	3		1	
2. No particles	3		2	3		1		1	2	2		2	
3. No risks	3		3	3		2		3	3	3		1	
TOTAL	Depend on ponderations												

ponderation coefficients

TABLE 1 : Tooling comparison example

Naturally the proposed evaluations are debatable, they may vary when taking in account a precise given tooling and they shall be expanded to better surround the actual characteristics of candidates.

III - MANIPULATORS SELECTION

Various operating concepts can be envisaged to handle the tools, we may summarize the alternatives in the following way :

- **Telemanipulator + Tool** = . power manipulators
  or
  . servomanipulator

- **Industrial Robot + Tool**

- **Task specific machine + Tool**

A specific machine is taylored to one type of task and only that one. It is clearly simpler but less versatile and adaptative than the other choices. No help will be provided in case of unscheduled situations and it is cetainly less applicable when more than one or two different tasks are to be achieved on site (operating different cutting processes - handling pieces of equipments after cutting...).

An "Industrial Robot" is understood as beeing developped and marketed to apply to the manufacturing industry, which is essentially non nuclear. It has however to be considerred as an alternative to equip dismantling machines because of following advantages : wide range of different types of machines on the market - large diffusion of the different models which offers a good guarantie with respect to availablity of the system and its subcomponents - low cost with respect to specially developed machines. Naturally important drawbacks will be found regarding its applicability ; we can mention : important total weight and low payload to weight ratio - no force control (or very limited) - poor "man in the loop" control possibilities - poor provisions regarding decontaminability and work under radiation fields.

Nuclear Telemanipulators are clearly dedicated to work in the type of environment we consider. Classicaly two families of such machines are available : the **power** manipulator which is **unilateral** (no force feedback) and generally controlled through simple push-button boxes. Power manipulators are used to handle large payloads without great dexterity. - ***Servomanipulators*** which offers unique characteristics in controlability through their bilateral Master-slave mode. Force feedback in the master controller authorizes to directly monitor the forces exerted on slave environment. Additionally recent Computer Assisted Teleoperation (CAT) controls enhance the capabilities of the system.

III.1 Assessment of performance parameters for a Manipulator

Desirable performances of the Manipulator are highly depending on the selection of the tool and the choice of a dismantling scenario. Following discussion is insisting on the critical parameters and the cross coupling that have to be taken in account in the selection process :

III.1.1 Internal characteristics

1. **Force/payload capacity** requirements are directly driven by the tool characteristics. Moreover additional tasks (waste handling) may add requirements on this point. Small values are obviously opening the field for Manipulator choice.

2. **Force control capabilities** : in case of operations involving contacts with the environment pure position control may not be sufficient (insertions - extractions - grinding tool...). Forces shall then be measured and controlled. This can be done automatically and/or through operator's control (thanks to Master-slave control as an example).

3. **Positionning accuracy** : achieved in open loop or in closed loop thanks to external sensors, the requirement is directly derived from tools positionning tolerances. Unfortunately light weight and compact tooling don't accept large tolerances leading to strong requirements on manipulator's accuracy.

4. **Velocity** has to be high enough not to disturb manual operations, 0,5 m/s is about a lower limit.

5. **Low speed** control stability characteristics has to match guided tools requirements. Thermal cutting processes on thick plates are generally very demanding in speed regulation.

6. **Work in all positions** with respect to gravity is required when the device is intended to be mounted on a highly movable transporter. Due to lack of balancing some machines are not able to work upside down or sideways particularly.

7. **Inherent protection againts shocks** is important to consider because collisions with the environment are unavoidable during dismantling operations. Rigid structures and actuation trains will suffer from shocks. "Inherent" means that built in protections have been provided so as mechanical compliances (cable driven mechanisms as an example).

8. **Range of motion** of the maniuplator shall be naturally as large as compatible with other requirements (resolution and total weight notably). Range of motion shall be considered in full space meaning 3 positionning and 3 orientating degrees of freedom of the payload. Real

usefull range of motion of manipulators is generally small compared to its dimensions.

9. **Versatility** : ability to operate number of toolings. This addresses the mechanics as well as the controls that shall comply with different operating conditions.

III.1.2 External parameters

1. **Total weight** of the manipulator will critically drive the design of transporters. Present experience make us think that a transporter weights roughly 50 times the manipulator's weight. Any improvement at end-effector level is valuable.

2. **Size** of the machine will condition its capabilities to access and to manoeuver inside the environment. Size parameter is different from range of motion for it includes all the volumes of the Manipulator. We shall consider size in minimum configuration (folded to pass through entry ports for example) and in average operating configuration.

3. **Supply line** shall naturally be as simple and small as possible. Hydraulics may cause some troubles and generally electrics are preferred. On board electronics would dramatically reduce the ombilical problem that considerably influences the design of transporters.

4. **Resistance to environment aggressions.** Primarily radiations, than possible corrosive surroundings (under water) and nuisances following upon tool operations (dusts, slags, grits, molten metal projections). Combined with decontaminability problems this may lead to prefer watertight designs.

5. **Reliability,** particularly resistance to tool induced vibrations.

6. **Maintainability** including decontaminability and adaptation to work in a glove box or through protective suits.

7. **Operability** which characterizes the easiness to control the machine by an "average" operator. Various types of controls can be evaluated with this respect as examples :

.. **Push button** type at actuator level is certainly the less friendly but the simplest and commonly used on power Manipulators.

.. **Robot like** control involves programming phases that may ask for computer sciences aspects, moreover it is not very well suited to permanently manually operated sessions often required in dismantling.

.. CAT control type on Master-slave servomanipulators is the most open for it offers a continuous spectrum of modes from the purely manual to the fully autonomous.

8. Cost : industrial Robots are two to three times less expensive than custom developments for nuclear applications. However, one shall consider the operating costs particularly when the machine is hardly maintainable or operated far away its nominal environment conditions.

III.2 Manipulators comparisons

A systematic approach as presented for tooling would not be here very significative unless a large number of different Manipulator designs are considered. Additionally the quotations cannot be made independently of the tool selection and the general application.

Specific machines are attractive because they are generally simple and can be optimized for a restricted set of parameters. They cannot represent total solutions for a complex dismantling problem, but shall be envisaged to solve particular points of the scenario (cutting entry port in concrete...).

Power manipulators are light weight and compact however they are very slow, lacking in force control capabilities and generally in advanced control features. They can be envisaged to handle heavy and autonomous tooling without great efficiency in time. A large and complex plant dismantling can hardly be based only on this type of tool handler.

Industrial Robots are very powerfull in programmed mode of operations and unexpensive. But programmed mode is not the nominal usefull mode in a non trivial dismantling environment. Additionally payload to total mass ratios are very low and the design may have to be significantly modified to comply with environment constraints (replace encoders to whitstand radiations - add bootings to protect against chemicals and contaminants). When the conditions are not extreme Industrial Robots are however to be considered.

Servomanipulators are certainly the most appropriate machines in complex situations. When large variety of tubing, vessels and welded equipments are to be cut and pieces handled in a reasonable amount of time, one have to consider that a Remote handling machine is generally "one armed", "legless", "one eyed", and operated through a control station that will add to all theses degradations. Consequently enhancement of all the elements of this system and notably the Manipulator are mandatory. In addition to the time figure, which is a prime justification element in favor of the servomanipulator, the multi purpose functions have to be highlighted : this is an exact answer to the multi-task aspect of complex dismantling operations.

IV - CONCLUSION

The ideal cutting tool would be light, compact, fast, easy to operate, clean and wireless ! The rapid overview presented in the paper shows how far we are today from this (unreachable) objective. Large efforts are still to be made on tooling developments in order to broaden the present choice which is very limited.

Reducing weight and augmenting operating speed of mechanical tools is a first objective because of their very attractive cleanliness. Working constraints of new processes (thermal cracking - arc sawing) shall be reduced in order to ease their operability with a Remote handling Manipulator.

New directions are pointing out from present analysis regarding Manipulators. The need for a hybrid type of Machine combining Robot like capabilities and Master-slave controlability is clear. First character is beneficial to precise guided-tools operations, the second is mandatory to address the numerous manual operations involved in dismantling.
A more advanced line is to work on the reduction of **size** and **weight** of the overall machines (easiness to install, to move and cost reductions).
Small, lightweight, and sensor based controlled manipulators shall be envisaged. Naturally implications on the total system will be important because the transporter will have to become more "clever" in order to recover the lost range of motions.

At a more general level one have to consider the work efficiency at the work site. Dimensions of the initial waste package transported and processed in the conditionning zone has some direct cost implications on additional installations. The segmentation process could advantageously be essentially performed **at the worksite.** This leads to consider highly efficient toolings and Manipulators in combination with a multiple (and small) machines concept, parallelising the process.

The dismantling project will then look like an "anthill" or a flexible workshop.

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TMI-2 EXPERIENCE WITH REMOTE OPERATIONS TECHNOLOGY

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ABSTRACT

This paper describes the remote operation technology developed for use in the TMI-2 cleanup. Such technology and experiences are considered to have applicability for a nuclear plant decommissioning. Specifically, it addresses:

- Mobile, remotely controlled vehicles for surveillance, sampling, light-duty, surface decontamination, and dismantling work.
- Stationary remotely controlled surveillance equipment.
- Remotely controlled boring, cutting, dismantling, and manipulating tools for damaged reactor defueling and disassembly.

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INTRODUCTION

The TMI-2 cleanup program has been faced with the challenge of removing large quantities of radioactive waste and fuel debris while maintaining personnel exposures as low as is reasonably achievable (ALARA). Early recovery tasks assessed the potential of robots or tele-operated tools to perform surveillance and cleanup tasks in those

areas of the plant considered too hazardous for personnel entry (Reference 1). Once this strategy was determined feasible, the appropriate tools were designed and tested. The result of these efforts is that remotely operated tools, sufficiently rugged for nuclear plant application, are now available for use in hazardous environments.

The remotely operated equipment for TMI-2 includes stationary and mobile surveillance, surface decontamination, remote cutting, defueling, and dismantling equipment. This equipment was developed for specific TMI-2 cleanup needs and to minimize overall worker radiation exposure for the cleanup. While plant decommissioning priorities are expected to vary, the tooling discussed herein should be applicable to a variety of the decontamination and dismantling tasks involved in conducting nuclear plant decommissioning.

MOBILE REMOTELY CONTROLLED TOOLING

Within this category are the various remotely controlled vehicles, mobile work platforms, and assorted attachments developed for completing the needed tasks. Vehicles which have been used at TMI-2 in specific cleanup tasks are the ROVER (3 vehicles total, each with a 3-camera vision system), LOUIE I (tracked base), and LOUIE II (6-wheeled base). Vehicles which have not been used to perform actual work at TMI-2 but are included because they were developed with some level of EPRI TMI funding (and because they are considered potentially useful in decommissioning) are the MOOSE and WORKHORSE.

ROVER (Remote Reconnaissance Vehicle, RRV)

ROVER, developed by CMU with support from EPRI, GPU Nuclear, DOE, and the Commonwealth of Pennsylvania Ben Franklin Partnership, is clearly the most versatile of the above-mentioned remotely controlled tools (see Figure 1). ROVER has performed the following operations in the highly contaminated TMI-2 reactor building basement: a) video surveillance, b) radiation surveys, c) floor sludge sampling, d) concrete wall core drilling and sampling, e) wash-down of walls and equipment, f) wall hydro-scabbling, and g) floor sludge vacuuming and removal. Due to high levels of radiation in the basement, (nominally 25-35R/hour gamma and up to 1000R/hour), manned entry has been precluded, demonstrating the relative independence of the vehicle once deployed.

The RRV was designed as a basic work platform specifically for use in the TMI-2 basement (Reference 2). The six-wheeled, water and contamination resistant platform with video cameras is controlled and powered through a 300-foot (91-m) tether. The 400-pound (181-kg) vehicle can handle up to a 400-pound (181-kg) payload. The ROVER is operated from a control console with video monitors and included are controls for a unique on-board tether

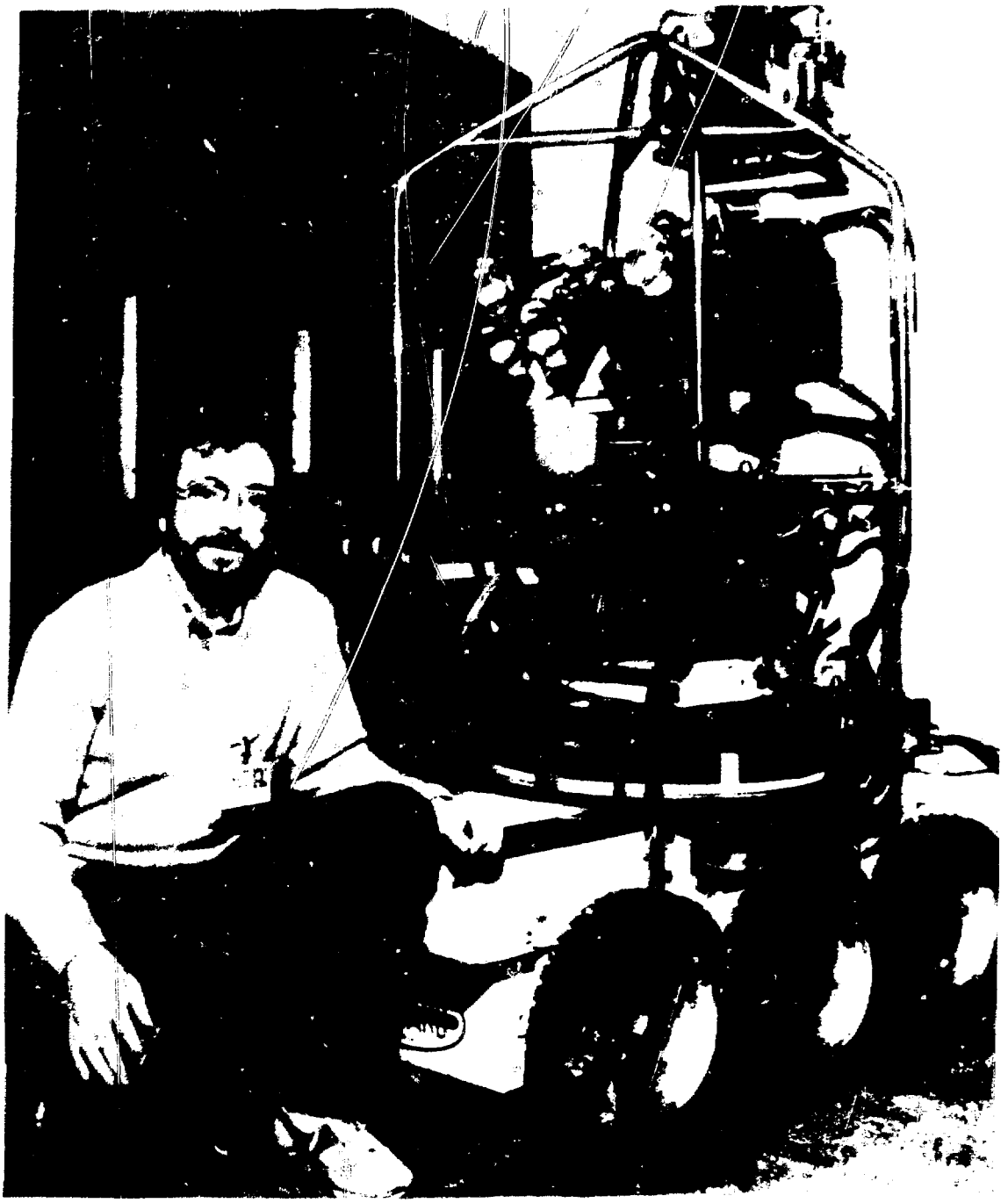

Figure 1. The original Remote Reconnaissance Vehicle, RRV or ROVER, with the basic control console shortly after delivery to TMI-2.

management system, controlling take-up and pay-out from the vehicle. The ROVER design was selected after carefully evaluating both commercially available and research vehicles. The basic evaluation criteria were: ease of implementation, reliability, and decontaminability.

In addition to the basic video surveillance capability, the ROVER has been outfitted with modules to perform radiation measurements, to obtain concrete coring samples from walls, and to obtain sludge samples from floors. Work modules have been added to perform high-pressure sprays and to vacuum sludge from floors. Additional details are below.

Remote Tooling Attachments (Reference 3). Development of tools for the RRV was initially based on the need to characterize the highly contaminated basement. Following the video and radiation surveys, sediment sampling was performed to determine the best approach for its removal with an eductor-type vacuum pump activated by air to effect transfer of the sediment from the scoop to the container.

The remote concrete core drill is an attachment developed to obtain samples of concrete from the basement wall to determine contaminant penetration. It could obtain up to four samples at heights up to about 8 ft. (2.4 m).

The remote flush device is an attachment mounted to the front of the base. The tool provided the ability to remotely direct high-pressure water through the spray nozzle at 25 gpm at a 7000 psi pressure drop.

The sediment collection attachment is a device to collect the sediment particles via a scooping action to where the material can then be transported. A positive displacement diaphram pump supplies the means to create the vacuum and transfer the sediment from the basement to the collection point on an upper elevation.

Operator Training Considerations. GPU Nuclear's extensive experience in robotic operations, especially with the ROVER has led to some interesting training principles. Briefly, these principles can be summed up in the following points:

- Know your machine
- Know your area
- Know yourself

The first item refers to the operator's need to be thoroughly knowledgeable of the machine and its capabilities. The second is for the operator to be thoroughly familiar with the area of operation, since the video cameras provide a limited perspective. The third item is particularly interesting since it is clear recognition of the need for operators to be aware of their own physical limitations. Operation of equipment via a video system requires intense concentration. Such focus begins to affect operator performance after about two hours. As a result,

operator tasks are exchanged periodically to provide some variation in their work activity with the robot.

Three ROVER units are now operating at TMI-2, one for training and equipment testing and two for cleanup applications. Through all operations, ROVER has been proven highly reliable.

LOUIE I (A Small Reconnaissance and Light Duty Vehicle)

LOUIE I, a small remote reconnaissance vehicle loaned to TMI-2 from Westinghouse, Hanford, has been used to perform video and radiation surveys in areas other than the basement (see Figure 2). It is much simpler than ROVER with a small, tracked base and a manipulator arm mounted to a telescoping column (Reference 4). Power and controls to the vehicle are transmitted through a tether that is dragged during movement. Its simplicity limits applications but does allow for rapid training of operators. While it has been used primarily to perform radiation surveys, it has also performed light-duty tasks such as moving bags of radioactive trash. LOUIE I's small size permitted its use in auxiliary building cubicles where ROVER was too large to even enter, let alone maneuver.

LOUIE II (A Small Remotely Controlled Scabbling Vehicle)

LOUIE II, designed by TMI-2 recovery personnel, is a 6-wheeled vehicle with a 3-piston scabbler head and a vacuum device for debris collection mounted to the front of the vehicle base (see Figure 3). This vehicle's small size also allows its use in the auxiliary building cubicles, for which high radiation levels (up to 100R/hour) precluded manual scabbling. For one cubicle, the seal injection valve room, up to 5 inches (13 cm) of grouted cement floor surface is planned for removal.

WORKHORSE (An Aggressive Decontamination and Dismantling Robot)

The WORKHORSE (also referred to as the Remote Work Vehicle or RWV) has not been used at TMI to date. It was designed and built by CMU with funding support from GPU Nuclear and EPRI. Its capabilities to dismantle concrete block walls and pipes were demonstrated during testing at CMU and during checkout and operator training in the non-contaminated TMI-2 turbine building.

The RWV is a tethered, mobile, tele-operated, electrohydraulic robot designed for wash-down, sampling, material packaging and transport, surfacing, and demolition work within a work envelope up to 23 feet (7 m) in height (Reference 5). In contrast, ROVER has taken coring samples as high as 8.5 feet (2.6 m) with its special coring attachment. Workhorse supports tooling that abrades, cuts, and transports for a variety of tasks that may be required to complete a mission. Reliability has been insured by essentially "mirroring" most subsystems, with each dual motor and control system supplied from its own power source. The control system even

Figure 2. The remote controlled transport vehicle (LOUIE I) used for light duty and reconnaissance.

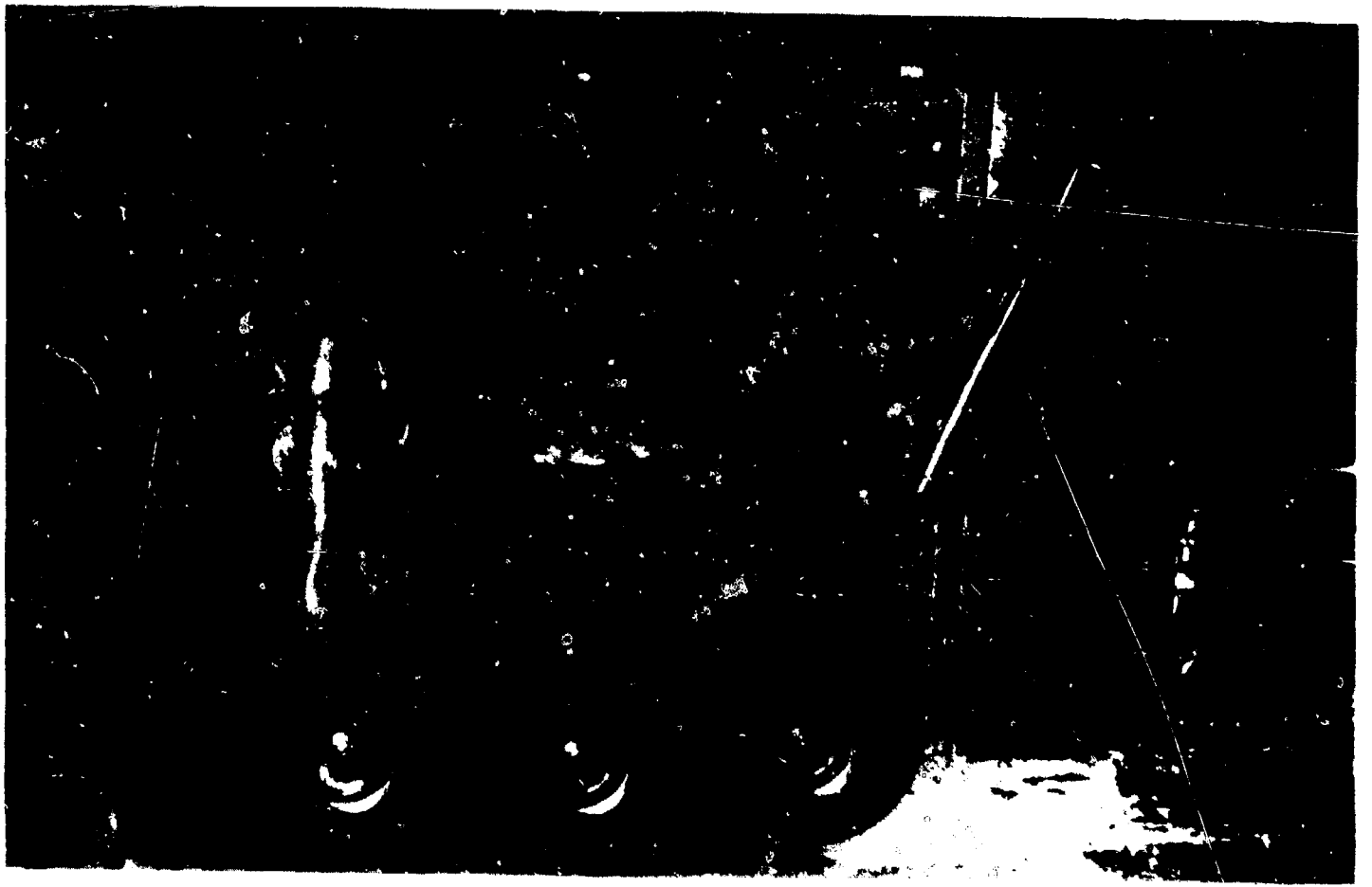

Figure 3. The remotely controlled scabbling unit (LOUIE II) with a 3-cylinder scabbling head.

includes teach-playback capability for executing repetitive tasks. In summary, the vehicle was designed for varied multiple capabilities, high reliability, decontaminability, maintainability, and reserve capability for add-on tooling. Such requirements were often competing and some trade-offs were made. However, this vehicle is remarkable in its overall capability, which is clearly beyond that of ROVER.

MOOSE (Remote Controlled Scabbler)

GPU Nuclear demonstrated the effectiveness of using manual scabblers to decontaminate concrete floors since contaminants normally were trapped within the paint or immediately underneath the paint layer. The MOOSE (see Figure 4), a remotely controlled scabbling unit, was developed with EPRI funding for TMI-2 use but was not used since recovery workers completed the required scabbling by manual means. The MOOSE has subsequently been demonstrated and is now in use at other nuclear facilities.

The vehicle is controlled by two operators at distances of up to 50 feet (15 m) from the work area (Reference 6). The vehicle can decontaminate up to 400 feet^2/hour (37 m^2/hour) at a surface removal depth of 1/16th inch (1.6 mm). The completed unit consists of a tethered chassis, a portable remote-control console, a seven-cylinder scabbler head, and a vacuum system to collect and store the scabbled debris.

STATIONARY REMOTE SURVEILLANCE EQUIPMENT

Remote Video Cameras

Remote video cameras were installed in the reactor building as one of the first steps in the cleanup. The system, initially designed and procured by EG&G, Idaho for DOE, consisted of standard commercial cameras with remotely operated pan, tilt, and zoom as well as the basic focus and light intensity controls. They were installed at strategic locations with overlapping views and the monitors were located in a containment work coordination or "command" center. The cameras are used to plan recovery tasks and for control and observation of the actual cleanup operations. The observation capabilities enhance task safety and help in problem resolution, while minimizing worker radiation exposure. Such video cameras or other temporarily installed cameras are also used to help control remotely controlled vehicles, as they provide a perspective which the vehicle's on-board cameras cannot. Since each reactor building camera can be viewed by another, an operable camera can be used to examine a malfunctioning camera's LED diagnostic lights and, thereby, help in camera repair and maintenance. Use of the cameras has been a significant factor in reducing the total worker radiation exposure for the cleanup.

Video cameras have also been used extensively for inspecting reactor vessel and other component internals, again feeding the signals back to the

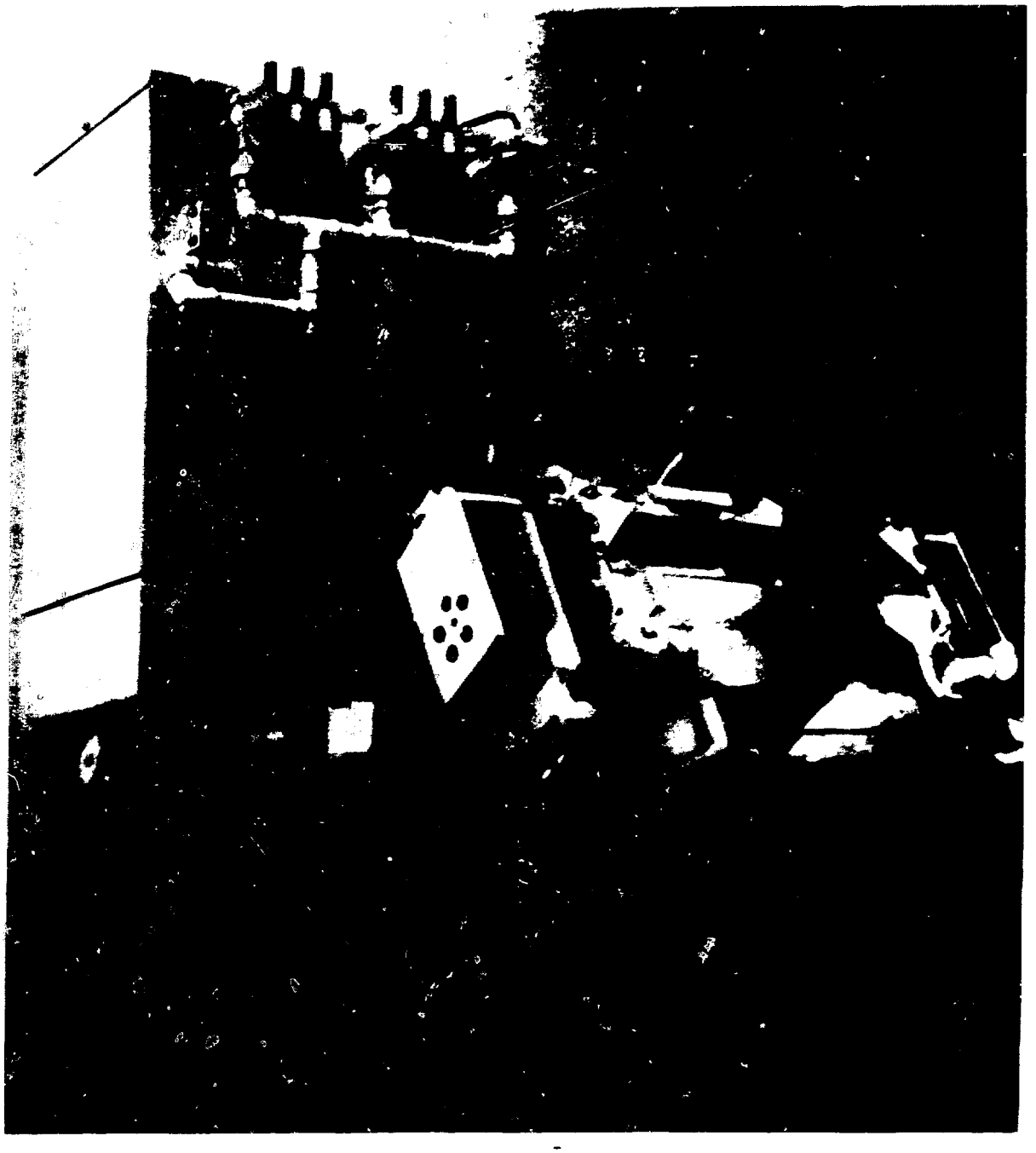

Figure 4. The remotely controlled scabbling unit (MOOSE) with a 7-cylinder scabbling head.

command center for recording. In using these cameras, waterproofing and auxiliary lighting have been the primary concerns. Camera manipulation is generally accomplished with simple pendant controls or lanyards, though special camera positioning tools have been used for some applications.

Core Topography System

A sonar-type device was constructed to obtain the first measurement of the extent of damage in the upper area of the reactor vessel (Reference 7). This remote controlled device, designed and built by EG&G, Idaho, was installed through a control rod drive tube and operated from the relay room. The information obtained on the size and shape of the damaged core void was essential for the ongoing defueling planning. The design and application of this core topography device minimized the worker exposure in obtaining the necessary core data.

The core topography system had a distance resolution of 8mm. This type device was originally designed for obtaining accurate dimensional and imaging information under conditions of poor liquid visibility, e.g., in nuclear facilities with liquid metal coolant.

REMOTELY CONTROLLED EQUIPMENT FOR DAMAGED CORE SAMPLING AND REMOVAL

Long-Handled Remote Tooling

Long-handled tooling to remove the fuel debris and damaged structurals from the core is the basic TMI-2 defueling approach. While a broad range of defueling tooling was designed and developed, workers have actually used a limited number of these tools for the defueling work (see Figure 5). The long-handled tools presently in use were designed and built by GPU Nuclear. While the specific tools may not be suitable for a decommissioning task, the following principles of their design and use may be: 1) such tooling is relatively low-cost, thus permitting a wide variety of tooling to be made available, 2) some simple standard tooling such as a vise grip can be adapted to a variety of useful configurations, 3) long handles with hydraulic-fluid connections can be fabricated for use with a variety of end effectors, and 4) video cameras should be used with these tools to observe their manipulation close up.

Examples of these end effectors are simple hooks, 3- and 4-point grippers, spades and scoops, heavy-duty tongs for gripping damaged fuel assemblies, and hydraulically operated shears (see Figure 6). The long-handled poles are segmented into 7-feet (2.1-m) and 22-feet (6.7-m) sections with the longest reach being 36 feet (11 m), which is beyond that nominal 15-25 feet (4.6-7.6 m) for most long-handled tooling used in the nuclear industry.

Reactor Core Drilling Machine

The major step to quantify the extent of damage to the TMI-2 core was to drill and remove samples from the once-molten region of the core. This was accomplished with a system designed and built by EG&G, Idaho using a

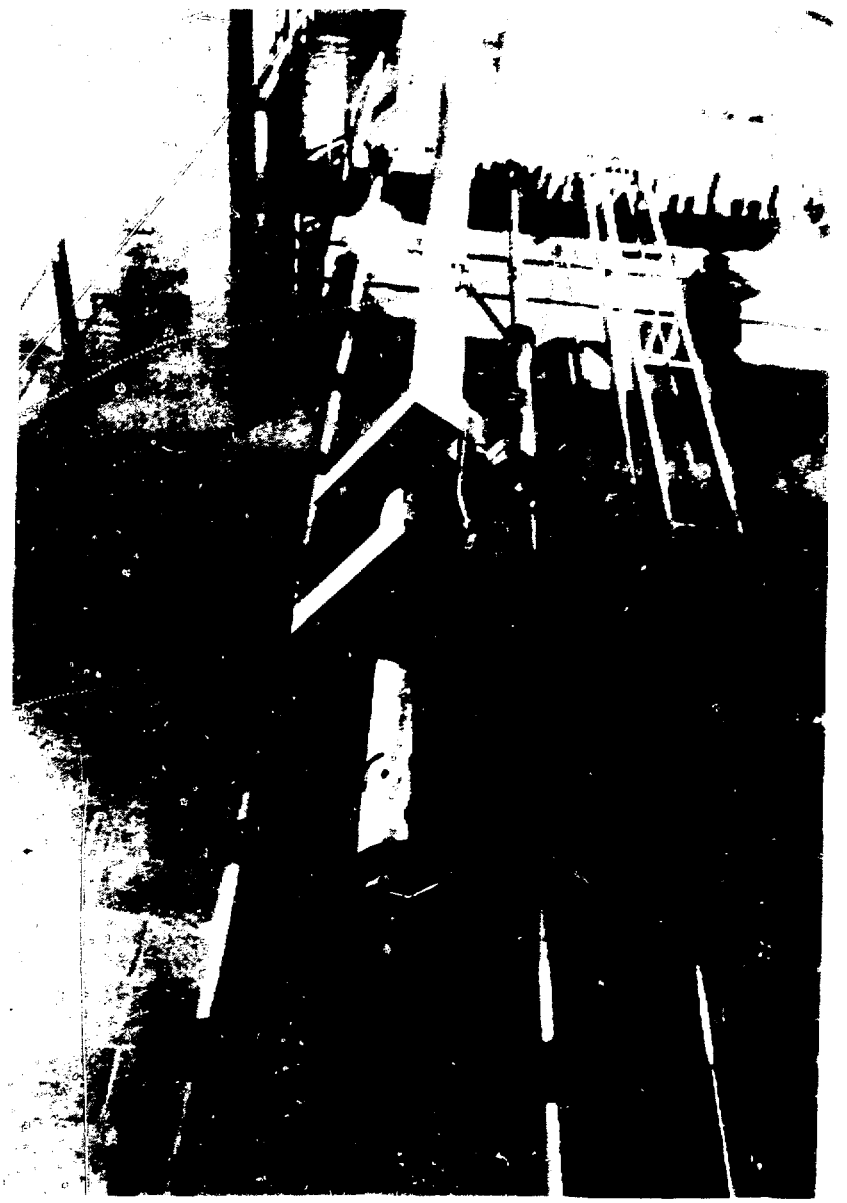

Figure 5. Various long-handled assemblies for attaching tools to defuel TMI-2.

