8C

ORGANISATION FOR ECONOMIC
CO-OPERATION AND DEVELOPMENT

NUCLEAR ENERGY AGENCY

RESTRICTED

Paris, drafted: 24th April 1986
dist: 28th April 1987

Addendum 2 to
NE(87)6

Scale 4

Or. Engl.

WM Record File
412.2.1

WM Project
Docket No.
PDR
LPDR

Distribution:
REB
GALSON
CARDONE
(Return to WM, 623-SS)

FOR INFORMATION

'87 MAY 11 P12:36
WM DOCKET CONTROL CENTER

STEERING COMMITTEE FOR NUCLEAR ENERGY

THE RECOMMENDATIONS OF THE DATA BANK MANAGEMENT COMMITTEE ON THE FUTURE ROLE OF THE DATA BANK

1. The Committee discussed the proposal in paper NE(87)6 and agreed that the Data Bank should increase its level of effort in support of the nuclear safety and radioactive waste management areas of work. However, reservations were expressed by the Committee about the balance of effort and schedule proposed between these new areas and the traditional areas of nuclear data and computer program collection and distribution.

2. The Committee reaffirmed the value to the nuclear power community in Member countries of the present activities of the Data Bank and considered that these would continue to require more effort after 1988 than proposed by the Secretariat. In particular the Committee noted :

(i) Nuclear data and computer programs continue to be essential for the safety design of nuclear reactors and more generally to the further development of nuclear energy.

(ii) It would be unlikely that the effort on JEF could be reduced after 1988 to the level suggested by the Secretariat. To continue to safeguard the past investment in nuclear data and to maintain the momentum on data evaluation work, the Data Bank Committee recommends that provision be made for continuing effort on JEF and related work beyond 1988.

(iii) The Data Bank by making good quality and well tested programs widely available contributes greatly to the overall productivity, reliability and safety of the nuclear power industry throughout the Member countries. Comprehensive testing of key selected programs by the Data Bank is an important activity in this field which should continue beyond 1988.

(iv) The Data Bank Committee considers that a natural and important development of these traditional areas of work is for the Data Bank to become an internationally recognised Repository of Quality Assured computer programs and data. Such a Repository, functioning in a way consistent with developments in technology would be a major contribution to nuclear power safety.

(v) Work in support of nuclear safety and waste management should form an important part of the Data Bank's programme. The Committee recognises that without an increase in budget the level of effort would be less than that proposed by the Secretariat.

04 232

3. In line with the above, the Committee considers that the proposals by the Secretariat are too drastic both in time schedule and resource shifting. The Committee recommends that for current planning the level of effort on nuclear data and computer program activity should not fall below the level proposed by the Secretariat for 1988 in Table D of paper NE(87)6. Any proposals for further reductions must be based on adequate experience of working at this reduced level in the light of circumstances then prevailing. The Secretariat is requested to keep under review and make recommendations on the resources to be devoted to each area of Data Bank work.

4. Finally, the Committee reaffirms its appreciation of the excellent work done by the Data Bank in the past. The Committee considers that a significant factor in this has been the close and effective working relationship between the Data Bank and the Committee, ensuring that the work of the Data Bank is complementary to national activity and accords closely with national needs : the Committee recommends that this close relationship should continue. To ensure a similar effective relationship in the nuclear safety and radioactive waste management areas the Committee recommends that its representation should be strengthened by national representatives knowledgeable in these areas.

ORGANISATION FOR ECONOMIC
CO-OPERATION AND DEVELOPMENT

NUCLEAR ENERGY AGENCY

RWM/DOC(87)3

RESTRICTED

Paris, drafted: 24th April 1987

distr. : 29th April 1987

English Text Only.

RADIOACTIVE WASTE MANAGEMENT COMMITTEE

Decommissioning: Progress within the International Programme on Information Exchange

1. The Co-operative Programme for the Exchange of Scientific and Technical Information concerning Nuclear Installation Decommissioning Projects entered officially into force towards the end of 1985. According to the requirements of the Agreement for the implementation of the Programme, the "Liaison Committee shall make annual reports to the Steering Committee for Nuclear Energy on the progress of work". The attached document CPD/DOC(87)2 is the first Annual Report prepared by the Programme to fulfil the above-mentioned obligation. This report refers to the period: September 1985-December 1986.

2. Since the issuing of this report, further progress has been made. The most important events during the first quarter of 1987 include another meeting of the Liaison Committee, the inclusion in the Programme of a new Project, namely the BNFL Co-precipitation Plant for the production of mixed plutonium and uranium dioxide fuel (United Kingdom); and the application by Belgium to be admitted into the Programme as Observer participant.

Another interesting achievement was the establishment of three Special Arrangements within the framework of the Programme. They are respectively:

- Co-operation between Canada and Japan on the Application and Development of a Computerised Code System for the Management of Nuclear Power Plant Decommissioning;

- Co-operation between Japan and the United Kingdom on Reactor Decommissioning Technology Developments;

- Co-operation between Japan, Sweden and the United Kingdom on the Evaluation of Radioactive Inventory.

3. The Technical Advisory Group, responsible (under the guidance of the Liaison Committee) for the actual implementation of the Programme, has also an active programme of work for 1987, including meetings and technical visits to

three Decommissioning Projects and other technical installations in the Federal Republic of Germany and to the Shippingport and West Valley facilities in the United States.

4. The Liaison Committee, besides its basic function as the steering group for the Programme, is playing a more general role of assistance to the NEA in matters concerning decommissioning. In particular, the Committee has carried out reviews and expressed advice (and continues to do so) on the technical aspects of the application of the Paris Convention on Third Party Liability to decommissioning.

The visibility of the Programme has been ensured through the presentation of papers in different international congresses by officers of the Programme and the Secretariat.

5. The Committee is invited to take note of the progress achieved, and to agree to the submission of the first Annual Report to the Steering Committee for Nuclear Energy.

THE OECD/NEA CO-OPERATIVE PROGRAMME FOR THE EXCHANGE OF SCIENTIFIC AND TECHNICAL INFORMATION CONCERNING NUCLEAR INSTALLATION DECOMMISSIONING PROJECTS

First Annual Report of Programme Activities to the Radioactive Waste Management Committee

September 1985 - December 1986

1 BACKGROUND AND PARTICIPATING ORGANIZATIONS

The NEA Co-operative Programme on Decommissioning was initiated on the basis of a proposal submitted by the US Department of Energy and involves participation in the exchange of scientific and technical information as well as other forms of technical co-operation between specific major decommissioning projects.

The Co-operative Programme, which officially entered into force towards the end of 1985, has been started with nine participating countries with seven of the countries having a total of eleven decommissioning projects registered in the agreement. The countries and the respective participating organizations are listed below:

Canada	Atomic Energy of Canada Limited
Federal Republic of Germany	Kernkraftwerk Lingen jointly with Kernforschungszentrum Karlsruhe
France	Commissariat à l'Energie Atomique
Italy	Ente Nazionale per l'Energia Elettrica jointly with Comitato Nazionale per la Ricerca e per lo Sviluppo dell'Energia Nucleare e delle Energie Alternative
Japan	Japan Atomic Energy Research Institute
Spain	Junta de Energia Nuclear
Sweden	Svensk Kärnbränslehantering AB
United Kingdom	United Kingdom Atomic Energy Authority
United States of America	Department of Energy

Sweden has undertaken the duties of the Programme Co-ordinator, while Spain is participating in an observer capacity without a current decommissioning project. The Co-operative Programme will continue to be open to new participants from member countries of the OECD.

2 ADMINISTRATION

A Liaison Committee, with representatives of all participants, has been set up to implement the Programme Agreement.

Under the Liaison Committee functions a Technical Advisory Group at whose meetings the progress of work at the various participating projects is reviewed and technical problems of common interest are examined. The Liaison Committee has also designated a Programme Co-ordinator whose duties are to assist the Committee in the implementation of the programme arrangements and procedures.

