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FOREWORD

Gas cooled reactors (GCRs) and other graphite moderated reactors which share a common technology have been an important part of the world's nuclear power programme for the past four decades. This includes a very wide spectrum of plants whose present status covers initial design to decommissioning. This wide diversity in status was a major consideration in the recommendation by the International Working Group on Gas Cooled Reactors for the IAEA to convene this Technical Committee meeting on the subject of GCR decommissioning, including spent fuel storage and radiological waste disposal. The IAEA staff member resposible for this publication was L. Brey of the Division of Nuclear Power.

EDITORIAL NOTE

In preparing this publication for press, staff of the IAEA have made up the pages from the original manuscripts as submitted by the authors. The views expressed do not necessarily reflect those of the IAEA, the governments of the nominating Member States or the nominating organizations.

Throughout the text names of Member States are retained as they were when the text was compiled.

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SUMMARY

1. OVERVIEW AND PURPOSE

The Technical Committee Meeting (TCM) on Gas Cooled Reactor Decommissioning, Fuel Storage and Waste Disposal was convened by the IAEA on the recommendation of the International Working Group on Gas Cooled Reactors (IWGGCR), and was hosted by Forschungszentrum Jülich (FZJ), in Jülich, Germany, from 8 to 10 September 1997. Approximately ninety participants and observers from twelve countries (China, France, Germany, Japan, the Netherlands, the Russian Federation, Slovakia, Spain, Switzerland, South Africa, the United Kingdom and the United States of America) and the European Commission were in attendance.

The purpose of the meeting was to provide an opportunity to review the status of decommissioning and associated spent fuel storage and component waste disposal programmes and issues related to gas cooled reactor (GCR) plants including facilities sharing common technological aspects such as other types of reactors which have graphite moderators; and especially to identify pathways which may take advantage of the opportunity for international cooperation on developments addressing these activities. The meeting focused on those aspects of decommissioning which are unique and distinctive to GCR plants rather than on generic nuclear power plant decomissioning processes and storage and disposal experiences which are already well understood.

2. PLANT DECOMMISSIONING PROGRAMME STATUS

The majority of the presentations addressed decommissioning activities associated with the high temperature gas cooled reactor (HTGR) and Magnox GCR plants. Many of these plants are shut down and are undergoing selected equipment dismantlement with the chosen decommissioning option of safe store for times ranging from 20 to 135 years. Specific to the HTGR, there are three plants in varying stages of decommissioning; the German Arbeitsgemeinschaft Versuchsreaktor (AVR), the Thorium High Temperature Reactor (THTR), and the Fort St. Vrain (FSV) station in the USA.

Defuelling of the AVR is now partially complete, with approximately 23% of the (pebble bed) fuel removed from the core. A strategy has been developed for dismantling of the plant which began with the turbine hall and other outside equipment and will progress to include dismantling of the reactor vessel, and, possibly, dismantlement of the containment with the goal of final restoration to "green field" status in 2011. The decommissioning project plan includes the ability to place the plant into the state of safe enclosure (SAFSTOR) following the completion of each dismantling step. Relative to the German THTR, defuelling was completed in 1995. This operation required approximately one year to accomplish and included weekly spent fuel shipments to the Ahaus fuel interim storage facility. Establishment of the THTR to SAFSTOR was then initiated for an intended time frame of 30 years. This will be followed by completion of the remaining dismantlement activities.

The Fort St. Vrain HTGR plant was shut down in 1989 after approximately ten years of commercial operation. The initial activity following shutdown of this HTGR was to construct a modular dry vault storage facility for the cores' 1,482 hexagonal graphite fuel

elements. Defuelling of the reactor to this facility then required approximately six months to complete. After assessing the different decommissioning options available to nuclear power plant license holders in the USA, the plant owner, Public Service Company of Colorado, chose to proceed with early dismantlement. This decommissioning method was selected in order to eliminate long term financial risks and mitigate extended environmental impacts to the company, its customers and neighboring communities. The major effort in early dismantlement was the removal of the primary system internals and subsequent cutting and sectioning of the prestressed concrete reactor vessel using diamond impregnated wire cutting equipment. Internal vessel component removal was accomplished through the use of underwater dismantlement techniques. Flooding of the reactor vessel provided an excellent radiological exposure shield for the workers who often performed their tasks underwater in diving equipment. On 5 August 1997, the US Nuclear Regulatory Commission determined that the plant site is available for "unrestricted use" and subsequently terminated the plants' reactor license.

The preferred decommissioning strategy in the United Kingdom for the Magnox stations is "Safestore". This strategy allows for a stepwise approach beginning with defuelling, then a period (~35 years) of "care and maintenance" and ending with deferral of final dismantlement of the reactors for an additional ~100 years to obtain the benefits of radioactive decay. Specific site and equipment decommissioning activities would take place at discrete intervals during the Safestore period. There are three twin Magnox power stations which have now been defuelled and their fuel removed from the plant sites. Initial efforts have been directed to the decommissioning pre-planning processes of defining the organizational needs and requirements, establishment of regulatory processes and the development of procedures to allow for the actual work to be implemented in a safe, efficient and economical manner. Included in this pre-planning is the necessity to show that significant degradation of the physical, mechanical and chemical properties of the graphite moderator will not occur over the Safestore period. Among the areas under examination are the rate of chemical oxidation, formation of explosive dusts, consequences of accidential exposure to moisture, the potential for gas-phase and particulate release including the biological degradation of the graphite.

Other plants of the Magnox type which were addressed at the TCM included the Vandellos 1 nuclear power plant in Spain and the Tokai 1 unit in Japan. Defuelling of the Vandellos reactor was completed in 1994 with subsequent dismantlement of the spent fuel ponds achieved early this year. Currently, nearly all of the radwaste generated during the operation and the waste (primarily graphite fuel support sleeves) from three silos at Vandellos have been extracted and packaged.

Japan Atomic Power Company's (JAPC) Tokai 1 Magnox type GCR will be shut down permanently in early 1998, after 31 years of power operation. Preparations, including studies and research on plant characterization, remote-cutting and waste disposal are now under way in order to assure the safe and economical decommissioning of this plant. The ongoing cooperative study by JAPC, NNC Ltd and Fuji Electric of the neutron flux profile and radioactivity around the pressure vessel will provide supporting information in determining the activity of neutron activated structures and dose rates anticipated during dismantlement.

The decomissioning strategy for the prototype nuclear power plant A-1 in the Slovak Republic includes safe enclosure of the confinement; however, an updated plan for the

decommissioning of this plant is to be issued in the near future. This heavy water gas cooled reactor plant was permanently shut down in 1977 following an accident which resulted in localized overheating of fuel and subsequent contamination of the primary and secondary circuits. The initial decommissioning related activities on this plant included dismantling of the secondary circuit and cooling towers and transfer to Russia of all spent fuel which could be manipulated. This included 440 fuel assemblies and was completed in 1990. In parallel, the technologies for radwaste treatment and conditioning were developed and bituminization of the operational radwaste concentrates was performed in 1995–1996. Processing of the spent fuel coolant and construction of new spent fuel handling equipment and a semi dry storage facility is currently taking place. Other equipment and facility decontamination and dismantling work will continue in conjunction with the safe storage of the confinement.

3. COMMON AREAS OF CONSIDERATION IN THE DECOMMISSIONING OF GCR PLANTS

Areas of common interest and discussion by the participants at the TCM in the decommissioning of GCR plants were predominantly focused on the treatment and disposal of the graphite components which constitute the major volume of the core and the requirements associated with the handling and final disposition of the spent fuel.

3.1. Spent fuel transportation and storage

Both the AVR and THTR have incorporated the CASTOR cask in the transportation and storage of spent nuclear fuel. The development of this cask began in 1982 with corresponding testing and licensing to assure its integrity and safety for the dual requirements of transportation and interim storage-direct disposal of the fuel. These stringent requirements include a leakage rate for each of the double lid barriers of not to exceed 10^{-7} mbar.l/s. A total of 305 casks were required for the THTR core and, to date, 95 for the AVR spent fuel. Unloading of the core at the THTR was achieved in a manner similar to the removal of fuel during normal operation. However, some process modifications to the fuel charging system were necessary prior to initiation of defuelling in order to meet the shutdown conditions of reduced temperature and pressure as well as application of nitrogen instead of a helium environment.

Research and development activities are continuing at FZJ on the intermediate storage and final disposal of HTR fuel elements. Safety studies involving the release of gaseous radionuclides and storage cask development have been the principle focus for intermediate dry storage of this fuel. Investigation of the final disposal has centered around the behaviour of the fuel elements and kernels in a concentrated salt brine and the design of storage containers for subsequent disposal in a salt mine.

Plans for the storage of spent fuel are being developed for the High Temperature Engineering Test Reactor (HTTR), the High Temperature Reactor (HTR-10) and the Gas Turbine-Modular Helium Reactor (GT-MHR). The fuel assembly in the Japanese HTTR consists of fuel rods in hexagonal graphite blocks. The core is to be replaced after approximately three years of operation. The spent fuel is then retained for about two years in the reactor building fuel storage pool prior to transfer to air-cooled storage racks which are designed to contain ten core inventories. The options available for the treatment of spent fuel include reprocessing, long-term repository followed by reprocessing and direct disposal. The

Japanese intention is to reprocess all spent fuel. The Chinese HTR-10 incorporates the pebble bed core which is made up of spherical fuel elements 6 cm in diameter. These fuel elements are recycled through the core until a burnup of between 74 300 and 87 100 MW·d/t U is achieved. At that time the spent fuel is discharged into lead-steel containers and transferred to a special compartment in the reactor building for storage. This storage area is sized to handle the full inventory of the HTR-10 throughout its lifetime. The method of final disposition of the spent fuel is currently under study.

The strategy under consideration in Russia relative to the disposal of spent fuel from the GT-MHR includes storage in a geological repository without additional processing. The GT-MHR is under consideration for the disposition of plutonium and utilizes ceramic coated fuel particles in hexagonal graphite blocks. The coatings provide an excellent engineered barrier for containment of radionuclides. This, coupled with the high level of plutonium burn-up capable in the GT-MHR, provides an effective mechanisn for disposal without the need for intermediate processing to assure non-proliferation and safeguards goals are achieved.

3.2. Graphite characterization and disposal

The treatment of irradiated nuclear graphite was a significant concern to the participants at the TCM and was the focus of a number of presentations. There are nearly forty graphite moderated reactors in Russia. A key influence in the operating lifetime of these reactors is the amount of degradation in the physical, mechanical and structural properties which has taken place in the graphite. The decommissioning programme for these reactors includes the development of safe handling of the irradiated graphite and investigation of different disposal technologies such as burning, volume reduction and impregnation by conservants.