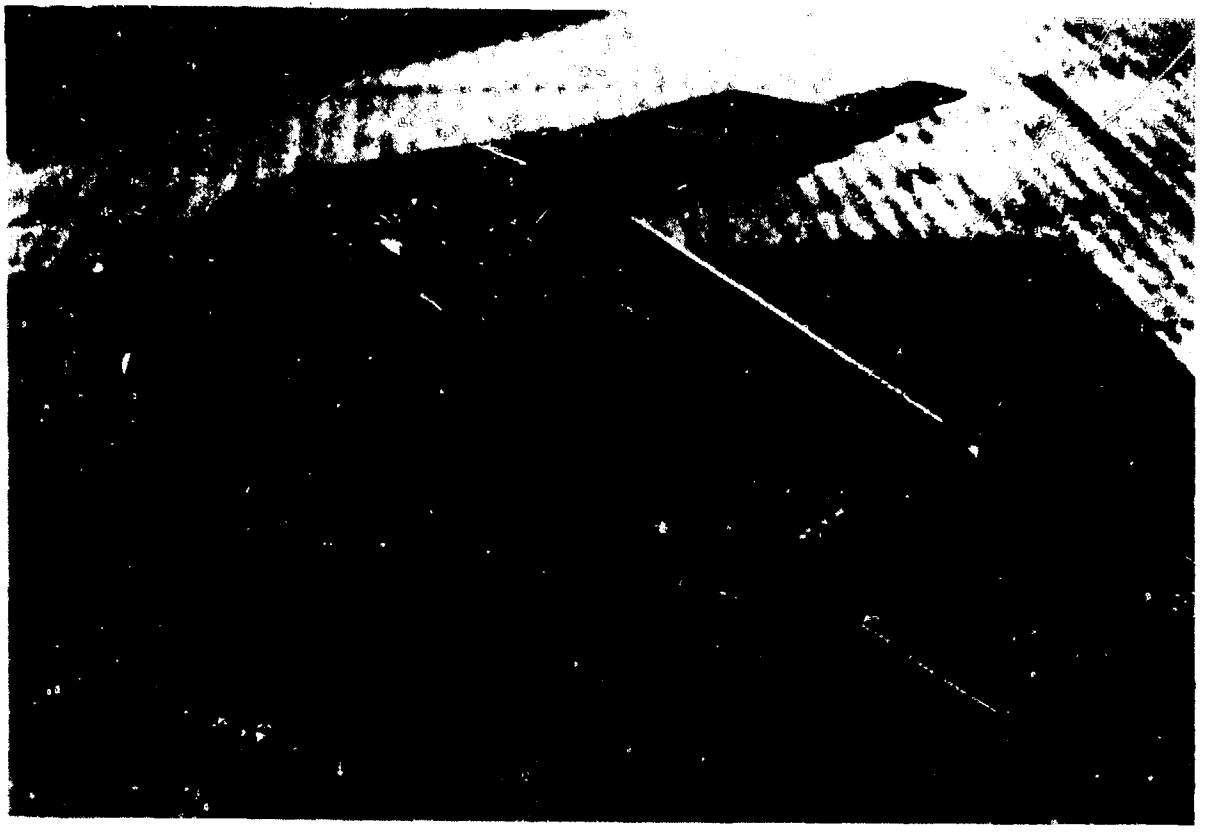

Figure 6. The spade bucket. One of the many end effectors placed on the end of long-handled tools for defueling TMI-2 debris.

combination of geologic drilling technology and precision measurement equipment to drill precise holes into the damaged core and extract coring samples of the core debris material (Reference 8). Actual drilling was controlled by the operator from a work platform above the damaged core. Later this system (see Figure 7) was used simply as a remotely operated tool to drill holes in the solidified/compacted region of the damaged core to facilitate material removal (Reference 9). Such a tool could be used for reactor vessel internals dismantling because of its precise positioning and programmable drilling capability. This was verified in a recent study which developed the procedures to dismantle the massive, stainless steel TMI-2 core support assembly as an alternative to plasma arc cutting.

ACES For Reactor Structural Cutting

GPU Nuclear has procured the ACES (Automated Cutting Equipment System), an industrial robot that can manipulate a plasma arc torch in five degrees of freedom for cutting stainless steel structures varying from 2.5-cm to 35-cm thick under 35.4 feet (10.8 m) of borated water. Once the fuel in the core region has been removed, ACES is being considered for use in dismantling the TMI-2 lower core support structure. This will permit full access to the fuel debris located on the bottom of the vessel.

The application of plasma cutting technology to the TMI-2 cleanup task goes beyond existing experience in both the severity of the cutting conditions (water depth and metal thickness) and the level of sophistication required for positioning. Feasibility tests for cutting under these conditions have been performed by EG&G, Idaho. This work will be applicable for both modifications to operating plants and to decommissioning (Reference 10).

MANFRED for General Manipulation and Tool Deployment

MANFRED (MANipulator For REmote Defueling), which was also procured by GPU Nuclear, is a double-armed remote manipulator being considered for use in conjunction with ACES to dismantle structures in the reactor vessel. A 40-inch (102-cm) arm would be used to stabilize the MANFRED, while a 72-inch (183-cm) arm would be used for actual dismantling. Position information from both ACES and MANFRED would be fed to a display system for use by the operators for collision avoidance between the two systems.

CONCLUSION

In conclusion, TMI-2's various mobile remotely controlled vehicles and the numerous attachments have clearly reduced worker radiation exposure while accomplishing decontamination and other recovery tasks. In addition, many of the cleanup tasks simply could not have been completed without use of such tooling. The remotely controlled viewing capabilities used at TMI-2 have resulted in further reduction of worker radiation exposure while improving task control. Finally, the defueling tooling for positioning,

Figure 7. The TMI-2 reactor coring machine, developed by DOE/EG&G, assembled for training in the TMI-2 turbine building.

drilling, cutting, and manipulation of end effectors on long-handled devices offers many options for dismantling the reactor vessel, its internals, or other massive steel components which have become radiation hazards. Extending the TMI-2 tooling concepts to reduce worker radiation exposure, improve decontamination techniques, reduce radioactive waste volumes, and accelerate dismantling, has great potential for significant savings in nuclear plant decommissioning.

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ENGINEERING DESIGN OF THE WINDSCALE ADVANCED GAS COOLED REACTOR DECOMMISSIONING SYSTEM

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ABSTRACT

The paper reviews the background to the Decommissioning System and the stages through which it was developed before arriving at the final concept. Included is the steel materials radioactive inventory which gives radioactivity and weight of steel materials to be size reduced and handled by the System. A detailed description of the System is given together with a typical operating sequence to provide an understanding of how the machine and its handling system work together. It describes how the problem of radiation shine where manipulator platform service lines pass through the floor shield was overcome by careful design. Key dates of important stages in the supply of the system are included.

INTRODUCTION

In the mid 1970's it had been realised that decommissioning of nuclear reactors would become increasingly important and because of its responsibilities for development the United Kingdom Atomic Energy Authority (UKAEA) decided it should initiate preparatory work. A Working Group was set up under the Chairmanship of Windscale Nuclear Laboratories (WNL) and studies started on the Windscale Advanced Gas Cooled Reactor (WAGR), see Fig 1, and continued until the reactor finally closed down in 1981. At this time studies had reached a stage at which the UKAEA board, with the approval of the Department of Energy and Electricity Boards, were able to give WAGR decommissioning project status and proceed with the decommissioning work.

ENGINEERING PLAN

During the course of the pre-project studies an Engineering Plan was developed for the complete decommissioning task which involved the removal of the Containment Building and all the nuclear and associated plant contained within thereby returning the site to green field status.

The plan identified an outline dismantling procedure for the reactor vessel and its internal equipment. Radioactive inventory calculations established that radiation levels existing in the vicinity of the reactor

vessel top dome, due to the effectiveness of the neutron shield, would be low enough for hands on techniques to be used for dismantling down to the hot box. At this stage the remaining dismantling work would have to be carried out using remotely operated equipment. After further development a procedure for working downwards for dismantling both reactor internals and reactor vessel was established and was made possible by transferring the reactor vessel support from the corbel region to the bottom of the vessel after removal of the internals, see Fig 2. From this procedure emerged a firm requirement for the remotely operated machine that was capable of dismantling/size reducing the equipment within the reactor vault working always in a downward direction. The machine would need a handling system to assist with the dismantling work and to transport the waste materials produced to a Waste Packaging Facility for encapsulation into concrete waste containers. These containers when filled would be placed into an intermediate store to await final disposal by Nuclear Industry Radioactive Waste Executive (NIREX).

The waste container emerging from the Plan had a size of 2.34m x 2.21m x 2.18m high and weighed some 50te maximum. This container complied with British Nuclear Fuels (BNF) rolling stock and the rail loading gauge of the Cumbria railway line. The container internal dimensions restrict the size of reactor vessel plate to 1.5m x 0.6m which is also the size of a thermal shield assembly. The latter weighs some 1.25te and this load is the maximum that would be transported from the reactor vault to the Waste Packaging Facility.

The study included an assessment of the radioactive inventory and specific activity of all the steel materials of the plant and equipment to be decommissioned. The radioactive inventory related to the more active plant components to be dismantled by the machine are shown in Table 1. The results of this study provided information for the design of the machine shielding.

TABLE I - PRINCIPAL ACTIVE STEEL ARISINGS, WAGR

Description	Mass tonnes	Material	Activity Ci/te at 7 yr decay (Bq/te in brackets
Pressure Vessel	210	MS	2.3 (8.5×10^{10})
Thermal shield	186	MS	34 (12.6×10^{11})
Neutron shield	13	SS	430 (15.9×10^{12})
Core support plate	18	MS	220 (8.1×10^{12})
Support bearings	2	SS	330 (12.2×10^{12})
Core restraint	17	Mixed	1500 (55×10^{12})
Loop tubes	4	SS	470 (17.4×10^{12})

After reaching the project stage the detail of the Plan was developed, particularly the dismantling sequence of the reactor vessel and its internals and at what stage the machine would need to be introduced. This sequence helped to determine the working requirements of the machine and its tooling needs and enabled WNL to prepare a Functional Specification for the machine and its associated handling system.

DEVELOPMENT OF THE SYSTEM CONCEPT

On receipt of the Customer Functional Specification design work started on the development of the concept that met all the requirements of the specification the residual radiation levels and the constraints on size and weight of assemblies imposed by the Containment Building, available lifting facilities. The UKAEA placed a contract with Strachan and Henshaw, because of their experience in the design of Electricity Board refuelling machines and active fuel handling equipment to assist them with the design of the system. The concept, see Fig 3 that emerged from this study met all the requirements of the specification and was developed through further stages of simplification, see Fig 4. This was achieved by reducing the machine flexibility and making use of existing facilities such as the 25 te Pile Cap crane for raising and lowering the mast assembly. This value engineering exercise was undertaken with the realisation that the machine has to operate only for a short period of time.

At this stage WNL decided that the manipulator, tooling and handling grabs be excluded from the system specification and that they would be responsible for the design and procurement of this equipment.

DECOMMISSIONING SYSTEM CONCEPT

The Decommissioning system Requirements.

The primary operating requirements for the machine as defined in the engineering specification are to provide; safe shielding containment thereby permitting the ventilation system to maintain a negative pressure in the reactor vault, the life expectancy required in the radiation environment and operate within the reactor vault and waste handling route areas by remote control with the aid of CCTV viewing systems. Additionally, dismantling of the reactor must generate waste of a specified size for optimum packaging in the waste container, and transport it through a pre-defined route to the Waste Packaging Building.

The Outline System Description - See Fig 4

The system consists of two interfacing machines which cover the reactor vault which is 9.2m diameter and extends 21m below the pile cap floor and the transfer routes of 14m into the adjoining sentencing cell and maintenance cell. The space for these cells was created by raising the heat exchangers within, some 12.5m from their installed position with the aid of a hydraulic jacking system. These two machines consist of a dismantling machine mounted at pile cap level and extending downwards to work in the reactor vault area to perform the dismantling activities, and a waste handling system located within the vault area and working either in conjunction with, or independent of, the dismantling machine.

This type of arrangement offers a considerable amount of flexibility of operation and limits the dismantling machine payload requirement to the dead weight of the manipulator and tooling, combined with the dynamic loads of the dismantling operations. The handling system supports the waste during dismantling/cutting and then transports it in one operation, without a change of support, to the storage location within the sentencing cell thus considerably reducing the risk of dropping the load. Pick and place operations can also be performed independently by the handling system with the dismantling machine manipulator performing close-up camera support work as necessary.

Safety Related Design Objectives.

The system must be capable of safely dismantling the reactor vessel and its contents within the beta/gamma radiation, graphite and asbestos particulate environment. This environment must be both shielded and contained, to permit access at pile cap level into specified maintenance areas of the machine.

The structural integrity for the machine is to be such that failure of a member is of low probability thus guaranteeing the system a high degree of integrity.

In the event of failure of any one sub-system there are sufficient alternative sub-systems to enable recovery of equipment from the vault area into the shielded maintenance area to permit maintenance activities and rectify the fault. It is not intended to continue operations in a failure case by providing redundant systems but to recover and repair.

Interlocking to safeguard against collision of the machines and/or equipment will as necessary utilise two diverse electrical channels, additionally, as necessary, some motions are mechanically locked in position to safeguard against spurious electrical operations. The need for these diverse methods is covered by a safety report which defines possible hazards and their safeguards.

Site Installation and Construction.

There are two principle site requirements which have significantly influenced the machine design. Firstly, the need to build the decommissioning module at the test facility approximately 12 months in advance of the complete system. Secondly, the physical restrictions imposed by entry into the containment building and the access and craneage capacity within the building.

Firstly, it is necessary to design the decommissioning module, comprising, the mast, mast support structure, manipulator platform and all the service supplies, as a separate assembly which can be built at the test facility, used for operator training and dismantling technique development. On completion of this activity it is transferred as one assembly to the rotating floor shield for final installation.

Secondly, the physical constraints of the WAGR building requires that all equipment enters the building via the goods air lock at ground floor level and the hoist well up to pile cap level utilising the pile cap crane impose a maximum size on assemblies of 3m square and weight of 25te.

DECOMMISSIONING SYSTEM DESCRIPTION

The Decommissioning Module - See fig 5

This main assembly comprises the mast, manipulator platform, module support structure and service feeds to the manipulator platform. This assembly is used to position the manipulator at specific work heights within the reactor vault and is mounted on the rotating floor shield with the mast at a fixed radius from the reactor centre line.

The mast is constructed as a rigid fabricated box section structure which can be progressively extended and lowered into the reactor vault by addition of appropriate sections as work proceeds. The plan area of the mast is filled at specific locations with steel blocks to interface with the floor shield and provide complete shielding. Support for the mast is by a retractable pin and lifting/lowering activities are carried out by the pile cap crane.

The manipulator platform which is guided on the mast by wheel assemblies is the mounting feature for the manipulator, tooling, CC TV systems and the termination point for all the service line requirements for the equipment. It is raised/lowered by a twin wire lift winch which permits easy retrieval of the manipulator and associated systems for maintenance and tool changing.

During dismantling operations the platform is lowered to the bottom of the mast where it sits on a mechanical dead stop. The twin wire lift system allows for redundancy of one drive wire without dropping the platform. The fall back recovery method for the equipment within the reactor in the event of both wires failing is to lift out the mast complete with platform and all associated equipment.

The lifting winch is provided with dual speed for fast retrieval/deployment combined with fine positioning, overspeed sensing, emergency braking and load monitoring to register platform loading conditions.

The Manipulator - see Fig 6

The manipulator is being ordered by WNL who are technically responsible and it will be supplied to Strachan & Henshaw for installation on the platform and interface testing. In addition to pick and place operations that would normally be required of a power manipulator in the nuclear industry, the requirements involve dynamic cutting and tooling operations. The manipulator was selected on technical and cost reasons and is being supplied by Taylor Hitec. It is a power manipulator based on a hybrid of Advanced and Warrior types developed for CEGB, carrying out "in Reactor" maintenance and repair work. The manipulator is electrically powered with a payload capacity of 35kg with the arm fully extended to 2.795m. The arm has six degrees of freedom and is expected to achieve a tool tip repeatability of some 0.5mm. The arm has sufficient reach, dexterity and payload to position the various tool packages to any point within the reactor vessel. It is required to manoeuvre the cutting tool including progression at predetermined speed, maintaining the tool at correct attitude to work face, maintaining distance from work face, defining the geometry of the item being cut and progression in linear and non-linear motion in any plane. In order to achieve these requirements the attitude and position of tool tip needs to be known at all times, either in the form of position co-ordinate or known axis positions. This is achieved by the use of a microprocessor control system. The manipulator is provided with secondary manually operated drives to enable it to be folded into its "park" position in the event of loss of power supplies. In the "park" position it lies below and within the periphery of the manipulator platform enabling it to be withdrawn into the maintenance area of the machine located above the rotating floor shield.

Tooling.

Requirements for the dismantling work are currently under review at WNL and development work is expected to continue through to verification of suitability during the dismantling development work in the Machine Test

Facility, See Fig 8. To date some tools have been identified for use with the manipulator gripper, they are:-

Oxy-Propane Gas Cutting Torch using powder injection. This tool package incorporates cutting gas supplies together with spark ignition, an ultrasonic ranging device and a flame failure device. This cutter is used for all flame cutting operations though not always using the power injection facility. (Fe/Al)

Insulation cutter for cutting the insulation on the external surface of the reactor vessel.

Nailing gun which is under consideration for fastening lifting features to graphite reflector bricks.

Disc cutter for miscellaneous cutting including pipes.

Cropper for cutting of small bore BCD pipework.

Drill for drilling holes in material.

All tooling needs adapting for handling by the manipulator gripper jaws and their reaction forces must lie within the manipulator payload of 35kg.

Platform Mounted Equipment

Other equipment on the platform includes:-

A BISTEM unit which carries a small TV camera mounted on a pan and tilt unit at its extremity. This device is capable of extending the camera position by up to 2.5 metres in order to view awkward areas such as the core restraint assembly. The camera is also used as part of the orthoganal viewing system.

A powder hopper to dispense the Fe/Al powder to the oxy-propane cutting tool. This also includes nitrogen gas circuitry and control valves.

A location feature to support the manipulator arm in its park/ recovery position.

Two high pressure sodium vapour lights to aid viewing of the oxy-propane cutting tool at work.

Omni directional microphone.

Gamma measuring instrument.

Temperature measuring instrument.

Service feed system.

To supply the manipulator, TV systems, lighting, cutting tools and other equipment at the platform a comprehensive system of service feed line is required. This requires approximately 300 electrical cores and 6 low pressure gas lines running from the containment area above the floor shield down the length of the mast to the platform. These lines are fed from a bank of 14 reeling drums which are capable of paying out sufficient cable to accommodate all platform working levels within the reactor. Some of these drums possess slip ring type take off but the manipulator control lines require non slip ring take off and some development is required to obtain a suitable solution.

The whole of this decommissioning module is mounted on the rotating floor shield with the mast locating at a fixed radius from the reactor centre line.

The Rotating Floor Shield.

This performs two primary functions, firstly it replaces the existing pile cap reactor shielding which is removed prior to machine installation providing safe shielding for the pile cap area.

To achieve this and reduce the beta/gamma radiation to specified levels it is designed as a 20mm steel shell iron shot filled with concrete. This concrete infill has a density of 5370Kg/m^3 and is nominally 360mm thick. The shield covers the entire reactor vault area and weighs 120te.

The floor also provides the main structural support/positioning method for the decommissioning module containment structure and needs to be capable of slewing the mast module about the reactor centre line thus enabling the manipulator to cover the whole plan area of the vault.

The area above the shield floor and within the containment structure is designated a maintenance area and is for servicing the platform mounted equipment.

A system of interlocked sliding shield doors, see Fig 7, provide access for the platform and manipulator between the vault and maintenance areas. This either totally shields off the reactor or opens to permit entry of the mast and platform. Then by clamping up around the mast and interfacing with its inner shield provides full continuous shielding.

The doors open a stage further to permit entry of the platform, associated equipment and service feed lines into the reactor vault. After entry is complete they return and clamp the mast whilst permitting entry and movement of the service lines supplying the equipment operating on the platform.

The whole of this equipment above the rotating floor shield, the decommissioning module and the maintenance areas are totally enclosed in a containment structure which also forms the first line of containment for the dismantling operations within the reactor vault. A negative pressure within this space of -250 pa is maintained by the ventilation system to prevent outflow from the containment. Entry into the maintenance areas is via an access air lock and high efficiency air filters are provided for air inlet flow and safeguards against the consequences of back pressurisation should it occur.

The complete structure comprising the rotating floor shield, decommissioning module and containment structure possesses an all up weight of approximately 200te and as stated earlier is capable of being rotated/slewed around the reactor centre line to permit positioning of the manipulator, cutting tools and TV systems at the work face. This is achieved by mounting the entire structure on a 9m diameter heavy duty slewing ring, driving via a peripheral ring gear. Drives to this gear are mounted at pile cap level with free access for maintenance. There are two diametrically opposite drives each capable of 3 speeds of operation giving a maximum floor slow speed of 0.5rpm and a minimum of 0.005rpm. This minimum speed is used to move the gas cutting tool during horizontal cutting of the reactor vessel wall and equates to a linear tool tip speed of nominally 100mm/min.

The Waste Handling System

The function of the handling system is waste transfer and to accomplish this the system must be capable of covering the whole plan area of the reactor vault and the sentencing cell.

A slewing beam is mounted below, independent from, the rotating floor shield and permits the 3te transfer hoist to plumb any areas within the reactor vault independent of the decommissioning module.

The transfer hoist is used to support the waste during the final dismantling/cutting activity, lift it out of the reactor and with the beam aligned with the exit route, transfer the waste out of the vault. Position control of the hoist and beam is via resolvers from a gear rack. This permits off line programming of the 253 reactor channel locations

thus permitting semi-automatic pick and place operations to be performed when unloading the large number of graphite core blocks.

The transfer hoist design is based on a commercially available electric runway hoist with considerable modification to incorporate the safety features necessary for this type of work, together with cable feed services to the variety of grabs that are required. Service feeds to the hoist are a significant design problem with a need to supply 2 x 72 core cables from the above rotating floor shield and allow for the independent slew motions of the reactor slew beam and further slew motions of the sentencing cell slew beam. A test rig has been built to simulate the cable feed system and to determine loading conditions that will be encountered on the cable and hoist during operation. It will also give some assurance of the life expectancy of the cabling and prove the ability to re-thread a replacement cable in event of damage.

On transfer of waste into the sentencing cell the transfer hoist is positioned on a centrally mounted slewing beam which enables the operator to position the waste at any location within the cell. On entering the cell the waste is assayed by presenting it to various monitoring/assay devices after which it is segregated into selected racks/baskets. These racks/baskets are located on the carousel floor and rotated to position them immediately in front of the operator viewing window. The combination of transfer hoist traverse, beam slewing and carousel floor rotation provides operational coverage of all the floor area within the cell and a significant amount of contingency fall-back condition in event of a failure of a motion.

The mounting of the reactor slew beam is similar to that of the rotating floor shield in that it is supported from a large slewing bearing approximately 8.0m diameter. The centre support is a universal pivot and is also the entry point through the floor shield for the cable feeds to the transfer hoist. This type of support and a 2/3 span beam greatly increases the flexibility of operation. The sentencing cell slew beam is a different type of construction being supported at the centre of a single column by a set of bearings and having the drive through this centre feature. Maintenance of these bearings can be undertaken from above in the hoist room which is fully shielded from the sentencing cell.

The carousel floor is a bolted construction supported on 8 wheel boxes and having a centre pivot bearing. The top is clad in stainless steel to permit ease of decontamination and all drive features are mounted outside the cell in shielded accessible areas.

After successful assaying and segregation of the waste into the racks/baskets they are loaded into the waste container for final

packaging. Eack rack has a maximum weight of 7te when fully loaded and is handled by the waste hoist situated in the hoist room above. When required the hoist room shield door is opened to permit access by the 4 fall 8te SWL hoist into the sentencing cell where it handles and lifts a rack. The carousel floor is then rotated and locked into location which aligns the exit route and the waste can be lowered into the waste container. This operation is repeated for a second load.

Grabs and Handling Tools

The manipulator gripper is used for handling and retrieval of small and lightweight objects and assisting the positioning of grabs.

Grabs deployed by the transfer hoist are:-

Graphite channel brick grab. An internal-expanding grab based on the ball-and-wedge principle.

Thermal shield plate grabs. Two designs are required: a shackle type to lift the top course of narrow plates by their fitted eyebolts, and a hook attachment to engage the 4 lifting trunnions of the full-sized plates. The captive shackle pin is inserted by the manipulator. The shackle grab is also used to lift reflector graphite bricks by engaging a feature attached by the nail gun.

Claw grab. A general purpose grab for lifting miscellaneous waste with irregular geometry eg BCD pipework, certain sections of the hot-box.

A magnetic grab for small ferrous components and for segregating ferrous material eg pieces of rebar from concrete rubble.

With the exception of the magnetic grab all are self-acting, they do not need sustaining power during operation - only for opening and release.

Viewing System

Remote control of dismantling is carried out with the aid of CCTV cameras, TV monitors and associated lighting the controls for which are incorporated in the main control panel. Cameras and lights are positioned to cover three operational situations:-

Overview. To provide a frame of reference for the manipulator and transfer hoist relative to a defined datum on the reactor. Cameras and lights mounted on pan and tilt units at 2 positions 90° apart underneath the rotating shield and moving with it.

Tool and grab guidance. Two cameras with pan and tilt units arranged orthogonally and mounted below the manipulator platform. An "anti-flare" camera mounted on the manipulator arm specifically for flame-cutting operations.

"Round the corner" viewing. One camera with light mounted on BISTEM (an extending boom) with 2.5m reach mounted underneath the mast platform. The camera is mounted on a pan and tilt unit. A demountable fibrescope or endoscope is fitted to this camera to permit operations in the thermal shield/RPV annulus eg. fish-plate removal. This camera would also be used to supplement views from the tool and grab guidance system.

Each camera system provides TV pictures with a resolution better than 500 TV lines per picture height limiting (TVL/PHL).

All cameras are monochrome, monoscopic with remote iris and focus control, and contained in dust-proof easily decontaminated housings. The overview cameras are fitted with remote-control zoom lenses.

Cameras for tool grab guidance and round the corner viewing are radiation resistant to 10^6 rad.

Service leads for all components of the viewing system except those associated with the overview cameras are connected at terminals on the mast platform.

Audio System

Power tools generate characteristic sounds which can be interpreted to indicate normal or abnormal operation. Development work has shown that this is also the case for a cutting torch. A Uni-directional microphone is attached to the manipulator and an omni-directional microphone to the mast platform. Services are fed via connectors on the platform to an amplifier and loadspeakers in the control room.

Control & Instrumentation

Control of the whole decommissioning process using the machine is via three interlocked control stations:-

Main control station housed in the existing reactor control room.

Sentencing cell operations controlled from directly outside this cell in the waste packaging building.

The upper loading cell operations controlled from outside this cell in the waste packaging building.

The machine is controlled primarily by man-in-loop methods with the aid of CCTV viewing systems. Sequencing of operations to ensure interlocking of specific motions to limit the risk of collision or other hazard is provided by programmable logic controllers (PLC). Direct position sensing of principle drives is via resolvers.

The machine functions are in sequenced, step by step fashion avoiding combined movements wherever possible and so greatly reducing control complexity and collision risk. Any interlock overrides are key switch operated.

In the event of CCTV camera failure, it is possible to return the machine to a known safe (parked) attitude under "blind" conditions using interlocks only.

The control system monitors all measured variables and uses this information together with machine self diagnostics to regularly verify safe operating conditions. These variables are also to be used to present the operator with a simple graphics display of machine position and geometry during operations.

DECOMMISSIONING OPERATIONS

The initial phase of reactor dismantling is expected to be undertaken by hands-on methods, this includes dismantling of the pile cap secondary floor shield, the reactor stand pipes above the pressure vessel, the pressure vessel hemispherical dome and its contents down to the top of the hot box. This is achieved because radiation levels as measured in these areas are low due to the neutron shield. At this stage the decommissioning machine is installed to continue dismantling operations by remote means.

There are three primary modes of operation of the decommissioning machine, these are:-

a. Operations of the transfer hoist and grab typical for transfer of waste into the sentencing cell. These are performed by the handling system while the dismantling machine performs auxiliary manipulation.

b. Cutting operations with combined use of the manipulator and handling system.

c. Loading of waste into the appropriate waste rack and transferring the filled rack into the waste disposal container.

The following is a typical operating sequence for hot box top plate section removal with combined use of manipulator and transfer hoist.

The manipulator is fitted with the oxy-propane powder cutting equipment in the maintenance area and powder flow, gas pressures, freedom of hoses, cables, etc. checked. The transfer hoist is fitted with a suitable grab and functionally tested in the maintenance cell.

The transfer hoist is moved on to the reactor slewing beam and the platform and manipulator are lowered to the bottom of the mast. The mast is rotated by the floor shield to the manipulator operating position. The reactor slewing beam, combined with transfer hoist traverse and lowering, will position the grab above the plate section to be cut. The grab is attached to the nominal centre of gravity of lift. The manipulator will assist if required by parking the cutting tool and using the gripper. Manual teaching of the manipulator may be used to profile the outline of the required plate size around the supporting grab and hoist cables. The tool co-ordinate position readout is used to measure the size of the waste being cut.

If desired; the repeat mode is used to prove the manipulator trajectory and size of waste; the torch is then ignited and the manipulator repeat program induced. The operator monitors powder flow, gas pressure, flame size, etc and observes through CCTV viewing system whilst the manipulator performs movements automatically, at the optimum speed, to minimise fume and aerosol generation.

The manipulator is then retracted and the transfer hoist lifts the plate to the fully raised position. The reactor slewing beam then moves to align with the exit beam leading to the sentencing cell.

Control is then taken by the operator at the sentencing cell, and the sentencing cell slewing beam aligned. The transfer hoist and waste is traversed into the sentencing cell and the transfer hoist moved on the slewing beam to align with the monitoring station. After rotation of the carousel floor to align the exit route, the plate is lowered into the upper loading cell for monitoring. Control is then taken by the operator at the upper loading cell, the plate monitored and control returned to the operator at the sentencing cell. The plate is raised, the carousel floor rotated to align the rack local to the operator position and the plate deposited in a suitable waste rack.

PROGRAMME

A contract was placed with Strachan & Henshaw in October 1986 for the design, manufacture and installation of the decommissioning machine.

Key dates and activities are shown in Table II.