For the first two years of the programme, the following have been chosen to act in the capacities named:

<u>Liaison Committee</u>

Chairman:	A Bertini, Italy
Vice chairman:	J M Liederman, Canada

<u>Technical Advisory Group</u>

Chairman:	H Lawton, United Kingdom
Vice Chairman:	R Lurie, France

<u>Programme Coordinator</u>

S Menon, Sweden

3 PARTICIPATING PROJECTS

The participating projects are listed in Table 1 and an overview of their time schedules is given in Table 2. The technical status of each of the projects is briefly described in Annex 1. *

---

* The Annex covers all participating projects except the West Valley Demonstration Project, which has only recently announced its taking part in the Co-operative Programme.

4 PROGRAMME ACTIVITIES

Hitherto, the Liaison Committee of the Co-operative Programme has met three times, the first meeting in September 1985 to launch the programme, followed by meetings in February and April 1986.

The Technical Advisory Group has held two meetings, an inaugural one in Paris in November 1985 and the first "working" meeting at Montreal in March 1986.

In addition to the above activities, the Japan Atomic Energy Research Institute held a workshop in November 1985, as part of the programme of work of the Co-operative Programme. At the workshop, the technological developments achieved to date by the Japan Power Demonstration Reactor (JPDR) Decommissioning Project were described and discussed.

The exchange of information on decommissioning technologies, which is the primary objective of the Co-operative Programme, takes place mainly at the meetings of the Technical Advisory Group (TAG). The Montreal meeting of the TAG was organized as a three-day conference in order to facilitate this information exchange. The first day was devoted to

- Progress reports from all participating projects

- Presentations by each participant on a few generic topics, which had been announced in advance.

A site visit to Gentilly I was organized on the second day, while programme formalities were discussed on the final day.

Given the positive experiences from the Montreal meeting, the future meetings of the TAG will be organized on similar lines, i.e. a day for discussion of generic issues, a site visit and discussion of programme organization and administration. The choice of the topical subjects will be made on the basis of directives from the Liaison Committee. Some examples of these subjects, whose discussion is scheduled for the next meetings, are:

- melting
- recycling levels
- techniques to measure very low surface contamination levels
- waste transports.

Two meetings of the TAG are planned for each of the years 1987 and 1988.

As mentioned earlier, the OECD/NEA Co-operative Programme on Decommissioning was organized to share the technologies and know-how arising from a number of major decommissioning projects, this in order to most effectively build up the data-base required for the decommissioning of large commercial nuclear facilities in the future. The Co-operative Programme is as yet in its initial stages. However, the active spirit of co-operation shown by the participants and the results so far indicate strongly that the objectives of the programme can and will be realized.

Table 1. List of participating projects

Facility	Type	Operation	Decommissioning option*
1. Gentilly-1, Canada	Heavy water moderated/ boiling light water cooled prototype reactor	1970-79	Variant of stage 1
2. Rapsodie, France	Experimental sodium cooled fast reactor	1967-82	Stage 2
3. G2, France	Electricity and nuclear materials production	1958-80	Stage 2 (followed by stage 3)
4. AT1, France	Pilot reprocessing plant for FBR oxide fuel	1969-79	Stage 3
5. Kernkraftwerk Niederaichbach (KKN), Federal Republic of Germany	Gas-cooled heavy water moderated	1972-74	Stage 3
6. Kernkraftwerk Lingen, Federal Republic of Germany	BWR (with super-heater)	1968-77	Stage 1
7. Garigliano, Italy	BWR (dual-cycle)	1964-78	Stage 1 for main containment
8. Japan Power Demonstration Reactor (JPDR), Japan	BWR, research	1963-76	Stage 3
9. Windscale Advanced Gas Cooled Reactor, United Kingdom	AGR	1962-81	Stage 3
10. Shippingport, United States of America	PWR	1957-82	Stage 3
11. West Valley Demonstration Project, United States of America	Reprocessing plant for LWR fuel	1966-72	

* The decommissioning options are defined according to the IAEA classification (see IAEA Technical Report Series No 230).

NS39 DE

Table 2. Time schedules of participating projects

OECD NEA CO-OPERATIVE PROGRAMME on DECOMMISSIONING

OVERVIEW TIME SCHEDULE

	CANADA	FRANCE			FEDERAL REPUBLIC of GERMANY		ITALY	JAPAN	UNITED KINGDOM	UNITED STATES of AMERICA	
	GENTILLY-1	RAPSODIE	G-2	AT-1	KWL	KKN	GARIGLIANO	JPDR	WAGR	SHIPPING PORT	WEST VALLEY
1985					Licence for Stage I						
1986	"Static State" Storage with surveillance	Safety report Pilot tests: decon.	Develop processes for decon, cutting, Waste management	4th cycle Wksp. decon/disassem. Vent. equip. Dismantl. machine for main cells Fabric.	Plant into Stage I	Installation of smelter Installation of equipment Dismantl. inactive systems	•Trpt fuel •Radwaste treat •Plan safestore containm. •Plan dismantl. decon turbine bldg etc	Mockup model tests Design, Fabric	Construct.equip Waste pack. bldg. Heat exchanger Work Order remote cutting mech.	•Compl. asbest. removal •Start liquid waste process •Start equipm. decon./removal •Start prim. syst. comp. removal, piping & equip. removal	•Compl D/D of Process Bldg Chem. Cells (CPC) •Equip. Remov •Concr. remov equip procure •Cont. Bal. of Plant (BOP) D/D
1987		Prim.circ. Decon-Dismantl. Main vess. cut. equip.	Reactortank sep. from circuits. Pret. work equipment.	952 cell dismantl. – Wksp decon, dismantl. Study-1 907 cell		Removal of contaminated systems	Lab scale decon tests.	Waste Contr. Dismantl. Equip. round reactor		•Start concr. & structures removal •Start reactor vessel prep. •Complete removal of prim. components	•Compl D/D of CPC •Compl D/D of Equip Decon. •Compl D/D of Irr. Vent. Duct. •Cont. BOP D/D
1988		Hot cell modif	Decon. Dismantl. circuits Pret.work Steam generators	Dismantl. machine tests. Study 2 902 cell		Installation of remote handl equipment		Dismantl. Reactor Inter. Pressure vessel Dismantl. Turbine Dump C etc	Waste contain. design frozen	Complete removal of pipes and equipm. from chambers	•D/D of Crane Rooms •Refurb. Gen. Purp Cell (GPC) •Start D/D of GPC Pre Mech. Cell (PMC)
1989			SG Decon Dismantl.			Removal of moderator tank, thermal shield.	Reactor containm placed in safestore Dismantl. Decon in turbine bldg	Demol. Biolog. shield	Delivery of remote cutting mech.	• Reactor pressure vessel lift • Compl. concr. removal • First back-filling	•Compl D/D of GPC and PMC •Cont. BOP D/D
1990		In vess. cut. (water) In cell sampl	Fuel handl. Packaging.			Remove biologic shield. Equipment rem. Decon of bldg		Dismantl. other equip. Demolition of contain.	Machine in reactor. Start of reactor demolition	Project completed	•Continue BOP D/D