The treatment of graphite is also a key consideration in the decommissioning activities at the Windscale (now Sellafield) site in the United Kingdom include the Windscale Advanced Gas-cooled Reactor (WAGR) and the Windscale Piles Nos 1 and 2. Decommissioning of the WAGR has included reactor dismantlement to the level of the hot gas collection manifold, and the removal of the top biological shield, refuelling standpipes, top section of the reactor vessel and the heat exchangers. AEA Technology is currently developing equipment and the methodology for the dismantlement of the core structure. The intended programme for the Windscale Piles is for the "safe storage" of Pile No. 2 and the dismantlement of Pile No. 1, which was damaged by fire in 1957. Evaluation of safety issues in these reactors has been predicated on data received through extensive investigation programmes including non-intrusive surveys, intrusive visual and swab identified blocked channels, assessments of residual contamination levels, and trepanning of the graphite with associated analysis of samples and sample holes. To date, there does not appear to be any obstacles to the continued storage of Pile 2 and the data received is being utilized in support of the safety case for Pile 1 including the development of handling, storage and disposal procedures for the graphite blocks.

A technique under study for the disposal of graphite is fluidized bed incineration. Of particular interest is the release and radiological impact of carbon 14. The analysis of the incineration of ~600 tonnes of contaminated and irradiated graphite per year over 50 years is expected to result in an increase in the atmosphere's carbon 14 inventory by 5×10^{-3} times that from natural sources. This is seen to be small in comparison to the natural fluctuation of

carbon 14 concentration in the atmosphere as determined in the last one hundred years.

3.3. Waste handling and disposal

Considerable experience is being gained from the dismantlement activities at the THTR and AVR. Preplanning of these activities required a thorough evaluation of the process requirements and physical limitations specific to each respective plant. A key aspect of this has been in the training of personnel prior to initiation of major activities. An example of this effort was with the disassembly of the THTR fuel burnup measurement facility which included training on a full scale mock-up primarily to reduce the radiological dose to personnel performing the disassembly. Extensive evaluation and preplanning for the dismantling and treatment of decommissioning wastes has also been an ongoing activity at the AVR. This has included studies for the packaging of waste materials so that they are safely transported and placed in storage in a manner which optimizes the use of materials to achieve high shielding efficiency and to meet or improve on associated regulatory requirements.

Nirex Ltd of the United Kingdom is developing a deep underground repository for the disposal of intermediate and selected low level wastes. GCR related waste material to be placed in the repository includes graphite fuel struts and sleeves, fuel boats and dowels, prototype reactor core blocks and Magnox fuel can components, activated steel components, sludges and ion exchange resins from clean-up activities. Excluded from the repository are the actual cores which will remain in Safestore for 135 years. The specifications for packaging these wastes continue to be developed in relation to their physical and chemical characteristics.

A waste handling center is being developed at the site of the Chernobyl nuclear power plant complex. This center is part of the remediation and restoration programme for the Chernobyl exclusion area and includes a facility for the melting and subsequent recycling of contaminated metallic materials. This technology is based on the separation effect that occurs to the radiologically dominant nuclides of radioactive metal during the melting process where the active nuclides transfer from the basic material into the waste process slag and filter dust. The results from this process include a significant reduction in radioactive wastes with the corresponding production of shielding equipment and the manufacture of casks and containers for use in further disposal of wastes.

4. CONCLUSIONS AND RECOMMENDATIONS

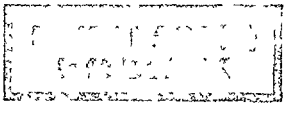
Concluding comments from a member of the IWGGCR referenced the need for further scientific work in the treatment of irradiated nuclear graphite. Also referenced as a need for further study were the benefits and drawbacks associated with the length of the safe storage period which varied significantly between Member States, and the need by many countries to establish specific decommissioning regulations and guidelines. It was further acknowledged that the TCM provided the opportunity for sharing of information between Member States in the areas of decommissioning, spent fuel storage and component waste disposal of GCR plants and other facilities having common technical aspects.

A review of ongoing IAEA activities associated with nuclear power plant decommissioning was provided including a listing of related publications which are available from the IAEA. It was also indicated that activities associated with the characterization,

treatment and conditioning of radioactive graphite from decommissioning of nuclear power reactors are being incorporated within the IAEA's waste technology programme.

Also under consideration by the IAEA is the coordination and development of a database for the preservation of irradiated nuclear graphite related data by Member States. This database would allow participating Member States the capability of access to graphite property information which would be of specific help in the preparation of decommissioning and licensing plans and associated waste disposal activities.

STATUS OF PLANT DECOMMISSIONING PROGRAMMES



DECOMMISSIONING OF THE THORIUM HIGH TEMPERATURE REACTOR (THTR 300)

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Abstract

The prototype Thorium-High-Temperature-Reactor (THTR 300) was decommissioned using the option of safe enclosure. Decision was made in 1989 and safe enclosure was reached in February 1997, followed by up to thirty years of operation of the safe enclosed plant.

I. Introduction

The pebble bed high temperature reactor THTR 300 was shutdown on 01.09.89 after more than 16,000 h in operation. The THTR 300 is a prototype reactor project that is jointly sponsored by the Federal Republic of Germany, the state North Rhine Westphalia and the operator Hochtemperatur-Kernkraftwerk GmbH (HKG). The public financiers of this prototype reactor and the operator could not solve the financial problems for continued operation of this technically intact plant. The decommissioning decision had not been expected at the time by the operator. This is why safe enclosure the German term for SAFSTOR turned out to be the only technical solution for quick decommissioning of the plant, apart from financial reasons and the non availability of a final repository . The plant is intended to be dismantled after about thirty years of safe enclosure, provided respective funds are available. The decommissioning was done in three steps that were mostly scheduled one after the other (FIG. 1), /1/.

II. Description of the Work

A. SHUTDOWN OPERATION

Step 1 has included the conversion of plant operation from the power mode to the shutdown regime to keep the operating costs of the plant low until the license required under the Atomic Energy Act for the core unloading has been granted.

In shutdown operation, the shutdown rods were fully inserted and locked to prevent withdrawal. Recriticality of the reactor core was thus precluded.

Owing to the long outage period, which started when the reactor was shut down for the scheduled maintenance on September 29, 1988, forced residual heat removal by operating systems was not longer required. These systems have been taken out of service by depressurization, removal of operation media, cutting off the energy supply and by blockage. These measures also apply to the prestressed-concrete reactor vessel (PCRV) with the primary system in which the helium was replaced by air/nitrogen.

This lead to a reduction in the number of yearly inservice inspections from about 4,000 to 2,000. Moreover, savings have been achieved in terms of insurance, plant security, maintenance and through labor reduction, so that the monthly operating costs of about 9 million DM in power operation could be decreased to 5 million DM in shutdown operation.

B. CORE UNLOADING

Step 2 was the core unloading, according to Section 7 (3) of the Atomic Energy Act a prerequisite for the establishment of the safe enclosure /2, 3/. For the THTR 300 this meant that about 580,000 irradiated fuel elements, which still were in the reactor core, had to be unloaded. This could only be done by the complete unloading of the core, including the absorber and graphite elements that remained there, too. A worldwide first of its kind activity to a pebble bed reactor (*FIG. 2*).