TABLE II - KEY DATES

Let Contract for Machine	October 1986
Let Contract for Manipulator	February 1987
Deliver Manipulator	August 1988
Deliver Decommissioning Module and install in Test Facility	March-July 1990
Deliver Rotating Floor Shield and install	September 1990-April 1991
Install Module on Floor Shield	May-June 1991
Commission and Active Operation	June-August 1991

Fig. 1 WINDSCALE ADVANCED GAS COOLED REACTOR

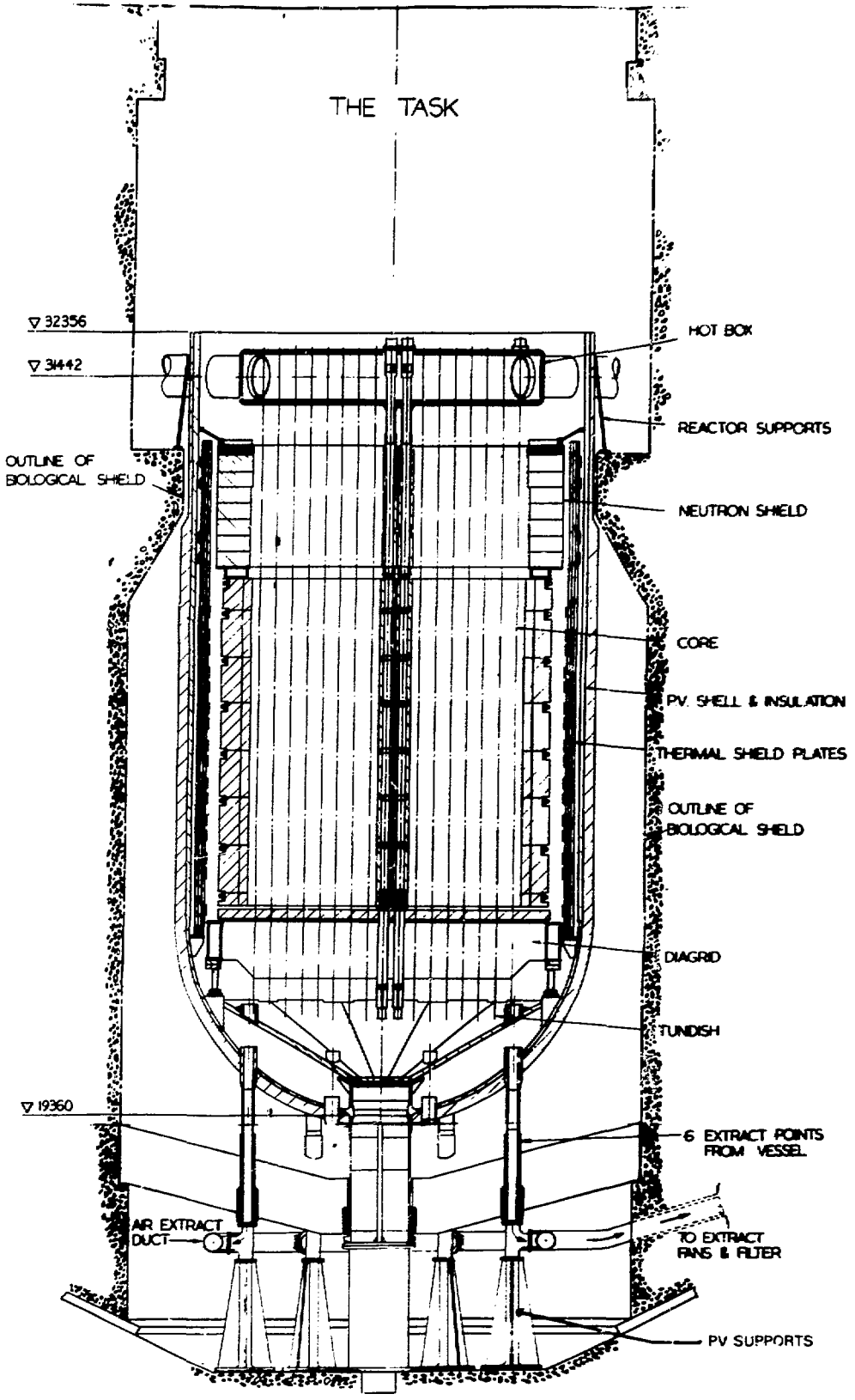

Fig. 2 WAGR REACTOR DISMANTLING TASK

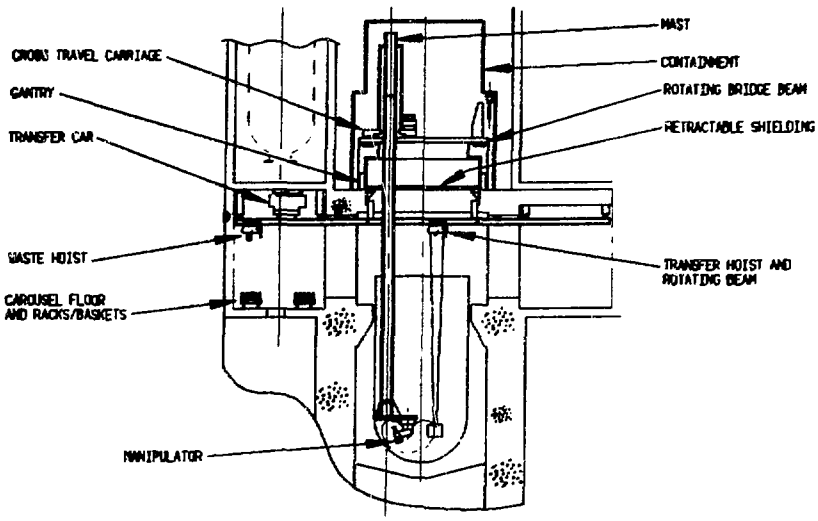

Fig. 3 ORIGINAL CONCEPT

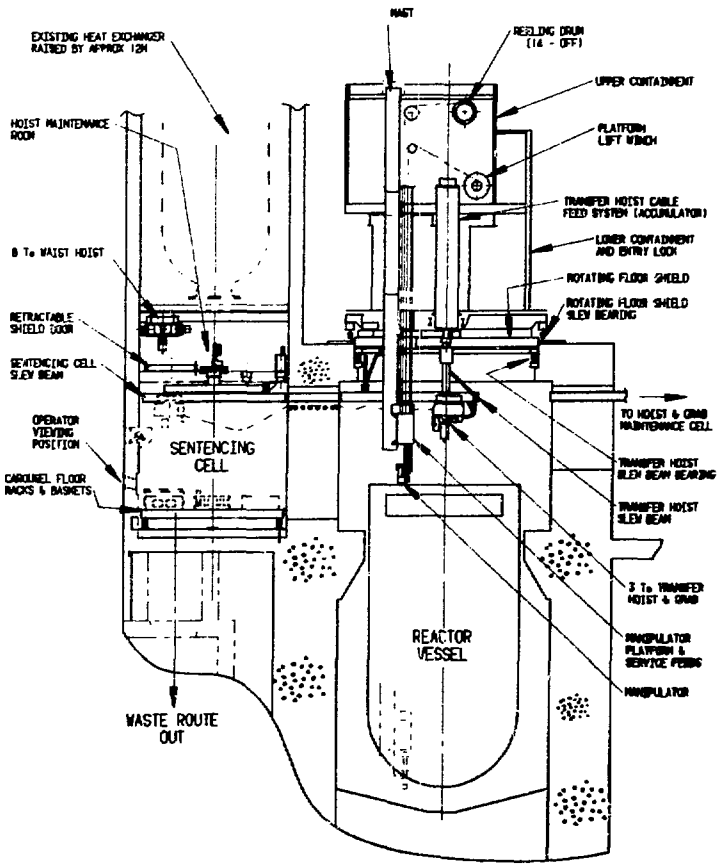

Fig. 4 THE DECOMMISSIONING SYSTEM

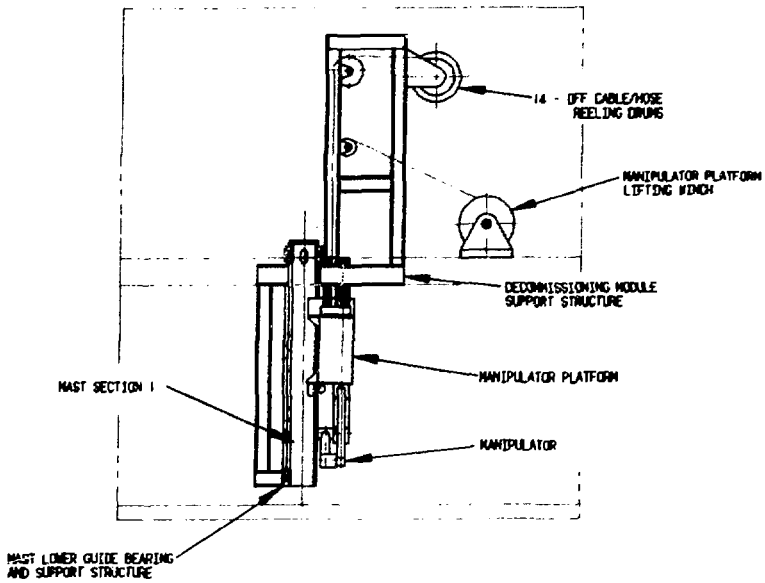

Fig. 5 DECOMMISSIONING MODULE

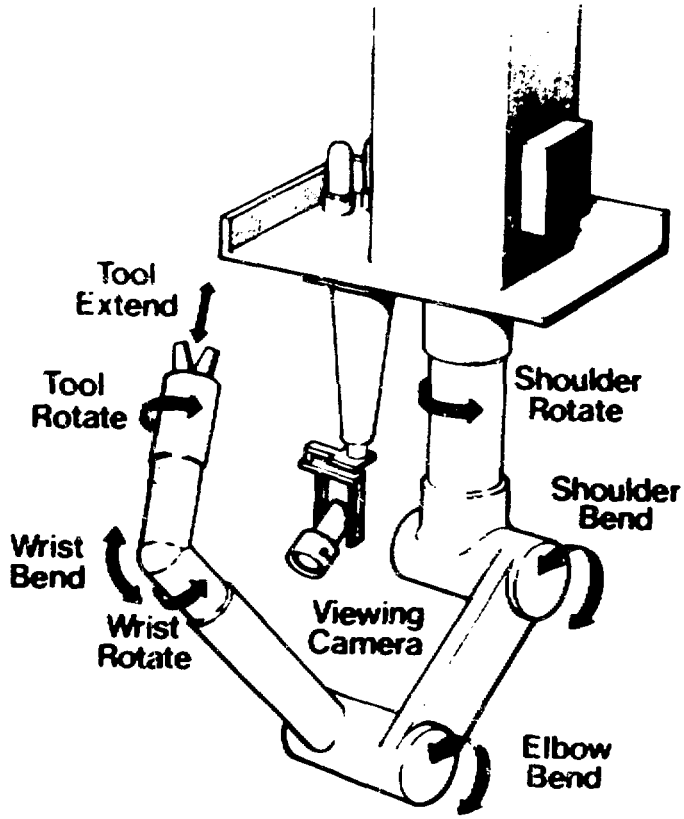

Fig. 6 MANIPULATOR

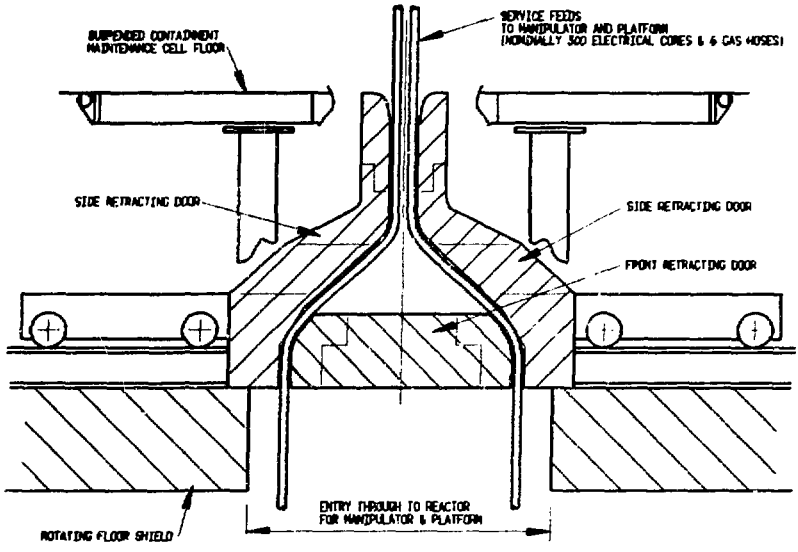

Fig. 7 RETRACTABLE SHIELD DOORS

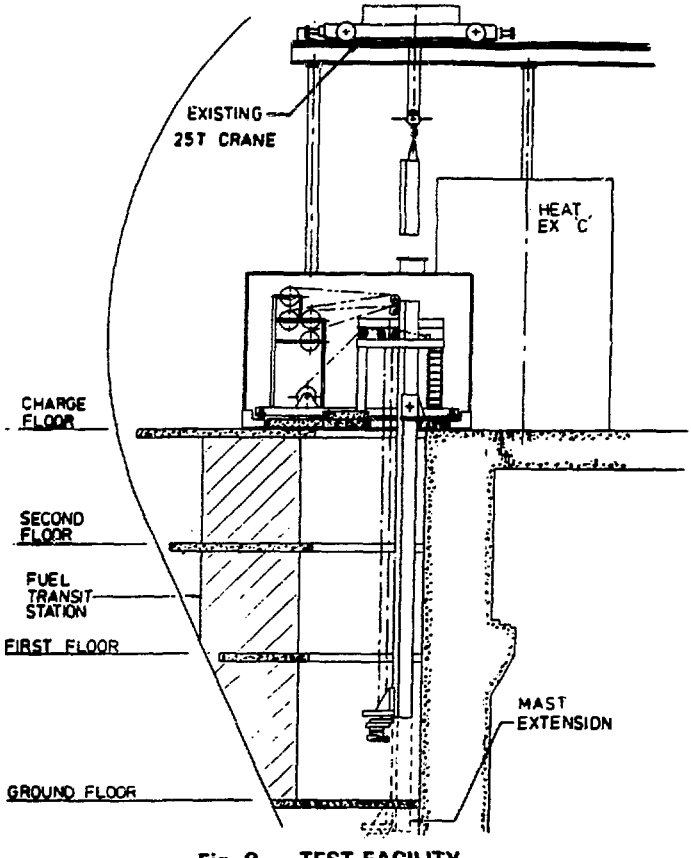

Fig. 8 TEST FACILITY

DEVELOPMENT OF CUTTING TOOLS FOR JPDR CORE INTERNALS AND PRESSURE VESSEL DISMANTLEMENT

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ABSTRACT

The development of remote cutting tools is one of the most important reactor dismantlement topics. Since the Japan Power Demonstration Reactor (JPDR) decommissioning program started in 1981, the Japan Atomic Energy Research Institute (JAERI) has developed an underwater plasma arc cutting system for dismantling core internals and an underwater arc saw cutting system for cutting the pressure vessel of JPDR. The final check-out tests for these systems* proved they have satisfactory performance for use on JPDR dismantlement.

INTRODUCTION

After it has been in service, reactor pressure vessel and its core internals are highly activated and contaminated. Thus, in dismantling them, it is necessary to apply remote cutting techniques to reduce radiation exposure to workers. JAERI reviewed existing cutting techniques for steel structures, and decided to develop an underwater plasma arc cutting system for core internals and an underwater arc saw cutting system for the reactor pressure vessel.

These were selected for good cutting performance, and more importantly, reduced worker exposure and ease of by-product treatment. The latter advantages are because operations are performed underwater.

First, many parametric tests concerned with cutting performance were performed to understand the basic characteristics of underwater plasma arc cutting and underwater arc saw cutting. Based on the results of these parametric tests, the plasma arc cutting system and the arc saw cutting system were designed and fabricated. Mock-up tests for both systems were performed, using JPDR simulated core internals and pressure vessel, prior to use in JPDR.

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* This work was performed by the Japan Atomic Energy Research Institute under contract from the Science and Technology Agency of Japan.

The objectives of the mock-up tests were:

(a) To confirm the operability of the underwater plasma arc cutting system and the underwater arc saw cutting system, and to modify them, if necessary.

(b) To determine the workable cutting conditions for the JPDR core internals and pressure vessel.

(c) To estimate management data such as manpower and worker exposure for the actual dismantlement of JPDR.

This paper describes the cutting systems and the results of the mock-up tests.

DISMANTLING PROCEDURE OF JPDR REACTOR COMPONENTS

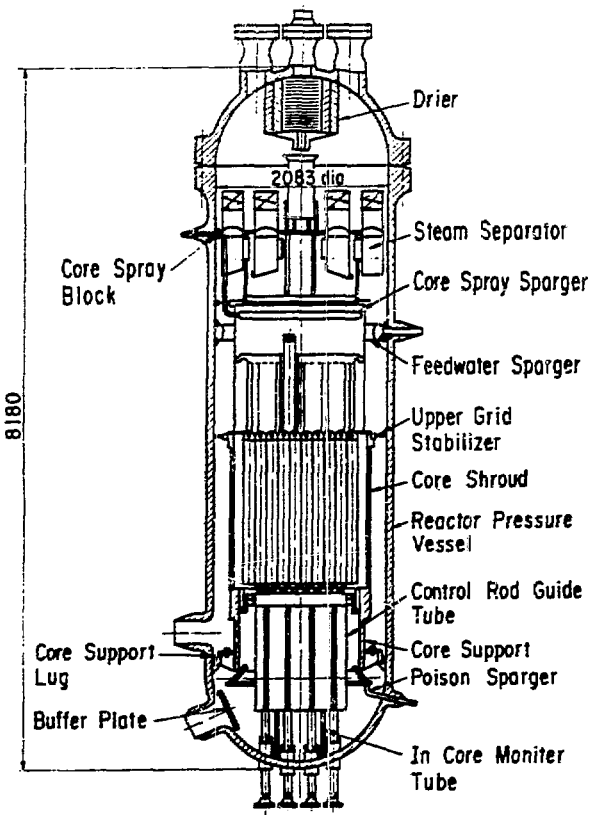

Fig. 1 Cross Section of Reactor Vessel and Component

The JPDR is a BWR demonstration reactor (90MWt), the first reactor to generate electricity in Japan. It was operated from 1963 to 1976 to acquire nuclear power reactor operational data.

A cross section of the JPDR reactor and its components is shown in Fig. 1. The core internals, piping connected to the pressure vessel and the pressure vessel are to be dismantled using the following techniques.

(1) Removable core internals (stainless steel, max. 50mm thickness)
- Cutting by plasma arc in spent fuel storage pool or on the operating floor. (Plasma arc cutting in air is not described here.)

(2) Welded core internals (stainless steel, max. 105mm thickness)
- Primary rough cutting in the pressure vessel and secondary cutting, sizing to fit containers, in the spent fuel storage pool with the underwater plasma arc.

(3) Piping connected to the pressure vessel (carbon steel, stainless steel, 50 to 320mm in diameter)
- Cutting by inside disk cutter, shaped explosive and gas cutting.

(4) Reactor pressure vessel (low alloy steel with stainless steel cladding, 73mm and 250mm thickness)
- Cutting by underwater arc saw.

The underwater plasma arc cutting system and the underwater arc saw cutting system are described in the following sections.

UNDERWATER PLASMA ARC CUTTING SYSTEM

Parametric tests confirmed that the underwater plasma arc cutting technique was useful to remotely cut off core internals in the pressure vessel. The basic specifications of designed underwater plasma arc cutting system are shown in Table I . In addition, the test provided the characteristics of by-products such as dross, gas and particles in air and water.

Table I Basic Specifications of Underwater Plasma Arc Cutting System

Tested Parameters	Range of Parameters
Arc Current	150 to 100 A
Supply Gas	$Ar + H_2$, $Ar + N_2$, Ar + He
	30 to 120 liters/min.
Torch Moving Rate	0 to 1000 mm/min. (Vertical)
	0 to 1000 mm/min. (Horizontal)
	0 to 1000 mm/min. (Rotational)
Material to be Cut	Carbon steel, Stainless steel
	3 to 130 mm
Atmosphere	Water

Mock-up Test Facility

The underwater plasma arc cutting system is shown in Fig. 2. The system consists of power supplies, a plasma arc torch and associated moving mechanism, control panels and other components.

The power supplies are a pilot arc power supply and a main arc power supply. The main arc power supply can supply electricity up to 1000Adc following the ignition by the pilot arc power supply.

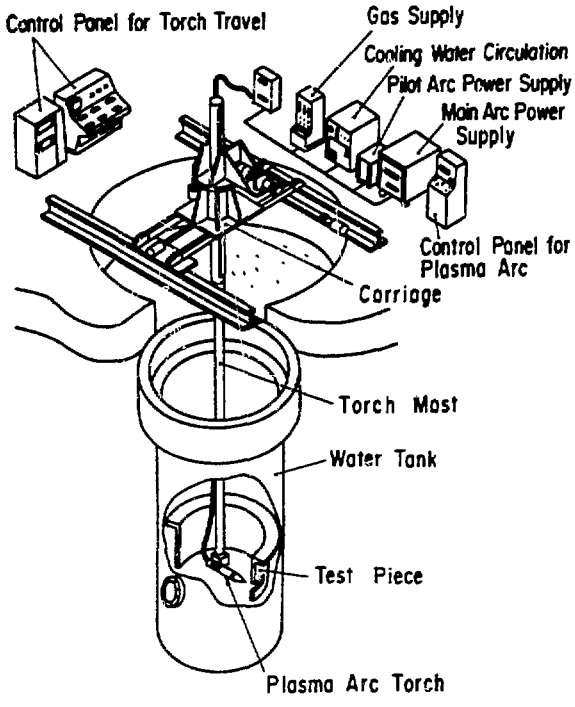

Fig. 2 Underwater Plasma Arc Cutting System

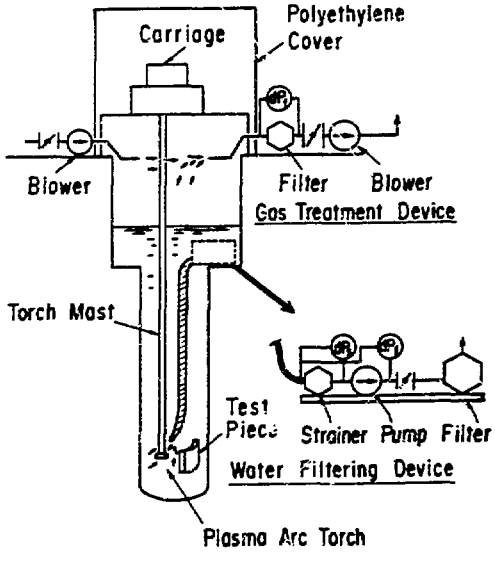

Fig. 3 By-product Collection System for Underwater Plasma Arc Cutting

The torch mast is vertically hung from a carriage which travels back and forth or right and left on the operating floor. The plasma arc torch is attached to the end of the torch mast. The mast, and thus the torch, can move up and down and revolve during cutting. The torch is supplied with electric power, supply gas and cooling water for the torch nozzle through a cable assembly. Plasma arc cutting is accomplished with an arc between the torch and the material to be cut, and by supplying ionized high temperature gas to the material. The 3 to 2 ratio mixture of argon and hydrogen is supplied at a flow rate of 50 liters per minute.

The underwater plasma arc cutting system is operated from the control panel on the operating floor. The motion of the torch is controlled in two modes, automatically by a micro computer and manually by the operator's observation of torch travel through an underwater camera. Special tools are provided to handle dismantled pieces.

The by-product collection system, shown in Fig. 3, consists of a gas treatment device and a water filtering device. The gas treatment device has two blowers and a filter, making an air curtain above the cavity to prevent the diffusion of by-products into the air. The water filtering device has a strainer, a pump and a filter, extracting by-products in the water through a flexible hose.

Test Results and Discussions

Before completion of the cutting tests, the accuracy of torch positioning was evaluated by measuring the deviation between the desired position and the real one. Then, torch speed, arc current, gas flow rate and torch path were sent to the micro computer. The torch was controlled, according to the designated conditions, keeping a constant stand-off (i.e. the distance between the torch nozzle and a test piece surface). The by-product collection system was operated during the cutting tests. Dismantled core shroud segments were hoisted by a special handling tool and were transferred to an adjacent spent fuel storage pool for secondary cutting. Long pieces, like pipes, were lifted by a pipe gripper. Small pieces, like blocks, were allowed to fall into a bucket hung from the operating floor.

The conditions and results of the cutting tests are shown in Table II. Cutting tests used materials and configurations selected to be typical of JPDR. In addition, the cutting tests were all performed under JPDR conditions, which included consideration of shape, thickness and surroundings. The results generally indicated that as the thickness increased, gas flow had to be increased and torch speed decreased. An arc current of 500A was enough to cut all test pieces.

Table II Conditions and Results of Underwater Plasma Arc Cutting Tests

Test Piece	Shape	Thickness of Cut (mm)	Arc Current (A)	Torch Speed (mm/m)	Gas Flow (ℓ/m)	Cutting Results
Core Sprary Block	Block	105	500	100	50	Successful*
Core Support Lug	Block	50,63	500	100,150	50	Successful
ICM Tube	Pipe	3.2	500	150	50	Successful
CR Guide Tube	Pipe	8.1	250	100	50	Successful
Poison Sparger	Pipe	5	500	100	50	Successful
Feedwater Sparger	Pipe	19(max)	250,500	100 ~ 300	50	Successful
Upper Grid Stabilizer	Rod	76 dia.	500	75	50	Successful
Core Support	Plate	55	500	100 ~ 150	50,80	Successful
Buffer Plate	Plate	25	500	150	50	Successful
Core Shroud	Plate	12	500	300	50	Successful

* The cutting was completed by arcing from each side of the block.

The torch stand-off was evaluated, 10 to 17mm was found to be best. Nozzle damage by dross attack resulted from shorter stand-offs, and the longer stand-offs made the arc unstable. The accuracy of torch positioning was within 4mm, therefore it is possible to maintain an arc without sustaining nozzle damage. Nevertheless, the nozzle and the electrode in the torch were gradually worn away, so that they had to be replaced. It is desirable to provide a spare torch for quick replacement of both nozzle and electrode.

As a result of the cutting tests, it was observed that the cutting performance was dependent on the expulsion of dross from the cutting vicinity. Because of this, arc current, torch speed and gas flow had to be chosen appropriately. Otherwise, a mass of dross was deposited between kerf surfaces. This bonded the pieces being separated. It occurred frequently in cutting thick or complicated structures. In this case, cutting was performed again to exclude the dross. When the dross could not be excluded, the test piece was forcedly pulled up with a handling device or down with a stick. It was recognized that visual observation of the cutting point was very important to confirm complete cutting. Therefore, effective water purification was also needed to maintain good visibility in the water. Confirmation of the torch stand-off distance by a more sensitive camera is also required.

The by-product collection system worked well in general. By-products generated during cutting were dross, particles in the water, gas and particles in the air. Since more than 98% of generated dross and particles in the water were vacuumed and collected by the strainer and the filter, the visibility of water remained almost the same as before cutting. In addition to supply gas, hydrogen (about 13 ℓ/min.) was generated by thermal decomposition or electrolysis of water. Particles in the air were less than 0.2 mg/ℓ. These by-products in the air were drawn into a suction blower. Because the hydrogen concentration in the cavity was less than 0.1%, the capacity of gas treatment device (140m^3/min.) proved to be satisfactory. All cuttings were completed under the conditions shown in Table . The cutting tests also examined the functions and operation of ancillary devices.

UNDERWATER ARC SAW CUTTING SYSTEM

The parametric tests confirmed that the underwater arc saw cutting technique was useful to cut the pressure vessel wall into rectangular shaped blocks. The system design reflected the results of the tests. The basic specifications of underwater arc saw cutting system are shown in Table III. The tests also provided the characteristics of by-products such as dross and particles in the water.

Table III Basic Specifications of Underwater Arc Saw Cutting System

Tested Parameters	Range of Parameters
Arc Voltage	10 to 50 Vdc
Arc Current	20 to 40 kA
Blade	914, 1000, 1100mm in diameter 12mm in thickness carbon steel 400rpm
Saw Head Moving Rate	0.5 to 10mm/s (horizontal) 0.5 to 20mm/s (vertical) 0.5 to 20mm/s (rotational)
Driving Mode	Constant Velocity Constant Current Constant Voltage
Material to be Cut	Stainless Steel, Carbon Steel 80 to 250mm
Atmosphere	Water

Mock-up Test Facility

The underwater arc saw cutting system is shown in Fig. 4. The system consists of a power supply device, a saw head and associated servomechanism, control consoles and other components.

The electric power for cutting is supplied from the power supply device, which transforms the commercial 3300Vac to the required 10 to 50Vdc through a transformer and a thyristor.

The main mast penetrates the center of a mast lift table. The saw head and associated servomechanism are attached to the end of the main mast which can move up and down and can rotate to allow the saw head to approach a cutting position. To suppress vibrations during cutting, the main mast is supported by a mast snubber which stretches its arms against the cavity wall. The numerically controlled, hydraulically powered mechanism controls the motion of the saw head during cutting as shown in Fig. 5. Therefore, the rotating saw can cut an object vertically or horizontally according to a designated cutting pattern. When changing the pattern, the saw blade is re-positioned using a blade angle change mechanism.

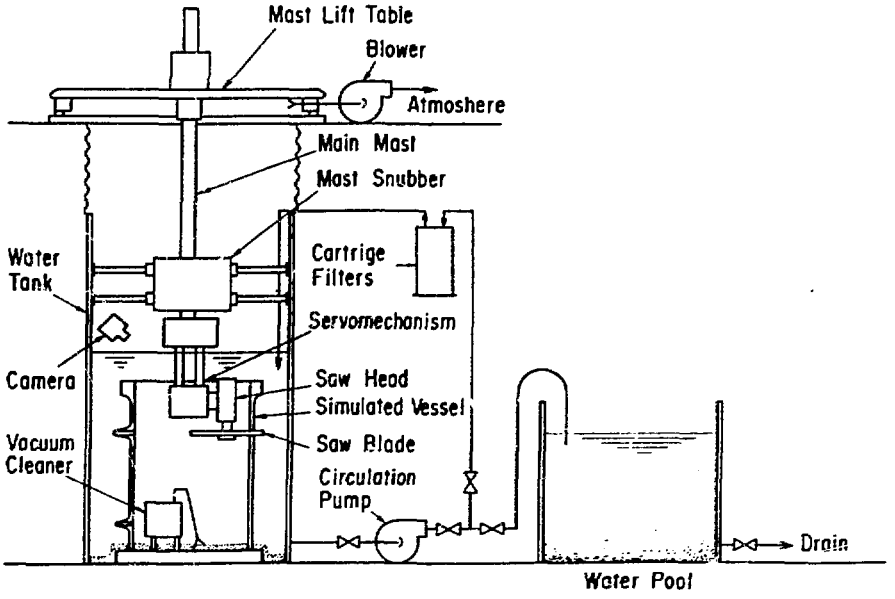

Fig. 4 Underwater Arc Saw Cutting System

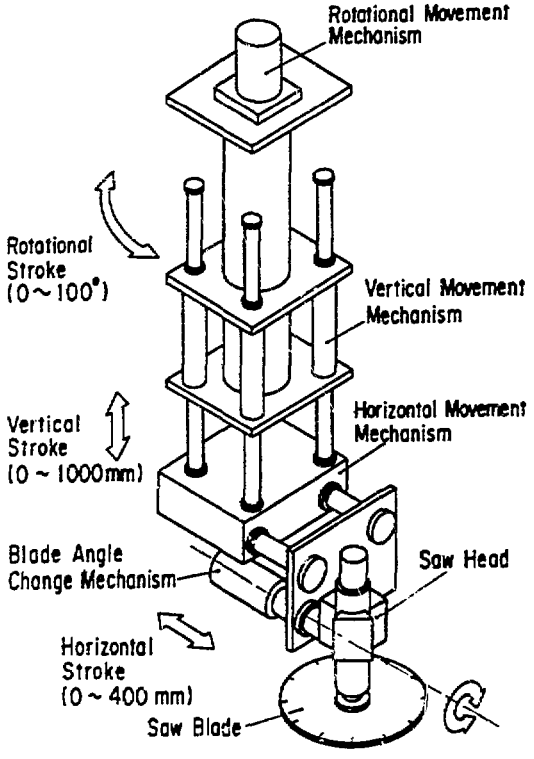

Fig. 5 Hydraulic Servomechanism

The saw blade is a plane disk with slits and is bolted to the saw driving mechanism head. The saw head is connected to the negative side of the power supply through a water cooled cable. Arc saw cutting is accomplished by arcing between the rotating blade and metal being cut, melting the metal by the arc, and excluding the dross produced with the blade. Therefore, the saw blade is gradually worn away, so the blade must be replaced at periodically. The blade change is performed remotely using a blade change device.

The underwater arc saw cutting system is operated from control consoles located outside the radioactive area. The saw head is driven in three modes; constant speed mode, constant voltage mode and constant current mode. The saw head vicinity is monitored by color TV camera mounted on the mast snubber.

The by-product collection system, shown in Fig. 4, consists of a blower, a dross vacuum cleaner, a circulation pump and cartridge filters, and a water pool temporarily prepared to flocculate particles with a coagulating agent.

Test Results and Discussions

The procedure is as follows for dismantling the simulated pressure vessel immersed in a water tank as shown in Fig. 6.

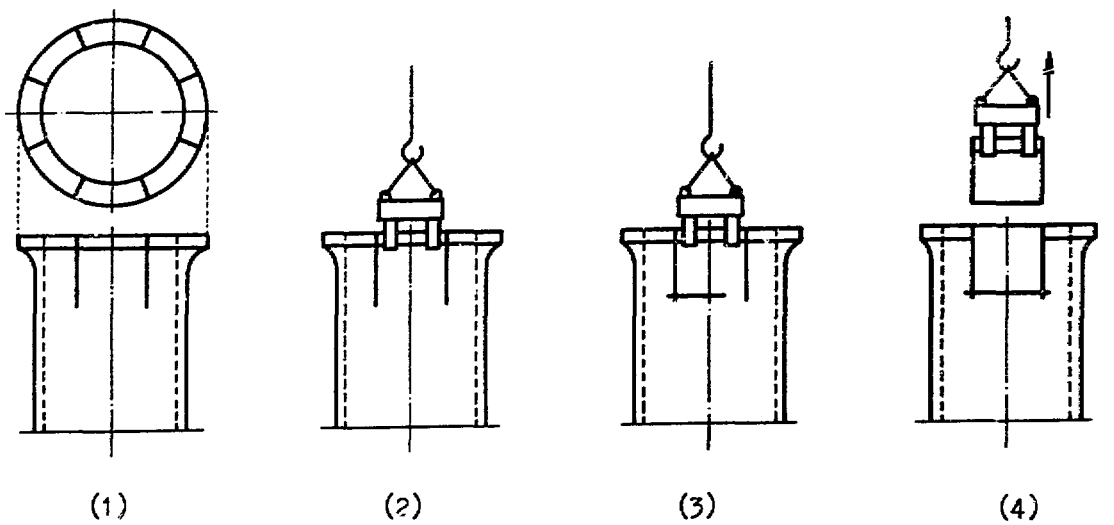

Fig. 6 The Procedure of Dismantling a Pressure Vessel

(1) First, vertical cuts are made from the top. The cuts are about 800mm in length and are spaced about 900mm apart.

(2) Next, a gripping device, hung from a ceiling crane, is attached to the piece to be cut off.

(3) The piece to be removed is then cut horizontally.

(4) When cut free, the piece (800mm x 900mm) is lifted from the vessel and put it into a container on the operating floor.

(5) Steps (1) through (4) are repeated.

Based on this procedure, various cutting tests required for an actual dismantlement were conducted. The operability of other devices was also examined. However, the following cuts were essential:

(a) Horizontal cutting of vessel shell (80mm in thick)

(b) Vertical cutting of vessel shell (80mm in thick)

(c) Vertical cutting of flange region (250mm in thick)

The conditions and results of the cutting tests are shown in Table IV.

Table IV Conditions and Results of Underwater Arc Saw Cutting Tests

Cutting Direction	horizontal	vertical			
Cutting Thickness	80mm	80mm		250mm	
Saw Blade	original	original	insulated	original	insulated
Cutting Velocity(mm/s)	3 ~ 5	0.5 ~ 5	3	0.5 ~ 3	0.5
Set Voltage (V)	50	50	50	50	50
Set Current (kA)	20	20 ~ 40	20	20 ~ 40	20
Blade Rotation Rate(rpm)	300 ~ 460	150 ~ 570	330 ~ 460	270 ~ 420	230 ~ 400
Results (completed/started)	31/33	11/32	7/9	0/11	4/7

Horizontal cuts of 80mm thickness were almost always completed by the original saw blades. But vertical cuts, especially of 250mm thickness, were not completed by the original saw blades. This was not a problem of blade wear, but was usually due to the tripping of the power supply caused by current overload. When a power supply tripped, the saw blade was retracted from

the cut so as not to stick to the vessel. After resetting the power supply, the cutting was tried again with different cutting parameters. But, the problem repeated itself. The current overloads, which probably came from short circuit, continued to trip the power supply. To avoid the trips, a number of causes for the short circuit were presumed, i.e. instrument error, lack of capacity of the power supply, unusual vibrations of saw blade, unsupported main mast, deviation of the saw head and so on. The causes of the short circuits were examined continuously in many ways.

A necessary characteristic of the underwater arc saw cutting system for JPDR is to cut the vessel wall obliquely twenty three degrees from a radial or right angle cut, as is shown in Fig. 7. This obliquity caused the saw blade to cut the vessel wall with a thicker section than if cut at right angles. This produced a disproportionate flow of dross around the blade. This effect might make the melting metal difficult to exclude from the kerf. To understand the influence of the oblique cutting, three pieces of vessel shell were arranged at right angles to the saw blade, as shown in Fig. 7. The right angled cuts were performed with original blade. The results were as expected, the three right angle cuts were completed. Unfortunately, it was not possible to modify the system to make right angle cuts in the pressure vessel. The JPDR pressure vessel is too small to be cut with right angle cuts.

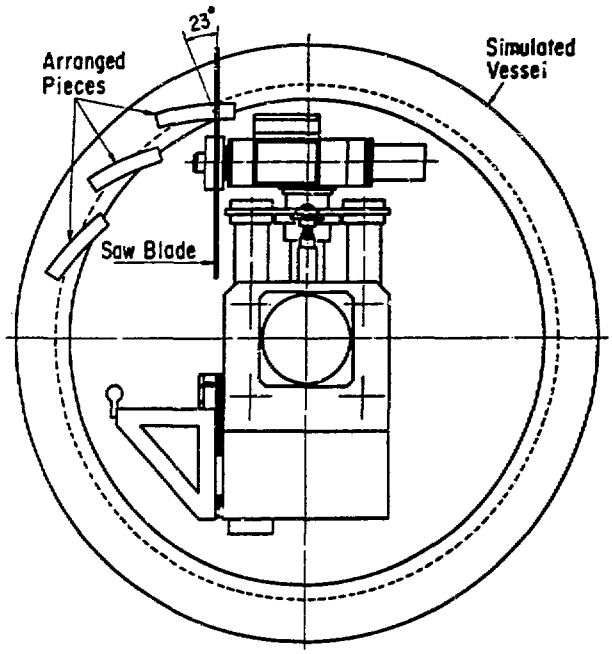

Fig. 7 Right Angled Cutting

Additionally, it was observed that the both sides of a used blade were sustaining damage from arcing. The arcing between the blade side and the cutting metal, called side arc, is undesirable, because it reduces current density at the cutting edge of the blade. It is probable that the reduction of current density causes a drop in cutting energy. When the energy drops sufficiently, the blade and metal make contact and cause a short circuit. To understand the influence of the side arc, some saw blades were modified to suppress the side arc. Vertical cuts were examined using these modified saw blades. These cutting tests indicated the vertical cuts should use a saw blade with insulated sides. The conditions and results of vertical cutting tests by the insulated blade are also shown in Table V.