TECHNICAL SUMMARY

Progress of Participating Projects

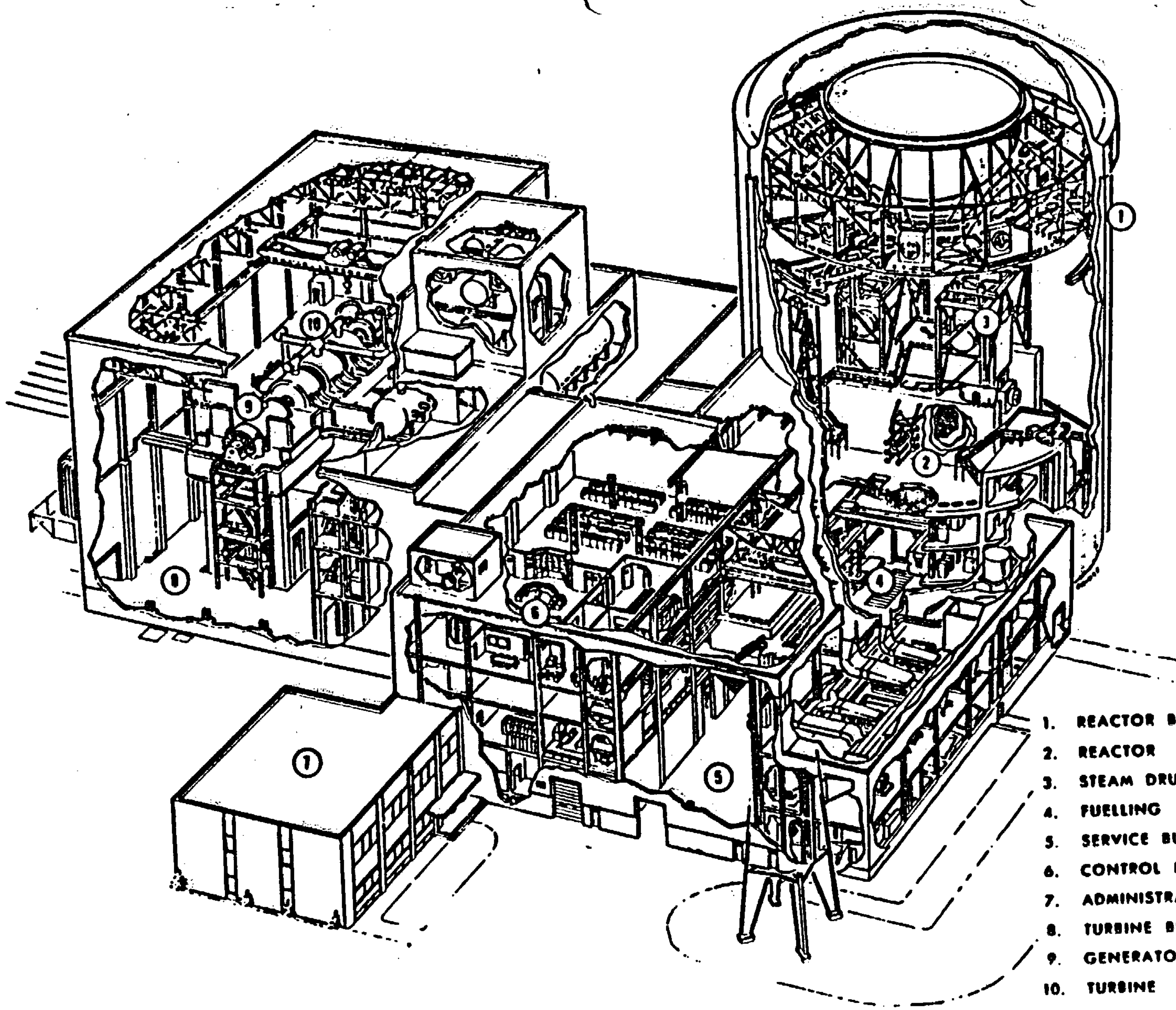
1 GENTILLY-1 (Canada)

Gentilly-1 is a heavy-water moderated, boiling light water cooled prototype reactor, which was shut down in 1979. (See Figure 1).

The object of the decommissioning project at Gentilly-1 was to place the facility in a "static state" - a variant of the "storage with surveillance" state. At the end of the project, the site licence was amended and it became the Gentilly-1 Waste Storage Facility.

The decommissioning project was started in 1984 and completed on schedule in March 1986. A major part of the work was in connection with the dry fuel storage facility, where all fuel irradiated during the operating life of the station is stored in above ground concrete canisters. The Service Building has been cleared of all installed equipment and decontaminated, as also part of the Turbine Building. These areas have been transferred to Hydro-Québec in a so-called ZONE 1 state where there are

- No detectable loose contamination
- No radiation fields > 0.25 mrem/h
- No fixed beta-gamma contamination levels > 22.200 dpm/100 cm^2
- No radiation sources > 0.25 mrem/h at 1 metre
- No fixed high toxicity alpha contamination level > 2.200 dpm/100 cm^2
- No fixed low toxicity alpha contamination level > 22.200 dpm/100 cm^2



1. REACTOR BUILDING
2. REACTOR
3. STEAM DRUMS
4. FUELLING MACHINE
5. SERVICE BUILDING
6. CONTROL ROOM
7. ADMINISTRATION BUILDING
8. TURBINE BUILDING
9. GENERATOR
10. TURBINE

Figure 1

Gentilly generating station

Low level waste generated during the decommissioning project is stored in designated areas of the Turbine Building. Other active waste is stored in the Reactor Containment Building, which is sealed, allowing only authorized entry.

The costs of maintaining the Gentilly-1 site before the decommissioning project was 10 MCAD/year since 1979, when it was shut down. The cost of the decommissioning project was 25 MCAD over two years. The expected costs of maintaining the "static state" plant is 0.5 MCAD/year.

2 RAPSODIE (France)

This is an experimental sodium cooled fast breeder reactor of the loop type, 40 MWth, see figure 2. In 1978 a small sodium leak was suspected in the primary vessel, then the reactor was operated at derated power (22.4 MWth) at which the fault was no longer detectable. After the detection of a nitrogen leak in the double casing lining the primary circuit, the reactor was shut down in 1982.It had then seen about 15 years of service.

The decommissioning level aimed at is Stage II, involving the complete disassembly and removal of the two primary loops outside the reactor block, which will then be sealed and confined as such. Prior to this, a technical appraisal programme will be executed. This will concentrate on non-destructive examination and sampling of the primary vessel with internals, the primary loop and its major removable components like pumps, intermediate heat exchangers etc.