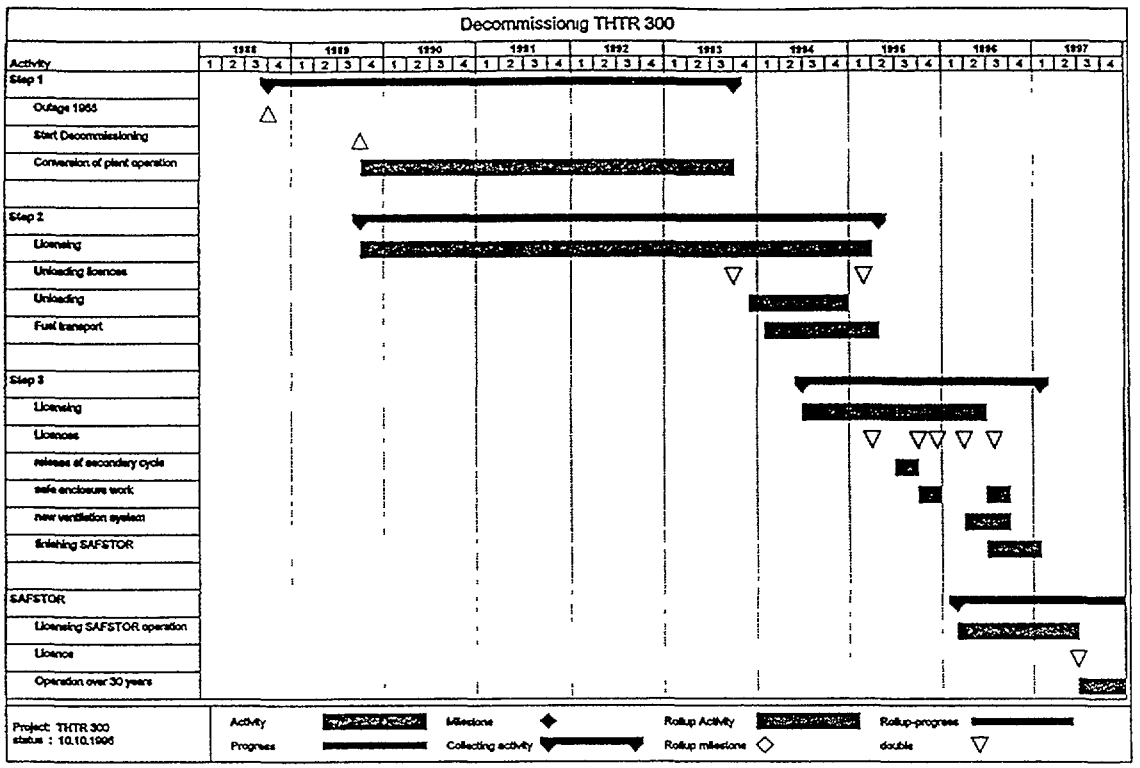


FIG. 1 *Time schedule, Decommissioning THTR*

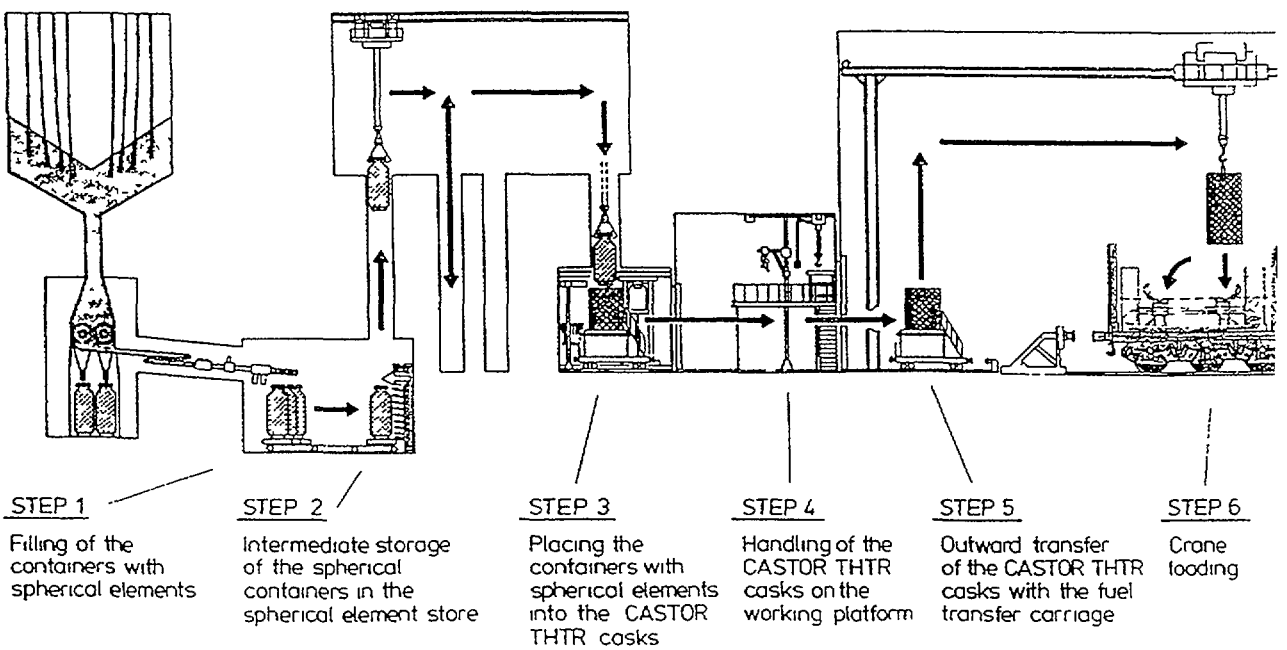


FIG. 2 *Core unloading, Management of the spherical elements*

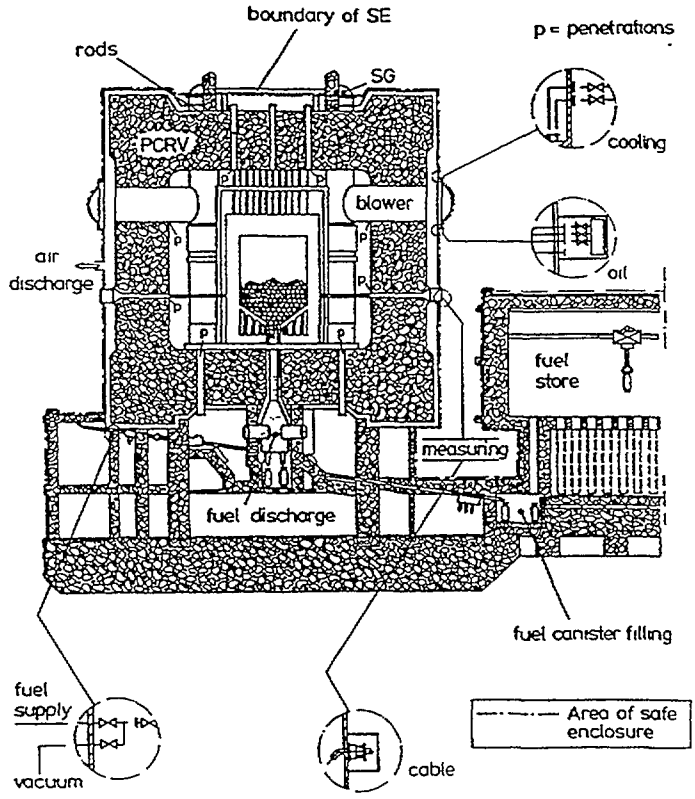


FIG. 3 *Closing scheme of penetrations through the safe enclosure*

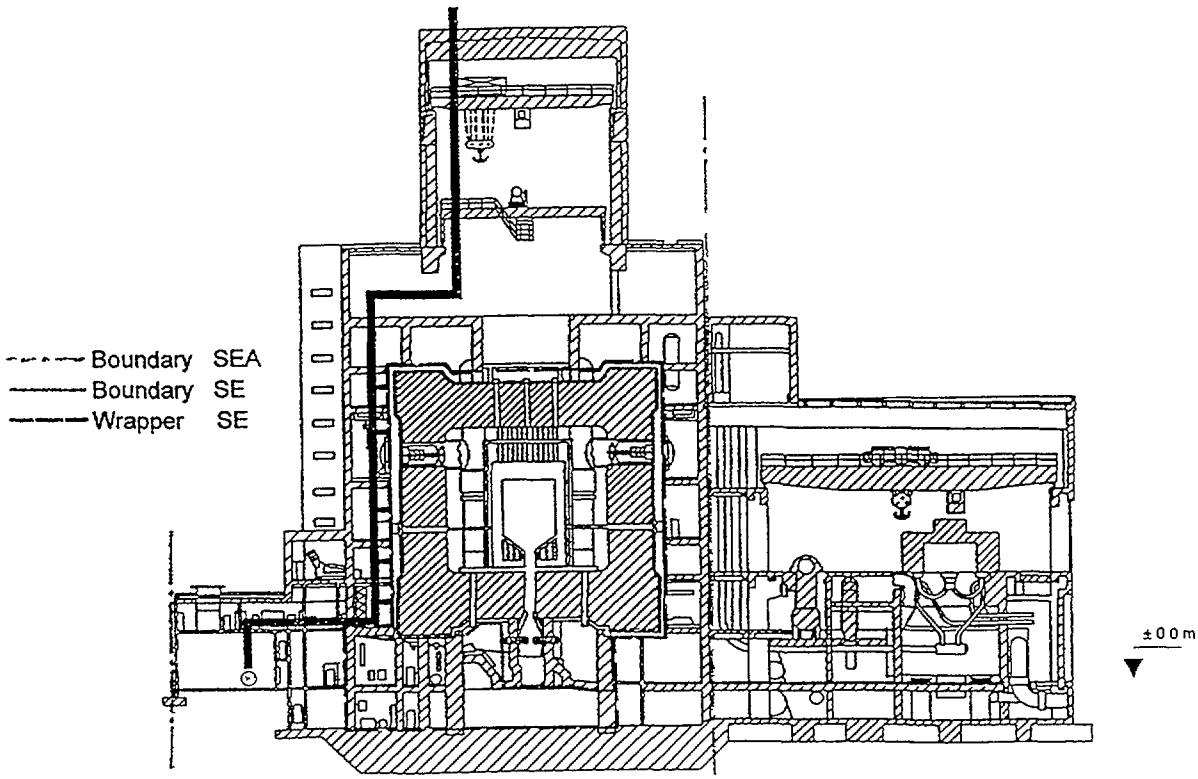
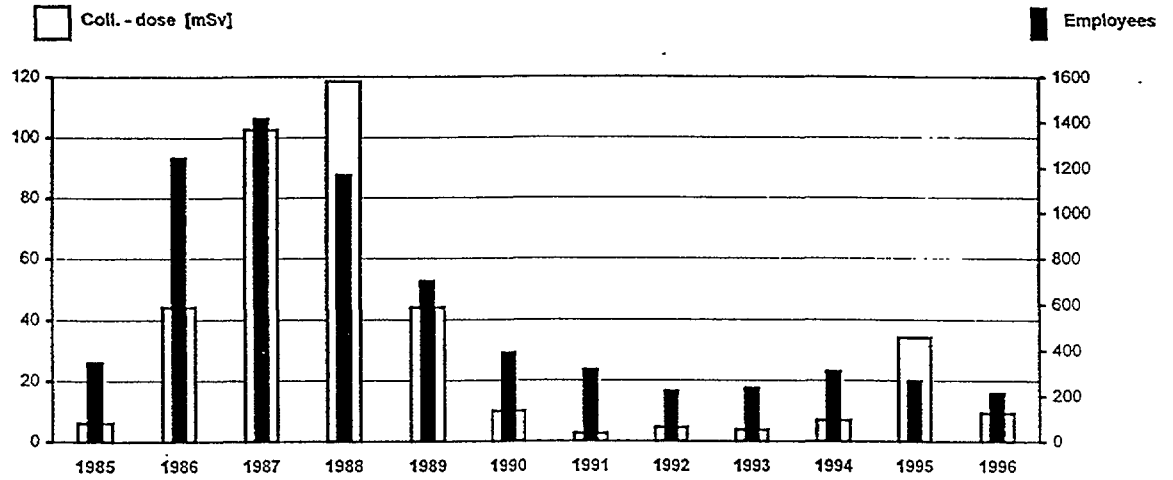


FIG. 4 *Safe enclosure concept (Reaktor building sectional drawing)*



1985	1986	1987	1988	1989	1990	1991	1992	1993	1994	1995	1996	
6,1	44,1	102,5	118,46	43,98	9,89	2,54	4,5	3,71	6,89	34,19	9,38	Coll. - dose [mSv]
352	1249	1418	1172	705	392	317	224	237	311	270	217	Employees

FIG. 5 *Collective dose from 1985 - 1996, Decommissioning during 1990 - 1996*

The respective license was granted after a four years long lasting licensing procedure in October 1993 (*TAB. I*) The unloading itself was executed in a one year period from Dec. 1993 till Dec. 1994 accompanied and followed by regularly (weekly) fuel transport campaigns with CASTOR casks to the Ahaus fuel interim storage facility (in total fifty-seven transports without any rumors as unfortunately on Gorleben-CASTOR transports).

Also the nearly hands-on decommissioning of the small burn-up measuring reactor, used for distinguishing fuel absorber and graphite elements and monitoring the burn-up of fuel elements containing 3.6 kg U^{235} in form of high enriched U-Al-fuel, took place just after finishing of core unloading in early 1995.

C. ESTABLISHING THE SAFE ENCLOSURE

Step 3, the establishment of safe enclosure, was started also in 1995 after applying for in 1994/95 and granting of attachments to the core unloading license in 1995 (*TAB. I*). The main steps undertaken and finished by a general contractor even in 1995 /4/ were

- enclosing the prestressed-concrete reactor vessel by cutting and sealing all approx. 2,000 penetrations (*FIG. 3*)
- sealing all primary circuit system components
- establishing of an additional enclosure for those sealed components by using the existing vented containment as a type of air flow guidance envelope
- release of the water-steam-cycle with turbine and generator and the four emergency diesel generators from the restrictions of the Atomic Energy Act
- preparation work for the establishment of a new ventilation system tailored to the requirement for the safe enclosure operation.