The final results obtained from workable cutting conditions for three essential cutting patterns are presented in Table V. The constant speed mode was most advantageous of the three modes for driving the saw head, because of blade wear and by-products generation.

Table V Workable Cutting Conditions for Underwater Arc Saw Cutting System

Body to be cut Cutting Direction	Vessel Shell horizontal	Vessel Shell vertical	Flange region vertical
Saw Blade	original blade	insulated blade	insulated blade
Material	carbon steel	carbon stel	carbon steel
Diameter	1000mm	1000mm	1000mm
Cutting Velocity	4mm/s	3mm/s	0.5mm/s
Driving Mode	constant speed	constant speed	constant speed
Set Voltage	50V	50V	50V
Set Current	20kA	20kA	20kA
Blade Rotation Rate	400rpm	400rpm	400rpm

Blade wear was measured remotely after each cutting pass. The blade wear rate was 4 to 6mm per one cutting pass. After specific wear occurred, the saw blade was replaced with a new blade using a blade change device. This device was hung from an existing ceiling crane. Based on this, a schedule for blade change is established. A series of handling tests to grip and lift cut pieces were completed without problem.

By-products generated by cutting were sorted according to dross, particles drifting in the water and hydrogen caused by thermal decomposition or electrolysis of water. After cutting started, water quickly became muddy due to the generation of dross and fine particles. Dross (100 to 1000 μm in size) accumulated at the bottom of the water tank. After the mock-up tests were finished, the dross was collected by an underwater vacuum cleaner without problem. Particles (0.1 to 2 μm in size), drifting in the water, were collected by cartridge filters. The life span of cartridge filters was shorter than expected because of unexpected oil leakage from a hydraulic servomechanism. Flocculation of drifting particles was time consuming but effective. A few hours were needed to deposit the flocculus in the water pool, so the treatment of water could not be performed continuously. The hydrogen concentration in the cavity was 0.1 to 0.3%. Thus, the capacity of the gas treatment device ($140 m^3/min.$) was satisfactory. Various functions and operations of other devices were also examined.

CONCLUSION

The results of mock-up tests of underwater plasma arc cutting system and underwater arc saw cutting system were satisfactory for JPDR dismantling. Information for dismantling JPDR, such as work schedule, manpower, radioactive exposure, radioactive waste product, required containers and utilities were also obtained. Such data will be acquired also in the actual JPDR dismantlement. These data will be useful for future dismantlement of commercial power reactors.

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TEST RESULTS OF CONCRETE DEMOLITIONING BY SMOOTH BLASTING INSIDE A REACTOR CONTAINMENT *)

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SUMMARY AND SCOPE OF TESTS

Explosive methods have been demonstrated earlier to be an adequate tool for dismantling of activated or contaminated concrete /1, 2, 3/. Specific advantages are pronounced in cases of large wall thickness and heavy steel reinforcement.

Here test results are reported where the blasting has been performed inside the former hot steam reactor HDR which is now in use as a nuclear reactor safety test facility /4, 5/. A comparison between HDR and standard type reactors with respect to main structural features is shown in Fig. 1. The HDR facility is described in detail in a poster contribution /6/.

The blasting experiments are aimed primarily at

- optimizing the method for different types of concrete as, in particular, in Niederaichbach (KKN) and Biblis (KKB) biological shield
- safety measures against dust and debris
- quantifying the dynamic loading of the blasted component, the reactor containment and its vicinity.

Large effort has been invested into characterizing the dynamic loading and structural response with respect to safety and predictability. For this purpose a variety of measurements have been performed. The response has also been calculated using finite element models. It is shown that the blasting method can be applied in accordance with safety and regulatory requirements.

*) Supported by the Federal Ministry for Research and Technology and by the EEC Decommissioning Programme

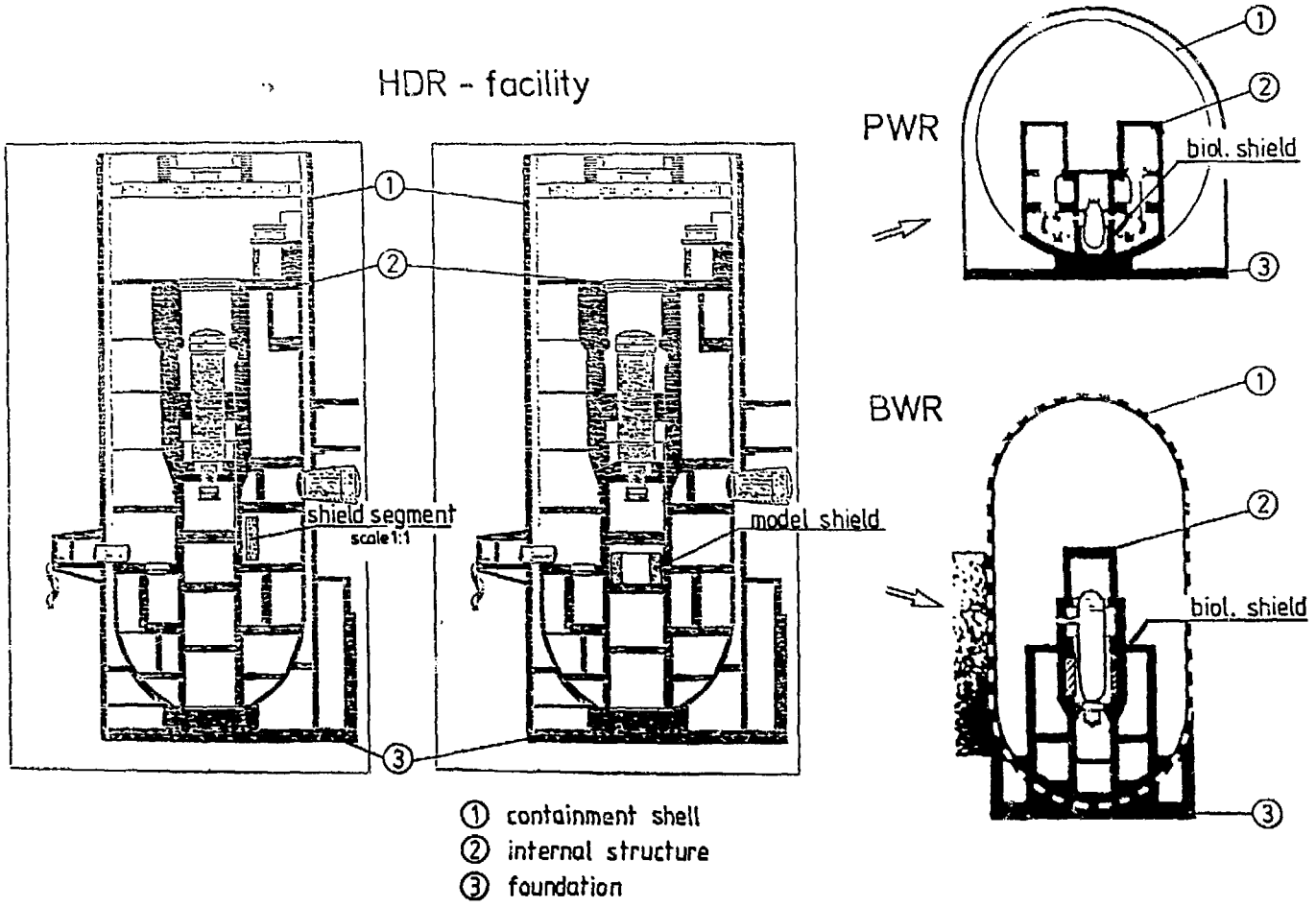


Figure 1: The HDR-facility as compared to german standard light water reactors

BLASTING METHOD AND CONCRETE STRUCTURES

Bore hole blasting is used with the aim to "peel off" predetermined concrete layers of thickness between 20 and 40 cm. Two types of concrete structures are used: a wall segment of dimensions 2 m x 2 m, and a model shield of 4,7 m diameter and 3 m height, see Fig. 2. Heavily reinforced high density concrete as in the Niederaichbach plant (KKN) has been used as well as normally reinforced normal density concrete as in the PWR standard reactor (Biblis type). The wall segments were bolted to the inner cylindrical column of the HDR. They were either mounted on the 4 bolts with a distance of 12 cm to the wall or mounted tightly surface to surface with a prestress of 800 kN. A maximum of 2,5 kg of explosive per blast has been used with the wall segments. Bore hole pattern and blast results are shown in Fig. 3 and 4.

Experiments with the model shield which is embedded into the inner cylindrical column of the HDR are presently in preparation. There a maximum of 15 kg of explosive per blast is scheduled for the removal of a full 360° ring layer of 20 000 kg of concrete.

In past experiments the bore hole pattern and charging has been varied widely, see Fig. 5, depending on differences in concrete density and in the amount of lateral and vertical steel bar reinforcement. The blasting parameters used during present tests are given in Table 1. The short bore hole method was given priority in the tests since it allows better to adjust for complex contours. Since it requires a larger amount of explosive and therefore causes higher dynamic loading, it is regarded the enveloping case with respect to the ultimate load bearing capacity of the ambient structure.

SAFETY MEASURES AGAINST DUST AND DEBRIS

The blasting area in the containment is enclosed by a plastic sheet tent which retains the gaseous explosive reaction products and the dust generated. The concrete debris is held back by rubber mats hanging about 1 m in front of the blasted wall surface. The air from the tent is cleaned through a mobile filter unit. The amount of dust which is set free varies widely depending on the blasting parameters, the content of humidity in the concrete and in the ambient air, see Fig. 6. In addition the particle size distributions of both dust and debris are measured. Details may be taken from the paper presented in the poster session /6/.

Table 1: Blasting parameters for shield segments scale 1:1 and model shield

type of shield	concrete density (g/cm^3)	steel reinforcement	bore hole pattern		specific explosive loading (g/t)*)	
			short bore holes	long bore holes	minor reinforcement	heavy reinforcement
Niederaichbach	3,3...3,6	4 x 26 mm dia St 420/500 RU	40cm x 40cm x 20cm (40cm)	30cm x 20cm x 300cm (20cm x 20cm x 180cm)	100-200	1500
Biblis	2.5	2 x 18 mm dia St 420/500 RU			70-150	500

*) see Fig. 5

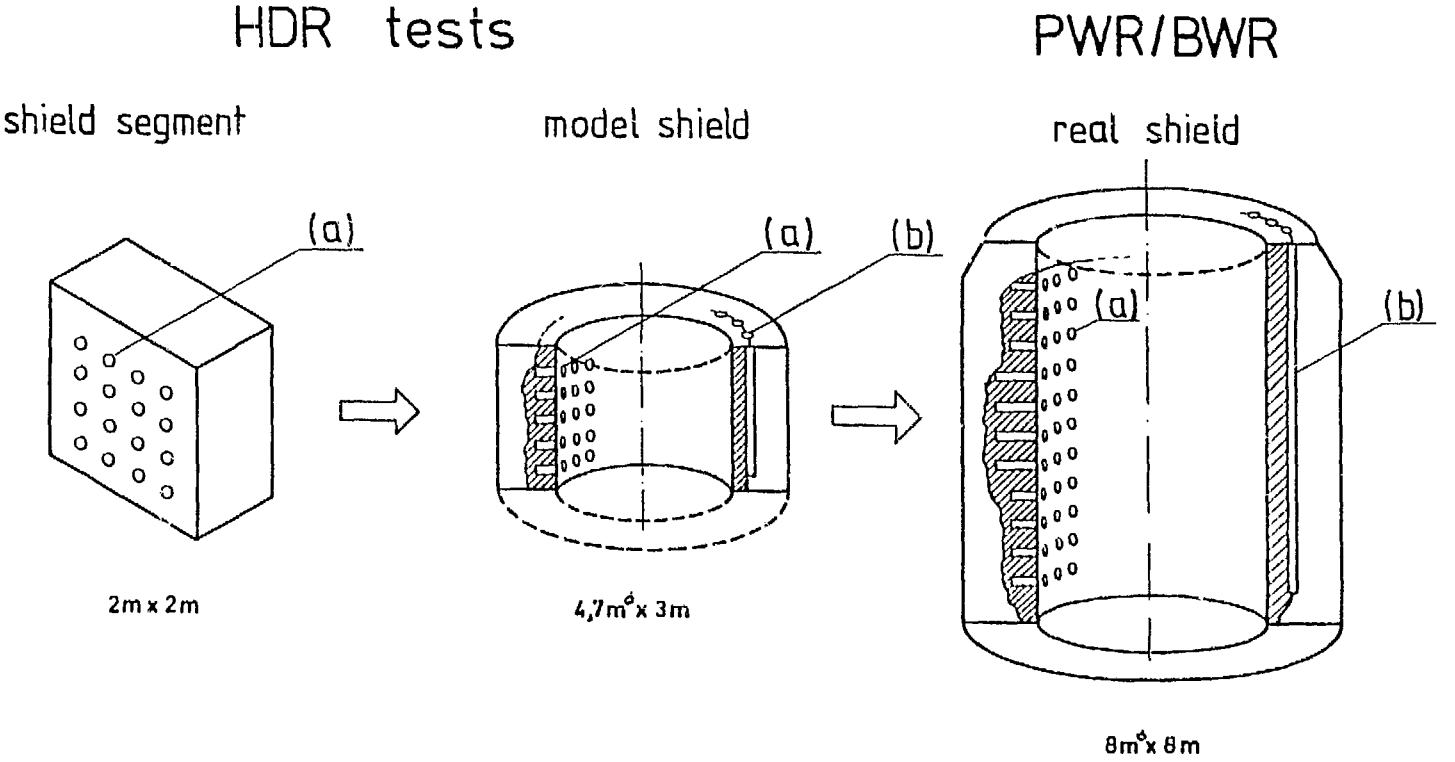


Figure 2: Explosive dismantling of the biological shield: steps approaching the real case, blasting methods: short bore holes (a) and long bore holes (b)

Figure 3: Bore hole pattern 40 cm x 40 cm applied to wall segment 2 m x 2 m x 0,7 m

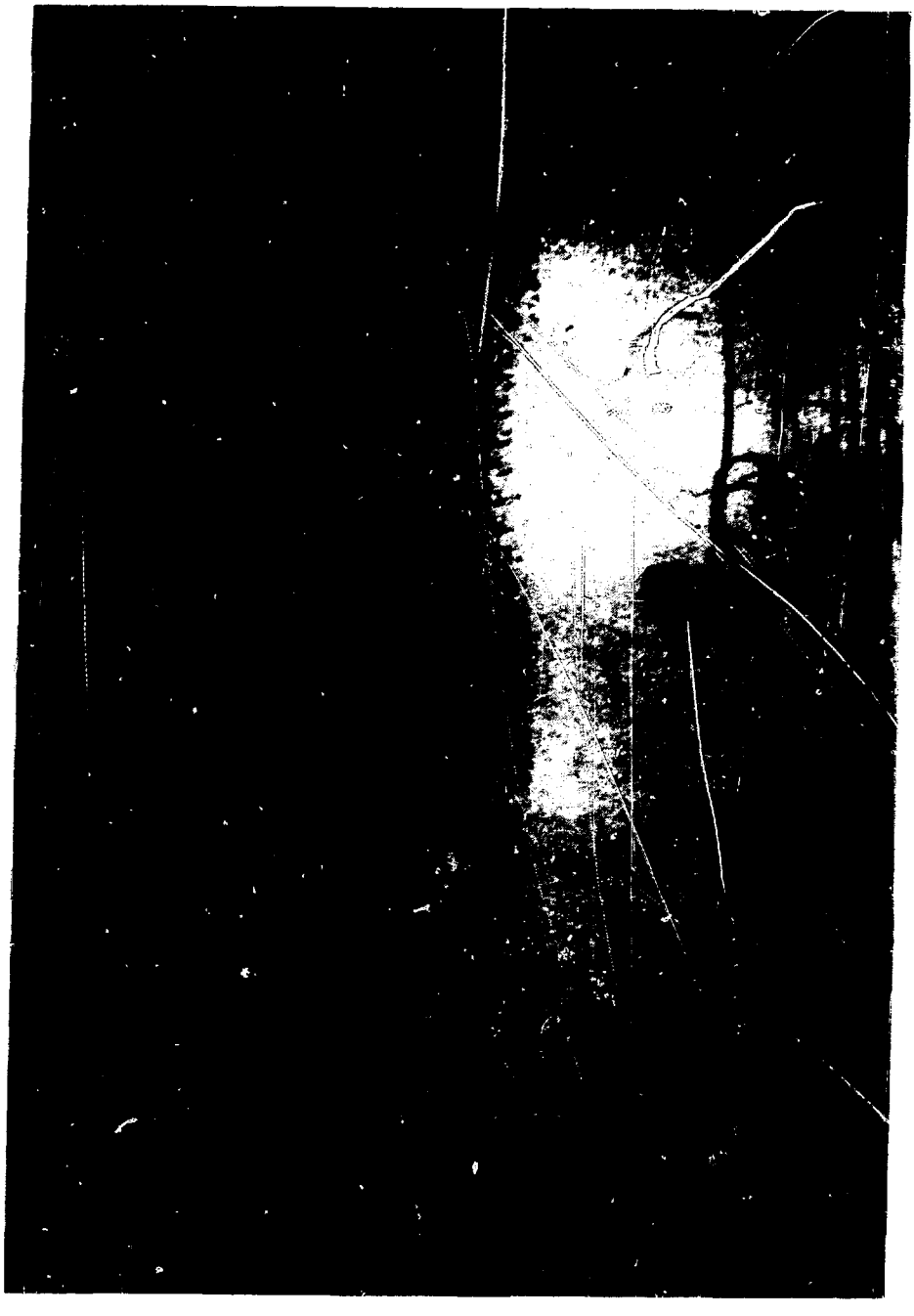


Figure 4: Result of "blast-peeling" a 40 cm thick concrete layer, steel reinforcement removed by mechanical cutting

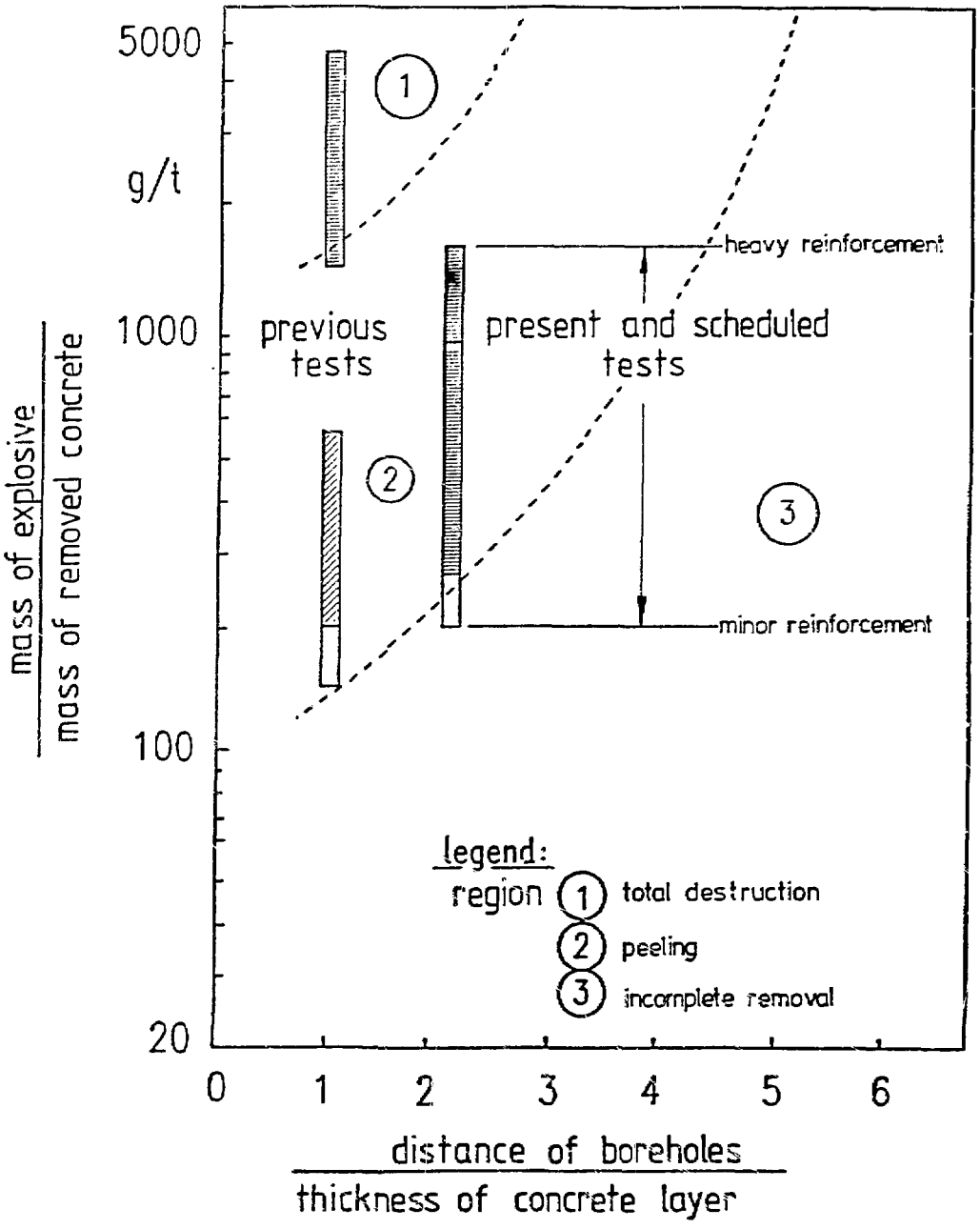


Figure 5: Blasting parameters

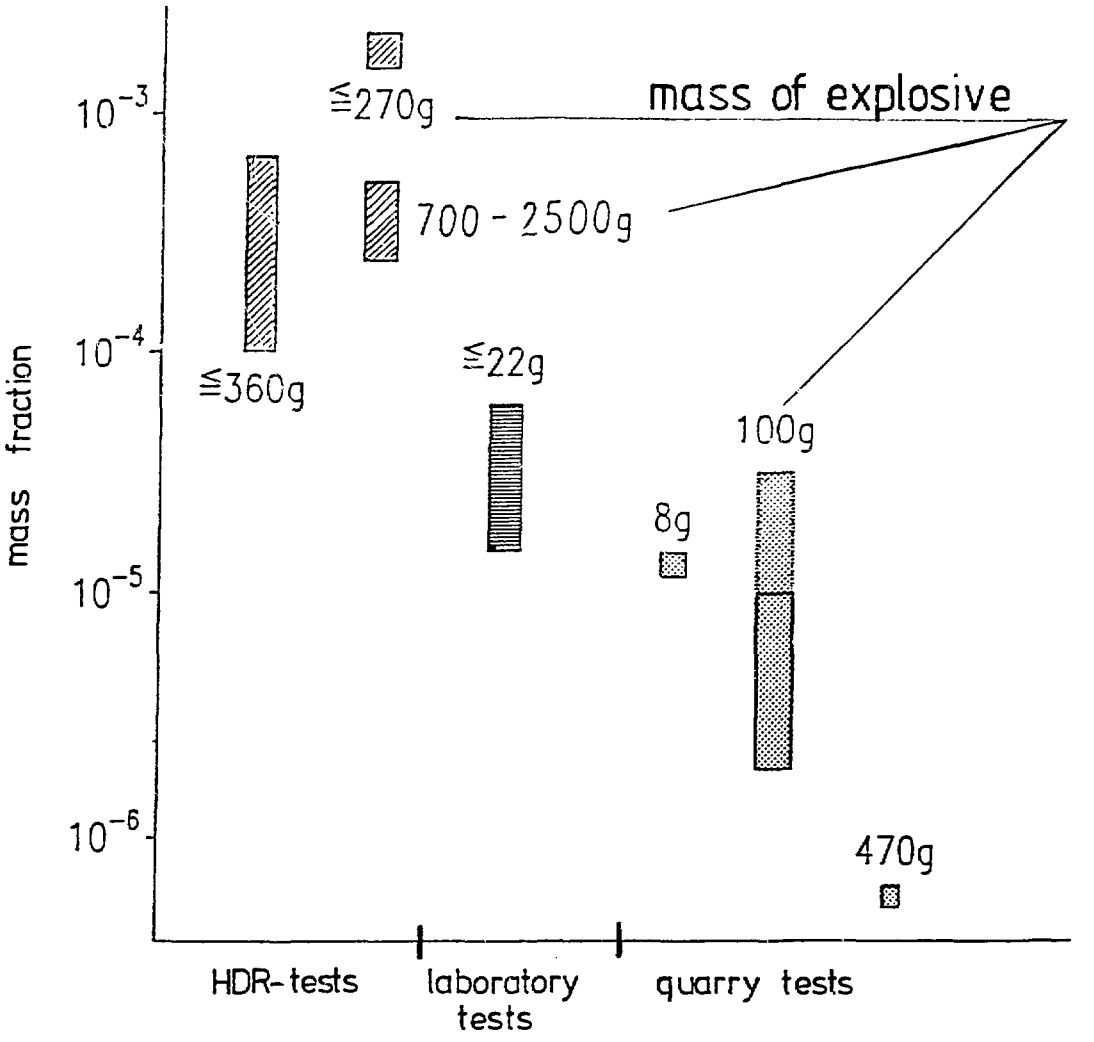


Figure 6: Aerosol fraction of total debris deduced from filter collection

DYNAMIC LOADING AND STRUCTURAL RESPONSE

Loading

Three effects have to be considered

1. the shock coupled directly into the structure
2. the blast wave generated by the gaseous explosive reaction products
3. the impact of falling debris.

All three contributions are measured and are used in an analytical form as input data for model calculations. It turns out that the third contribution - falling debris - is only of minor importance for the containment loading.

In the immediate vicinity of the blast i.e. around the bore holes in the concrete and at the blasted surface, the contributions 1 and 2 cannot be separated. The loading pulse is of short duration (< 1 ms), its amplitude exceeds the material strength of the concrete (see Fig. 7a). The loading pulse propagates from the blasted segment into the containment structure (see Fig. 7b) with a speed of 2000 to 4000 m/s. At the same time the pulse flattens in amplitude and stretches in duration. The blast wave propagates only with slightly over 300 m/s. Its amplitude is significantly lower than from an unconfined detonation. It affects the structure primarily in the near zone of the blast.

The measurement of the time-dependent loading parameters is performed with special sensors for high dynamic pressure and force.

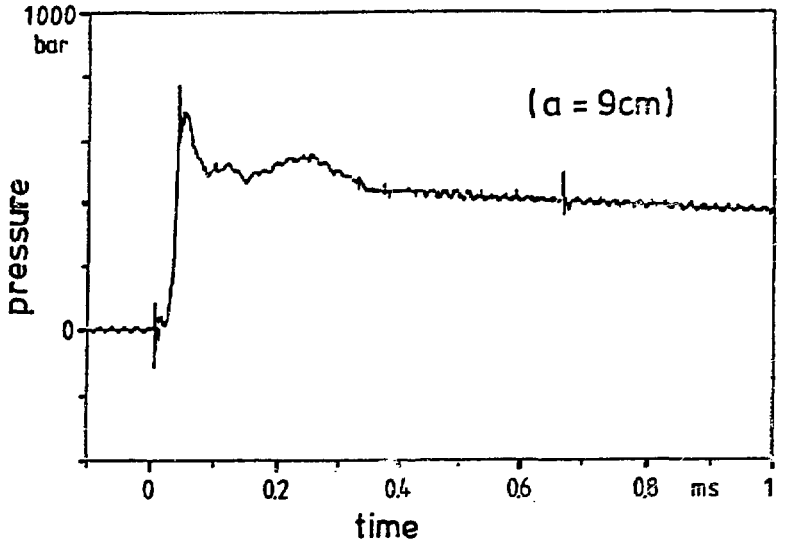
Structural response

Local effects. Just outside the destruction zone the concrete material is still subject to intense loading. Wall surfaces are shaken by high frequency accelerations with amplitudes exceeding 10^4 ms^{-2} (see Fig. 8). This leads to cracking and weakening of plugged mountings.

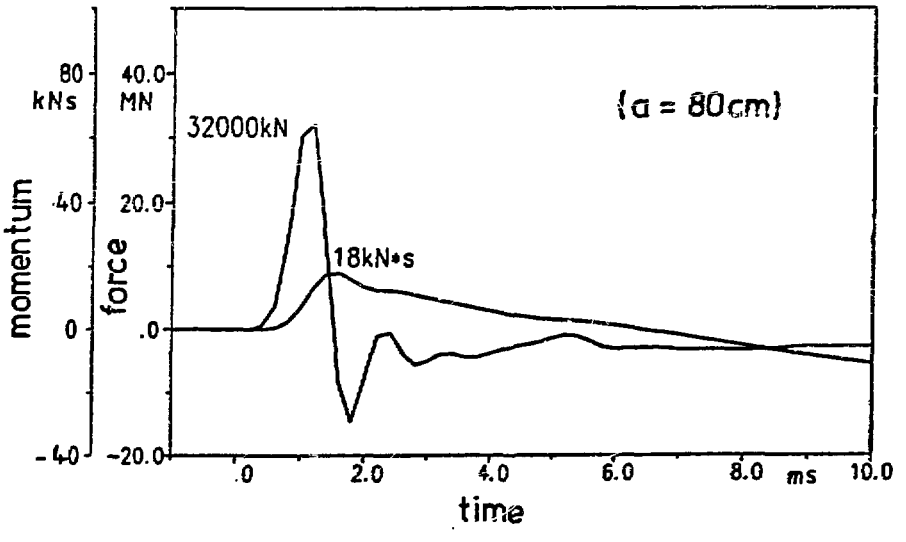
Response of containment and components. During all tests the blast response has been recorded at selected positions. The accelerations at the following points are regarded as safety control data:

- crane runway
- reactor platform
- reactor foundation

In Fig. 9 the acceleration is shown for a 2,5 kg blast at two positions as time history and as frequency distribution. The transient character of the excitation is particularly apparent in the platform



a)



b)

Figure 7: a) Pressure in the concrete 9 cm away from blasted bore hole charge, 45 g

b) Force and momentum transfered from the blasted segment to the reactor column, 2,5 kg of explosive

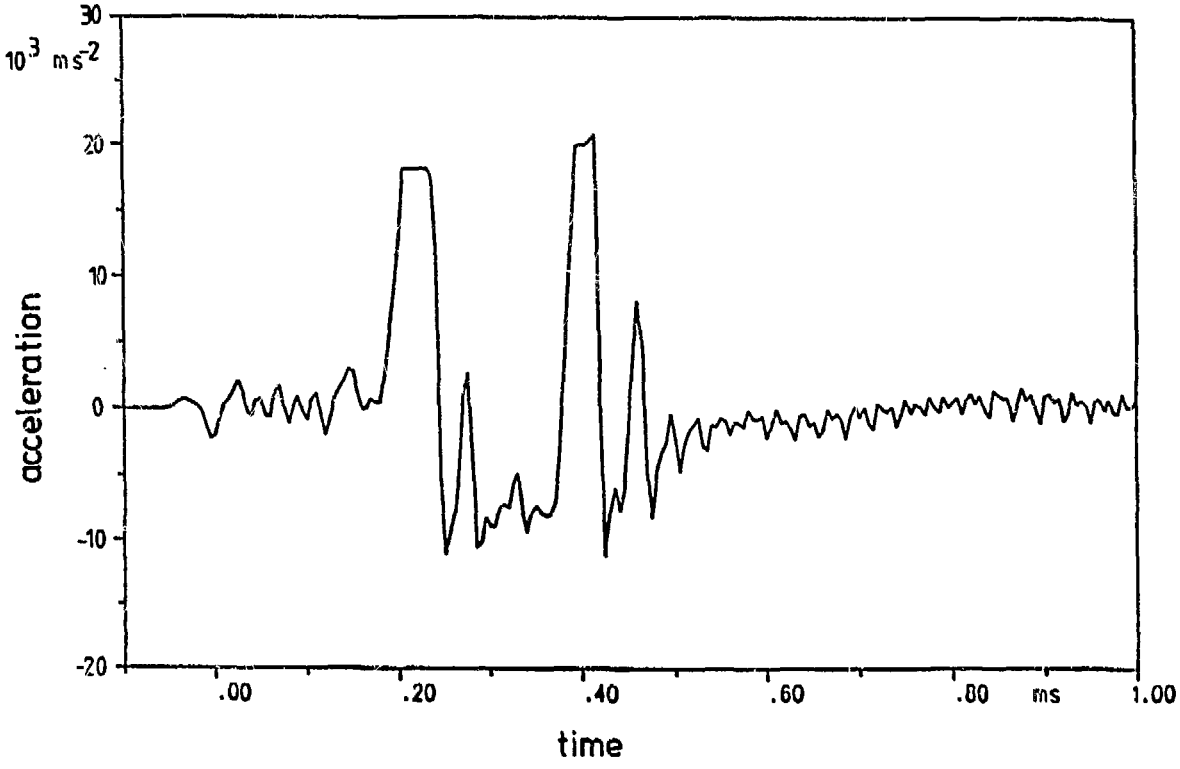


Figure 8: Acceleration on the side surface of the blasted concrete segment, 0,72 kg total explosive charge

Figure 9: Acceleration at crane runway and reactor platform following a blast of 2,5 kg of explosive

response, similarly in the foundation response (not shown). A general feature is the dominance of frequencies above 100 Hz which is a regime normally regarded irrelevant for damage considerations. While the platform response is due to direct structural shock coupling the crane runway response reflects in part the air blast wave shaking of the reactor dome. It is pointed out, however, that the maximum acceleration in all cases is well below critical values (see following section).

It is also of interest to know the response of the outer containment shell which constitutes the outermost safety barrier. The coupling to the blasting area occurs via the central column and the foundation. Besides marginal excitation of low frequency bending (below 10 Hz), the main acceleration amplitudes here too lie above 100 Hz, with an amplitude of 2 ms^{-2} which is only one tenth of the value at the reactor platform.

Response of the adjacent reactor building VAK. At a distance of 100 m a second reactor - the VAK - is located. Its response was recorded through acceleration sensors on the foundation which had been installed for earthquake simulation measurements.

Within the noise of the sensor system no response could be measured. The limits are as follows:

frequency region 0 to 80 Hz: below $0.4 \cdot 10^{-2}$ ms^{-2}

0 to 30 Hz: below $0.05 \cdot 10^{-2}$ ms^{-2}

The fact that the neighboring reactor remains practically unaffected by the blasting inside the HDR can be attributed to two reasons:

- Momentum compensation takes place to a large extent inside the reactor containment
- The soil damping is high for the predominantly transmitted frequencies above 100 Hz.

Comparison between experiment and calculations. The containment loading has been simulated by calculations using two different finite element models of the HDR: a planar multiple beam model and a cylindrical shell model. The models are expected to represent oscillations of the containment building up to frequencies of about 30 Hz and 70 Hz respectively, due to the element size. This requires that the mathematical loading function correctly resembles the real dynamic loading. When using a two dimensional loading function as derived from experiment the agreement between experimental and calculated peak acceleration values is very satisfactory. This is documented in Table 2.