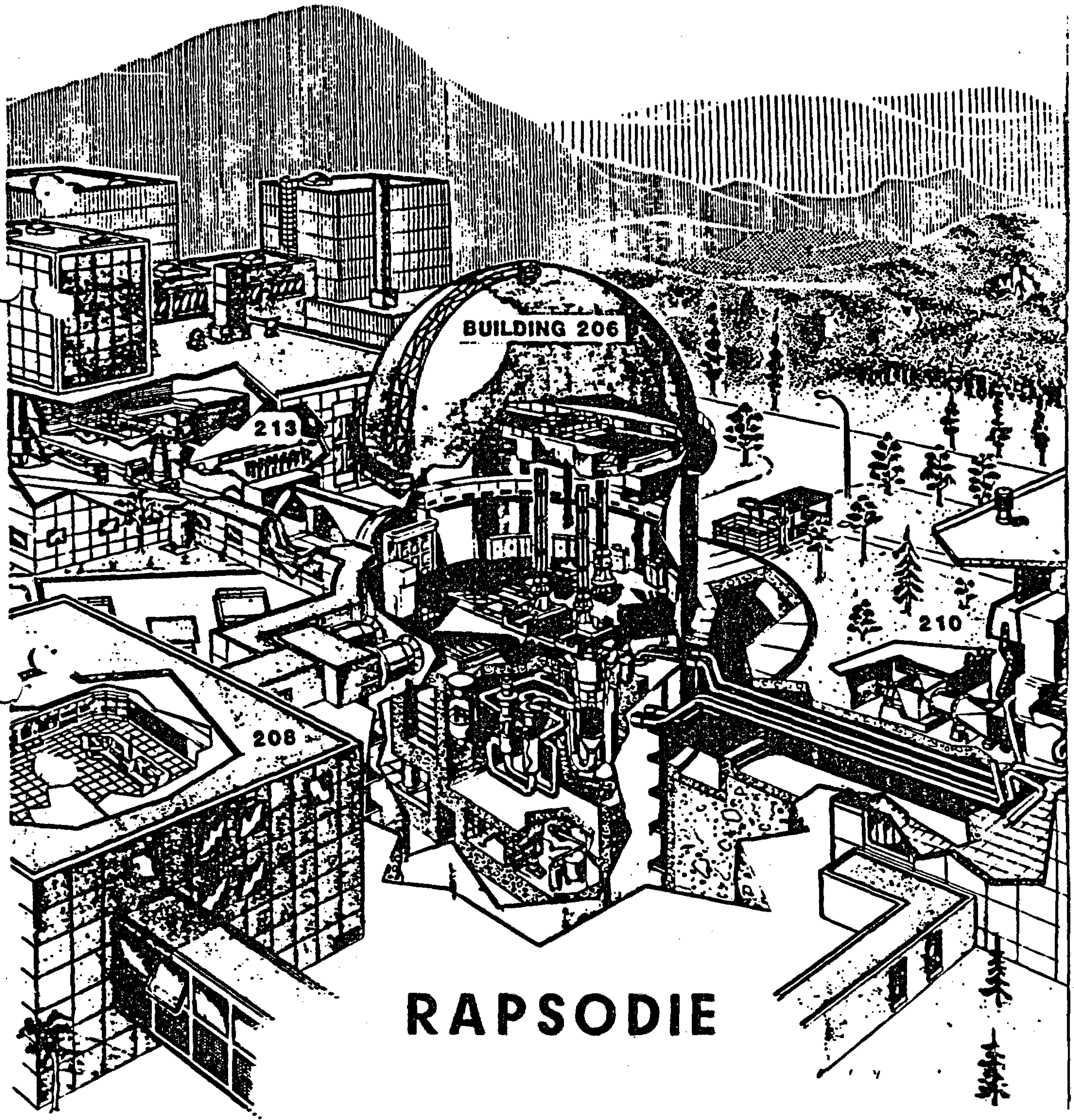


Figure 2

II10 MLL

The plant is at present in a state of final shut down. Fissile/fertile assemblies have been removed. The primary sodium has been flushed out to storage tanks. After purification to reduce the levels of Cs-137, it is intended to re-utilize the 37 tons of Sodium in another test facility or reactor.

After dismantling all removable components from the reactor block, the primary system will be washed with heavy alcohols and then rinsed with water. Disassembly of the loops from the reactor block is planned for early 1989.

Parallel to this work, mamipulators and a cutting/sampling machine will be designed and built. Sampling is expected to take place between 1991 and 1993, after which the reactor block will be dried and sealed. These operations are scheduled to be completed by the end of 1994.

3 AT-1 (France)

AT-1 was a pilot plant for reprocessing mixed oxide fuel elements from the FBR programme, using the PUREX process. It was operated for 10 years from 1969 to 1979. After the current decommissioning project, the buildings of this complex facility, which were highly contaminated with fission products and α-emitters at the end of operations, will be available for unrestricted re-use.

The facility consists mainly of

- shielded hot cells where spent fuel was cut up and dissolved and U/Pu were extracted

- α-cells and glove boxes with less shielding for U/Pu separation, concentration/transformation to UO_2 and PuO_2.

At the end of the operations, some of the plutonium built up in the plant was recovered by first rinsing with acid solutions and then treating the solutions by solvent extraction. This was followed by decontamination of the process systems. In spite of the Pu-recovery and decontamination, the doserates and activity levels in the shielded cells as well as the level of α-contamination are too high to allow direct access work. The decommissioning work is thus basically structured around the following:

- a remote controlled machine for dismantling the equipment in the main (hot) cells (see Figure 3)
- a modular α-tight workshop to dismantle the glove boxes and the cells devoted to Uranium and Plutonium separation and processing
- separation and activity measurement of waste arising from dismantling to comply with new disposal site regulations regarding the content of α-emitters.

The situation at present is as follows:

- a concept has been chosen for the design of the equipment for dismantling work in the hot cells
- a modular workshop has been designed and built to fit with glove boxes of various sizes and with the available room around the various α-cells.
- a large quantity of α-bearing wastes has been generated. These are contained in 200 l drums and are awaiting measurement.

The dismantling machine is scheduled to be in operation by the end of 1988 and dismantling of the main cells is to take place during 1989-1992.