In April 1996 the first part of the next license (safe enclosure establishment and pre operational tests) was granted (see also *TAB. I*) concerning mainly the erection of the new ventilation and the exhaust air measuring system. That work was finished on schedule in September 1996.

TABLE I. LICENSING, DECOMMISSIONING THTR 300

step	application	license
Core unloading	19.12.89	22.10.93
Sorting of some operating elements	14.01.94	09.02.95
Closure of PCRV, steam cycle	13.07.94	23.05.95
Closure of wrapper SE	29.06.95	02.10.95
Dismantling the He-purification	04.09.95	27.10.95
Erection of new ventilation	28.06.94/ 06.12.95	26.04.96
Establishment of SE	28.0694/ 01.02.96	15.07.96
Operation of safe enclosed plant	14.05.96	21.05.97

TABLE II. OVERALL COST, DECOMMISSIONING THTR FROM 1990-2009

	Mio. DM
Waste	253.0
Experts	55.0
Contractors	112.0
Operation 1990 - 2/1997	288.5
Operation 3/1997 - 2009	35.0
Financing	30.0
Total	773.5

TABLE III. SOLD EQUIPMENT OF THTR 300

	Mio. DM
Secondary cycle - steamturbine - generator - auxiliaries	15.0
Transformer	4.1
4 emergency diesel generator sets	3.0
Spare parts, tools etc.	1.3
Electrical-, communication-, radiation monitoring equipment	1.2
Total	24.6

The second but more important part of this license was granted in July 1996, containing the main steps for the establishment of safe enclosure and allowing to:

- dismantle the liquid waste store and evaporation system, decontamination shop and the like
- adapt the power supply
- dismantle contaminated equipment outside safe enclosure that doesn't fulfill the requirements of this area later concerning contamination limits
- adapt the drainage of the building
- decommission all other systems that are not needed for operation of safe enclosure
- install new control equipment fitting with the new operation tasks
- release all buildings of the site (except the three buildings of the safe enclosed plant: reactor hall, reactor operating and auxiliary building) from the restrictions of the Atomic Energy Act.

One important issue of this phase was the conversion of the major part (≈ 80%) of the controlled area inside the safe enclosed plant into an "operational supervised" area with a dose level less than 2μSv, which can be entered for maintenance purposes without health physics monitoring. This area is the area outside the "envelope of safe enclosure" but inside the safe enclosed plant (*FIG. 4*), /4/.

This last but one part of step 3 took approximately eight months for execution and ended with the THTR 300 in safe enclosure (*FIG. 4*), comparable with the US-SAFSTOR or the IAEA passive SAFE STORAGE option at end of February 1997.

The last part of Step 3 was given on the way for licensing in May 1996. The applying documents like final safety analysis report, operating manual for thirty year operations of the safe enclosed plant and the like were checked by the experts. The license was granted on May 21, 1997.

Results

Work executed since 1990, even core unloading, resulted in yearly collective doses of personnel less than those in the years of operation (*FIG. 5*). The highest value during decommissioning occurred in 1995 due to the hands-on decommissioning of the small burn-up measuring reactor and the enclosing of the PCRV-penetrations.

Operating personnel could be reduced during step 1 + 2 only from 10 to a 8 men shift. Starting step 3 a further reduction to 5 men was allowed and at the end of second part of step 3 (safe enclosure established) only one control panel has to be checked by the site guard (24 hours a day). The personnel will then consist of the operator's plant manager plus one engineer and four additional standby service engineers on a call and contract basis. Necessary inspections will be done by specialized and certified companies on contract basis.

Then the yearly operating costs are reduced from more than 50 million DM per year during step 1, step 2 and first and second part of step 3, to 1.5 million DM per year. The overall costs of the decommissioning (1990 - 2009) sum up to 773.5 million DM and include costs of fuel transport and storage and also other waste handling and mandatory financing of final storage and financing of the project during 1990 - 2009 (*TAB. II*). The design of the THTR 300 that has for the secondary cycle a similar layout as fossil fueled power plants enabled the operator to sell many of the used components and spare parts to make financing of the decommissioning easier (*TAB. III*).

Starting into decommissioning of a nuclear power plant without chances of preplanning causes two to three years additional project execution time, equivalent to approximately 250 million's DM in the case THTR. This is why latest schedules for decommissioning up to green field include preplanning phases of up to four years.

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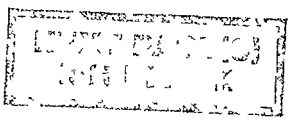

DECOMMISSIONING OF THE AVR REACTOR, CONCEPT FOR THE TOTAL DISMANTLING

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Abstract

After more than 21 years of operation, the 15 MWe AVR experimental nuclear power plant with pebble bed high temperature gas-cooled reactor was shut down in 1988. Safestore decommissioning began in 1994. In order to completely dismantle the plant, a concept for Continued dismantling was developed according to which the plant could be dismantled in a step-wise procedure. After each step, there is the possibility to transform the plant into a new state of safe enclosure.

The continued dismantling comprises three further steps following Safestore decommissioning:

1. Dismantling the reactor vessels with internals
2. Dismantling the containment and the auxiliary units
3. Gauging the buildings to radiation limit, release from the validity range of the AtG (Nuclear Act), and demolition of the buildings

For these steps, various technical prrocedures and concepts were developed, resulting in a reference concept in which the containment will essentially remain intact (in-situ concept). Over the top of the outer reactor vessel a disassembling area for remotely controlled tools will be erected that tightens on that vessel and can move down on the vessel according to the dismantling progress.

1 Introduction

The 15 MWe experimental nuclear power plant with helium cooled pebble-bed high temperature reactor of the Arbeitsgemeinschaft Versuchsreaktor Jülich (AVR) GmbH was one of the first nuclear power plants developed in the Federal Republic of Germany (Fig. 1). In 1987, the dismantling was decided and in 1988, the reactor was definitively shut-down after more than 21 years of operation /1/.