Safety and regulatory requirements. Safety regulations have to be met which deal with the following aspects:

- handling of explosives: charge preparation and ignition

Table 2: Maximum values of acceleration in m/s, comparison of experimental and calculated values, 0,36 kg of explosive

Position		Experiment low pass filtered		Calculations	
		0 - 30 Hz	0 - 80 Hz	beam model *)	shell model **)
crane runway	h	0,45	5,2	0,41	1,5
	v	0,17	0,42	0,055	0,37
reactor platform	h	0,11	0,36	0,12	0,12
	v	0,07	0,32	0,035	0,22
reactor foundation	h	< 0,05	0,09	0,05	0,035
	v	< 0,08	0,15	0,03	0,025

h ... horizontal component

v ... vertical component

*) model calculations by F. Rischbieter; Battelle-Institut

**) model calculations by R. Zinn; Zerna, Schnellenbach und Partner, Bochum - W.Germany

- handling of radioactive material: protection of personnel, prevention of contamination
- dynamic loading of buildings and structures

The handling of the explosives has to follow the specific state regulations. They may be supplemented by additional technical requirements e.g. to reduce the possibility of a misfiring of explosive charges.

Specific problems of handling the radioactive material arise from the necessity to avoid any escape of material to the outside of the tent confinement which encloses the blasting area. This has been solved by using highly flexible PVC-foil for the tent /5/.

Since the turbulent flow of dust-loaded gaseous reaction products and air causes uniform spread of the dust within the tent confinement, no clean surfaces can be preserved. However, taking advantage of this flow a short flush of dust-binding liquid immediately during and after the blast reduces the aerosol content by a factor of 10 or more and keeps the dust attached to the surfaces thus reducing the danger of contamination /2/.

The requirement to guarantee structural integrity focuses on two areas:

1. immediate vicinity of the blast
2. containment and components

In the immediate vicinity damage is caused by

- shock waves in the concrete exceeding the material strength and/or
- air blast exceeding the ultimate load bearing capacity of structural elements

It was found that for the 2 m x 2 m concrete segments the following simple formula holds for the ultimate load. It relates the explosive mass per unit area and the static stiffness of the segment:

$$\frac{m_{expl.}}{A \cdot W^3} \quad 0.8 \text{ kg} \cdot \text{m}^{-5}$$

with

$m_{expl.}$ ... explosive masse
A ... loaded area
W ... thickness of segment

An estimate of the extent of damage can also be made using graphs when both the peak pressure p_s and momentum density i_s of the air blast wave are known, see Fig. 10.

For the containment and components a safety estimate of the dynamic loading response can be made using an adequate mathematical model for the building structure and a realistic loading force F(t) as described before. The calculations will yield oscillation velocities or accelerations which may be compared to the limits imposed by technical regulations or by state authorities. In Fig. 11 this is shown for the HDR with respect to the German technical standard DIN 4150. As can be deduced, the HDR will safety bear an explosive loading of 15 to 20 kg. Such loads will be employed in the final dismantling tests with the model shield.

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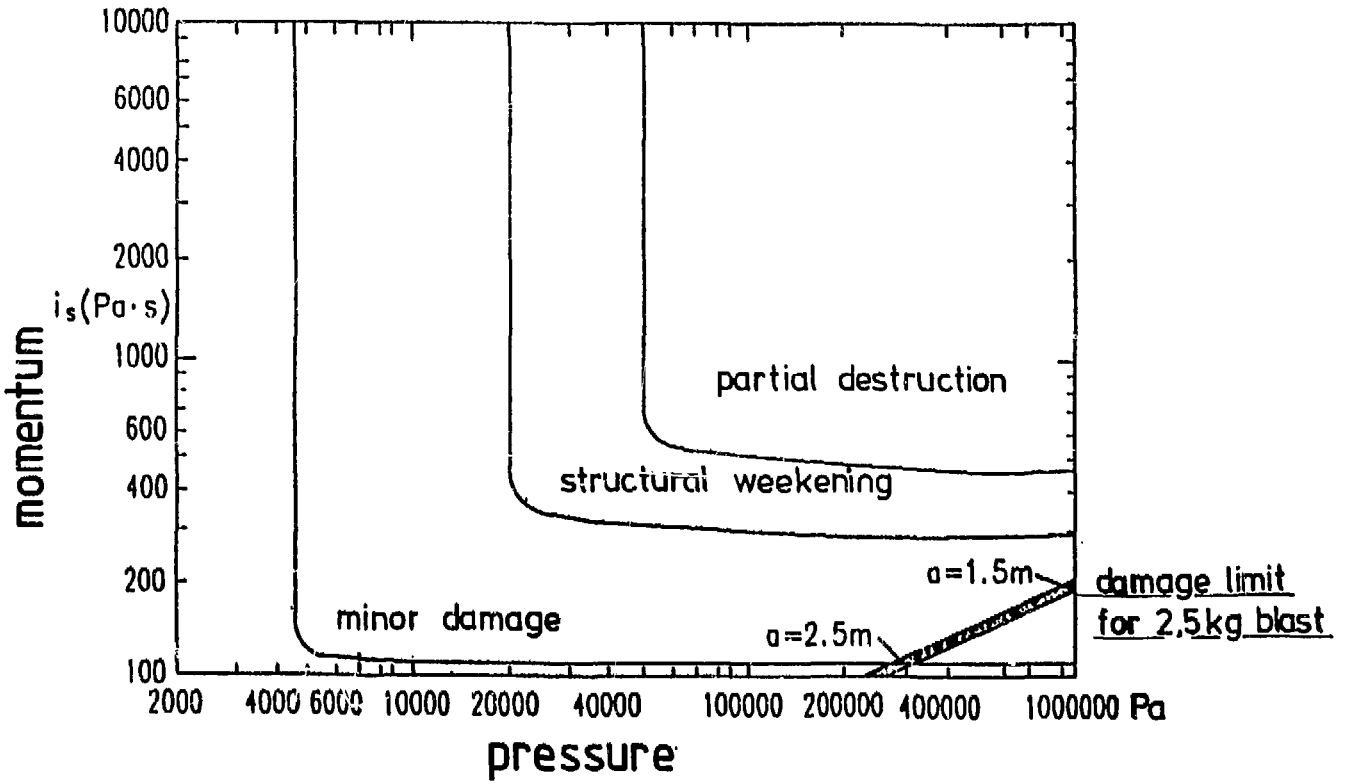

Figure 10: Iso damage curves for blast loaded concrete structures

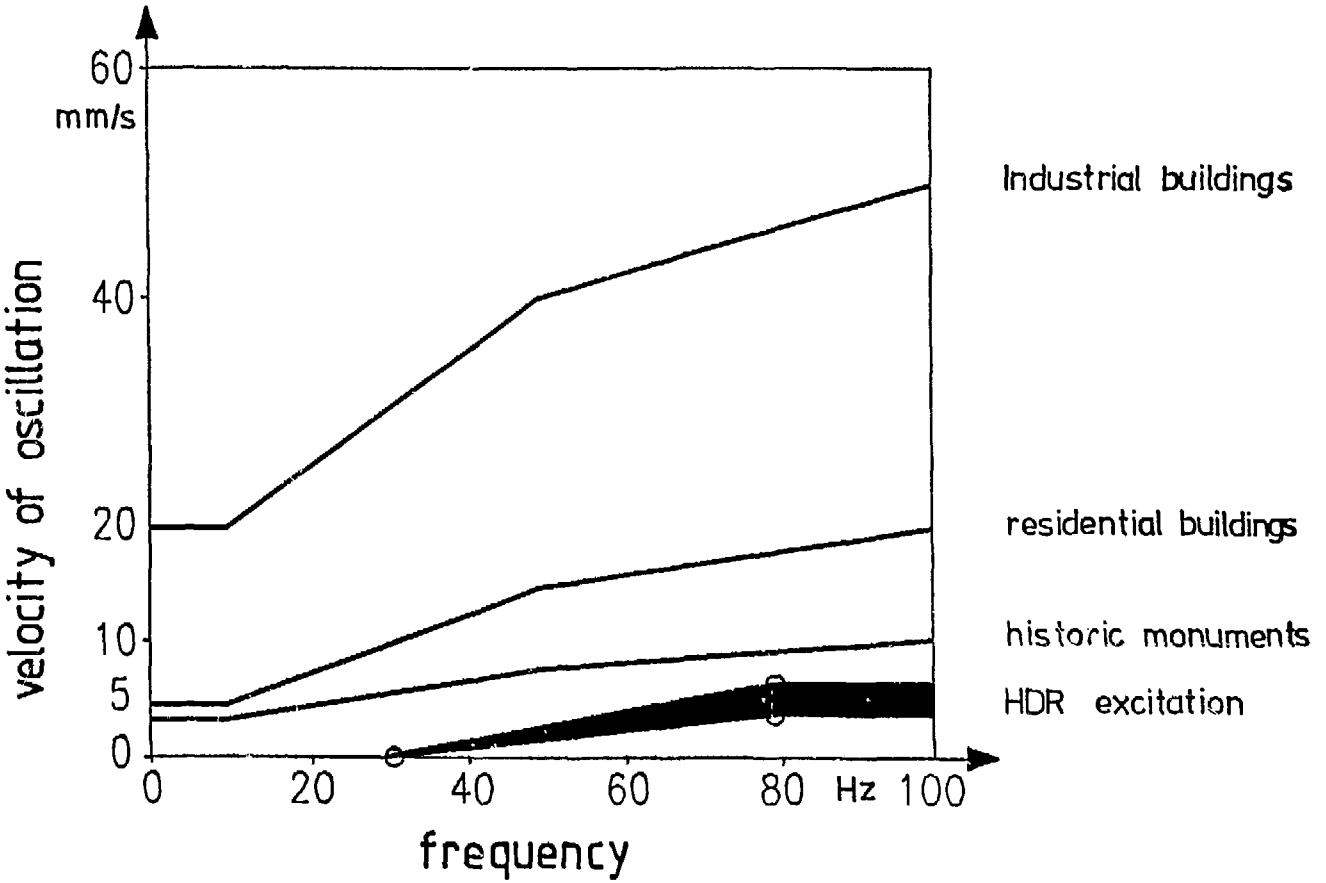

Figure 11: Velocity of oscillation-limits according to german technical standard DIN 4150
HDR-results for 2,5 kg explosive mass, foundation velocity (experimental and theoretical)

ABRASIVE-WATERJET CUTTING OF THICK CONCRETE AND WATERJET CLEANING FOR NUCLEAR FACILITY DECOMMISSIONING AND DECONTAMINATION

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ABSTRACT

Two tools have been developed for use by the nuclear industry: the Deep Kerf tool and the Cleaner/Scarifier tool. The Deep Kerf tool is designed to cut through thick, reinforced concrete structures to facilitate their decommissioning. It employs the abrasive-waterjet (AWJ) cutting technology. The basis of the system is a rotary nozzle that makes a slot in the concrete wide enough to accommodate the cutting tool as it advances. In this program, concrete as thick as 1.5 m was cut through from one side. A shroud and vacuum system covers the opening of the slot during cutting to contain the spoils with greater than 99% efficiency. The Cleaner/Scarifier tool was designed for removing the surface layers of contaminated concrete and decontaminating metal surfaces. It uses ultrahigh-pressure waterjets mounted on a rotating arm to remove or clean the target surface. Spoils recovery with a shroud and vacuum system is over 99% complete for both horizontal and vertical surfaces.

INTRODUCTION

At the end of their useful lives, nuclear reactors, support facilities and reprocessing plants require decommissioning. This involves the orderly disposition of a nuclear facility, taking into account the environment, waste management, and safety. The procedure can range from minimal removal of radioactive material to the complete disassembly of a facility and its unconditional release for other uses.[1]

A major portion of most decommissioning jobs is the decontamination and removal of concrete. In many facilities, large thick sections of activated reinforced concrete need to be removed along with contaminated surfaces. To reduce the volume of waste that must be placed in controlled storage, it is desirable to separate the contaminated portion for isolation, while allowing the remainder of the structure to be demolished by conventional techniques. Concrete surfaces are also difficult to decontaminate because concrete is porous and often contains numerous cracks that hold the contaminants. In addition to the problem of radiation, nuclear structures typically contain a large volume of concrete with reinforcing bars. Basemats may be 8 m thick, and biological shields up to 3 m thick. Conventional concrete removal methods cannot remove these structures efficiently while containing contaminants from release into the environment.[2]

Another facet of decommissioning a nuclear facility is the decontamination of metal parts. The surfaces of such metal parts may have been painted to contain smearable contamination or coated as a result of accidental

releases. Efficient techniques are needed to remove surface contamination, reduce personnel exposure and enable less restrictive proposal requirements.

This paper describes a research and development program sponsored by the U.S. Department of Energy. In this program, two systems were designed, built and tested for use by the nuclear industry: a Deep Kerf tool and a Cleaner/Scarifier tool. The Deep Kerf tool will cut through thick concrete structures to facilitate their decommissioning. It is based on the abrasive-waterjet (AWJ) cutting technology. The Cleaner/Scarifier tool uses ultrahigh-pressure waterjets to remove the surface layers of contaminated concrete and to decontaminate metal surfaces. Both systems are equipped with a shroud and vacuum system to contain and recover the cutting and cleaning spoils for proper disposal.

BACKGROUND

Decontamination and Decommissioning Equipment Requirements

There are certain unique requirements for concrete cutting and surface removal techniques in a contaminated environment.[3] The ideal equipment characteristics for performing this work include:

- Total containment of the cuttings, with
  - no release of airborne contamination, and
  - no recontamination due to the cutting operation.
- Generate no additional waste in the removal process.
- Remove only the contaminated surface, leaving radiation-free concrete.
- Cut concrete and reinforcing bars simultaneously.
- Perform all cuts on all surfaces (ceilings, walls, floors, contoured surfaces, etc.) without tool changes.
- Easily adapt to remote operation and automation and to use in confined locations.[4]
- Generate no shock, vibration, or excessive noise while operating.
- Easy to operate by personnel in protective clothing.
- Easy to repair.
- Economically feasible.

Reviews of conventional concrete demolition and concrete surface removal equipment are found elsewhere.[3-7] Conventional demolition and surface removal equipment does not meet many of the ideal features listed above. A study[8] on the use of conventional AWJ techniques for the dismantling of thick concrete structures by cutting out prism-shaped blocks, 0.6 m on a side, showed that such an approach is possible but slow for dismantling thick concrete structures.

Abrasive-Waterjet Cutting

The AWJ technology appears to have great potential for concrete demolition and surface removal. Abrasive-waterjets are formed by mixing small-diameter (0.25- to 0.8-mm), high-velocity (up to 750 m/s) waterjets with abrasive particles. The mixing process occurs in a specially designed

mixing and accelerating chamber. The abrasive particles exit the acceleration section with high velocities and become capable of cutting even the hardest material.[9-12]

DEEP KERF TOOL

Equipment Description

The Deep Kerf tool used in this study consists of a stem, swivels, a traverse mechanism, an obstacle detection system, a shroud and catcher, and a collection and storage system. Figure 1 shows some of these components.

Nozzle Stem. The concept of a single angled jet in a circular tube was chosen as the basis for the Deep Kerf tool because of its versatility, ease of manufacture and ease of application. It was also selected because it is the most simple configuration for producing a slot of uniform depth over the entire length of traverse. Figure 2 shows a cross-sectional view of the abrasive-waterjet stem and nozzle. Abrasives flow to the nozzle through the annulus between the high-pressure conduit and the outer stem. The waterjet nozzle is machined such that the jet exits at nominally the same angle as the inclination of the mixing tube. The life expectancy for a mixing tube depends on a number of factors, such as waterjet pressure and diameter and abrasive flow rate. On average, the mixing tubes used during testing lasted about one hour before being changed. Increased mixing tube life is an area requiring further development.

Theoretically, the stem and nozzle assembly could be any length desired; however, it is limited by stiffness and strength. The outer stem for the Deep Kerf tool was 22 mm in diameter, which was found suitable for depths up to 1.5 m. For kerf depths exceeding 1.5 m, a larger diameter stem would have to be used.

Abrasive and High-Pressure Swivels. High-pressure swivels are relatively well developed, and there are a number of acceptable products for use in this application. An abrasive swivel was developed in this project to allow the feed of abrasives from a stationary hopper to the rotating stem. In this swivel, the abrasives are introduced into the annular area between the high-pressure tube and the outer stem through an angled port. At the upper end of the swivel is a seal and bearing assembly that allows for relative motion between the high-pressure tube and the housing. At the lower end of the swivel there is another seal and bearing assembly that allows for relative motion between the outer stem and the housing.

It is estimated that during the test program there were 20 hours of rotary running time on the abrasive swivel and, when disassembled, there was little detectable wear on the seals.

Traverse Mechanism. Figure 3 shows the Deep Kerf traverse mechanism designed and manufactured for this project. The mechanism provides three axes of motion to the stem and nozzle assembly. The range of traverse motion for this mechanism is 1.5 m, and the traverse speed range is 0.5 m/min. Depth

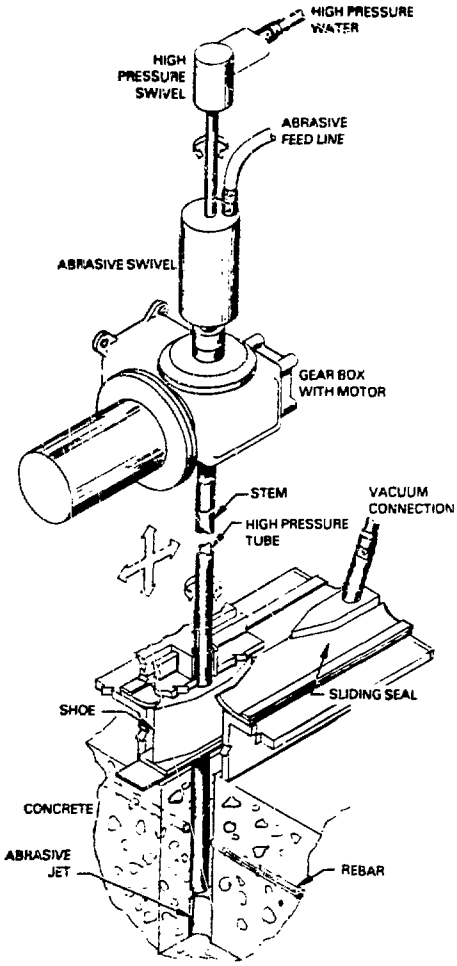


FIGURE 1. Abrasive-Waterjet Deep Kerf Tool

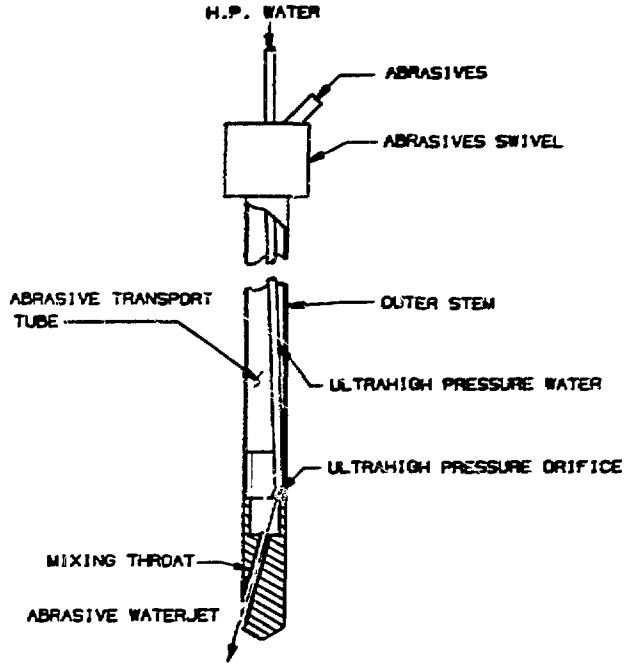


FIGURE 2. Rotating Nozzle

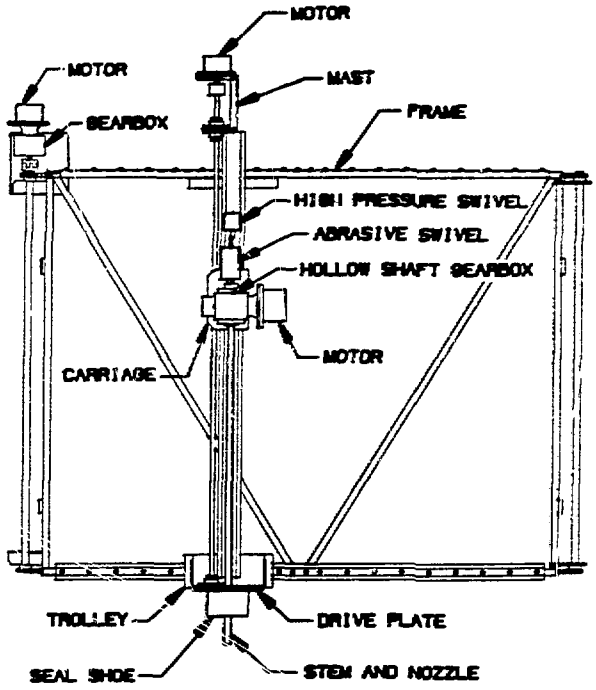


FIGURE 3. Deep Kerf Traverse Mechanism

motion advances the stem and nozzle into the kerf. The range of depth for this mechanism is 1.5 m, and the speed range is 1.5 m/min. The third axis of motion is stem rotation. The mechanism is capable of rotaring the stem and nozzle through a range of 0 to 60 rpm. Each axis is powered by a stepping motor, which facilitates the use of a simple open-loop control system when the system is under computer control.

Obstacle Detection. The stem and nozzle are connected to the drive plate on the trolley (Figure 3) through a spring-centered bushing. If the stem encounters an obstacle during operation, the side load pushes the bushing off center, and proximity switches positioned adjacent to the bushing detect that an obstacle has been encountered. The control circuit is wired such that the traverse mechanism will not move in the direction in which an obstacle has been detected but rather a different routine will be followed to remove the obstacle, which could be a rebar or an aggregate.

Shroud and Catcher System. To effectively contain the water and cuttings generated by deep kerf cutting, there must be no path through which the jet backsplash can escape. The shroud and catching system designed for deep kerfing relies on sealing the cutting area or slot in its entirety. The catcher frame is sealed to the concrete with neoprene rubber lip seals that run the length of the frame. Also running the length of the catcher is a pair of seals which can be parted to create an opening in the top of the frame. A wedge-shaped seal shoe fastened to the traverse mechanism parts the seals as the stem is traversed along the length of the deep kerf slot and maintains closure of the catcher cavity. The stem is free to move up and down through the seal shoe for cutting at various depths. Two suction tubes were built that could be slipped through the seals of the catcher to the bottom of the kerf, and these kept the slot free from water and debris. When the stem approached one end of the slot, the suction tool at that end would be turned off and removed allowing the stem and nozzle to cut to the end of the slot. The shroud and catcher system collected over 99% of the spoils produced.

Spoils Collection. Figure 4 is a diagram of the spoils collection system developed for this project. Water, air and solids from the shroud and catcher system enter the primary drum through a cyclone baffle cover. Most of the solids settle to the bottom of the drum leaving water and air to exit through the other port in the lid. This mixture enters the secondary drum through a cyclone baffle causing the water to separate from the air. In the bottom of the secondary drum is a sump pump with a float switch to maintain the water level below the air outlet. The sump pump discharges through a check valve into a tank, where the remaining solids can settle out over a period of time. The outlet of the secondary drum has an automatic shutoff float valve to prevent damage to the vacuum system should the sump pump fail to operate.

The vacuum system used for this project was an Invincible Model 700 with a HEPA filter. The cyclone baffle covers were Invincible Type A2. The sump pump for liquid removal from the secondary drum was a Teel submersible sump pump.

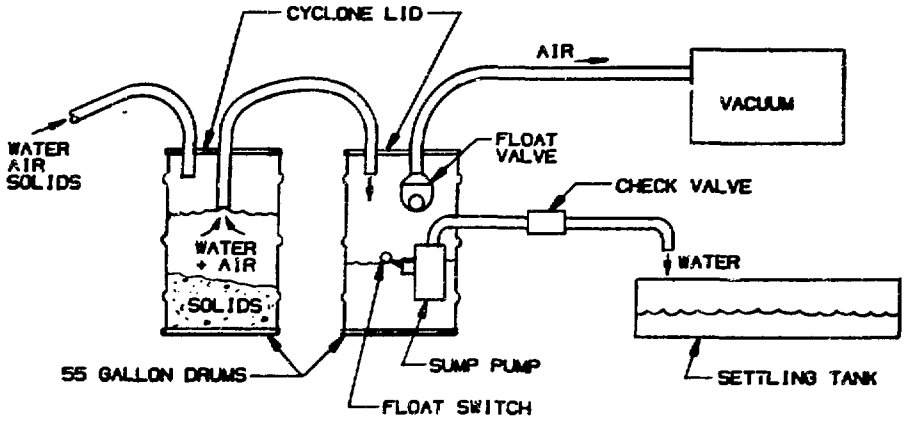


FIGURE 4. Spoils Collection System

Testing Results

Linear Cutting Tests. Parametric linear cutting tests were performed to determine the effect of a number of variables including:

- o Abrasive flow rate
- o Traverse rate
- o Standoff distance
- o Mixing tube diameter
- o Jet pressure
- o Abrasive size
- o Abrasive type

The aim of the linear cutting tests was to determine the optimum operating parameters to maximize the volume removal rate of material in the kerf. It was found that most of the material removed from the kerf is due to the direct cutting action of the abrasive-waterjet, and little material is removed by breakage between adjacent cuts.

Table I compares the volume removal rates for the different abrasive materials used in the tests. As can be seen, garnet sand and steel abrasives performed comparably well. The potential advantage to using steel as the abrasive material is that it can be recycled more easily than garnet sand.

TABLE I. Abrasive Material Comparison

Abrasive Material	Removal Rate (liter/hr)
Garnet sand	2.42
Silica sand	0.59
Steel shot	2.27
Steel grit	2.22

Deep Kerf Testing. The emphasis of the Deep Kerf test program was on creating nozzle motion algorithms that would produce an acceptable kerf through the range of depths specified. Much effort was expended on coping with obstacles, either rebar or pieces of hard aggregate, in the kerf and on the removal of spoils from the bottom of the kerf. In all, some 28 tests were performed producing an estimated 3 m^2 of kerf area.

The results of the testing produced an optimum sequence of nozzle motions as follows to produce a deep kerf:

- The nozzle is rotated at 50 rpm and traversed the length of the slot to be cut.
- At the end of each pass, nozzle translation and rotation are stopped. The nozzle is then oscillated 180 degrees from two to four times to slightly overcut the slot end. This ensures that the ends of the slot do not slope inward.
- The stem is then indexed downward slightly less than the average depth of cut for the previous pass.
- Every fourth pass the jet is raised up to the height where the first pass of the sequence was made. The nozzle is traversed back and forth to remove the projections from the walls of the slot.
- The nozzle is then lowered to the next depth index and the process is repeated.

The following sequence was developed to remove a detected obstacle:

- The stem is backed up 12 mm to free it from the obstruction.
- The stem is raised up 150 mm.
- The jet is oriented to one side of the kerf and traversed for about 40 mm.
- The jet is oriented at the other side of the kerf and traversed back.
- The stem is returned to the position where the obstacle was encountered.

The Deep Kerf system generated kerf face at a rate of 0.1 to 0.5 m^2/hr depending on concrete strength, hardness of aggregate, and density of rebar.

Abrasive Recycling. It would be desirable to recycle abrasive material to limit the amount of abrasives used and to limit the amount of radioactively contaminted waste that is produced. The abrasive most commonly used for AWJ cutting, garnet, is nearly totally pulverized during use, thereby rendering it unsuitable for recycling. Steel grit was investigated as an alternative abrasive material. The magnetic nature of the steel provides a way of separating it from the concrete spoils.

The steel grit, being much more dense than garnet, was not moved to the slot ends by the splashing action of the jet to a satisfactory degree. The suction had to be moved along closely behind the nozzle to remove grit adequately from the slot. This would have to be accounted for if steel grit were used. It was found feasible to use steel grit as the abrasive medium for deep kerf cutting with some modification of the cutting system.

The reuse of steel grit showed that 93% of the unused grit was retained by the 20-mesh screen. On successive cycles, 78% and 71% were retained, respectively. This represents about a 10% loss of particles of a particular size for each time the material is used. This, coupled with a reasonable estimate of 90% recovery from the spoils for a production-type system, yields an overall recovery of 80% of the abrasive material per use.

Linear test cuts made with unused, once-used, and twice-used abrasives showed a change in cutting performance. The test cut made using once-used grit was 13% deeper than the test cut made using unused grit probably because the particle size became closer to the optimum value. The test cut made using twice-used grit was 4% shallower than that for unused grit. A test cut was made using 50-grit (smaller particles) abrasives, and the depth of cut was 17% greater than for the 25-grit abrasives. This shows that the particle size used for this test was larger than the optimum for cutting concrete. For recycling, it is desirable to use grit that is larger than optimum to increase the number of times it can be reused such that optimal performance is obtained during the recycling process instead of only at the beginning.

Economic Evaluation of Steel Grit Abrasive Recycling. The recycling of steel grit abrasive will result in a decrease in the solid waste generation rate and a resulting reduction in waste disposal costs. This is offset by the increased purchase price of the steel grit abrasive over garnet; the reduction in cutting efficiency when using steel grit, and the cost of the purchase and operation of the recycling system.

The results of the laboratory testing were incorporated into an economic model to evaluate the utility and cost-effectiveness of steel grit abrasive recycling for the Deep Kerf tool. The model includes provision for various costs including capital equipment, consumables, maintenance, labor, and waste disposal. Model outputs are waste generation rates, hourly costs, and total costs. The total costs are given both in terms of cost per hour of operation and cost per square foot of kerf produced.

The model considered two cases. Case 1 is for garnet abrasive without recycling with a kerf generation rate of 0.37 m^2/hr. Case 2 is for steel grit that is recycled an average of 10 times but produces a kerf generation rate of 0.28 m^2/hr. Table II shows the hourly cost of operation for the two cases based on typical capital and maintenance costs.

On the basis of total cost per hour, the garnet, at \$161/hr, is more costly to use than the recycled steel grit at \$126/hr. However, on the basis of cost per square meter of kerf produced, the garnet is more economical at \$430/m^2 with the steel grit costing \$505/m^2.

The results are sensitive to the cost of solid waste disposal. For example, a cost of \$350/m^3 was assumed. If this cost was increased to \$635/m^3, the two methods wouls be approximately equal in overall cost per square meter of kerf generated. At over \$635/m^3, recycling of steel grit abrasive is the more economic alternative.

TABLE II. Hourly Cost of Deep Kerf Tool Operation

	Case 1 Garnet	Case 2 Steel Grit
CAPITAL EQUIPMENT		
Pump	8.14	8.14
Cutting System	8.62	8.62
Abrasive Recycling System	0	2.87
Financing	4.47	5.24
Total ($/hr)	21.23	24.87
OPERATION		
Abrasives	24.00	6.00
Carbide Nozzles	7.50	2.50
Recycling System Fuel	0	0.55
Pump Fuel	6.03	6.03
Total ($/hr)	37.53	15.08
MAINTENANCE		
Pump	2.44	2.44
Cutting System	1.72	1.72
Abrasive Recycling System	0	0.57
Total ($/hr)	4.17	4.74
LABOR ($/hr)	36.00	36.00
RADIOACTIVE WASTE DISPOSAL		
Water Disposal	24.00	24.00
Solids Disposal	25.56	5.25
Total ($/hr)	49.56	29.25
TOTAL COSTS ($/hr)	148.48	109.94
TOTAL COSTS ($/m^2)	399.00	438.00

Rebar Detection. An abrasive-waterjet cuts concrete more easily than steel. As a result, when cutting steel-reinforced concrete, the reinforcing bars remain, even after the neighboring concrete has been removed. The rebar protects the concrete directly under it, creating a pillar.

In normal field cutting of reinforced concrete, using standard AWJ equipment, it is common practice to cut through the concrete and then return to the locations of the now-exposed rebar to provide additional cutting time to cut through the bars. Alternatively, the concrete is cut in a single pass, with the traverse rate sufficiently slow to ensure the rebar is cut. These inefficient cutting techniques increase costs and spoils generation.

The Deep Kerf tool was designed to sense the presence of uncut rebar or other obstacles through impact. The system then provides additional cutting time to remove the obstacle. While this system works, it is inefficient. An AWJ cuts more efficiently if the jet is constrained by the sides of the kerf. In the case or remedial rebar cutting, all of the surrounding concrete has been removed, and the jet is unconstrained. A more efficient approach would be to sense the presence of the rebar during the normal cutting operation and then immediately adjust the cutting parameters to address the rebar. What is needed is a method of detecting the rebar during cutting.

Several possible means of detecting the presence of or contact with the rebar were considered. Any method that uses a signal derived from direct contact of the rebar with the nozzle was rejected. The severity of the cutting environment is such that any mechanical probes or fingers simply would not survive for any reasonable length of time.

Any solutions that use ferrous metal detectors such as eddy current coils or capacitance sensors were also rejected due to the need for wires running from the nozzle to a pair of slip rings and finally out to processing electronics. While such sensors would likely work, integrating them into the nozzle end and providing for signal wire routing was thought to result in a system too fragile for field operation. However, such an approach would likely be capable of providing a vivid indication of rebar presence.

The method chosen is based on the observation that there is a noise level difference when the impinging jet is cutting concrete as opposed to striking rebar. If such is the case, simple monitoring of the sounds of cutting should provide the sought after signal. This seemed to be a promising direction in which to proceed, and plans were made to make exhaustive studies of cutting sounds.

A test plan was made for rebar detection studies. It was decided that two basic sounds would be measured. The first basic sound was simply the sounds emanating from the slot being cut. A high-frequency-response microphone was placed 0.3 m (1 ft) away from the top of the concrete and 7 to 15 cm (3 to 6 in.) away from the rotating axis of the nozzle. The second sound was measured through the concrete itself. The signal was generated by a piezoelectric accelerometer cemented to the concrete. Both signals were recorded on high-frequency tape recorders for later analysis. The analysis consisted of operating on the signals with various filters and spectrum analyzers. An advantage of the tape was permanent storage of the experiments and the capability of playback at 1/10 speed. The reduced speed enabled the filters and analyzers to operate with more resolution and narrower bandwidths than the signals as presented.

The results of the rebar detection tests can be summarized as follows:

- Rebar can be detected when the nozzle head is rotating by noting changes in the acoustic spectrum of the accelerometer signal.
- The microphone provided no useful signal.
- Rebar cannot be detected when cutting narrow slots. It is too well supported to "ring" with its characteristic signature.
- There is a weak dependence of standoff distance and the acoustic spectrum.
- Simple filtering (low-pass, high-pass and bandpass) is all that is needed to detect the rebar automatically.

CLEANER/SCARIFIER TOOL

Equipment Design

The Cleaner/Scarifier tool removes material from a surface by rotating high-pressure waterjets at a radius, creating a circular pattern, and traversing the rotating mechanism across the surface to be cleaned or scarified. A photograph of the Cleaner/Scarifier tool is shown in Figure 5.