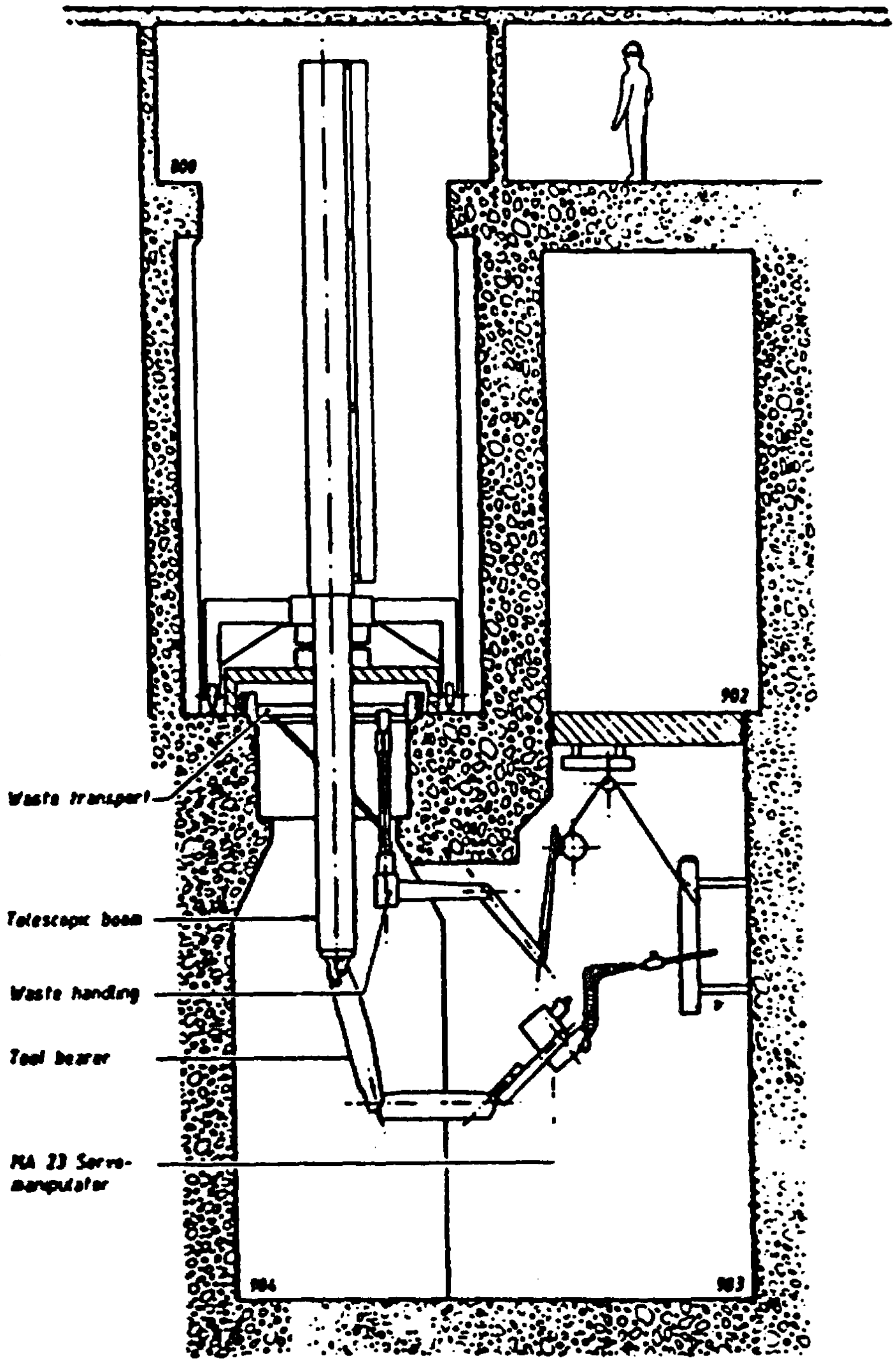


Figure 3

AT-1. Hot Cell Dismantling Principle

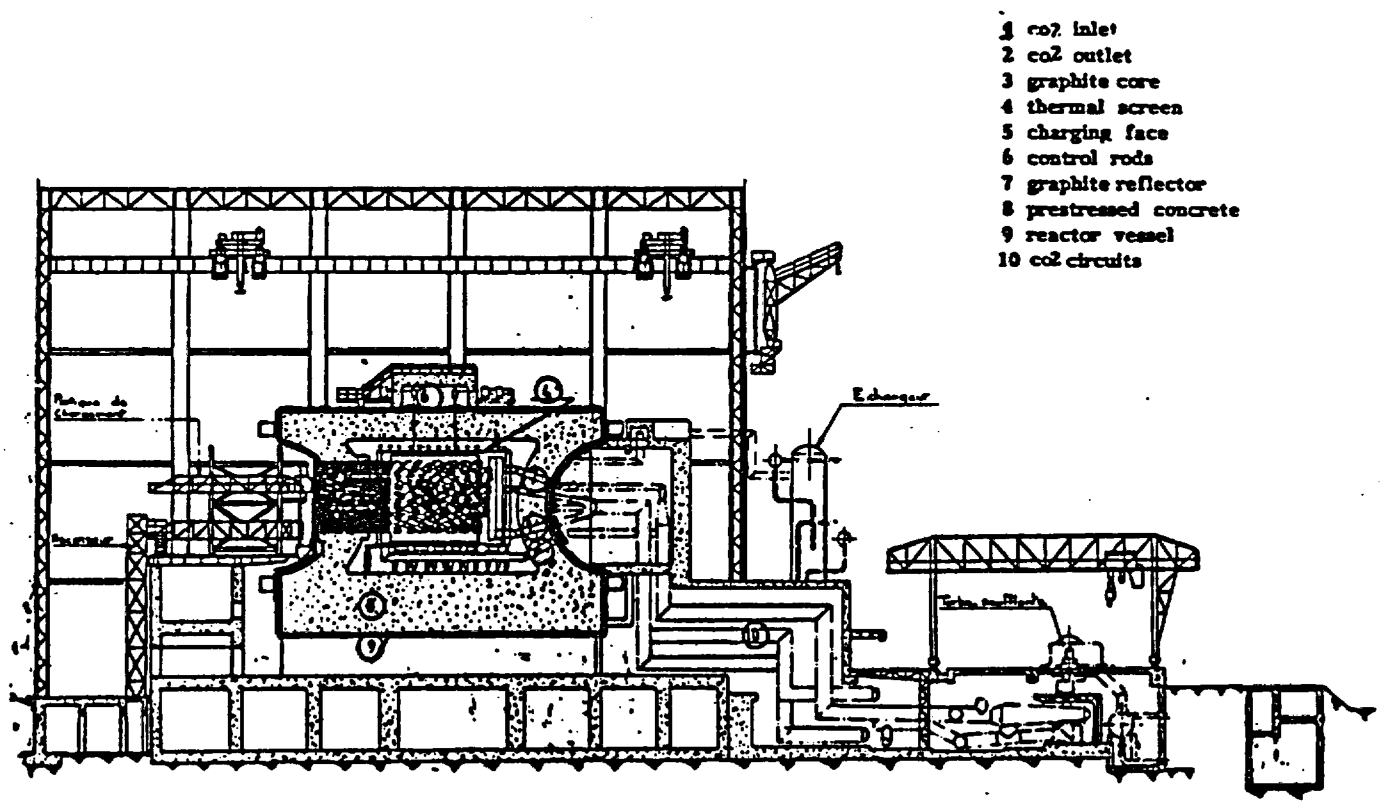
4 G2 (France)

G2 is a gas graphite reactor of 260 MWth (40 MWe) that operated between 1958 and 1980. The main cooling circuits and steam generators are outside the reactor vessel, which is a pre-stressed concrete pressure vessel (see Figure 4). This arrangement makes it convenient for a stage 2 de-commissioning for the time being.

The reactor vessel contains an octagonal graphite core surrounded by a 0.8 m graphite reflector, in its turn shielded by 12 cm thick steel plates. The pressure vessel is a horizontal cylinder of 14 m inside diameter and 18 m long, with a wall of 3 m thick concrete. No decommissioning activity is planned at present for the pressure vessel or its contents.

The CO_2 circuits consist of about 1700 m of pipes, 1000 m of which are over 1 m in diameter - a total weight of 1100 t. There are 4 steam generators, each of which is 3.5 m in diameter and 32 m long and weighs 300 t. This large quantity of steel has the very low contamination level of 10^{-3} $\mu Ci/cm^2$ (37 Bq/cm^2). So the main direction of the decommissioning project is the decontamination of this material to a level low enough to allow re-use or re-sale as scrap iron.

Currently studies are being made on whether to decontaminate the systems in situ or in a central facility after dismantling and cutting up the pipes. Work is also going on the choice of decontamination techniques. Pilot scale decontamination tests have been carried out on samples cut out from different parts of the primary systems. A number of processes have been found satisfactory. Larger scale testing is being planned for inproving these processes as well as for determining the type and amount of effluents produced and their future treatment.



<u>Figure 4</u>

G2. Reactor Core and Circuits

5 KWL LINGEN (Federal Republic of Germany)

Lingen is a 240 MWe dual-cycle BWR, with a fossil-fired superheater, that was taken into operation in 1968 and shut down in 1977. The uncertain costs for obtaining a new operating license after installation of new steam converters with associated backfitting led to a decision to permanently shut down the station. A cross-section of the reaction building is shown in Figure 5.

Changes in the operating license for the reactor were sanctioned in 1981, whereby the owners could carry out a number of alterations to adapt the plant to existing circumstances. Thus a number of pipes leaving the controlled area could be cut and sealed. All cooling water pipes to and from the reactor building were disconnected. Permission was granted to reduce power supply, instrumentation and shift personnel. The permitted rates of release of radioactivity to air and water were also reduced.

In November 1985 Lingen received a decomissioning license by which

- the plant can be finally shut down
- it can be brought to a status of safe enclosure (stage 1) by the end of 1986
- it can continue to be in that status for a period of 25 years
- the conventional part of the plant will be released from consideration under the Atomic Energy Law.

A small airconditioning system will be in operation to keep the air between 30 to 50 % relative humidity to reduce corrosion of the plant.