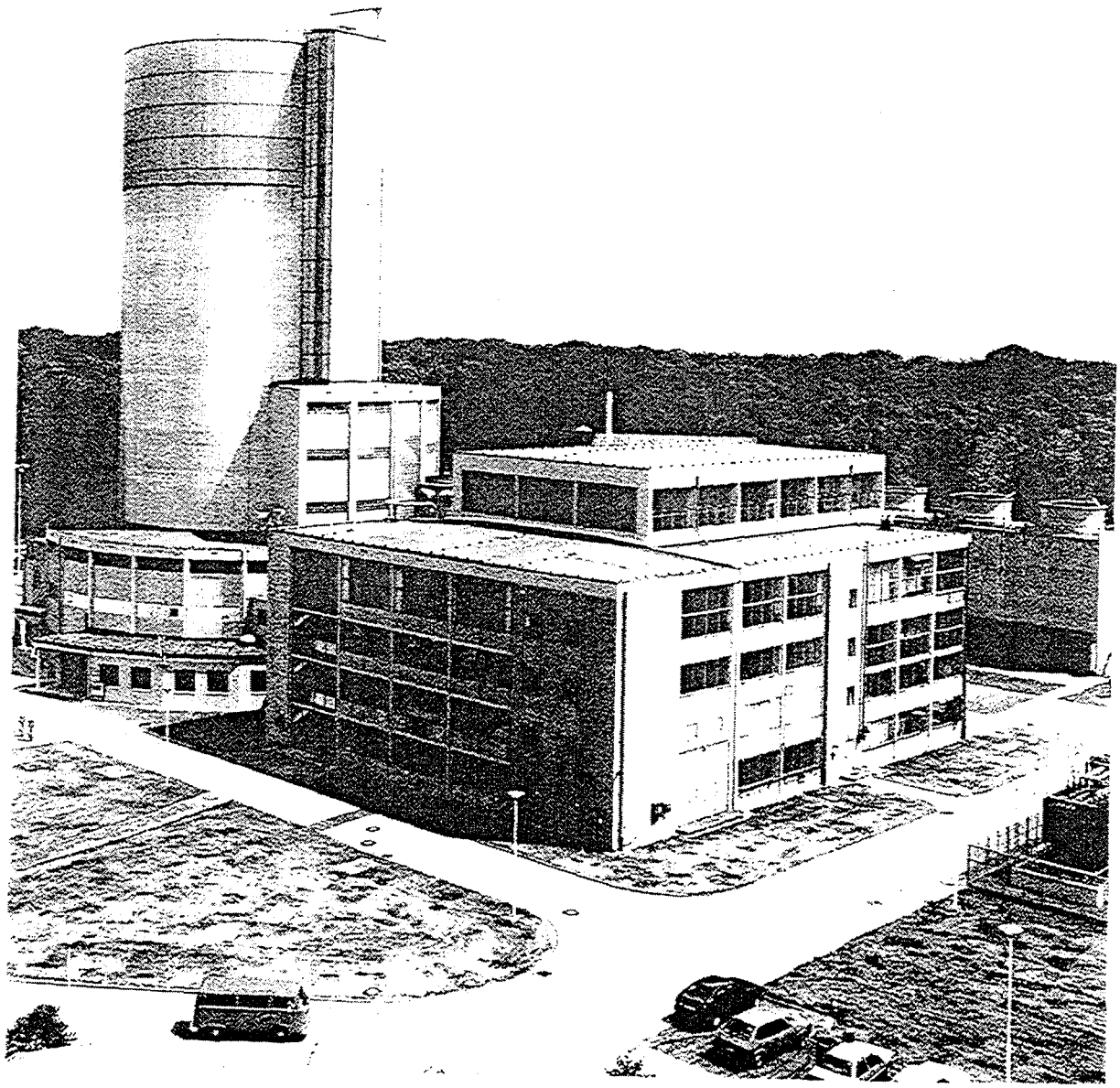

Fig. 1. AVR Experimental Nuclear Power Station

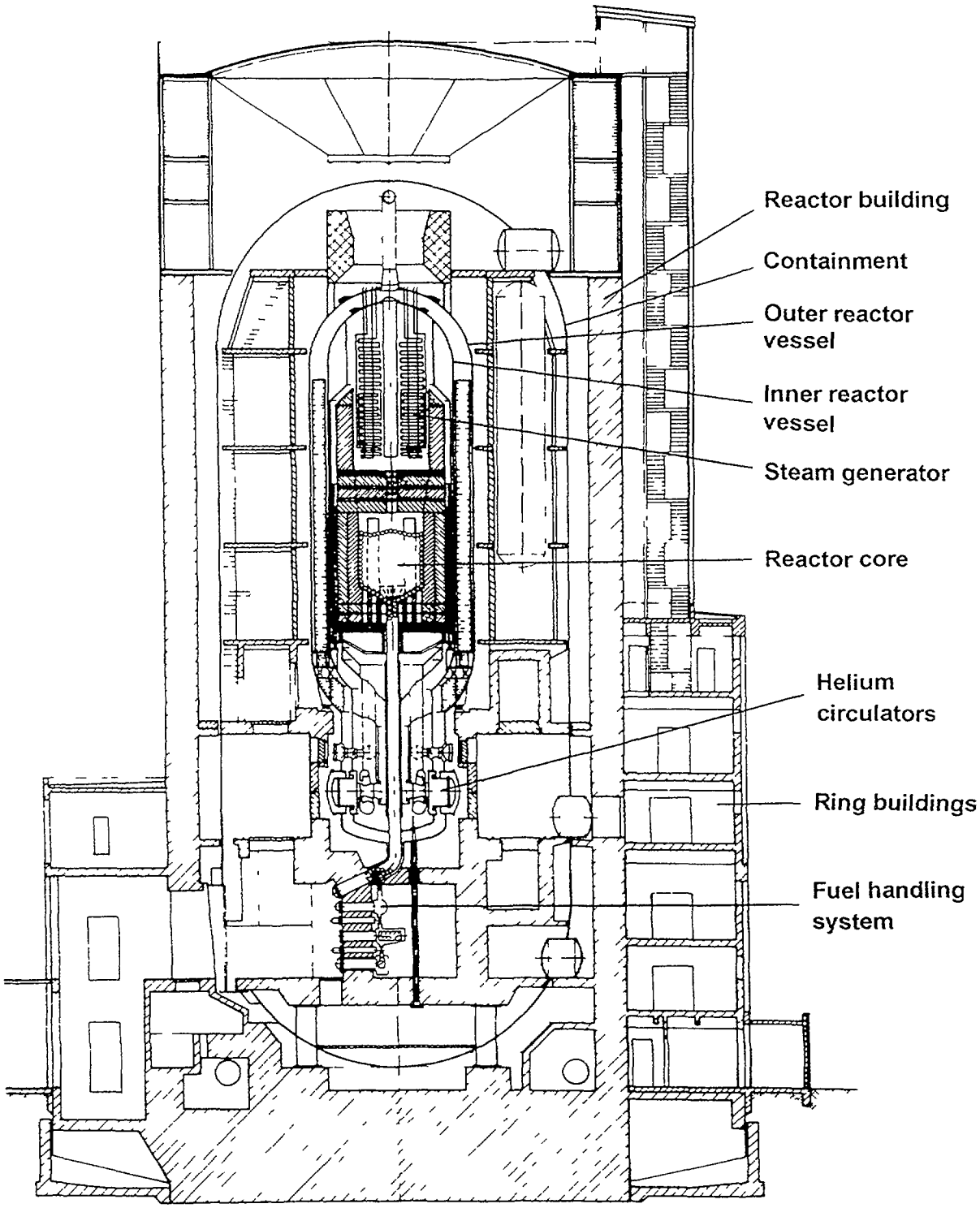

Fig. 2. AVR reactor building

1.1 Description of the AVR plant

The sectional view of the reactor building in Figure 2 shows the important components of the primary loop of the AVR plant

The important technical data of the AVR experimental plant are:

General data

Electrical output (gross)	15 MW
Electrical house load	1,65 MW
Heat output	46 MW
Number of fuel elements	92 000
Primary coolant	Helium
Mean Helium outlet temperature	950 °C

Fuel elements

Sphere diameter	6 cm
Fuel	U and Th
Max. fuel temperature	1 350 °C

Ceramic internals (Fig. 3)

Reflector	Graphite
Insulation, shielding	Carbon

Steam generator

Number	1
Type	Once through
Superheater outlet temperature	505 °C
Steam throughput	56 t/h

Reactor vessels

Inner reactor vessel	
Diameter	5.78 m
Wall thickness	40 mm
Height	24.91 m
Outer reactor vessel	
Diameter	7 6 m
Wall thickness	30 mm
Height	26 05 m

Containment

Diameter	16 m
Wall thickness	12 mm
Height	41.5 m
Design pressure	3 bar

1.2 Existing systems and facilities of the reactor operation

The systems and facilities of the reactor operation, as shown in Table 1, are available to sustain the basic operational functions during the dismantling works, i. e. control, supply and disposal functions. The operation of these systems takes place in accordance with the operating instructions of the existing decommissioning manual (Stillegungshandbuch SHB).

Table 1: Existing systems and facilities for the dismantling of the AVR plant

Vent systems 1 and 2
Vent systems 1 and 2 WW
Exhaust air control systems 1 and 2
Liquid waste disposal systems 1, 2, 3 and 4
Drain pump system of vessels 21 and 22
Compressed air supply system
Power supply system
Fire water supply system
Radiation monitoring laboratory
Clean rooms
Personnel locks +5 m, +11 m and +38 m

1.3 The radiological starting position of the plant

The inventory of radiological activity has been calculated for the year 1992 and is compiled in Table 2 listing the important nuclides.

Table 2: Activity of the principal nuclides of the reactor vessels including internals

Nuclide	Steam generator	Thermal shield	Biological shield I	Reactor vessels	Ceramic internals	Primary loop	Total
Co 60	3.1E+12	1.2E+14	2.0E+09	3.1E+14	2.8E+15		3.2E+15
Sr 90	3.6E+13					1.3E+13	4.9E+13
Cs 137	2.5E+13					6.6E+11	2.6E+13
C 14					1.2E+13		1.2E+13
Tritium					1.5E+15		1.5E+15

Besides the activation products and the activated corrosion products (e.g. Co 60, Fe 55, Ni 63), there exist dust-bound fission products (Sr 90, Cs 137, Cs 134 etc.) and partly nuclear fuel fines caused by abrasion.