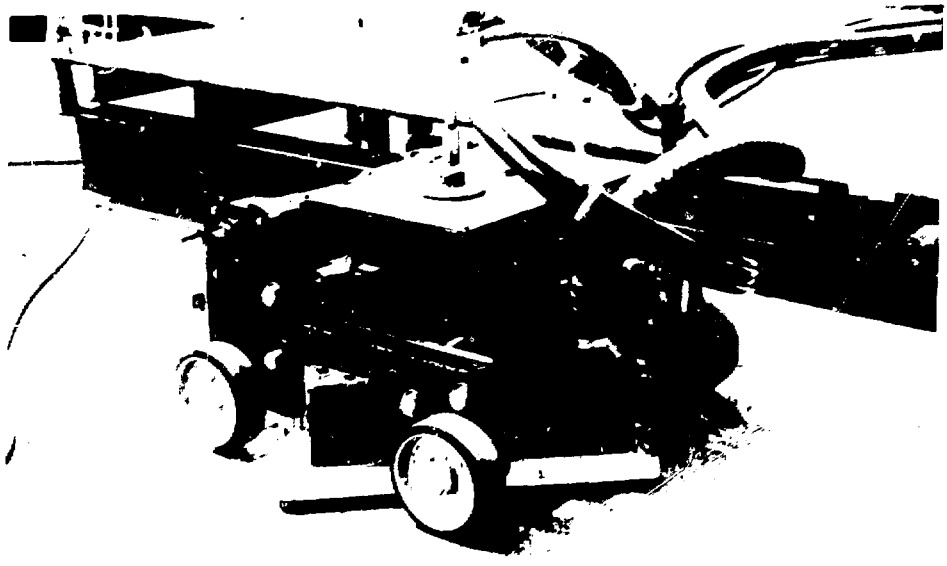


FIGURE 5. Waterjet Cleaner/Scarifier Tool

High-pressure water is fed to a central tube through a high-pressure swivel. The tube carries water through bearing supports to a manifold, which distributes the water to four arms leading to nozzle holders. Each nozzle holder has two nozzle ports that can accept nozzles of various sizes. The manifold can be rotated up to 1000 rpm.

The manifold and nozzles are encased in a steel box with a brush sealing system around its perimeter. A vacuum port and suction tube are located in one corner of the box for connection to the spoils collection system described earlier.

Parametric Testing

A series of parametric tests was conducted to determine the optimal configuration and settings for the Cleaner/Scarifier tool. The parameters varied included rotational speed, traverse speed, number of jets, size of jets, water pressure and target material.

The parameters described below produced light concrete scarification (approximately 3 mm of surface removal) and provided good cleaning of rust and paint from metal surfaces. The tool had four nozzle heads, each equipped with two nozzles. Two of the nozzle heads were mounted on a diameter of 350 mm, and the other two were mounted on a diameter of 250 mm. The outer jets were 0.25 mm in diameter, and the inner were 0.2 mm in diameter.

- Water Pressure: 241 MPa
- Water Flow Rate: 9.7 liter/min
- Jet Power: 38 kW
- Rotational Speed: 1000 rpm
- Traverse Speed: 20 mm/s
- Production Rate: 26 m^2/hr

Figures 6a and 6b show examples of a concrete surface and a rusted steel surface after operation of the tool.

Greater depths in concrete surface removal were accomplished by increasing the water flow rate, decreasing the rotational and traverse speeds and reducing the number of jets. Concrete was scarified to an average depth of 7 mm under the following conditions:

- Water Pressure: 241 MPa
- Jet Diameter: 0.356 mm
- Water Flow Rate: 11.5 liter/min
- Jet Power: 45.5 kW
- Rotational Speed: 50 rpm
- Traverse Speed: 8.5 mm/s
- Production Rate: 11 m^2/hr

Figure 7 shows an example of this depth of surface removal on very hard concrete (69 MPa compressive strength). The Cleaner/Scarifier vacuum system did not provide enough air velocity to remove the pebbles on horizontal surfaces. Even after being freed of the surrounding grout, they tended to remain in place, protecting the underlying concrete from attack. After slow passage of the tool, the concrete surface was covered with loose, well-cleaned pebbles.

The Cleaner/Scarifier tool was used on vertical surfaces to test the ability of the shroud to contain and remove the water and spoils. On both steel and concrete, the tool left the surfaces nearly dry and created no observable mist or spray.

FIELD DEMONSTRATION AT WEST VALLEY NUCLEAR FACILITY

The Deep Kerf system and the Cleaner/Scarifier tool were taken to the West Valley Nuclear Facility in West Valley, New York, to demonstrate their capabilities. The test specimen for the deep kerfing was a concrete slab that measured approximately 1.5 x 1.5 x 0.3 m. The slab was set upright on one edge with the Deep Kerf tool mounted above it. The test cutting was done vertically into the block. A kerf measuring 1.2 m deep and 1 m long was cut into the block. This kerf was made at an average rate of 0.46 m^2/hr.

The Cleaner/Scarifier tool was demonstrated on a rusty steel surface, a painted steel surface, and a painted concrete slab. Additionally, two stripes of duct tape were placed on the steel surface for removal. The majority of the tests were conducted under the following conditions:

o	Water Pressure:	241 MPa
o	Flow Rate:	9.8 liter/min
o	Jet Hydraulic Horse Power:	39 kW
o	Rotational Speed:	1000 rpm
o	Number of Jets:	8, 2 jets per arm
o	Translational Speed:	20 mm/s
o	Production Rate:	26 m^2/hr

Under the above conditions, the steel surface was cleaned of rust and paint, and the duct tape was removed. The concrete was cleaned of paint, and the concrete surface was removed to a depth of 3 mm in the center of the swath to 4.7 mm at the outer edge of the swath.

CONCLUSIONS

The following conclusions can be made from the results of the study presented in this paper:

- Deep kerfing with abrasive-waterjets is technically, economically and environmentally feasible for many nuclear decommissioning applications. Kerfing rates up to 0.6 m^2/hr were achieved and can be improved upon by further optimization.
- The recycling of abrasives is technically feasible for steel grit or shot. However, the economic feasibility depends upon the cost of disposal. An economic evaluation showed that recycling will be feasible if the cost of disposal is greater than \$635/m^3.
- Rebar detection can be accomplished with either an obstacle detection arrangement or through forecasting based on noise level changes. Both methods were demonstrated, and the latter proved more efficient.
- The cleaning of metal surfaces and shallow scarification of concrete were demonstrated with a lawn mower-like waterjet system. Rates of cleaning of around 33 m^2/hr were achieved, while scarification of concrete to 7-mm depths can be accomplished at rates of 11 m^2/hr.
- The containment and catching efficiencies of the Deep Kerf tool and Cleaner/Scarifier tool are greater than 99%. However, monitoring of the surrounding air during operation on radioactively contaminated material remains to be conducted.
- Further work is needed on hardware improvement regarding:
  - Wear of mixing nozzles
  - Reduced stem diameter versus stiffness
  - Quick change of mixing nozzles
  - Development of high-pressure fan jets for rapid surface cleaning.

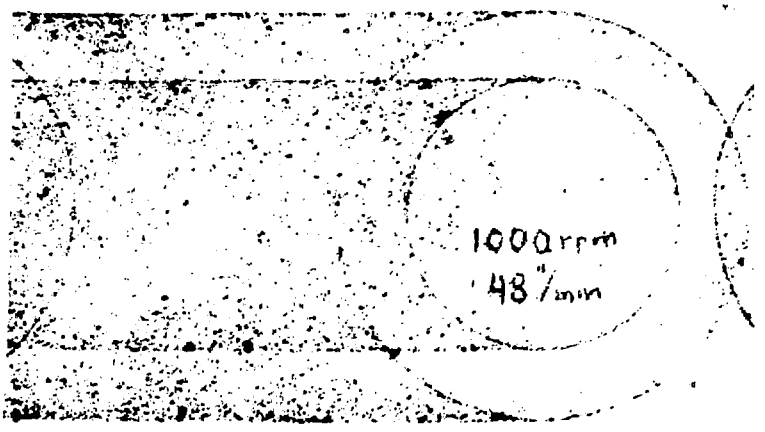
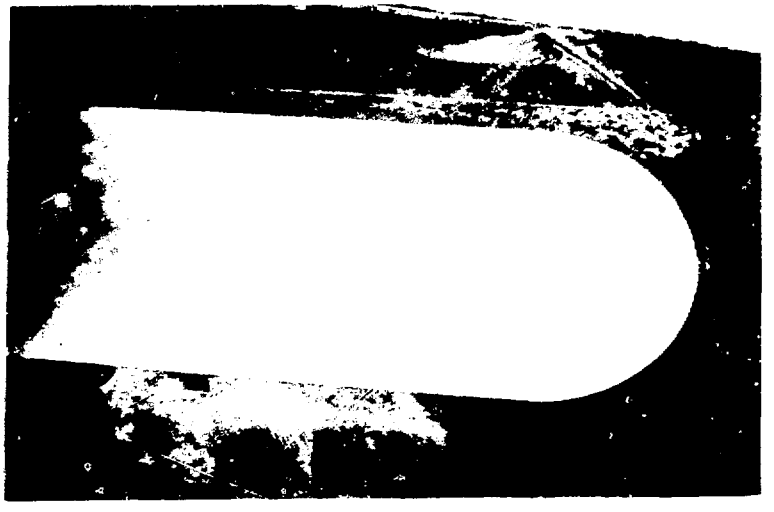


FIGURE 6. Surfaces Cleaned with Cleaner/ Scarifier Tool

a. Concrete



b. Steel

FIGURE 7. Concrete Surface Scarified to a Depth of 7 mm

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SECTION VI

CHANGEOUT OF LARGE COMPONENTS/SYSTEMS EXPERIENCE APPLICABLE TO DECOMMISSIONING

Investigations Related to a One-Piece Removal of the Reactor Block in the Frame of the JRR-3 Reconstruction Program

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ABSTRACT

In the Japan Atomic Energy Research Institute (JAERI), an outdated research reactor (Japan Research Reactor NO.3; JRR-3) was removed to a storage facility between October 14th and November 7th, 1986. The removal of the 2250-ton reactor block (10×10×10 m) was performed as a part of a program to replace the JRR-3's core (10-MW thermal) with an upgraded research reactor core.

The heavy-water and fuel elements were taken out from the JRR-3 before removal work began. The reactor block was raised about 3.7 meters, using a 12-cubic meter steel frame and a center-hole jack system. The reactor block was then transported horizontally about 34 meters on steel rails, using four 100-ton jacks, to a storage facility. Finally, the reactor block was lowered 14 meters into the storage facility.

After the reactor block is stored, a new 20-MW thermal, light-water moderated and cooled JRR-3 core will be built, with criticality targeted for 1989.

INTRODUCTION

The JRR-3 started operation as a research reactor in 1962 and was the first domestically built reactor in Japan. The research reactor, with a thermal output of 10-MW was used in performing neutron beam experiments, material irradiation and other purposes. Recently, because it had become functionally difficult to adapt the reactor to various research uses, the reactor operations were suspended for reconstruction work in 1983.

This removal work involved leaving the reactor building as it was and removing all major components, including the reactor core. The work involved cutting the 2250-ton reactor block away from the reactor building structure, raising it above the floor using a steel frame and then moving it 34 meters, and storing it in a storage facility.

The entire one-piece reactor removal method used in the removal work has many benefits, including reduced radiation exposure to workers and minimized waste production. The technical knowledge related to this removal process will be useful in future reactor removal work.

Safety was the most important factor in this removal process, and was confirmed through the use of several earthquake and other accident simulation test. Also, half scale mock-up experiments were used to test and evaluate various removal processes before actual work began.

After the JRR-3 is removed, a new 20-MW light-water moderated and cooled, and high-performance reactor, which consumes 20% enriched uranium, will be targeted for criticality in 1989.

REMOVAL PLANNING

Basic Planning

The reactor block to be removed has fairly high radio-activated substances around the core even after the removal of the fuel elements and heavy-water from the reactor core. Many reactor removal methods have been studied for the JRR-3. Each has considered some following conditions:

① to prevent the public from radio-active pollution
② to reuse the reactor building structure
③ to use only proven technologies
④ to confirm the safety of the romoval method beforehand

The results of these studies, based on the above, led us to select the so-called "One-piece removal method" as the most effective technique.

Outline of One-Piece Removal Method

The "one-piece removal system" is designed to remove the whole reactor structure as one block and permanently store it, under surveillance, in a storage facility. Walls and slabs of heavy-weight concrete, surrounding the reactor for radiation shielding, remain as permanent structures after the core removal. In other words, the following results can be achieved:

① Removal work can be performed without any contact with the reactor highly radio-activated core.
② The biological shield walls can be used as a temporary cask in the transportation of the reactor core.
③ The biological shield walls can also be used permanently as a surveillance barrier.

The removal process incorporates many conventional and widely used technologies with special precautions taken in order to control the radio-active areas.

The method and sequence are as follows:

① Remove fuel assemblies and heavy-water coolant, which is followed by the decontamination of the reactor.
② Prepare temporary structures to support the reactor weight during transportation.
③ Cut the reactor block away from the main structure of the reactor building.
④ Raise the 2250-ton reactor block, using the center-hole jack system, to an over-floor height for horizontal transportation.
⑤ Transport the reactor block horizontally by roller system for a distance of 34 m. The transportation rate is automatically controlled by lateral jacks (100-ton capacity each) and moving direction is monitored using laser theodolite.
⑥ Lower the reactor into the storage facility using the center-hole jack

system.

Fig. 1 shows a sketch of the concept. The following are the main advantages of this method.

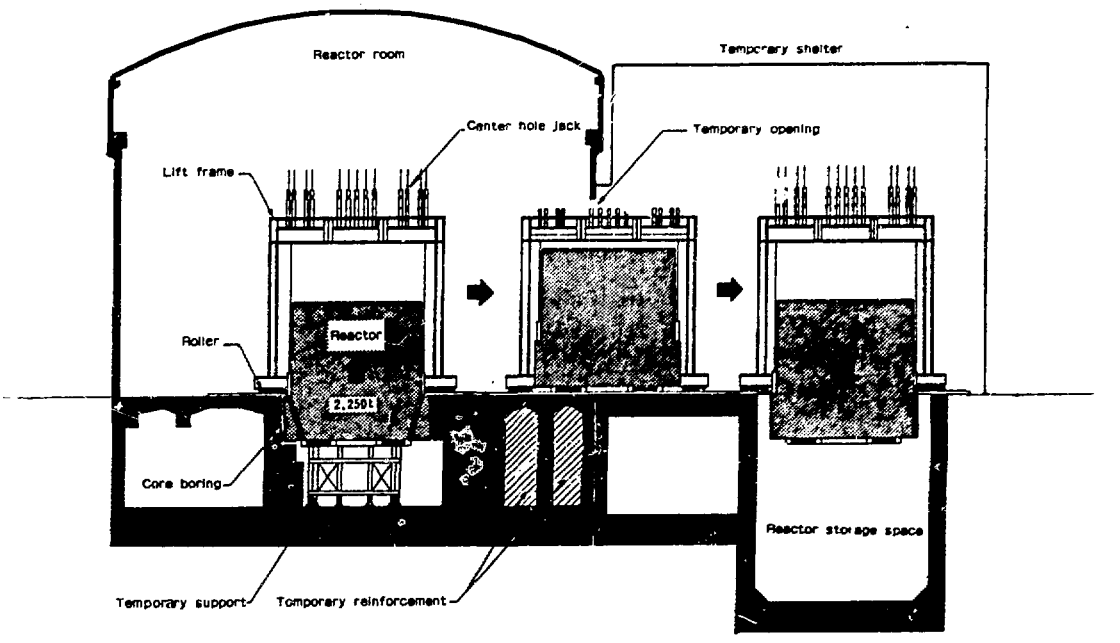


Fig. 1 Sketch of the Concept (Cross Section)

① Minimized environmental pollution.
② Minimized radiation exposure.
③ Negligible radio-active waste.
④ Simplified waste disposal.
⑤ Reusable reactor building structure.
⑥ Simple system using all reliable and proven technologies.
⑦ Cost and time efficiency.

Design Process

The basic concepts of system design and structural design are as follows.

① The "fail-safe" concept is used in the center-hole jack system. When one jack breaks down, the reactor will still be supported by the other jacks. The load is immediately redistributed evenly among the remaining jacks.

② The spring dampers are installed between the steel frame and the reactor block to prevent the reactor from excessive shaking during transportation process.

③ The safety factor for temporary steel structures should always be more than 3.0.

④ The base shear factor, for this steel frame, for the seismic force is 0.2, and 0.1 for the reactor block during suspended. (The base shear factor 0.2 is based on the code applied to ordinary structural design in Japan.)

⑤ The impact factor, applied to the center-hole jack design, during "jack-up" and "jack-down" should be 0.2.

According to these basic concepts, the basic design and feasibility design are completed. Then confirmation test is carried out using a half-scale "mock-up" model. Since this removal is the first of its kind, there are no established standards or codes to be followed. Therefore, the criteria used in the structural design depends on the test results.

Confirmation Test

All the techniques to be used for the removal work are conventional, however, quite a few data are available for planning this work. Therefore, a confirmation test using a half-scale model is performed before the actual removal.

The objective of this test is to confirm if our conceptual method would work well, and to obtain some data for the actual work. During the test, a concrete slab with a dummy weight is separated from the concrete structure, raised, transported and lowered as actual would be.

The steel suspension frame is about a half of the actual size, and the 280-ton dummy weight simulating the reactor block is 1/8 of the actual weight. The confirmation test is performed between February and July 1983. Fig. 2 shows a sketch of the confirmation test.

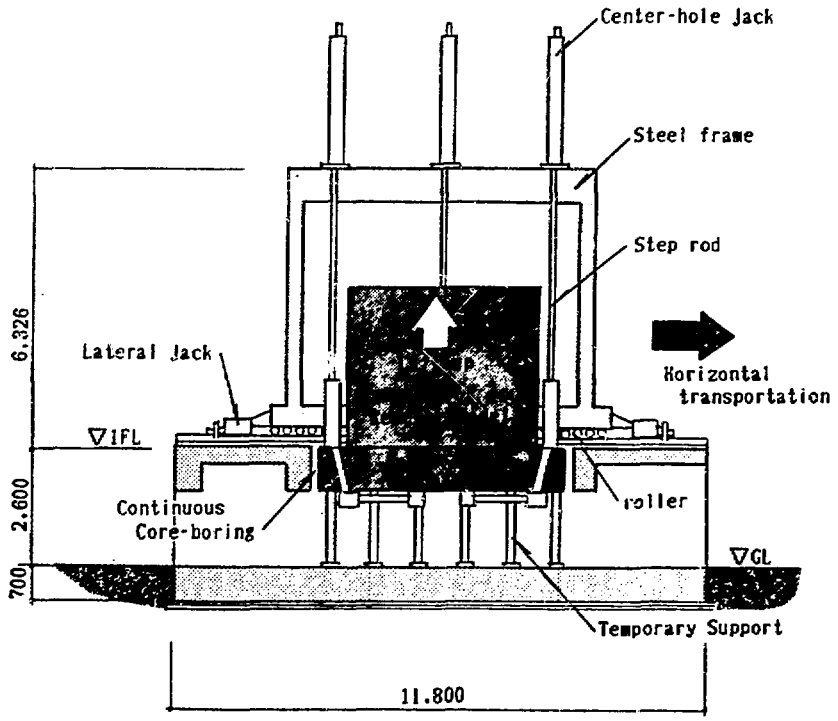


Fig. 2 Confirmation Test (1/2 Scale Model)

The main test items are as follows:

① to evaluate working efficiency of core-boring.
② to monitor load value on the temporary supports when the reactor block is separated from tne reactor building structure.
③ to monitor impact of the center-hole jacks during jack operation.
④ to comfirm "fail-safe" concept by simulating pipe-break down accident of the center-hole jack system.
⑤ to gauge step-rod strength in tension and in bending.
⑥ to confirm the efficiency of the load control system.
⑦ to measure the deviation value from the orbit during horizontal transporation.

The test results were all useful for actual work, and some of them were eventually reflected in actual system design.

REMOVAL WORK METHOD

Work Process

First, the storage facility made of reinforced concrete is constructed underground, and over the facility a temporary shelter to prevent leakage is provided. Under the reactor block, a temporary-support system to recieve the reactor weight during the separation is prepared.

Rail beds and temporary reinforced concrete walls to support the reactor block weight during the removal are built at the same time. On the rail beds, the rails and rollers to remove the reactor block are installed, and then a steel frame to suspend the reactor block is assembled over them. The center-hole jacks to raise the reactor block are attached on the top of the steel frame and the reactor block is suspended by step-rods, using a coupling-and-anchoring system.

When the above is prepared, core-boring process to separate the reactor block from the building begins. After the separation, the reactor block is raised about 3.7 m using center-hole jacks. Then the reactor block is transported out of the reactor building, passing through the temporary opening. The reactor block travels 34 m, and arrives at the temporary shelter over the storage facility. The steel frame carrying the reactor block is fixed by stoppers, and the reactor block is lowered to the final position for permanent storage, then the top opening is closed with reinforced concrete.

Among these processes, the reactor block separation process and the reactor block transportation process (both vertical and horizontal) are the most singnificant processes to be controlled strictly. Details are as follows. Fig. 3 and Fig. 4 show sketches of the concept.

Reactor Separation

The reactor block (the reactor, shield walls and 3 m-thick slabs) is separated from the main structure of the reactor building, using the core-boring method. The separation is performed both horizontally and vertically. Before the separation, temporary supports are installed under the slab structure, which is the deepest part of the removal block, in order to support the separated reactor block weight (2250 tons) from the beginning of the core-boring process to the commencement of the jack-up process.

The temporary supports were already experienced in the confirmation test.

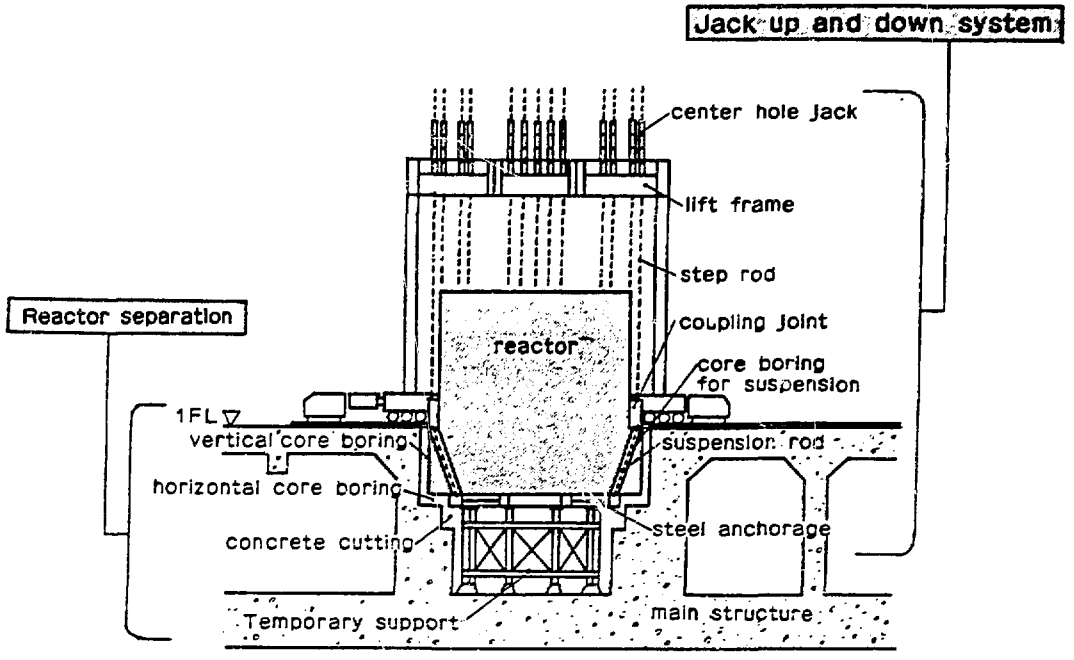


Fig. 3 Sketch of the System

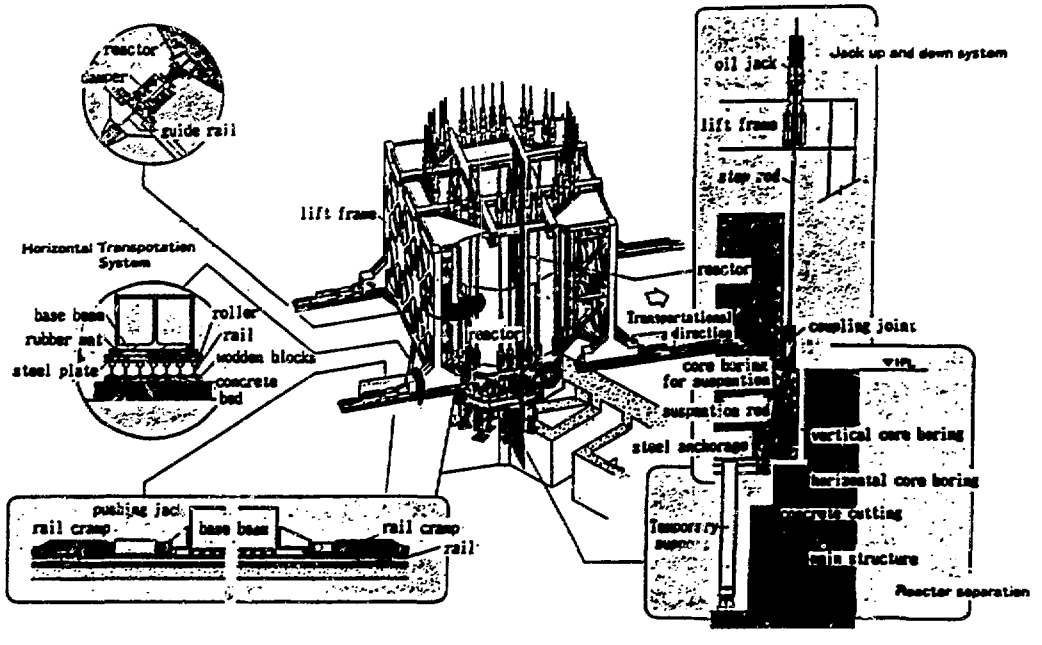
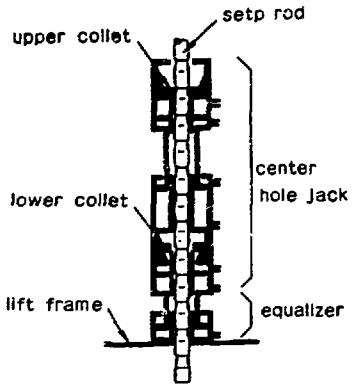


Fig. 4 Detail of the One-Piece Removal System

Monitored load values, increasing gradually in accordance as separated length, showed the efficiency and there occured no harmful impact force.

Jack-up and Jack-down

The separated reactor block is raised and lowered by center-hole jacks (36 jacks of 100-ton capacity each) which have a well arranged load control mechanism and many experiences for jacking-up and jacking-down of heavy objects.



load of equalizer(No1) increase

⇩

oil of equalizer(No1) flow to subcylinder(No I)

⇩

lower oil of subcylinder(No1) flow to other subcylinders (No II～VI)

⇩

equalizer stroke(2～12) get longer

⇩

load is equalized

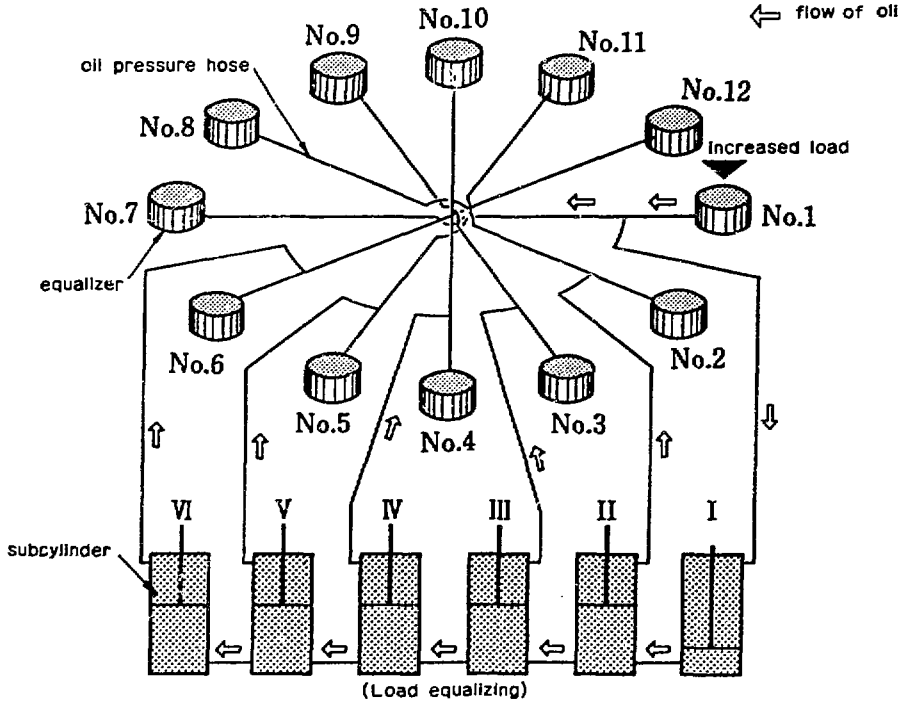


Fig. 5 Load Equalizing System

Because the reactor block is a rigid body, a load equalizing system, shown in Fig. 5 is employed for distributing load to each jack uniformly. The fail-safe concept is embodied by grouping the center-hole jacks into 6 systems placed in symmetry. The safety of the system is thus secured even if one system breaks down.

This center-hole jack system consists of many steel structures. Among these, the step-rod is the most important device to be designed carefully, excluding center-hole jack. The step-rod, a steel rod which has a notch to be bited by collet in the center-hole jack, has been chosen as a most critical failure mode device, because of its notch and its high-tension material. Therefore detailed analysis is conducted about the step-rod strength.

The lift-up device collet holds the reactor block weight, biting the notch of the step-rod. The first experiment of the step-rod strength does not consider this loading condition around notch. So, a detailed analysis using FEM (Finite Element Method) is conducted to check the concentrated stress, and it is found that the step-rod notch stress level is very large.

This analysis is conducted using the assumption that the stress-strain relation is linear (only elastic range), but actual stress-strain relation is bi-linear and the stress would be redistributed. For this reason, a loading test and a fatigue test are conducted in order to assure its safety.

A loading test (simulating the actual loading condition using collet) and a fatigue test (0-80 ton, 4500 times) are done using three test pieces each. And the criterion is that the step-rod safety factor should be more than three. And all test results satisfies this criterion.

During the actual work, nondestructive inspections were carried out to assure the strength of the step-rod. First, spectrum analysis testing was performed to screen inconsistent rods. Secondly, magnetic particle testing was performed to scan for cracks. In the actual removal process, both of these test were satisfactory and no problems occured during the reactor removal.

Horizontal Transportation of Reactor Block

The suspended reactor block in the steel frame is horizontally transported by using rails, rollers and lateral jacks, which make it easy to correct the transportation direction and have many accomplishments proved to be economical. The strokes of the lateral jacks are automatically controlled while the reactor block is moved. The steel frame moving direction is also checked, using a laser theodolite.

As measures against the inertia force during horizontal transportation or against the shaking caused by an earthquake, four spring dampers are installed between the reactor block and the steel frame.

In roller design, strength criterion is usually determined according to the experiment results and job experiences. And each roller is expected to carry the road evenly, but several rollers do not because of the gap between the steel base (a member of steel frame) and rollers. When the stress is calculated excluding free rollers (because of the gap, these free rollers would not support the reactor block weight), it would exceed the strength criterion. In this reason, a test is performed using the same size test piece.

The test resulted that no harmful roller deformation occurs and the safety factor is more than four in ordinary loading condition.

MEASURES FOR SAFETY

Many measures are employed to this system. For example, the load equalizing system and the measures against earthquake are shown below.

Load Equalizing System

In order to minimize overload to the step-rods which suspend the reactor block, a load equalizing system is employed to distribute the load uniformly to each jack. The fail-safe concept is also considered for the center-hole jack system. Even if one jack would break down, the reactor will still be supported by the other group jacks, and the unbalanced load is redistributed symmetrically and immediately.

It is found in the confirmation test that the maximum step-rod force is recorded 1.6 times of the average axial force. In the basic design, it is assumed that the maximum axial force would be 1.2 times of average value. So the improvement of this equalizing system has to be finished before the actual work starts.

To reduce the axial force's deviation from the average value, the sub-sylinder system is employed. After loading test, it is confirmed that the improved system would work well.

Measures against Earthquake

The temporary structures, which are used during the removal period, are designed for many loading conditions including seismic force. But there are no standards or codes to be applied to this seismic design.

A seismic analysis of the JRR-3 reactor building structures is done in order to find out the base shear factor. In this analysis, the seismic wave (acceleration level is 50-70 gal at ground surface) is used, and the base shear factor is calculated as 0.2. This is the same value of the Japanese building code which is used in ordinary building seismic design.

So temporary structures are designed using this seismic force (base shear factor 0.2), and the strength criterion in seismic condition is decided to be under yielding stress.

The spring dampers are installed between the steel frame and the reactor block to prevent the reactor block from excessive shaking during transportation. The spring damper consists of several corned-springs, and the spring stiffness is decided according to the seismic analysis results.

REMOVAL WORK RESULTS

Removal of the Reactor Cooling System

In order to remove the reactor block, the cooling system must be drained, disconnected, and sealed before other removal steps can be taken. Once the heavy-water coolant has been drained and the system being rinsed with light water, the piping is cut away from the reactor block, using suitable tools, for example a saw, under wet condition. The pipes are then completely filled with resin and sealed with steel alloy plates on the outside wall of the reactor block. At this point, the reactor block is ready for other removal processes.

Reactor Block Separation

For horizontal concrete cutting, three core-boring machines were symmetrically placed and operated, with thorough control for cutting depth. Horizontal core-boring of approximately 210 meters long in total was completed in 11 days. The cooling water was recirculated to reduce an amount of radio-active wastes to be processed, and only the slurry in the water was deposited in the storage tank.

The reactor block was vertically cut by continuous core-boring of the 3 m-thick floor around the reactor block. The vertical cutting was performed by 7 core-boring machines with good balance by monitoring the load transfer to the temporary supports. It took nearly a month to complete 259 core-borings of approximately 820 meters long in total.

During these separation, there occured no harmful cracks and deformations.

Jacking-up and Jacking-down of Reactor Block

The reactor block weighting nearly 2,250 tons was raised 3.7 meters in a time as long as approximately 15 hours. The jacking-up was performed by controlling the strokes of all jacks, equallizers and subcylinders which were the devices to distribute load to each jack uniformly and by monitoring the axial force of 12 representative step-rods.

The reactor block was lowered 13.5 m, under control similar to the jacking-up process, into the storage facility in three days.

In actual removal, jack-up and jack-down were performed using this system, and the axial forces were monitored by means of strain gauges and a pen-recorder. Each axial force's deviation was within 10 % of the average when the jacks were working, making the suspended reactor very stable with small fluctuation.