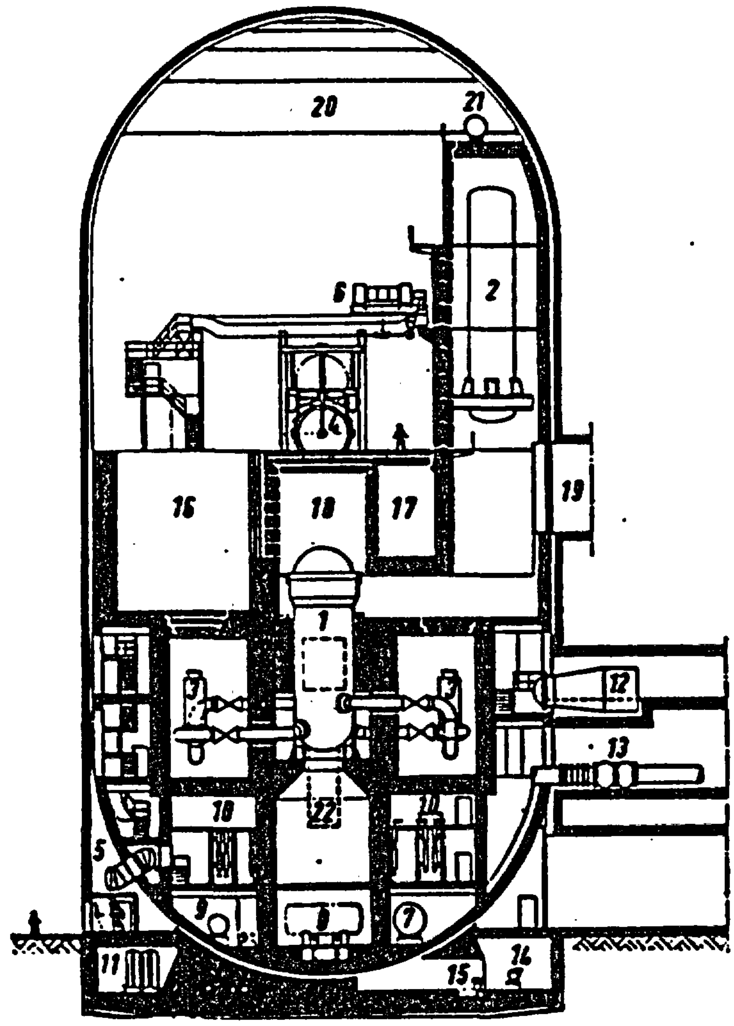


Fig. 1 Cross Section of Reactor Building. **1** reactor pressure. **2** steam converter. **3** forcing pumps. **4** materials lock. **5** accessory lock. **6** reactor building crane. **7** inactive drain tank. **8** active drain tank. **9** intermediate tank. **10** control rod drive system. **11** Waste gas retarding plant. **12** personnel air lock. **13** ventilation lines. **14** core spraying pump. **15** sump water pump. **16** fuel element storage area. **17** equipment storage area. **18** reactor well. **19** pipe support bridge. **20** containment spray system. **21** high level cooling water tank. **22** control rod drives.

Figure 5

Lingen: Cross Section of Reactor Building

6 KKN NIEDERAICHBACH (Federal Republic of Germany)

KKN Niederaichbach is a 100 MWe gas cooled heavy water moderated reactor that was taken into operation in 1972 and shut down in 1974. During that period, the reactor operated for 18 full power days (at 40 % nominal power). The plant has since been in a stage 2 decommissioning status. The fuel and heavy water have been removed from site and all radioactivity is at present confined to the safety containment in the reactor building. (See Figure 6).

An application for a permit to totally dismantle the plant (stage 3 decommissioning) was filed with the authorities in March 1980. A license to proceed with the decommissioning was granted in June 1986. However, objections have been raised by intervenors and hearings are planned to be held in the spring of 1987.

The long period between the first application for the decommissioning permit and its being granted has been caused by requirements for further information on a number of issues such as

- definition of the waste management concept that includes recycling of contaminated steel components by melting
- maximum credible accident during decommissioning
- significance of certain nuclides such as Fe 55 and Ni 63 which have an importance during decommissioning that they did not have during station operation
- definition of release procedure for large quantities of non-active waste.

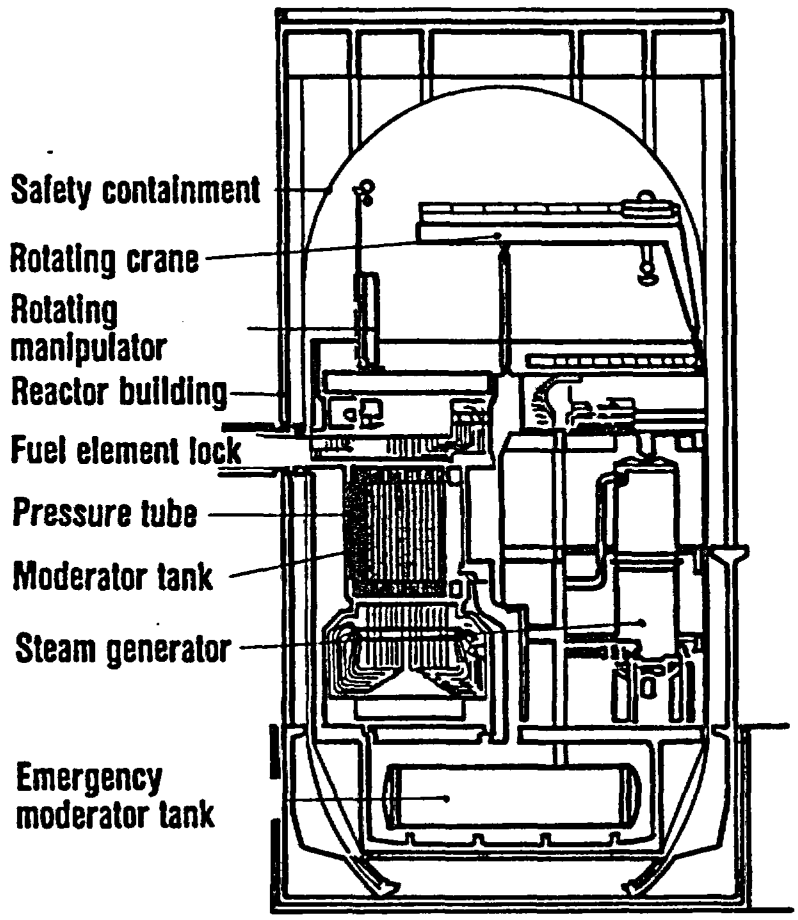


Figure 6

Niederaichbach. Reactor Building

In order to avoid the spreading of contamination, the dismantling of the facility is planned to be performed, as far as possible, inwards from areas of low activity to those with high activities. The dismantling of the equipment within the biological shield will be carried out almost exclusively by remotely operated manipulative devices without water shielding.

The metal melting facility for contaminated waste will be located at the Karlsruhe Nuclear Research Centre. A construction/operation license has already been granted. It is scheduled to begin operation in the middle of 1987 and is expected to recycle the 1700t of contaminated metal waste from KKN during 2 years.

According to existing plans, the dismantlement and removal of active components is scheduled for 1987 and 1988, while the more advanced remotely handled dismantlement will take place in 1989 and 1990. This will be followed by building decontamination by the end of 1992. However, the delay in the permission to start may affect the above schedule.

7 GARIGLIANO (Italy)

The Garigliano Nuclear Power Plant is a 160 MWe dual-cycle BWR that started commercial operation in 1964. It was finally shut down in 1978.

The current decommissioning project is to

- place the reactor containment with its contents in a Stage 1 decommissioning status
- dismantle and decontaminate to release-able levels the contaminated equipment and facilities in the Turbine and other buildings.

Prior to this, the ca 335 fuel elements will be transported off site and the accumulated rad-waste will be treated for disposal.

A feasibility study regarding the dismantling and decontamination of the turbine building systems and components has given the following results:

Some 32 000 m^2 of contaminated metal surfaces will be decontaminated, whereby it is expected to remove about 980 mCi of an estimated inventory of 1020 mCi of radioactivity. Most of the material will be decontaminated by electropolishing. Other methods used will be

- high pressure water with abrasives
- ultrasonics in a strong acid bath
- freon

Analysis has shown that decontamination after dismantling is more advantageous than in-situ decontamination. In order to evaluate the possibility of reducing the extent of decontamination, studies are being planned on melting and re- cycling materials for controlled use.

According to present plans, the station will be placed in a Storage with Surveillance status by 1992 or 1993.

8 JPDR (Japan)

The Japan Power Demonstration Reactor is a 90 MWe direct cycle BWR which operated from 1963 to 1976. The current project is a Stage 3 decommissioning of the reactor whereby all parts of the station, except for the administrative building and a storage facility, will be removed to 1 m below ground level and the entire site landscaped.

The dismantling of the JPDR is aimed at providing techniques and the data base on systems engineering, such as manpower, radiation exposure and waste streams necessary for the future decommissioning of large commercial reactors. In addition, the current project should give valuable information on regulatory control for use in the future.