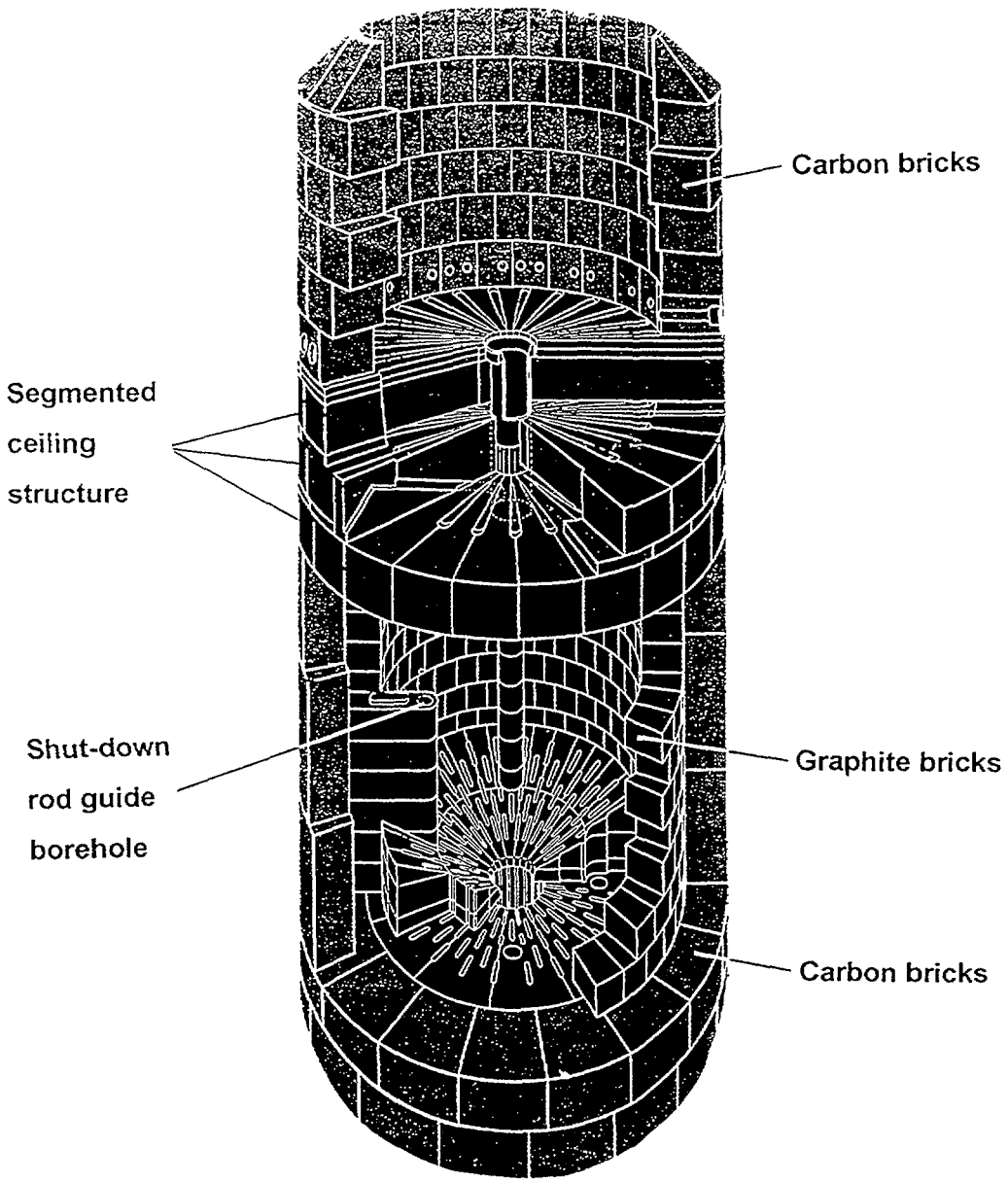

Fig. 3. Structure of AVR ceramic internals

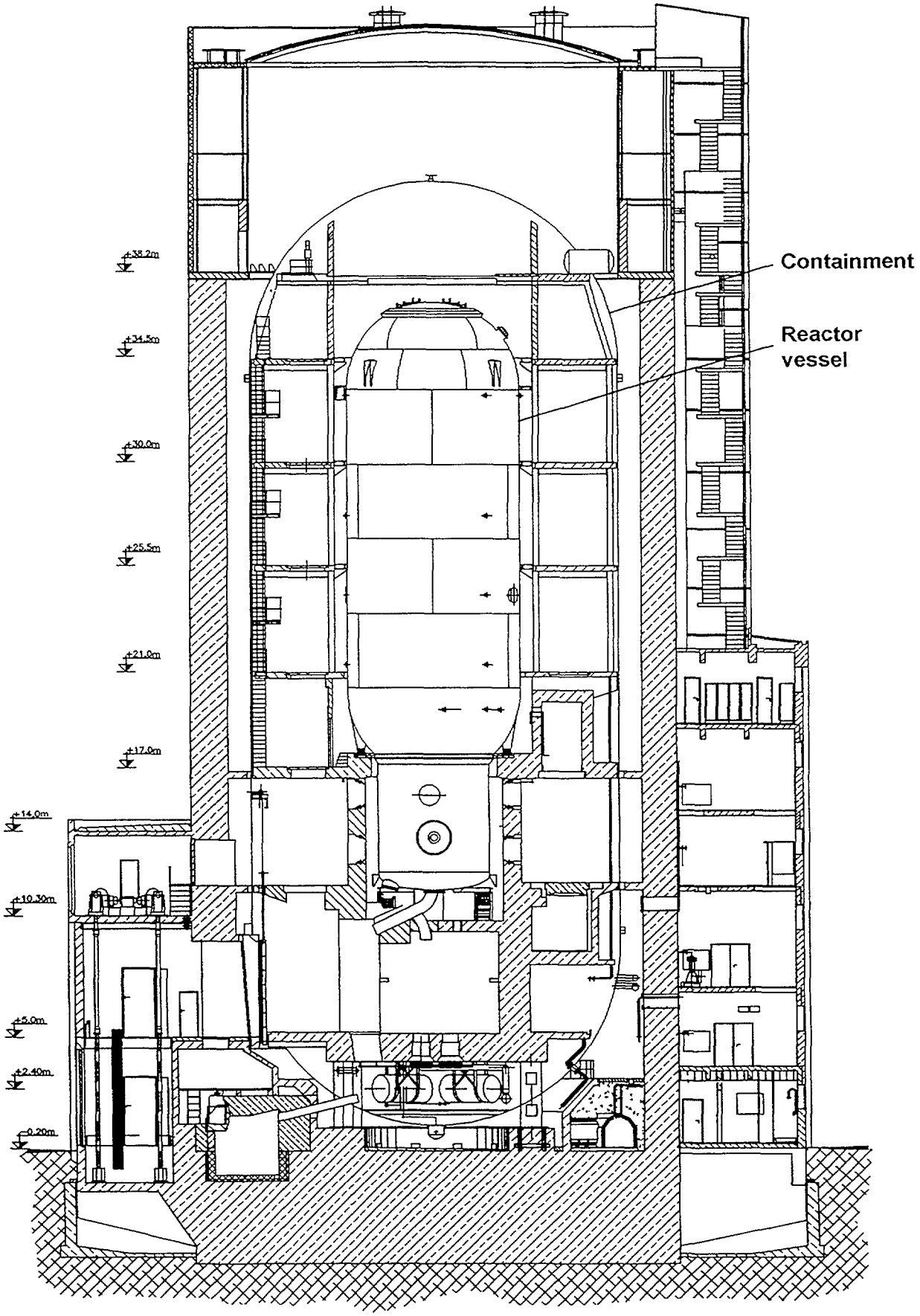

Fig. 4. Status before dismantling the reactor vessels

In order to determine the radiological starting position (nucleonic vectors, contamination atlas, dose rate atlas, inventory of activity) and to develop the data used in the application documents, a sampling measurement programme will be performed after unloading of the fuel elements is completed.

2 Status of the project

In December 1986, the application was filed to achieve and operate the AVR plant in the state of safe inclusion. The respective permit was granted in March 1994. This project was split into two safestore decommissioning phases.

- First Safestore decommissioning phase
  It encompasses essentially the defuelling of the reactor and the dismantling of the secondary loop outside the reactor building.

- Second Safestore decommissioning phase
  It encompasses the dismantling of components, the alteration of components, the installation and operation of new facilities for the state of safe inclusion as well as the operation in the state of safe inclusion.

In March 1994, the first safestore decommissioning phase has started with the unloading of the fuel elements, which is expected to be completed beginning of 1998. The secondary loop outside the reactor building has been dismantled to the most part.

The second safestore decommissioning phase is planned to start in 1998 after the fuel elements have been removed from the core and after this has been confirmed by visual inspection of the core interior. The originally envisaged dismantling measures shall be modified and enhanced on the basis of three supplementary permits in such a way that the number of capping heads in the AVR plant will be reduced to a minimum. The first supplementary permit was granted in March 1997, the second supplementary permit was applied for in August 1997 and the third supplement shall be applied for at the end of 1997. The status of the plant after termination of the dismantling measures in safestore decommissioning is shown in Fig. 4.

In parallel to safestore decommissioning the variant 'Complete Removal' is continued to be investigated. For this purpose, a strategy of step by step dismantling was developed so that, after each step, there is the possibility to transform the plant into a state of safe inclusion.

Subsequent to the two approved safestore decommissioning phases 1 and 2, the continued dismantling encompasses three further decommissioning project steps:

1. Dismantling of the reactor vessels with internals
2. Disassembling of the containment, step by step dismantling of the remaining facilities and decontamination
3. Gauging the buildings to radiation limit, release from the validity range of the AtG (German Nuclear Act), demolition of the buildings, and recultivating the site area.

3 Concept and pre-engineering

For the further decommissioning project steps 1 to 3, various technical procedures and concepts were developed. These studies were aimed at showing feasible ways to dismantle the AVR plant, emphazising the dismantling of the reactor vessels with internals, and assess the costs in order of magnitude. On this basis, the AVR GmbH has requested one consortium to provide the engineering of the disposal and two consortia to provide the pre-engineering of the following concept variants:

- Concept variant 1: Dismantling of the reactor with extension of the containment
- Concept variant 2: Dismantling of the reactor without structural alterations of the containment (In-Situ Concept)

AVR requested that two conditions are to be adhered to during the pre-engineering:

- The Two-Barrier-Concept is to be maintained; i. e. the barriers containment and outer reactor vessel are to be preserved or to be adequately replaced
- The dismantling work in the inner reactor vessel is to be performed under an inert atmosphere.

3.1 Concept variant 1

The concept variant 1 was investigated by ARGE BABCOCK/STEAG-DETEC and is characterized by the following criteria /2/:

- After the dismantling of the roof, the containment will be enlarged (Erweiterter Schutzbehälter, ESB)
- Installation of a disassembling area inside the ESB
- Dismantling of the steam generator and disassembling in the disassembling area
- Dismantling and disassembling of the reactor vessels in parallel to the disassembling of the steam generator
- Use of Master-Slave manipulators

3.2 Concept variant 2

The concept variant 2 was investigated by ARGE NOELL-LENTJES and shows the following criteria:

- Installation of a disassembling area without enlargement of the containment
- Step by step lifting of the steam generator and disassembling of the tube bundle by use of a power manipulator
- Installation of a large manipulator with tools to dismantle the reactor vessels
- The dismantling of the steam generator and the reactor vessel are executed sequentially.

3.3 Disposal

Simultaneous to the two pre-engineering concepts, ARGE WTI/SGR has developed a concept to dispose of radioactive materials which are generated during the dismantling of the reactor vessels /4/.

3.4 Further concepts

Supplementary to the a.m. pre-engineering work, DETEC has provided an In-Situ-Concept for the dismantling of the reactor vessels envisaging the use of Master-Slave-Manipulators. Furthermore, the DETEC concept provides an essential feature in terms of using a platform structure which closely connects to the outer reactor vessel and which descends with the progress in dismantling the reactor vessels.

Based on the total of the concept variants and pre-engineering work, AVR has developed own variants and further modified variants respectively.

4 Reference concept

All plans were assessed in view of compliance with the regulating act, the dismantling technology and the feasibility. The investigation was conducted by AVR and ISE GmbH. The results were compiled in a reference concept which provides the basis for the further project work.

4.1 Licensing procedure

For the dismantling of the AVR plant, a strategy for the licensing procedure was developed /5/aiming at:

- Activity enclosure and retention
- Protection of the environment (minimising the release of radioactive materials)
- Protection of the dismantling personnel (minimising the dose rate and protection of labour)

According to § 7 Sect. 3 AtG (German Nuclear Act) the following applications will be filed:

1. Interruption of safestore decommissioning and dismantling the reactor vessels with internals
2. Dismantling of the containment, step by step dismantling of the remaining facilities and decontamination

The demolition of the building and the recultivation of the site will be applied for according to conventional laws and acts.

4.2 Technical concept

It is the essential objective to dismantle the AVR plant within the constraints provided by the regulatory body, the budget and the a.m. protective aims. In order to ensure this general requirement, the reference concept is based on the following superimposed engineering requirements which are to be adhered to in any case

4.2.1 Engineering requirements

In-Situ-Concept

The dismantling of the reactor vessel is to be performed without any - from outside - visible alterations of the reactor building and under keeping the containment. If need be, the containment inside the reactor building may be altered. These alterations, however, may only be of insignificant nature.

Two-Barrier-Concept

The dismantling of the reactor vessels with internals is to be performed under the restraints of the Two-Barrier-Concept. I. e. during the dismantling of the reactor vessels, the two barriers

- containment and
- outer reactor vessel

are constantly to be maintained by appropriate measures in order to warrant a safe activity enclosure.

Emission of radioactive materials with exhaust air

The limiting values for the emission of radioactive materials with exhaust air in safestore decommissioning are also to be adhered to during the dismantling of the reactor

Waste treatment

Due to the limited space available, the waste treatment and conditioning of dismantled parts are to be performed in the Hot Shop. The dismantled parts are thus to be packed into appropriate containers on location and to be transported to the Hot Shop for further treatment.

Packing

The packing of the radioactive wastes has to abide to the receiving conditions of the possible future disposal site KONRAD. The packing of radioactive materials has to abide to the receiving conditions of the neighbouring Research Centre (REBEKA facility) and external waste disposal companies (e g Siempelkamp)

Regulations from the safestore decommissioning licence

The design and licensing for the dismantling of components has to consider the relevant clauses and regulations of the approval for safestore decommissioning /2/.

Utilisation of existing facilities

The existing facilities in the AVR plant are to be used whenever possible during the dismantling of the reactor vessels.

4.3. First dismantling step: Dismantling the reactor vessels with internals

4.3.1 Dismantling concept

The engineering performed so far has resulted in the following conceptional procedure. The dismantling of the reactor vessels shall be executed inside the unaltered containment in a disassembling area, which is created by the enlargement of the outer reactor vessel. To achieve this, a disassembling area is built over the top of the outer reactor vessel, supported by a rack and tightening on the cylindrical wall of the vessel in such a way that in the course of dismantling the disassembling area serves as the first and the containment as the second barrier. In accordance with the dismantling progress the disassembling area is step by step descending. The facilities of the remote control dismantling are descending in the same way so that the operating conditions (approach position and handling area) for the remote controlled facilities remain essentially the same.