Horizontal Transportation of the Reactor Block

The suspended reactor block together with the steel frame (approximately 2500 tons in total weight) was horizontally transported 33.6 m to the place above the storage facility in 7 days at speed as slow as about 5 meters a day. During the horizontal transportation, the propelling force and the strokes of lateral jacks at right and left were controlled and the straightness of the steel frame was monitored by laser theodolite.

The transport direction was corrected by means of adjustment of the propelling force of lateral jacks at right and left and by means of correction of roller direction. The reactor block was transported accurately to the fixed location.

The propelling force required for pushing the reactor block and the steel frame was about 80 tons and the coefficient of rolling friction of the rollers was 0.128 cm.

Although small earthquakes occured during the actual work, shaking of the reactor was very small.

The radiation exposure to workers was negligible during the removal work of the reactor block.

CONCLUSION

In order to update the JRR-3 reactor with current technologies, decision

was made to remove the outdated reactor, equipment and piping, and to install a new reactor.

With this decision, a one-piece removal system design was necessary. The method needed to be efficient and safe, and this has been proved with the successful completion of the JRR-3 removal project-the first of this kind in the world.

ACKNOWLEDGEMENTS

The authors would like to express their appreciation to the staffs in the Research Reactor Development Division of Tokai Research Establishment of JAERI, and the staffs of Nuclear Power Division of Shimizu Construction Co., Ltd.

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LARGE SCALE FUEL CHANNEL REPLACEMENT PROGRAM

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The CANDU reactor, Figure 1, has been designed so that any or all of the components which make up the reactor channels can be removed and replaced. It has always been expected that at least once during the lifetime of a CANDU nuclear power station, the reactor channels would require replacement, and that this feature of being able to replace the reactor channels may allow extension of the life of a CANDU nuclear station far beyond the normal amortization life. The components of the reactor channel are shown in Figure 2.

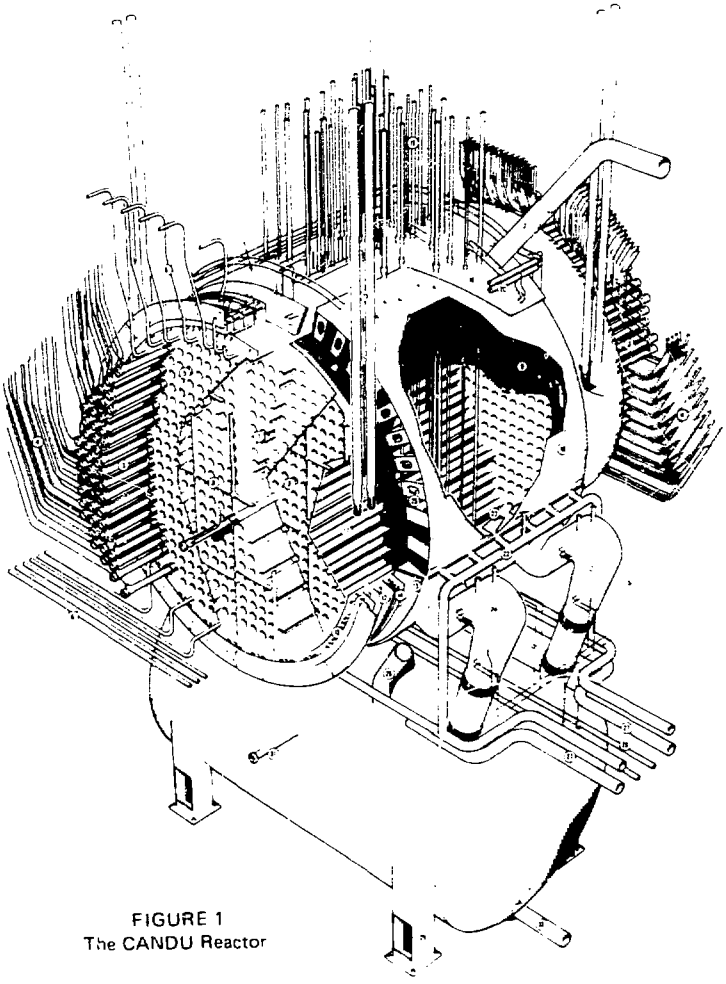

FIGURE 1
The CANDU Reactor

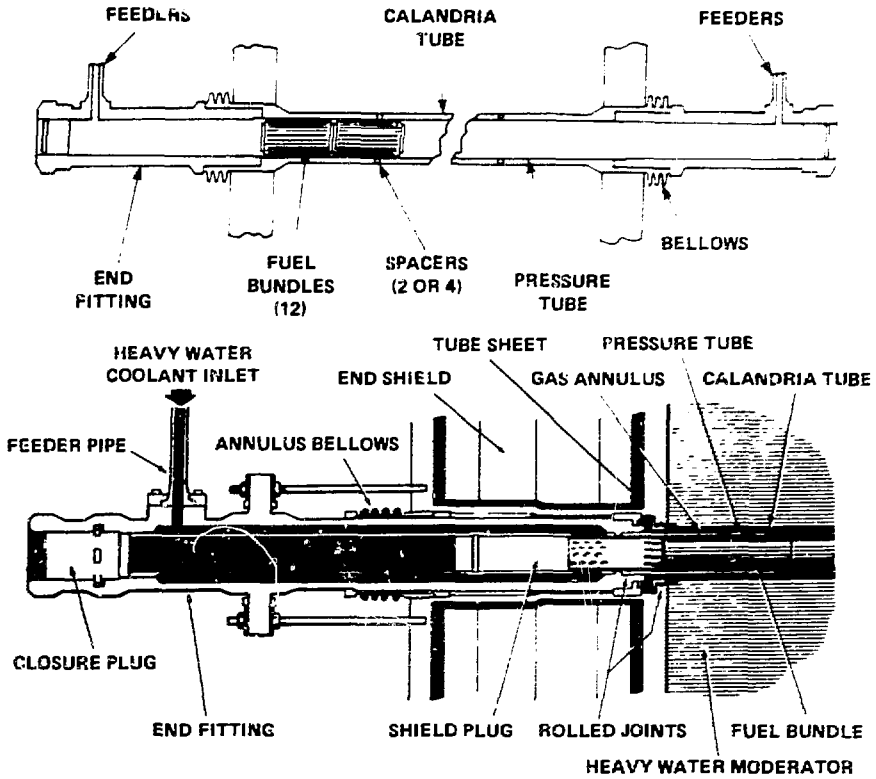

FIGURE 2
REACTOR CHANNEL

Preliminary planning to develop capability to retube an entire reactor began in 1975, when it was first realized that the pressure tubes in Pickering were growing axially at a greater rate than had been anticipated during the design of the station. Up to that time, replacement of individual pressure tubes in Pickering had been successfully undertaken following leakage through tiny cracks which developed in the tube walls at the rolled joint in a number of channels due to improper rolling in procedures. This retubing concept involved an articulated arm operated remotely and was called the Large Scale Fuel Channel Replacement Program (LSFCRP).

On August 1, 1983, a pressure tube failed in Pickering A, Unit 2, releasing coolant to the reactor vault, and leading to an orderly shutdown and depressurization of the unit by the station operators. The failed tube had a crack along the bottom stretching from the end fitting to about two meters along the channel. A team of operating staff, research scientists, design engineers from Ontario Hydro and Atomic Energy of Canada Limited (AECL), were quickly assembled to identify the cause of the failure and to plan and recommend remedial action. The subsequent investigation of units 1 and 2 revealed that hydrogen concentration had built up in the wall at points along the bottom of a number of the Zircaloy pressure tubes and that this concentration had lead to the failure.

After eight months of intensive investigation involving site inspections of a number of channels in both units 1 and 2, laboratory work in hot cells at Chalk River, detailed analysis and tests and careful safety and materials studies within Ontario Hydro and AECL, it was decided that both units should be retubed before they were returned to power. Both units 1 and 2 had completed about the same number of full power operating hours and both were fitted with Zircaloy 2 pressure tubes. All other CANDU reactors built after Pickering unit 2 were constructed using Zirconium-Niobium alloy pressure tubes which is not as susceptible to the build up of hydrogen and zirconium-hydride under the same conditions as existed in the Pickering unit 2 tube which failed.

It was a difficult decision. While we had done preliminary planning and engineering and development work on equipment and facilities for full scale retubing, we were not well prepared to take on, for the first time, the full scale retubing of a reactor on such short notice, and we were faced with the task of taking on two units immediately.

In the late fall of 1983 when it appeared that at least one reactor may have to be retubed we began studies on the options available to us.

RETUBE STUDY

The Retube Task Group set out to determine if it was feasible to proceed with the first stages of the disassembly manually, with the provision of phasing in a remote retubing scheme at an appropriate time, if necessary. The Task Group showed that an early start to retubing was indeed feasible, if a remote operated Retubing Tool Carrier (RTC), Figure 3, was utilized in a shielding cabinet This RTC, operated by remote control from outside the vault, would convey radioactive components from the shielding cabinet at the reactor face to a shielding flask on the vault floor. With the RTC concept under detailed engineering, the Task Group extended their studies to develop a retubing scheme that did not include removal and replacement of the calandria tube, since it was felt at that time that retubing after only 12 years of reactor service without removing the calandria tube was the most logical and attractive approach.

The results of this study, which was completed in the first quarter of 1984, showed that an early start on the retubing was indeed practical and in fact, by using the shielding cabinets, remote viewing and manually operated tools, that complete retubing in this manner could be undertaken in stages on one reactor in about 30 months. The second reactor would follow the first by about six months.

The Board of Directors of Ontario Hydro made a decision at their March 1984 meeting to proceed immediately with the above plan to retube units 1 and 2 at Pickering.

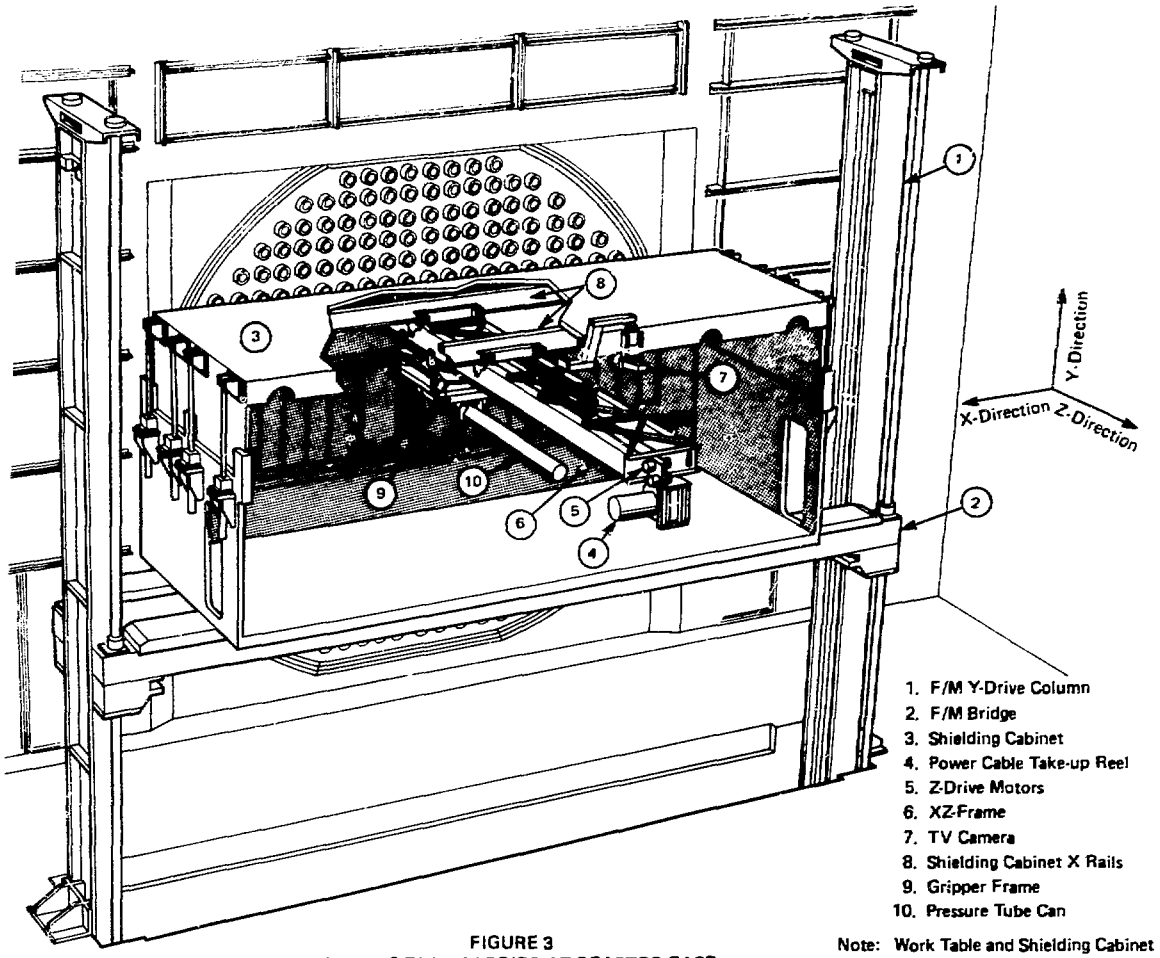

FIGURE 3
RETUBING TOOL CARRIER AT REACTOR FACE

RETUBE PROGRAM

The Retube Program began shortly after the March 1984, at which time unit 2 had been down for 8 months and unit 1 had been down for about 4 months.

The main steps taken from this point onward in this Retubing Program and the particular tools or equipment employed are as follows:

1. De-fuelling

All CANDU stations are fuelled on-power using two remotely operated fuelling machines. Each machine attaches to an end fitting of the same channel, makes a seal capable of withstanding full heat transport system pressure, removes the sealing plug and shielding plug which normally resides in the end of the channel, and then replaces a given number of the 12 irradiated fuel bundles in the channel with new bundles. The shielding

and sealing plugs are then re-installed and the machine is removed. De-fuelling of Pickering Units 1 and 2 was done by the operations staff using the on-power fuelling machine, but with the heat transport system full of cool and depressurized heavy water. To remove the 12 fuel bundles from all the 390 channels in Unit 1 it took 60 days and 65 days to remove all the fuel from Unit 2.

2. Decontamination

A major concern in extensive maintenance and repair of any nuclear reactor is the radiation exposure to the forces undertaking the work. Special tools, training and rehearsal facilities, protective equipment, and well developed procedures were all part of the planning on this retube project to minimize the exposure to the workers.

In addition it was recognized that if the radiation fields could be reduced significantly at the reactor face that the work could be done with lower exposure and fewer costly and time consuming protective measures. The heat transport systems of each of the reactors were decontaminated using the CAN-DECON process developed jointly by AECL and Ontario Hydro to reduce radiation fields in the reactor coolant systems of CANDU reactors. Unit 2 was decontaminated twice, once in January 1984 during the investigation phase and a second time in April 1984. Overall decontamination factors at the reactor face of up to ten were achieved, lowering the radiation fields to 100 to 140 mR/hour.

Unit 1 was decontaminated in May of 1984, resulting in even lower residual fields of from 40 to 110 mR/hour.

These reductions in the radiation fields at the reactor face where a substantial number of man hours of work were required in the manual/semi-manual process were very important to the success of this process. With these reductions in the fields at the face and the use of the steel shielding cabinets it has been possible to have workers undertake many operations manually while staying below an acceptable total man-Rem exposure level.

3. Draining And Drying The Heat Transport System

The reactor coolant in CANDU reactors is heavy water which when it is irradiated contains tritium, a radioactive isotope of hydrogen. Any leakage or spill of this coolant during the retubing operation would raise the airborne tritium level in the area where work has to be performed and could result in an increase in the internal radiation dose to the workers involved in the retubing. Removal of as much of this water as possible was considered desirable. The reactor coolant system was drained and then flushed with clean demineralized ordinary water. After draining this ordinary water to the maximum degree the piping within the feeder cabinets was heated by electric heaters and the reactor coolant system was vacuum

dried. This provided a dry and relatively contamination free reactor coolant system, and has minimized the tritium exposure to workers during the retubing.

4. Rapid Induction Heating Of Rolled Joints

The retubing plan was to retain the east end fitting in place. This required the pressure tube rolled joint to be separated in such a manner that the end fitting did not suffer any damage. A rapid induction heating technique developed by the Ontario Hydro Research Division made this possible, Figure 4.

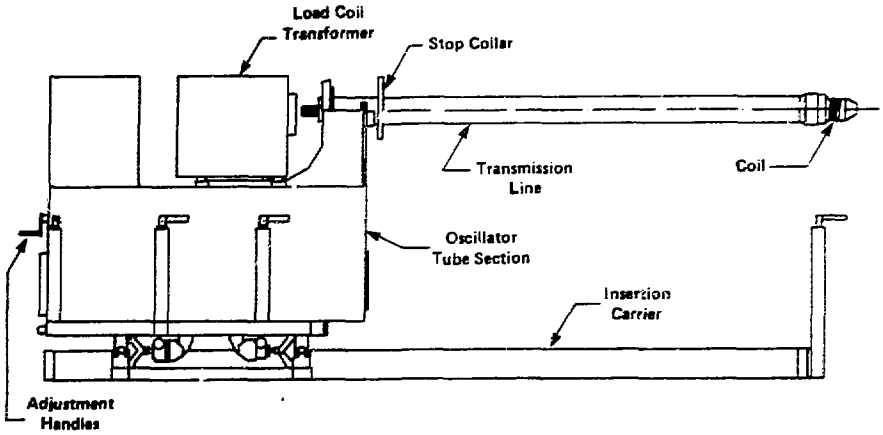

FIGURE 4
RAPID INDUCTION HEATING TECHNIQUE

In this process the inside surface of the pressure tube is heated up rapidly by induction heating from a coil placed inside the pressure tube at the rolled joint carrying a surge of current. The inside of the pressure tube tries to expand but is prevented from doing so by the cold heavy end fitting, which results in high local stresses in the pressure tube. As the pressure tube cools, it contracts to release these stresses and shrinks the outside diameter of the pressure tube. This process leaves an undamaged end fitting inside surface ready, with some minor burnishing, for installation of the new tube.

5. Retubing Tool Carrier (RTC)

The Retubing Tool Carrier (RTC) shown in Figure 5 mounted on the inside roof of the West shielding cabinet at the reactor face, was designed to remotely move materials axially, radially and vertically to transport radioactive components from the reactor face area to a trough of a loading device which sits on the vault floor, and which pushes articles in the trough into a shielding flask previously lined up with the trough.

FIGURE 5
RETUBING TOOL CARRIER AT REACTOR FACE

The RTC is a specially designed, gantry mounted manipulator able to operate as a wire rope hoist or as a rigid support. It is operated in a local manual mode while handling non-radioactive materials. When handling radioactive materials, control is from a remote panel located in the retubing control centre. There is feedback via closed circuit video on the control panel.

The main tasks visualized initially for the RTC were removal of the inboard stubs of the end fitting and the pressure tubes in waste cans. These would be moved back axially by the RTC, rotated and lowered to the trough on the flask loading device. However, the RTC proved to be so versatile that it was used to transport almost all the equipment from the vault floor to the shielding cabinet on the west side of the reactors.

6. Tooling

All tools were designed to minimize radiation exposure by self shielding where possible and by simplicity of design. The design work to meet the requirements of shielding, simplicity, effectiveness and flexibility required close cooperation between the research, design, project engineering, and the operations and construction staff.

RETUBE ORGANIZATION

The retubing of the two Pickering units was a major new undertaking for Ontario Hydro involving innovation in design and development and testing of equipment, the training of workers to perform unfamiliar tasks in plastic suits, the execution of repetitive work in the middle of an operating station, the management of a large work force made up of construction and operations staff and trades, and involving a radioactive environment and components.

Ontario Hydro has extensive experience and capability in the design and construction and operation and maintenance of nuclear power stations. However, a unique arrangement was called for in this instance. We established a project team of 55 technical staff at the site, drawing on experienced staff from design, construction and operations organizations. This team comprised:

- 10 design engineers
- 30 construction engineers and technicians
- 15 operations engineers

The field forces performed the retubing activities using the tooling, training and procedures provided by the project team. These procedures integrated Operations standards of radiation safety and dose reduction with Construction quality and production control. The integration of these capabilities into a combined workforce with one set of procedures was an important factor in the successful management of this project.

REHEARSAL FACILITY AND MAINTENANCE OF TOOLS

Ontario Hydro took the decision to build a replicated rehearsal facility in a warehouse on site, Figure 6, to train all workers involved in every action to remove and replace pressure tubes. This process included signals to cause quality engineering checks, radiation protection procedures and physical options to resolve on-core difficulties. A full-scale mock-up of the core was built to properly rehearse procedures, and included the fuelling machine bridge, shielding cabinet and tooling to be used. All workers were fully rehearsed many times, in radiation protection to ensure confidence on the job.

Also, the importance of tool maintenance and decontamination for use when needed was recognized. A dedicated team and workshop was set up for this purpose. Additionally, the Head Office project group, with support from the supply industry, provided timely spare equipment so that delays in back up equipment was minimized. Equipment was provided as needed by the schedule of core-face activities.

FIGURE 6
MOCK-UP OF RETUBING TOOL CARRIER
IN WAREHOUSE AT PICKERING NGS

RETUBING OF PICKERING A UNITS 1 AND 2

The actual work, which consisted of about 40 different operations, can be grouped into four phases: preparation, removal of reactor components, re-assembly and re-commissioning Throughout each of these phases, the project team, supported by specialists where required, provided direction and management of the retubing program, with major emphasis on radiation dose control.

1. Preparation of the Units

The 4680 fuel bundles were removed from each reactor with the normal on-power fuelling machines and with the reactor coolant system depressurized and cold, over a period of about 90 days.

The decontamination of the reactor coolant systems using the CAN-DECON process proceeded during short intervals during the time when the reactors were being defuelled, reducing the fields at the face of the reactor about a factor of ten. The shielding cabinets provided a further factor of from 5 to 10, so that the radiation fields within the shielding cabinet were about 5 mR/hour. This allowed a working time per quarter of about 200 hours in the cabinet for each worker, and eliminated the need to hire and

train trades staff specifically for their dose allowance. In fact early on in the job the total predicted dose uptake per reactor was reduced from the preliminary estimate of 2000 man-Rem to 1000 man-Rem based on the lower than expected fields.

The total preparation time was about 6 months during which all other plant systems of both units were placed in a lay up state.

2. Removal of Reactor Components

The removal of the reactor components involved handling highly radioactive materials when the tools and equipment were being withdrawn from the reactor core. Although this was done remotely it was considered desirable that Operations trades people assigned to this removal activity have experience in radioactive work.
During the removal phase an unexpected radioactive contaminant - Carbon 14, in fine dust form, was discovered on the outside of the pressure tubes. Extensive special precautions had to be introduced for Carbon 14 control and dosimetry. In spite of the Carbon 14 problem and some labour interruptions the removal phase was completed in about 14 months. The tooling worked very well and the integration of the engineering, construction and operations expertise in one site team proved to be highly successful.

The radiation up-take during this phase was well below the dose estimates despite the presence of the Carbon 14 problem. This latter problem contributed less than 10% of the dose during the removal phase.

3. Reactor Re-assembly

Once the radioactive components had been removed and the calandria tubes cleaned of any foreign material the re-assembly phase began. This phase involves handling and installation of non-radioactive components in a stable background radiation environment of 5 to 10 mR/hour. Radiation work expertise was not a major requirement. The work force to undertake the re-assembly (approximately 160 tradesmen) was obtained from the construction organization. These workers, who were not trained in radiation work were all, given extensive training in radiation work practices and on the specific re-assembly activities on the full scale mock-up before being allowed to work on the actual reactors in the vaults.

The quality assurance aspects of the re-assembly phase require verification to Nuclear Construction Standards. This capability was readily available from the Design and Construction Branch of Ontario Hydro.

The Fuel Channel Reinstallation Phase for Unit 1 began in February 1986. Progress was slow at first, but increased dramatically as further experience with equipment and procedures was gained by the construction tradesmen. Installation rates peaked at about three channels per day, as confidence and lessons were learned.

Work was carried out around the clock employing three installation crews of about 45 construction tradesmen and technicians each, and a smaller fourth crew dedicated to Bellows Welding. These crews worked a four days on, four days off/11 hour shift schedule, rotating days to nights.

All of the work was performed in low radiation fields in the shielding cabinet, with beams emanating from the reactor of up to 10 Rad/hr. In addition, airborne C-14 particulate (a long-lived Beta emmiter) was always present. All personnel wore full protective plastic suits with breathing air supplied. The record of radiation exposure management has been outstanding on this project. The total whole body exposure to gamma radiation for all workers since the project began in March 1984 has been 290 Man-Rem on Unit 1 up to the point of heat transport system hydrostatic testing. A total end of job exposure of 300 Man-Rem per unit is predicted. This compares to original estimates of approximately 2000 Man-Rem per unit and the subsequent revised estimate of 1000 Man-Rem. The total exposure from C-14 for all personnel to date, which was an unexpected situation, is the equivalent of six Rem whole body.

In the conventional safety area, which is a well established goal in Ontario Hydro projects, performance has been better by a factor of two than the station target, with only two lost time injuries in over two million man-hours worked.

In addition to this good overall performance, there has been no individual that has exceeded any exposure limit.

The completion of the channel reinstallation in Unit 1 during December, 1986, marks the first time that all of the fuel channels in a CANDU PHWR have been replaced. Detailed procedures, tool proving on full scale mock-ups and thorough training were used to achieve quality, with independent verification of work performed on each crew by trained technicians working along side of the tradesmen. Approximately 56,000 independent QC checks were performed during the reinstallation phase.

The original outage schedule called for the completion of the installation phase in August 1986 as outlined in Figure 7. The actual completion of the last fuel channel in December 1986 represented about a four month delay. The causes of this delay to that point were primarily the impediments caused by C-14, protection provisions, labour disputes and fixed equipment problems. The Unit 2 retubing was completed about mid-year 1987.

At the start of this major rehabilitation to Units 1 and 2, the estimated cost prior to in-service, without re-commissioning was $519.5M. Currently, and due to our successes, the projected costs were $418.7M. We are pleased with this actual and lower cost. However, we are not complacent, since we know that based on the Pickering Units 1 and 2 record using the manual retubing approach, retubing and energy replacement costs are unacceptable for retubing future units. We are studying new methods to involve more appropriate remote robotic and manual methods, based on the lessons learned

at Pickering NGS.

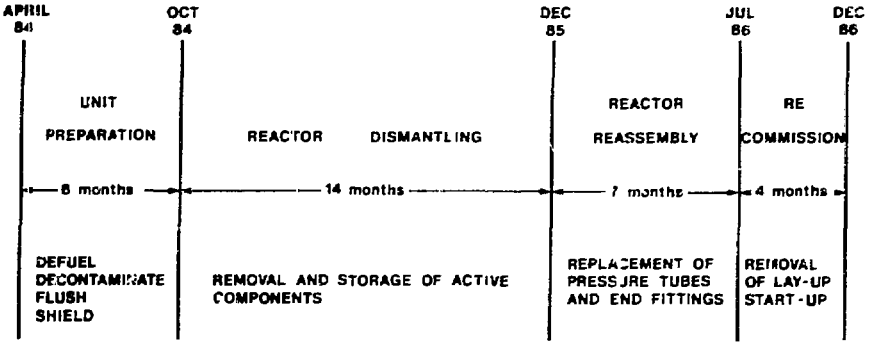

FIGURE 7
RETUBING SCHEDULE

The following section describes the re-commissioning of all major reactor systems, plant components and necessary modifications required to ensure future reliable operation.

RE-COMMISSIONING OF PICKERING NGS - UNIT 1

When the decision was made in March 1984 to retube both Units 1 and 2 at Pickering NGS, another program was instituted to be performed concurrently. This program designed "PICK-UP" (Pickering Upgrade), had the objective of ensuring a further 30 years of safe, reliable unit operation - post retube. It consisted of four components:

1. The Lay-Up Process

Where possible, systems were maintained in an operational state. If that was not practical, an environment was maintained within the system to inhibit degradation. Typical examples of actions taken were:

a) Service water, air and electrical distribution systems were maintained in limited service

b) Steam generators, feed heaters and other systems were filled with hydrazine treated water

c) Primary heat transport and main system steams were filled with nitrogen gas, and

d) Selected service water heat exchangers were drained and dried.

The inspections performed during the re-commissioning phase have shown that the lay-up program was generally successful in preventing any major degradation during the four year outage period.

2. The Inspection and Maintenance Program

Seventy inspection programs of the major systems were performed. These systems can be characterized into two main groups: Nuclear Systems and Conventional Systems. Our findings were as follows:

a) The Nuclear Systems

The nuclear systems were generally found to be in good condition, although components subject to wear or dry-out were prudently replaced. The Steam Generators tubes and drum internals were in good condition, however, a tube sheet sludge removal test was performed with just passing marks.

Extensive inspections, using remotely manipulated video cameras and ultra-sonic devices, were performed to assess the condition of nuclear systems within the Calandria vault, with particular attention paid to the biological shield cooling system.

b) The Conventional Systems

The conventional systems were generally in an acceptable condition but required extensive cleaning, re-calibration and particularly seal replacement.

The turbine/generator main steam, feedwater lines and the electrical systems were in good condition and no particular actions were needed.

The main steam safety valves and the pumps and motors associated with boiler feed and service water systems required some repairs.

In general, heat exchangers in contact with lake water showed signs of degradation and the main condensers, shield cooling and biological shield heat exchangers were retubed.

3. Station Safety System Modifications

Ontario Hydro had committed two major safety system improvements to the Pickering NGS A Station prior to the G-16 pressure tube failure. The first modification was associated with an enhancement of the emergency coolant injection (ECI) system, to align it with the capability of the Pickering B Station which is a higher pressure system. The second was to increase the depth of the shut-down system and retain the moderator as a major heat sink in the advent of a large loss of coolant accident.

These improvements represented an update of the earlier Pickering A plant to the increased "safety-in-depth" inherent in the more recently built nuclear units. The outage occasioned by the Pickering Unit 2 pressure tube failure, and the decision to retube both Units 1 and 2, provided the economic opportunity to advance these desirable actions prior to 1990 as previously

agreed. We wish to stress that these modifications are enhancements to an existing "defence-in-depth" against improbable accidents.

4. Re-commissioning

Re-commissioning and performance tests are carried out on both old and new systems to ensure that all could meet their design intent. Special emphasis was placed on ensuring that the quality of the special safety systems was up to present day standards.

Problems of long term inactivity became evident on the older systems. Relief and control valve settings had drifted, and valve seals were damaged, electrical relay contacts required cleaning, large isolating valves had become stuck closed, and valve packing required tightening or replacing. Almost three years to the day since defuelling began, fuel loading was performed from within the retubing shielded cabinet to minimize radiation dose to the workers.

Virgin heavy water was added to the primary heat transport system and a hydrostatic pressure test performed at 2000 psig. This proved somewhat difficult due to dried out valve packing (a result of the three year dry lay-up period), leaking pump seals and passing relief valves.

The reactor building was pressure tests and the leakage rate was less than the original 1970 commissioning tests. The bulkhead, isolating the unit from the station containment envelope for retubing, was removed and the moderator system filled with the original charge of tritiated heavy water. A period of hot commissioning was then performed. First criticality of unit 1 was achieved on June 23, 1987.

5. Lessons Learned

Generally, our experience from retubing Pickering NGS Units 1 and 2 has been positive. Our successes can be summarized as follows:

The method of defuelling the reactors was faster than predicted and the CANDE-CON decontamination process was more effective than we had assumed.

The method of flushing the heat transport system and subsequent vacuum drying eliminated tritium radiation dose concerns to the core-face workers. The sequencing of removal tasks over the reactor face, rather than row by row by both removal and installation, simplified integration of tooling and technology, resources, work planning and training.

The Retubing Tool Carrier (RTC) worked reliably to remove radioactive components from the reactor, and then transfer them to a conveyor system to load into flasks, whose contents were ultimately stored in the concrete canisters on site.

We found that many manual jobs could be performed faster than expected with correspondingly less man-hours and radiation exposure. This taught us that a rolling shift organization and highly motivated core face work crews, were more important than the philosophy previously adopted in the development of the large, remotely controlled arm to handle highly active components. Of course, this would not have been true, if the decontamination and worker radiation protection measures had not been as successful.
Our analysis of experience gained, suggests that improvements to shorten the schedule of future reactor retubings can be made in the following areas:

1. Quicker methods of draining and flushing the primary heat transport system.

2. Better methods of dealing with the Carbon-14 problem.

3. Faster radioactive component handling together with improved access control.

4. Tool refinement to improve their durability and reliability.

5. Better feeder pipe control.

6. Faster new pressure tube assembly.

7. Improved tool calibration to avoid excessive maintenance.

CONCLUSIONS

The retubing of the Pickering reactors has been an extraordinary challenge and successful experience for Canada's CANDU industry. It has demonstrated the versatility of this reactor concept and the ingenuity and talents of the large workforce involved in this unique project. Removal and replacement of the innermost pressure containing components of a large power reactor, the restoration to power and extension of unit life, is an outstanding achievement.

EXPERIENCE OF PARTIAL DISMANTLING AND LARGE COMPONENT REMOVAL OF LIGHT WATER REACTORS

by Michel DUBOURG

FRAMATOME S.A.

PARIS LA DEFENSE

FRANCE.

● ABSTRACT.

Regarding the relatively youthness of french operating commercial PWR built by FRAMATOME for ELECTRICITE DE FRANCE, none of these reactors needs to be decommissioned before the next decade or early 2.000 years.

However feasibility studies of decommissioning have been undertaken and several dismantling scenarios have been considered including the dismantling of four PWR units and the on-site entombment of the active components into a reactor building for interim diposal.

In addition to theorical evaluation of radwaste volume and activity, FRAMATOME has conducted several operations of partial dismantling of active components and decontamination activites in view of dismantling for both PWR units and BWR.

By analyzing the concept of both 900 and 1300 MWe PWR's, it appears that the design improvements taken into account for reducing occupational dose exposure of maintenance personnel and the development of automated tools for performing maintenance and repairs of major components, contribute to facilitate future dismantling and decommissioning operations.

1. FEASIBILITY STUDIES OF PWR DECOMMISSIONING.

Under the auspices of ELECTRICITE DE FRANCE, a feasibility study was performed by FRAMATOME concerning the dismantling of four PWR units and the on site entombment of the active components into a reactor containment building.

This study was conducted on the BUGEY plant site and two main options were considered :

1st Option : The use of the reactor vessel as a primary barrier of all irradiated components.

2nd Option : The dismantling and the cutting into pieces of irradiated components for transportation in lead shielded casks.

In the first option the various connections of the reactor vessel to the primary loop are sectionned at the level of nozzle safe ends. The tightness of the reactor vessel is obtained by the means of welded plugs for large size penetrations or by explosive plugs for instrumentation lines.

Control rod drive mechanisms are disconnected and the penetrations are sealed off by explosive expansion plugs. The reactor internals hold down spring is cut into pieces and located inside the vessel in order to release the tension of reactor vessel closure studs. The water contained in the reactor vessel is drained by an instrumentation pipe prior sealing off.