Preparatory to the actual decommissioning, the JPDR Decommissioning Programme has during a first phase from 1981 to 1985, studied, adapted and/or developed a wide range of dismantling and decontamination techniques. During the second phase of the programme, these techniques will be applied during the actual dismantlement of the plant. The dismantlement will thus be a full scale verification of some of the techniques developed. The dismantlement phase has been started in December 1986 and will be completed in 1991. The dismantlement procedure is shown in Figure 7. The reactor internals and pressure vessel will be dismantled and cut up from late

1987 through 1988, while the biological shield will be removed in 1989.

One feature of the JPDR programme is that the components and structures will be cut up into pieces rather than be removed in one piece, in order to demonstrate the typical decommissioning procedure for a commercial power reactor. Another feature is the large number of techniques demonstrated, as, for instance:

- Underwater plasma arc technique is used for detaching reactor internals from the reactor vessel and for cutting them into smaller pieces for packing. The plasma arc mechanism is handled by a mast-type remote mechanism as well as a robotic manipulator.
- Pipes to the reactor vessel will be cut by shaped explosives or by a disc cutter.
- The reactor vessel itself will be cut up using an underwater arc saw.
- Diamond sawing and coring as well as abrasive water jets will be used for removing the concrete biological shield. Controlled blasting will be used for the demolition of low radioactive concrete structures.
- New techniques such as microwave cracking will be used for decontaminating concrete surfaces.

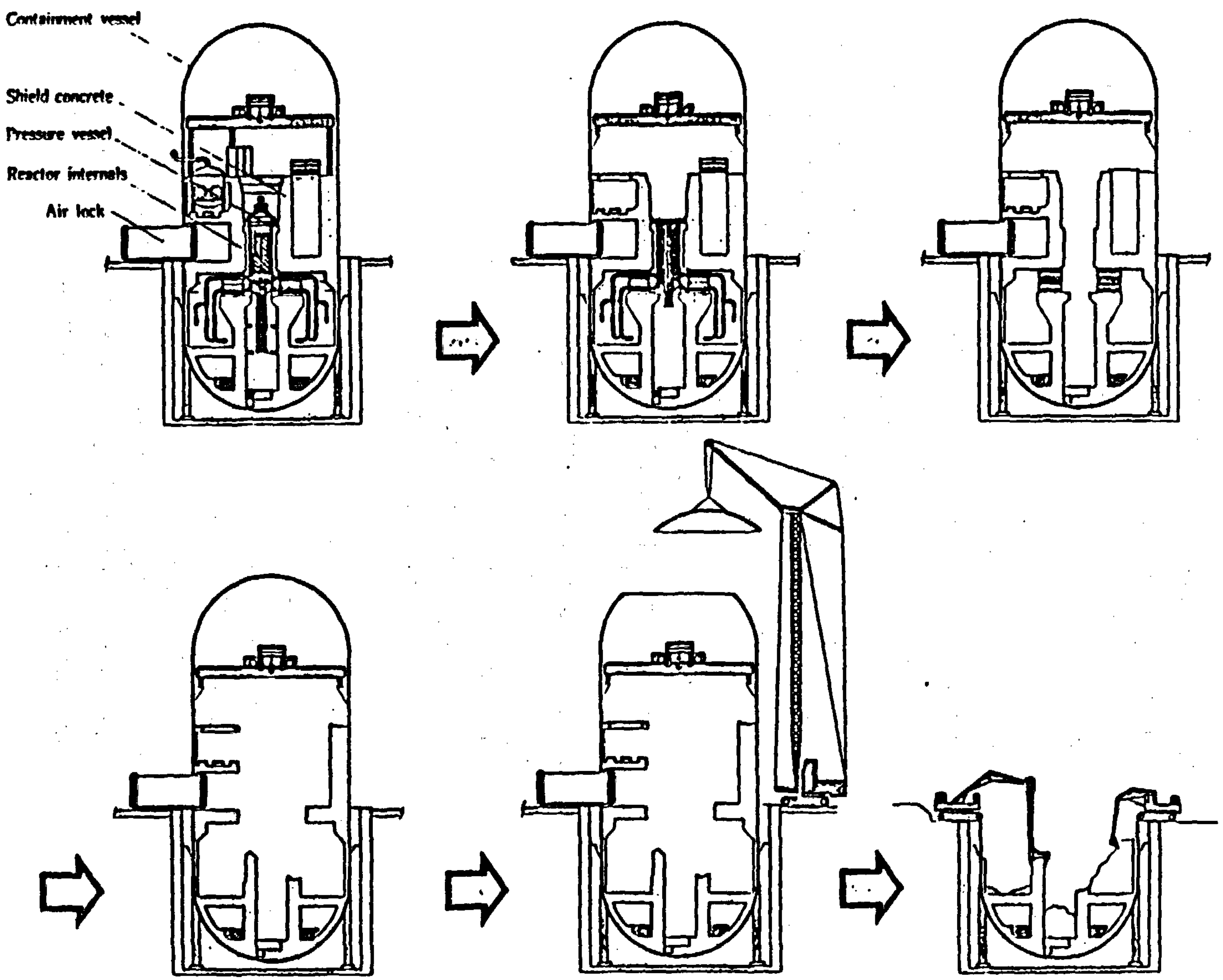


Figure 7

JPDR. Dismantling procedure of reactor, its surroundings and containment.

9 WAGR (United Kingdom)

The Windscale Advanced Gas Cooled Reactor is a 100 MWth prototype AGR that operated for 18 years from 1963 to 1981. The decommisioning project aims at achieving a Stage 3 status by 1996.

All fuel has been removed to dry storage. The nonactive turbine hall has been cleared.

The main radioactive dismantling work is to be performed by a remotely operated machine, which will be placed on a rotating shield above the reactor vessel, after its top has been removed. The machine will consist of a vertical mast and a manipulator to which a variety of cutting tools can be attached. (See Figure 8).

Approval has been received for the design of this remote cutting machine. Robot tests for handling oxy-propane cutting equipment are going on as is planning work for the installation of an active drain for effluents.

One of the four steam generators has been jacked up about 12 m in order to utilize an existing shielded route to the waste packaging building. A standardized steel-reinforced concrete waste container has been designed for meeting the requirements of site storage, transportation and future permanent disposal. Prototype containers have been successfully produced and tested.