The important dismantling steps are:

- Dismantling of the remaining but now unnecessary facilities in the containment and the platforms above 17.1 m.
- Installation of support rack inside the containment
- Installation of the disassembling area with the facilities of the remote controlled dismantling, connecting the disassembling area to the outer reactor vessel, commissioning of auxiliary systems (e.g. vent systems)
- Remote controlled dismantling of the steam generator
- Remote controlled dismantling of the reactor vessels with internals including the outer vessel
- Dismantling of the biological shield 1
- Dismantling of the facilities for the remote controlled dismantling and the disassembling area

4.3.2 Disassembling area

The disassembling area (Fig. 5) is mounted on a support rack. Its upper level serves as working area with disassembling and conveying systems for the remote controlled dismantling. For this dismantling, preferably master-slave manipulators should be used hanging from a crane. Additional lifting systems, support manipulators and conveyors serve to remove the dismantled components. The disassembling of the vessel internals preferably takes place on location or on an auxiliary disassembling area respectively by using remote controlled cutting tools as well.

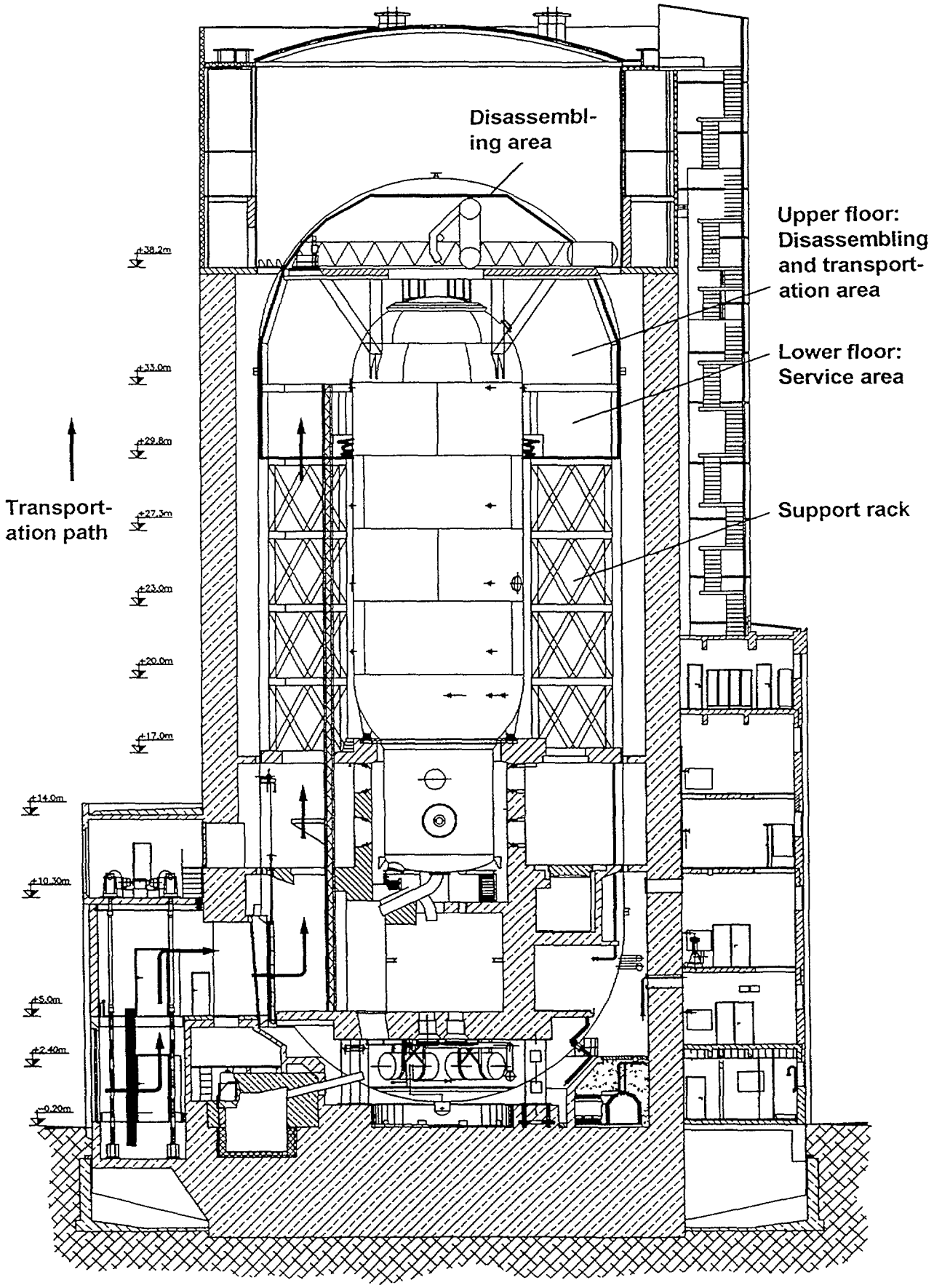

Fig. 5. Installation of disassembling area

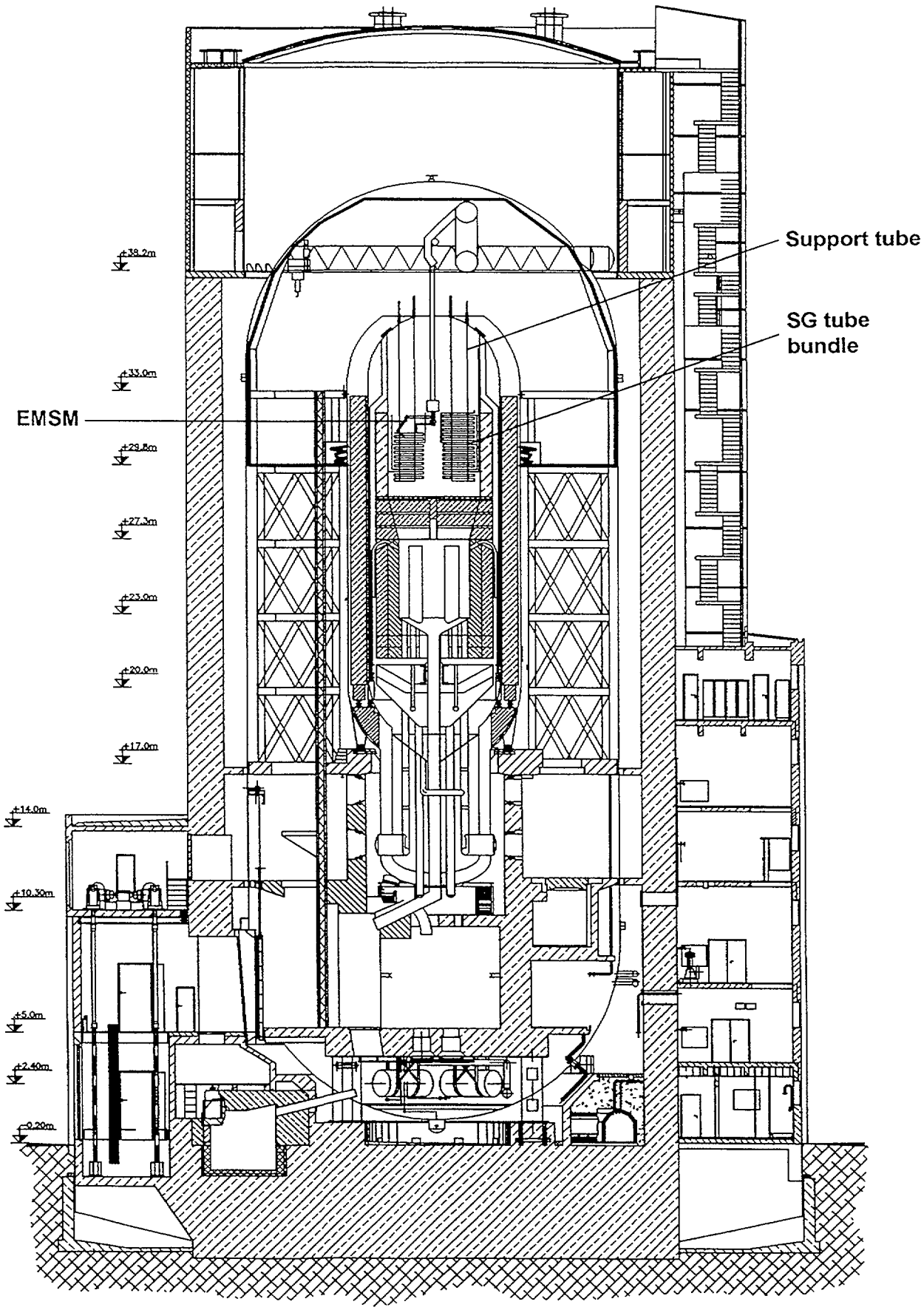

Fig. 6. Dismantling the steam generator (SG)