The total weight of the reactor vessel containing its own reactor internals support structure exceeds 490 tonnes. The reactor vessel handling and removal could be performed either by the containment building polar crane which needs some reinforcement or by a huge crawler crane as shown in fig. 1. The "Maintowoc" crane, the boom feet of which is supported by a circular ringer can handle 600 tons with a crane boom length of 91 m at a

distance of 25 m. The use of a Manitowoc crane can reduce the personnel exposure from 600 man rems down to 50 man rems for the handling and the diposal of the reactor vessel.

For the second option, it is anticipated a sectioning of all irradiated materials i.e. : reactor internals, thermal shield and reactor vessel by the means of underwater flame cutting with multiple thermic lances or by the use of arc saw technique.

2. PRACTICAL EXPERIENCE OF PARTIAL DISMANTLING OF ACTIVE COMPONENTS.

In the past, FRAMATOME has gained practical experience related to the partial dismantling and decontamination of active components of nuclear reactors.

2.1. Chooz Thermal Shield Removal. The CHOOZ reactor is a 280 MWe PWR and this reactor went to operation in 1968.

Due to flow induced vibrations roblem it was necessary to undertake a partial dismantling of the reactor internals and a complete removal of the degraded and irradiated thermal shield protecting the reactor vessel.

The thermal shield is a thin walled cylinder placed between the reactor core and the pressure vessel to reduce total irradiation effect on the vessel wall. For the CHOOZ reactor, the thermal shield was designed as a segmented cylinder (3 segments attached by welded clamps) to permit installation in the reactor vessel because the diameter of the shield was larger than the opening at the vessel flange.

During plant operation, coolant flow induced a rigid body movement to the thermal shield which generates fluctuating pressures on the core barrel and causes damage by fatigue. So it was decided to remove the thermal shield from the vessel and some peripherical fuel assemblies.

An electro discharge machine (EDM) using a vibrating graphite electrode was built and used for cutting under water the welded clamps of the thermal shield segments. The time required for cutting one welded clamp was about 10 hours. In order to remove the thermal shield segments which, while some repair work was carried out on the reactor internals in the reactor pool, a special dam was provided and fixed to the reactor vessel flange. The thermal shield segments were loaded in a lead shipping cask and then transferred for storage at La Hague Center (see fig. 2).

This operation was conducted successfully and after repair, the reactor power output has been upgraded of 10 % and the reactor operates satisfactorily since 1970.

2.2. Control Rod Guide Tube Centering Pins Replacement.

More recently the discovery of scarce defects affecting centering pins of control rod guide tubes located in the upper reactor internals of 900 MWe plants has initiated the connstruction of specific equipments and several "hot stands" for the removal and transportation of irradiated guide tubes and for the systematic replacement of these centering pins by new improved pieces.

During plant shut down for fuel reloading, the reactor upper internals are placed on their support stand and all control rod guide tubes are removed and replaced by new guide tubes equipped with centering pins of modified design.

Irradiated guide tubes are transported to the PIERRELATTE facility in FRANCE where centering pin replacement operation is taking place on hot stand. The replacement is carried out in air under lead shielded protection and a revolving table brings all the necessary tools for performing the replacement.

The attached figure N° 3 shows a general view of the "Hot Stand" and the equipment for removing the centering pins by EDM process.

This procedure allows to minimize the impact of the replacement of pins on plant outage and it can be said that such replacement is carried out during the normal refuelling outage of the plant.

Seven "Hot Stand" have been built by FRAMATOME for performing "off line" the centering pin replacement.

An operation was conducted in the United States on the R. GINNA PWR plant for Rochester Gas Electricity. A special designed "Hot Stand" was built for the removal of centering pins of 14 x 14 Guide Tube geometry.

This operation was carried out in 24 days in February 1986 during the normal outage for refueling of the GINNA plant.

2.3. Replacement of Damaged Steam Generators.

FRAMATOME has developed all the necessary tooling and established the procedures for the removal of corrosion damaged steam generators of PWR plants.

A large extensive research program including, development of decontamination process and the construction and full scale testing of automated tooling is undertaken.

The development program includes the following technical subjects :

- Soft chemical decontamination process of the lower part of the Steam generator to be replaced and the associated portions of primary pipings (see figure N° 4).
- Proper selection of a primary piping sectioning techniques based on a thermal process (Plasma) or on a mechanical cutting process.
- Bevelling of the primary piping by using a single point machining device (see figure N° 5).
- Automatic welding process with video follow up of the weldig junction between the new steam generator and the existing primary piping.

In addition to the technical research and demonstration, operational research is also performed in order to reduce man rem exposure and to shorten the execution time of steam generator replacement.

3. ON SITE DECONTAMINATION OF RECIRCULATION LOOPS PRIOR LOOP REPLACEMENT OF THE MUHLEBERG BWR PLANT.

To reduce the radiation exposure of the personnel dismantling the old piping system, "Bernische Kraftwerke AG" the plant utility, entrusted FRAMATOME with the decontamination of the MUHLERBERG recircultion loops.

For performing the decontamination operation, a mobile decontamination equipment consisting of several skid monted containers was built. The process is based on the use of dilute chemical decontamination solutions (<1 %) and the contamination is transferred from active components surfaces to ion exchange spent resins.

FRAMATOME uses, during the reduction phases, a mixture of organic acids and chelating agents and during oxydation basic potassium permanganate.

The mean features of this process are :

. A low concentration of reagents about .12 % is employed.
. The reducing agent is regenerated using cation exchange resins.
. A sequence of reduction/oxydation phases is employed.
. At the end of each phase reagent is removed by mixed bed exchange resins.

Before the Mühleberg decontamination operation went ahead, FRAMATOME conducted experimental studies to determine working conditions and to analyse reagent effects on materials.

These studies determined the decontamination sequence finally adopted at MUHLEBERG : a reduction phase, then an oxidation phase followed by a further reduction phase. A temperature of 120°C was adopted for the reduction phases.

The same sequence, but at twice the phase length, was then applied to the analysis of the circulating pump casing material.

Overall, general corrosion remained below 3 microns and no pitting or inter-granular corrosion was observed. Stress corrosion cracking tests did not show any quantifiable reagent-induced effect.

The decontamination "skid" designed and built by FRAMATOME includes three modules :

. Module 1 : ion exchange resins, and filters.
. Module 2 : reagent circulating lines and pumps, heaters, coolers, reagent injection line.
. Module 3 : Mobile laboratory for chemical and radiochemical analysis.

A temporary hall adjoining the reactor building was used for the installation of the three modules. Flexible tubing was used to interconnect the decontamination modules and recirculation loops and to provide connections to the plant auxiliary systems.

When the preliminary work and a 12-bar hydrotest had been completed, it was possible to go ahead with decontamination of loop B.

The results obtained from subsequent analyses showed that in three days of decontamination, 1,6 kg of oxides were dissolved and a Cobalt-60 equivalent activity of 14 Ci was removed from loop B.

During the decontamination of loop B, loop A was prepared for decontamination along the same lines. Decontamination of a loop was also successfully completed in three days : 1,3 kg of oxides were dissolved and a Cobalt-60 equivalent activity of 15 Ci was removed.

With the exception of one single point where the decontamination factor on contact was only six, all the points sampled during decontamination of the two loops exhibited a high decontamination factor : the average was 90.

It was calculated that the decontamination reduced the average dose rate in the drywell by a factor of ten and cut by 200 to 250 man-rem the collective dose to personnel involved in loop replacement.

4. DESIGN IMPROVEMENTS TO FACILITATE FUTURE DISMANTLING OPERATIONS.

By analyzing the concept of both 900 and 1300 MWe PWR's, it appears that the design improvements taken into account for reducing occupational dose exposure of maintenance personnel and the development of automated tools for performing maintenance and repairs of major components, contribute to facilitate future dismantling and decommissioning operations.

4.1. Reducing Activity. The overall activity at the time of dismantling results essentially from the activation of irradiated structures during reactor operation and from the deposition of activated cruds on the internal surfaces of plant components at the end of the plant life.

In order to reduce such activation and contamination, a strict control of materials submitted to neutron flux is performed and residual elements such as Cobalt 59 or other tracing long life radioelements are strictly limited.

Cobalt 59 content is restricted as follows :

- Reactor vessel shell base material : ≤ . 03 %
- Reactor vessel stainless steel cladding : ≤ . 15 %
- Reactor internals material : ≤ . 08 %
- Steam generator Inconel 600 tube bundle mat. : ≤ . 10 %
- Steam generator Inconel 690 tube bundle mat. : ≤ .035 %

+ N4 plant (1400 MWe)

For the manufacturing of reactor internals, stabilized stainless steel materials which contain tracing elements such as Niobium, the irradiation of which produces long lived radioelements are strictly prohibited.

In fuel assemblies, cobalt 58 (71 days of half-life) is produced by activation of inconel grids. In the FRAMATOME developed Advanced Fuel Assemblies AFA, inconel grids are replaced by Zircaloy grids with inconel clip springs. Activity calculations show potential reductions of dose field about 30 %.

In addition, investigations are underway to find an alternative to stellite (55 - 60 % Cobalt) in valve parts directly in contact with the primary coolant.

In particular, studies and tests are being carried on the use of Nickel base Alloys such as Colmonoy (77 % Ni, 11,5 % Cr-) or AISI 440 C martensitic stainless steel and Cobalt free thin hard faced coating of critical components, such as Control Rod Drive Mechanism grippers and CVCS valves.

Some practical measures have been taken at the design stage and during operation for minimizing the residual contamination of surfaces .

These measures include :

- Primary coolant coordonate B/Li chemistry.
- Specifications for surface conditions of steam generator materials.
- Large purification flow capability during plant shutdown for entrapping activity.
- Periodical decontamination of steam generator channel heads or isolated components for facilitate inspection and maintenance work.

4.2. In Service Inspection and Mandatory Periodical Inspections.

In service inspection of pressure boundary components as defined by the French regulations and Licencing Authorities implies to define access to critical areas such as welded joints for periodical inspections of activated circuits. Supports or guide rails are generally provided for the installation of automated tooling for inspection and in some case for repairs.

These supports will also facilitate future dismantling operations.

4.3. Plant lay-out design.

Several features of the plant lay out design have been defined to facilitate plant operation and consequently plant dismantling, these features include :

- Isolation between active and non active areas in nuclear buildings active and non active zones are existing.

- Shortening of the piping lay-out of primary and auxiliary systems of the 1300 Mwe plant. In addition some efforts were devoted for minimizing the number of valves.

- Erection of monorails and hoists for facilitating the replacement of componenets such as the RHR pump motors, or heavy valves or man hole covers of steam generators.

- Collection of the floor drains has been carefully review for avoiding stagnation or spillage of active waters.

- Implementation of a neutronic protection liner of the Reactor vessel concrete well for limiting and reducing the activation of concrete and iron structures.

This modification has been included in the new N 4 type reactor design (1400 MWe).

4.4. Updating of documentation.

The dismantling of active areas of a nuclear power plant must lead to a minimal collective exposure dose for the operating personnel. For carefully planning, the work in hostile environment, it is absolutely necessary to collect and to update all information related to the plant design (drawings, material specification, various modification and implementation....) and also information related to the plant operating performance during the plant life.

For performing such tasks, EDF has set up some procedures and systems for the collection, update and the recording of this information through out the plant life for making available such information when the plant dismantling operation will occur.

5. FUTURE EVOLUTION AND CONCLUSIONS.

The past experience gained by FRAMATOME in the field or repairs on irradiated components indicates that there is no more major obstacle for performing significant repair or modification on irradiated components and partial dismantling operation on Nuclear power plant.

The development of remote controlled equipment and robots operating in hostile environment can open the door for large modification or improvement on operating reactors.

The real limitation seems to be the availability of time and funds required for performing large modifications.

At the present time, FRAMATOME has developped and built more than 20 remote controlled devices for performing specialized repairs on reactors. Universal arms and robots are in used for increasing the flexibility and the adaptability to various operations.

In addition FRAMATOME is presently actively studying possible options consisting either to extend plant life reactors beyond its initial licence life and reviewing all economic and technological aspects.

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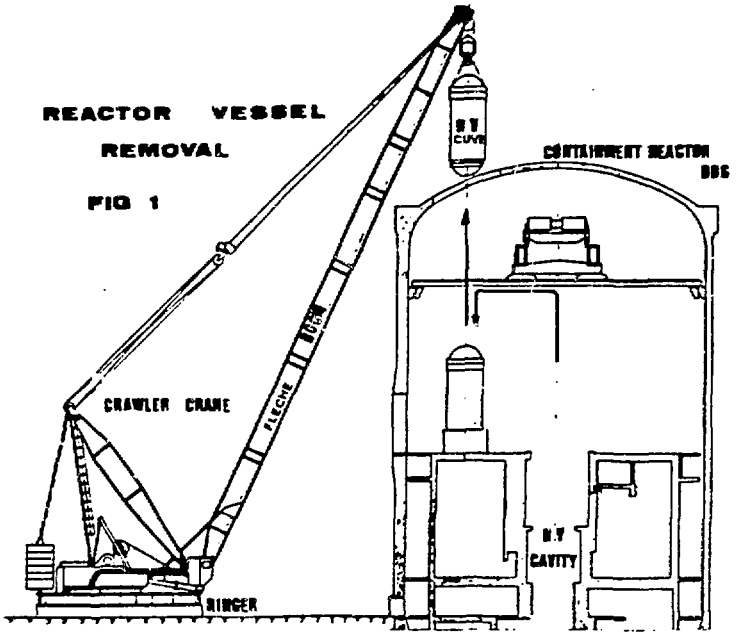


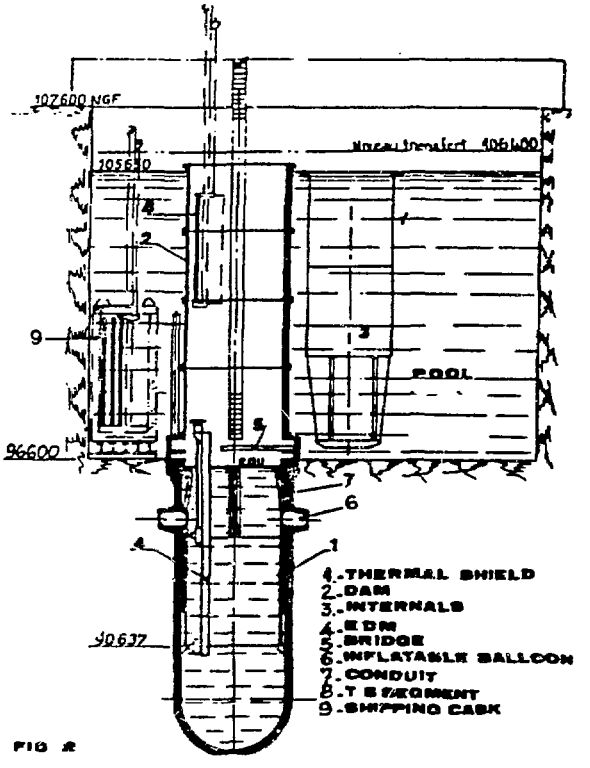
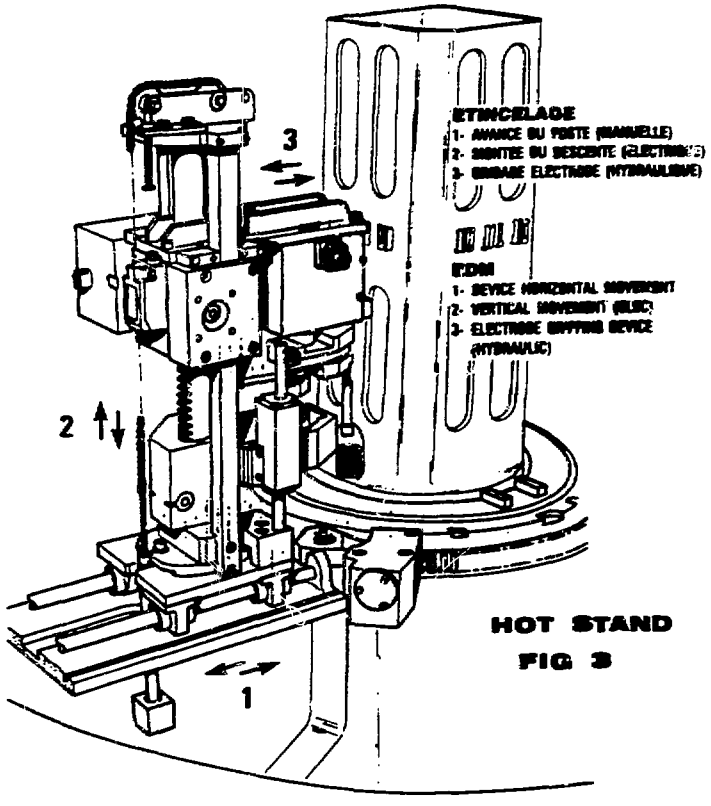


FIG 2

CHOOZ THERMAL SHIELD REMOVAL



HOT STAND

FIG 3

SGR - SOFT CHEMICAL DECONTAMINATION

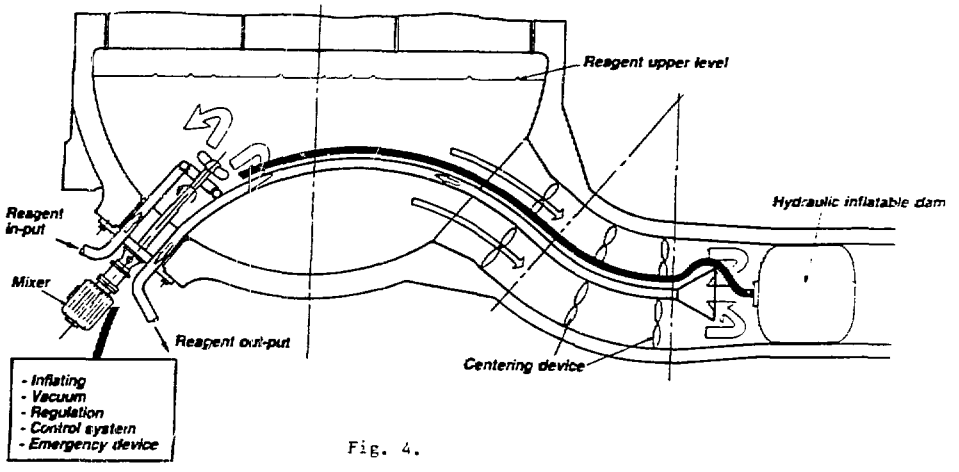
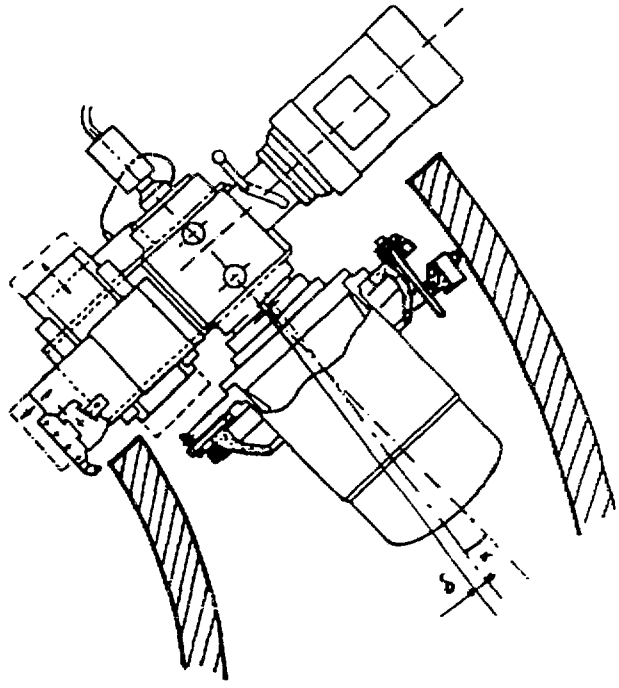


Fig. 4.



STEAM GENERATOR REPLACEMENT
MACHINING OF PRIMARY PIPING.
Fig. N° 5.

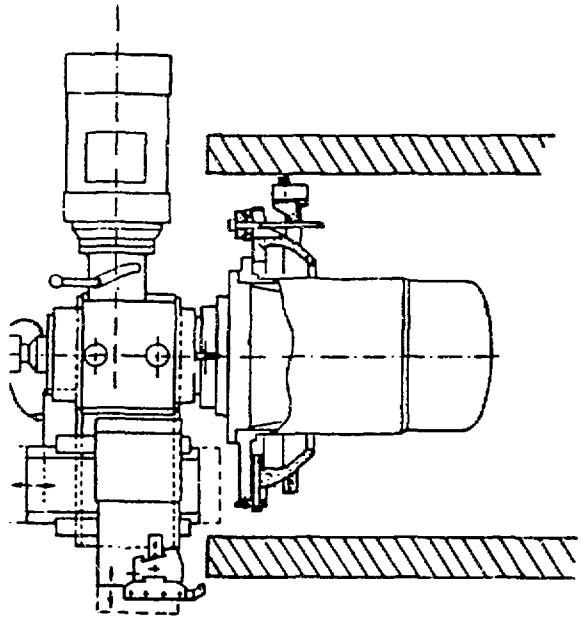
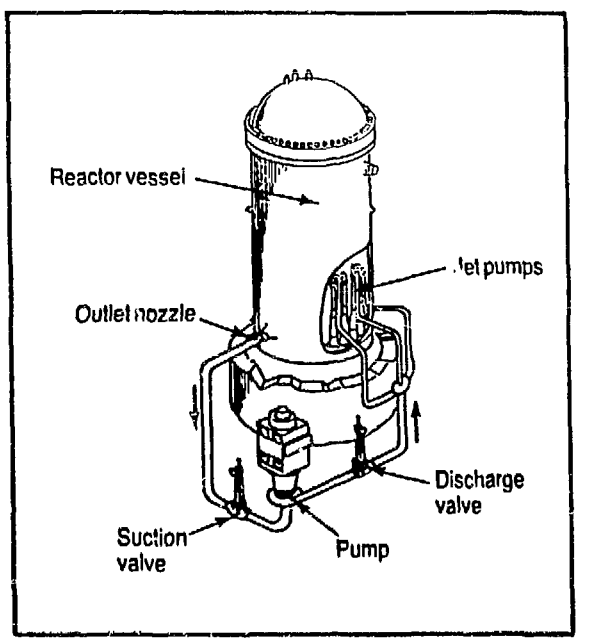


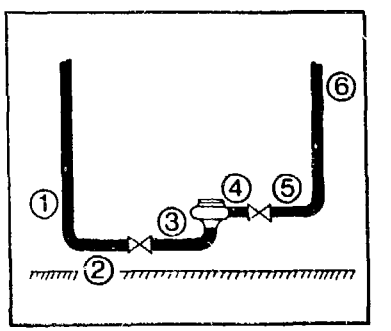
Table N° 1.

Dose rate readings on the Mühleberg recirculation pipework before and after decontamination, mrad/h

Measuring point	B loop						A loop					
	Before decontamination		After decontamination		Decontamination factor		Before decontamination		After decontamination		Decontamination factor	
	Contact	30cm	Contact	30cm	Contact	30cm	Contact	30cm	Contact	30cm	Contact	30cm
1	600	180	5	25	120	7	1 200	450	4	7	300	64
2	600	250	100	40	6	6	820	320	30	17	27	19
3	750	350	15	35	50	10	1 000	520	50	35	20	15
4			not measured				1 700	700	15	25	113	28
5	450	150	15	15	30	10	1 400	720	10	20	140	36
6	500	150	7	10	71	15	710	350	6	6	118	58



The Mühleberg BWR has two external recirculation loops (only one is shown in the diagram), six risers and twelve jet pumps.



The six points on the recirculation system at Mühleberg where dose rate measurements were taken.

STEAM GENERATOR REPLACEMENT AT SURRY POWER STATION
TECHNIQUES APPLICABLE TO POWER STATION DECOMMISSIONING

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ABSTRACT

During the Surry Steam Generator Repair Program it was necessary to cut the steam generator into two (2) pieces and to remove sections of the reactor coolant piping, feedwater piping, main steam piping, and instrumentation lines. Many of the cutting techniques and measures taken to minimize personnel exposure are directly applicable for use in nuclear power station decommissioning. This paper discusses these techniques as they were used at Surry.

INTRODUCTION

The purposes of the steam generator repair program at Surry Power Station were to repair the tube degradation caused by corrosion-related phenomenon and to restore the integrity of the steam generators to a level equivalent to new equipment. The repair program consisted of (1) replacing the existing lower shell assemblies (the portion of the steam generator containing the steam generator tubing) with new ones and (2) adding new moisture separation equipment to the upper shell assemblies. These tasks required that several pieces of reactor coolant piping, feedwater piping, main steam piping, and the steam generators be cut out of the system and be refurbished and/or replaced after the new steam generator lower shell was moved into place.

Many of the techniques used in the removal of the steam generators are directly applicable to the process of disassembling a reactor system for power station decommissioning. The areas to be covered in this discussion are as follows.

1. Cutting Techniques Used
2. Steam Generator Removal
3. Solid Waste Disposal
4. ALARA Techniques Used and Personnel Exposure

CUTTING TECHNIQUES AND STEAM GENERATOR REMOVAL

In order to remove the steam generators from the containment at Surry, the steam generators had to be cut in two above the tube bundle region and all of the reactor coolant, feedwater, main steam, and instrumentation piping had to be cut. A number of techniques were considered but flame cutting was determined to be the best as in most cases the equipment at the cut location would not be reused. The idea was then to make the cut as quickly as possible to reduce personnel radiation exposure.

First the main steam and feedwater piping and instrumentation lines were cut on the upper shell. This cutting was done with an oxygen-acetylene torch mounted on a track. The main steam pipe was cut where the pipe enters the crane wall and just above the steam outlet nozzle on the steam generator. The feedwater piping was cut at the crane wall and on the feedwater nozzle.

A track was then set up around the steam generator transition cone, and the steam generator was cut with an oxygen-acetylene torch. The criteria established for the location of the cut were (1) the outer diameter of the lower shell must be smaller than the diameter of the equipment hatch and

(2) the inside diameter of the upper shell must be large enough to allow the installation of the new moisture-separation equipment.

The polar crane was then rigged to existing lifting lugs on the upper shell, which was lifted off the lower shell and then temporarily set down on the transport-system rails. The upper shell was then rerigged to the two hooks of the polar crane such that one hook was attached to the man-way trunions (which were designed to bolt onto the upper shell where the man-way covers are found normally), and the other hook was attached to the existing upper-shell trunions after being passed around an inverting saddle that is lashed to the upper shell. The rollover of the upper shell was accomplished by raising the hook attached to the upper-shell trunions and lowering the hook attached to the man-way trunions. After the upper shell was upended, it was placed on a storage/work stand in the containment. The upper shell is 4.6 m in diameter, 7.6 m long, and weighs 116 Mg (128 tons). The diameter of the upper shell prohibited it from being removed from the containment in one piece through the equipment hatch.

When the upper shell was removed from the lower shell, the three swirl vanes (part of the moisture-separation equipment) remained behind. These swirl vanes were removed by flame cutting and disposed of. A one (1) inch thick steel cover plate with provisions for securing shielding was then lowered onto the top of the lower shell and was welded in place to seal the lower shell assembly.

The reactor coolant piping was then cut. A plasma arc torch mounted on a track was chosen to cut the reactor coolant piping because of the rapidity of the cut, thereby limiting personnel exposure. Typically, a cut on the 30 inch diameter, 3 inch wall thickness pipe could be made in 15 to 20 minutes. The removed piping consisted of an elbow of the hot-leg piping connecting the steam generator nozzle and the reactor coolant system isolation valve and two elbows and a straight section of the cold-leg piping. After the

reactor coolant piping was removed, shielded caps were welded over the steam generator nozzles, and the piping sections were moved to the containment basement for decontamination before refurbishment and reuse (on Unit 1 the pipe was discarded and new pipe was used). The secondary side water was then drained from the steam generator lower shell, and the blowdown piping and miscellaneous instrumentation lines were removed.

The polar crane was then attached to the steam generator with the swivel lift beam, the steam generator supports were disconnected, and the lower shell was lifted slowly out of its cubicle. The steam generator lower shell is 13.5 m (44 ft) long, 4.37 m (14 ft, 4 in.) in diameter on the transition cone and weights approximately 231 Mg (approximately 255 tons), including shielding. The swivel lift beam used to lift the lower shell is a lift rig that attaches to both hooks of the polar crane and is attached to the lower shell by two lift rods that attach to the two lifting trunions on the lower shell. The swivel lift beam allows the steam generator to be easily rotated while hanging from the two polar crane hooks.

After the steam generator lower shell was lifted out of the cubicle, it was moved to an area adjacent to the equipment hatch where the upending shoe was attached to two of the four steam generator feet. The steam generator was then set down on the upending grillage located just inside the crane wall adjacent to the equipment hatch. The upending grillage supports the bottom of the lower shell during the laydown operation.

The laydown of the steam generator was accomplished by simultaneously lowering the hoists of the two polar cranes while moving the crane trolley towards the center of the containment. Great care must be taken to ensure that the crane cables remain vertical so that no lateral loads are imposed on the steam generator during the laydown. Prior to laying down the lower shell, lifting cables forming a basket hitch were put into the transport cradle, and the steam generator was then laid down into the cradle.

The swivel lift beam was then unhooked from the steam generator, and the main hooks of the polar crane were rigged to the basket hitch. The steam generator was then lifted up in a horizontal attitude, rotated 180°, and set back down on the transport tracks so that the transition cone faced the equipment hatch. That is, the steam generator was positioned so that the transition cone was transferred through the equipment hatch first.

This transport configuration was necessitated by the configuration of the equipment hatch and the operating deck. The equipment hatch is circular with a small part of the circle below the operating deck. A small section of the operating deck had to be chipped out adjacent to the equipment hatch, so that the steam generator could be lowered into position for transport through the hatch. If the steam generator had been oriented in the opposite direction, a trench the full length of the steam generator would have been required in the operating deck.

Just before removing the steam generator from the containment, a final decontamination of the outside of the lower shell was performed to remove any loose surface contamination. In general, levels of 200 to 600 dpm/cm^2 were achieved. The only area that could not be decontaminated to acceptable levels was the area immediately around the reactor coolant nozzles on the channel head. This channel head was therefore wrapped in Herculite for transport to the waste storage facility. During the second unit (Unit 1) steam generator replacement the whole lower shell was painted to fix the contamination to the shell rather than decontaminate the whole shell.

Using the steam generator transport system, the steam generator lower shell was winched out on the equipment-hatch platform. Because of the clearance between the steam generator lower shell and the equipment-hatch barrel, the transport tracks could not go all the way through the hatch. Consequently two sets of tracks were installed, one set of tracks inside the containment and another set outside the containment. Transfer through the

hatch was accomplished by moving the steam generator part way through the hatch on two sets of rollers to a point where the cradle extended through the hatch and out onto the platform. A third set of rollers were then installed on the platform tracks, and the rollers inside the containment were removed. The steam generator was then winched completely out of the equipment hatch. During the removal operation, the clearance around the transition cone was approximately 0.05 m (2 in.).

The steam generator was then lifted off the equipment-hatch platform with a luffing derrick and lowered onto the haul vehicle for transport to the on-site engineered storage facility.

At the engineered storage facility, the steam generator was lifted off the trailer with jacking towers, set down on storage cradles mounted on rollers, and then winched sideways into the building. The engineered storage facility was then sealed. The lower shells will remain on-site until plant decommissioning. (One of the Unit 2 steam generators has since been removed and shipped to Hanford, Washington for use in an NRC sponsored research program.) By that time the radioactive inventory inside the steam generators will have decayed from approximately 1200 Ci each to between 12 and 20 Ci.

The engineered storage facility is a reinforced concrete structure with 0.91 m 3 ft-thick walls to provide shielding and to meet the criteria of 40 CFR 190. The building is sealed to prevent water intrusion. An internal sump collects any water that might get inside the building. Ventilation to allow for expansion and contraction of air in the building is provided through HEPA filters. Several two (2) inch plugs have also been provided to allow radiological surveys without entering the building and to monitor the water level in the sump.

SOLID WASTE DISPOSAL

As would be the case with power station decommissioning a large amount of solid waste was generated during the disassembly phase of the steam generator replacement.

The solid waste generated during the steam generator removal effort consisted of (1) contaminated insulation, structural materials, and components not intended for reuse, (2) solidified liquids used for decontamination, and (3) contaminated paper waste, disposable protective clothing and contamination control materials. During the steam generator removal phase of the repair program approximately 26,600 cubic feet of solid waste was removed for the Unit 1 repair effort and approximately 26,300 cubic feet for the Unit 2 repair effort.

During the steam generator repair efforts a packaging station with a compactor was set up in the basement of the containment to minimize the handling of the waste.

As was discussed earlier the largest component to be handled were the steam generator lower shells. The engineered storage facility was decided upon after a number of studies at the time that on-site storage was the best option both from an economic and personnel exposure standpoint.

ALARA TECHNIQUES USED AND PERSONNEL EXPOSURE

During the steam generator replacement most of the normal dose reduction techniques were used to try to minimize personnel exposure. In addition to these, two techniques were used which worked fairly well for this application.

During the work on the old steam generator lower shell, the water in the secondary side was maintained at a level covering the steam generator tube bundle in order to fully utilize its shielding quality. The shielding effect of this water provided an approximately 10 to 1 dose rate reduction for the area of the steam generator above the tube sheet.

During the cutting operations a concern was raised about airborne contamination. During the first unit steam generator repair (Unit 2), small tents and glove boxes were used to control any potential airborne contamination. Upon analysis it was found that more exposure was being expended in installing and removing the glove boxes than the exposure for actually performing the pipe cut. In the second outage (Unit 1), the steam generator cubicles were sealed and turned effectively into large contamination control tents.

Portable ventilation units continuously withdrew air from these enclosures through appropriate filters whenever cutting or grinding operations were in progress. Personnel working inside the cubicles were required to wear respiratory protection equipment during these operations, and until the ventilation flow effectively reduced the airborne radioactivity to acceptable levels. The benefits observed for these ALARA techniques were threefold.

a) The use of small glove boxes for individual pipe cuts was eliminated. These devices required considerable time and exposure to install and remove, and during the Unit 2 SGRP were found in many cases to be counter-productive to ALARA due to this fact.

b) The use of temporary containments, especially as applied to the lower steam generator cubicles, largely eliminated delays and interference with work being performed in adjacent areas while cutting and grinding of contaminated piping was in progress.

c) The overall potential for airborne contamination and problems associated with contamination control was minimized by effective use of temporary containments and portable ventilation.

The personnel exposure for the steam generator removal phase of the steam generator repair program was as follows:

	Steam Generator Removal (Man-Rem)	Total Repair Program (Man-Rem)
Unit 1	592	1758
Unit 2	946	2070

CONCLUSIONS

In summary, many of the activities which were performed during the steam generator repair program are directly applicable to a power station decommissioning. Further information on the details of the Surry Steam Generator Replacement is available in the documents referenced below.

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1. McKay, H. S., "Steam Generator Replacement At Surry Power Station", Nuclear Safety, Volume 23, pages 72-84, January-February 1982.

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3. Virginia Electric and Power Company, "Steam Generator Repair Program For The Surry Power Station Unit No. 2: Final Report For The Period February 3, 1979 Through December 31, 1979", Docket Nos. 50-280 and 50-281, Available NRC Public Document Room.

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