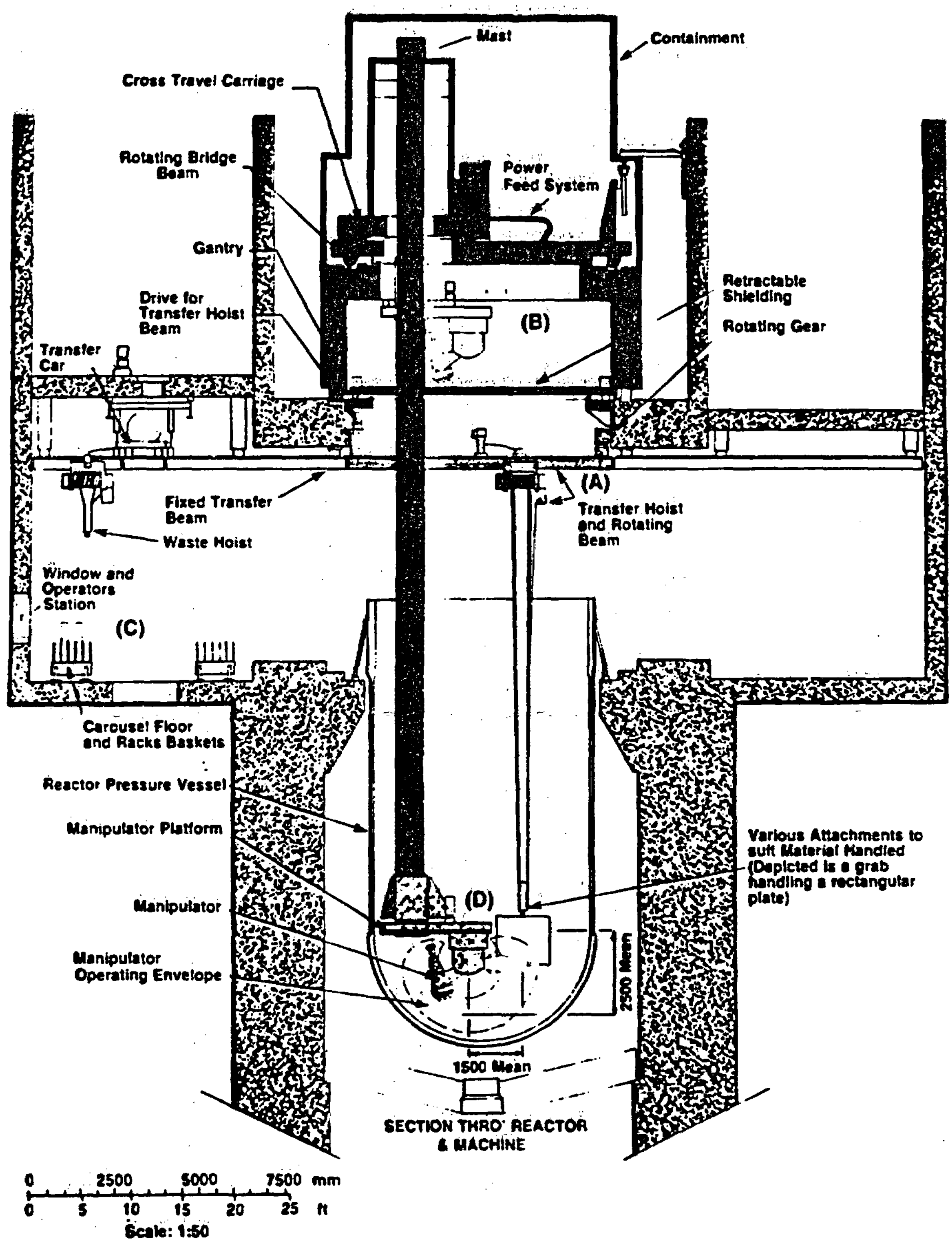


Figure 8

WAGR. Remote handling and transfer hoist equipment (designed in conjunction with Strachan and Henshaw Limited)

10 SHIPPINGPORT (United States of America)

The Shippingport Atomic Power Station was a 72 MWe PWR that operated between 1957 and 1982. From 1977 onwards, it was operated as a light water breeder. The project is organized as a demonstration of the safe and cost effective dismantlement of large scale nuclear power plant. A large number of subcontractors is being utilized in order to increase the technology base and experience.

One interesting feature of the Shippingport Station Decommissioning Project is the one-piece removal of the reactor vessel. The vessel with internals will be lifted out of the plant and placed in a shipping cradle. The neutron shield tank (outside the vessel), which was water-filled during operation, will be filled with concrete as shielding prior to transport. The reactor vessel package contains over 98 percent of the estimated total plant radioactive inventory and will have 6460 Ci at the time of shipment (See Figure 9).

The reactor vessel package together with the steam generators, other large contaminated components and piping will be shipped on a special barge down the Ohio and Missisippi rivers, through the Panama Canal, and up the Columbia river to Hanford. There they will be buried in a approved disposal facility. It is estimated that this single large transport would replace 80 truck loads over land, which would be necessary for transport following a segmenting of the vessel.

Site operations were started in late 1984. The flywheel generator bulding has been demolished. The pressurizer and the flash, blowdown and injection tanks have been removed. A large quantity of asbestos has also been removed.

Removal of primary components will be completed in 1987 and work on concrete structures will be started. The reactor vessel package will be lifted off for transport in 1989. In the same year, concrete removal and backfilling will be done. The project will be completed in 1990.

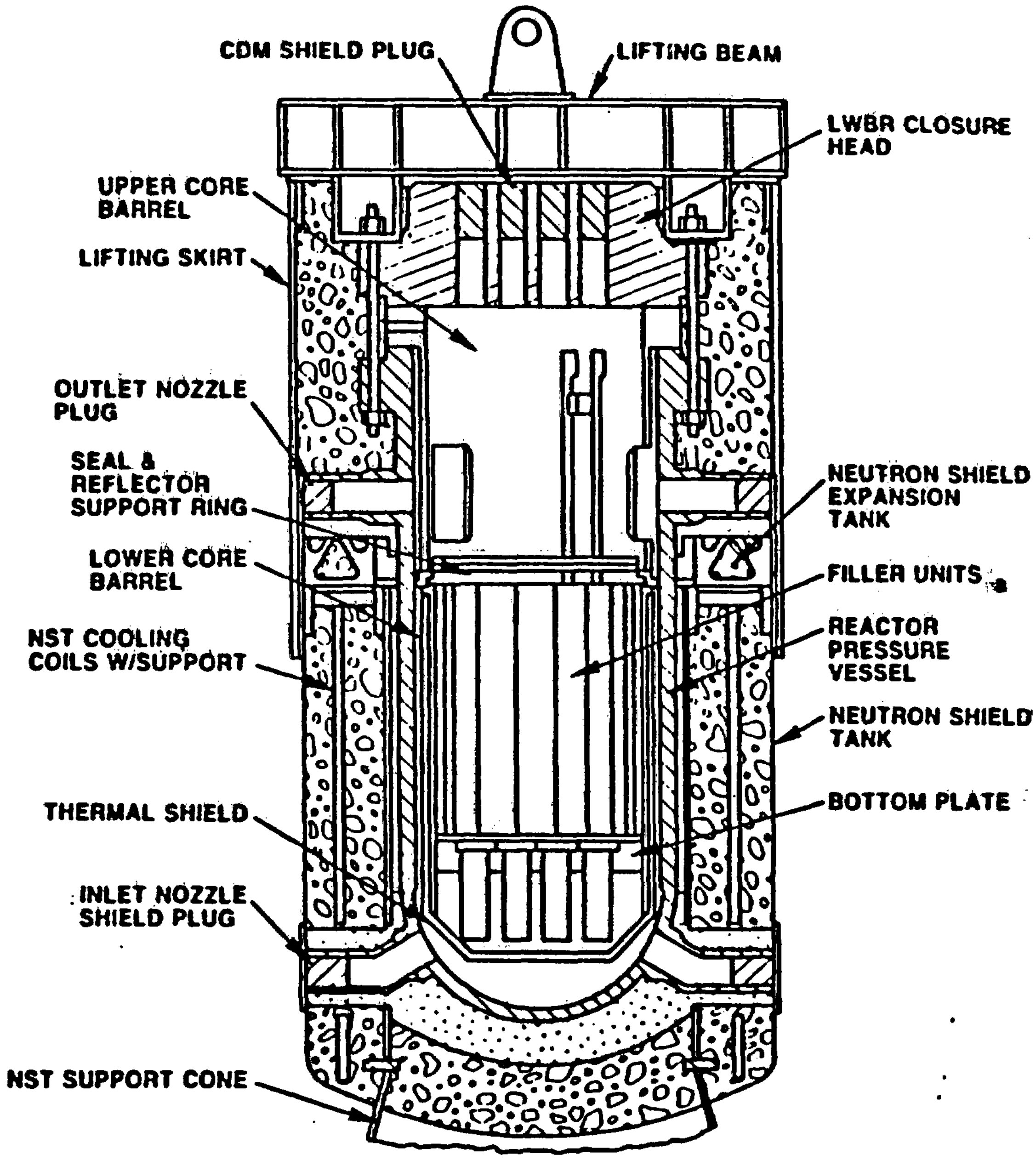


Figure 9

Shippingport station decommissioning project.
Reactor vessel package