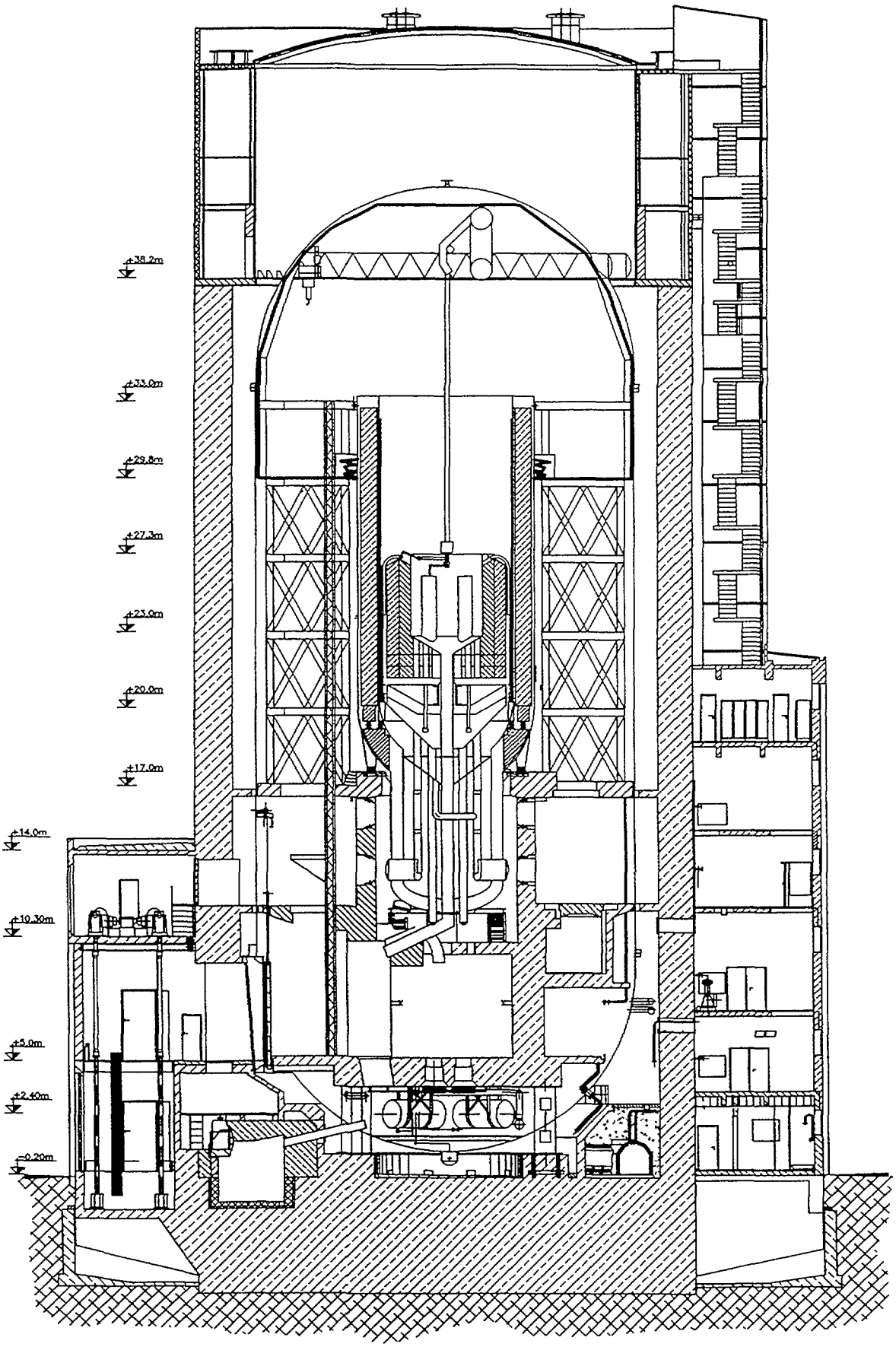

Fig. 7. Dismantling the core cavity

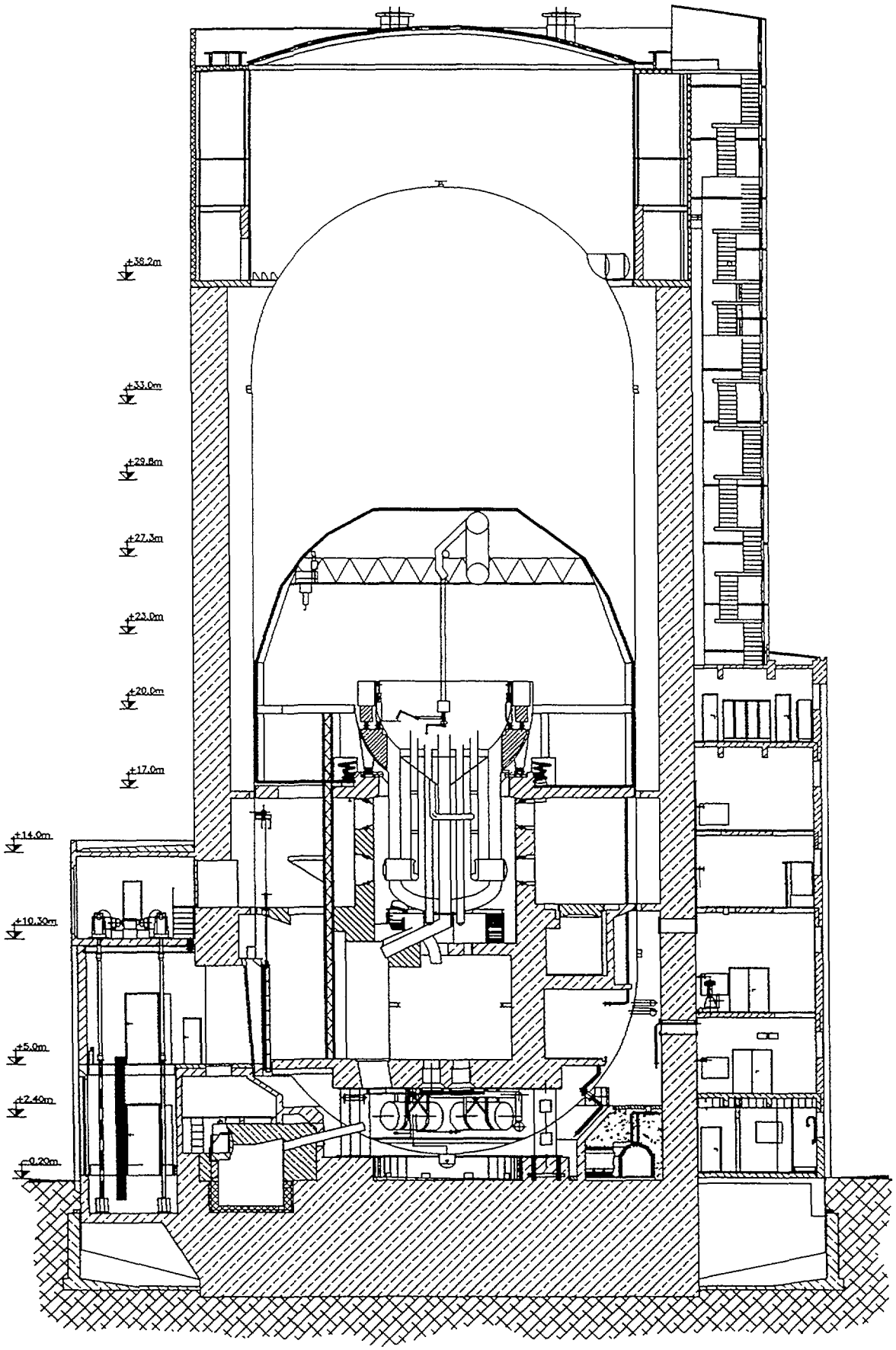

Fig. 8. Dismantling the lower part of the reactor vessels

For the dismantling of the steam generator and the reactor vessels, the following facilities shall be installed on the upper level:

- Electric powered Master-Slave-Manipulators (EMSM), carrying capacity approx. 100 kg
- Power manipulator, carrying capacity approx. 100 kg
- Auxiliary manipulator (power manipulator)
- Manipulator support system and bearings for parts up to approx. 1000 kg
- Lifting magnet for dismantled parts
- Disassembling facilities with guiding and mounting devices for milling and sawing tools used to disassemble the vessel walls and the thermal shield.
- Devices to dismantle the steam generator tubes and the fuel discharge tube
- Facility to transport the dismantled parts between manipulator and polar crane
- Auxiliary disassembling area

The lower level contains the service area with a lock area for removed materials, tooling machines, tools, supplementary means and persons during interventions and a measuring area for removed materials. Transportable shielding for interventions is stored at appropriate locations in the disassembling area.

4.3.3 Dismantling of the steam generator (Fig. 6)

The disassembling of the steam generator is to be performed in the mounted stage, whereas the load transfer continues to take place via the bracing tubes and the inner reactor vessel lid. For the disassembling, the central opening for the displacement tube of the steam generator will be enlarged only to the size necessary to bring in the EMSM. Cutting tools used shall be a hydraulic cutter for the vertically oriented steam generator tubes and a double disk saw for the horizontally oriented steam generator tubes. The cutting of the steam generator progresses from top to bottom collecting the cut-off tube pieces in a transportable bin.

The disassembly of the tightly coiled steam generator tubes, equipped with spacers and bracing tube fixations will have to be demonstrated on a model during the 'design and licensing phase'.

4.3.4 Dismantling of the reactor vessels with internals (Fig. 7, 8)

The dismantling concept provides to dismantle the reactor vessels with internals successively from top to bottom. In order to dismantle the ceramic internals the upper vessel domes will only be opened as much as necessary to bring in the manipulators.

The four time lowering of the disassembling area by approx. 3 m each time permits the disassembling of the cylindrical reactor vessel walls including the biological shield 1 moving from the outside to the inside simultaneously to the disassembling and dismounting work inside the inner reactor vessel. This permits a flexible way to proceed.

Due to the step by step moving of the disassembling area, the remote control personnel may always work more or less under the same space situation. This is an advantage for the personnel who shall execute their work with growing routine and work thus more time and cost effective.

- Partial disassembly of the upper calotte of the inner reactor vessel to prepare for the dismantling of the ceramic internals
- Dismantling of carbon bricks, layers 37 to 32
- Dismantling of the central tamper, layers 31 to 27 ('cake piece' segments)
- Dismantling of carbon and graphite segments, layers 31 to 25
- Dismantling of carbon and graphite bricks, layers 30 to 1/1
- Disassembling of remaining inner gas baffle plates and the reactor barrel down to the bottom plate
- Dismantling of the remaining outer and inner reactor vessel calottes
- Lowering of the disassembling area
- Loosen and remove by suction the fill of the biological shield 1 (in several steps)
- Dismantling and disassembling of the thermal shield
- Disassembling and dismantling of the coolant gas ducts, the upper part of the fuel discharge tube, the support and bottom plates
- Dismantling of the remaining internals and decontamination

After the remote controlled dismantling of the complete inner reactor vessel, the inner side of the remaining outer reactor vessel and the disassembling area will be cleaned and the manual dismantling of the outer reactor vessel along with the remains of the fuel discharge tube will take place. Prior to manual dismantling the inert gas and the below atmospheric pressure stage in the disassembling area will be released.

4.4 Second dismantling step: Dismantling of the containment, the remaining internals, and decontamination

4.4.1 Dismantling of the containment (Fig. 9)

After termination of the measures to dismantle the reactor vessel, the containment will be completely cleared of objects and fully cleaned inside. In the lower area, at the elevation of the manipulator (+ 5.0 m), a provisional platform will be installed in the containment. On this platform a ring-shaped rack will be erected on the inner wall of the containment. In the upper area, it will be adjusted to the shape of the containment so that each location of its inner side is accessible. From this rack decontamination of the containment shell will be performed, so that it can be free released and the steel be conventionally recycled.

For the dismantling of the containment shell, appropriate lifting and conveying devices are to be installed. In order to disassemble the containment a thermal cutting method is advantageous for cost reasons.

The transport of the dismantled containment parts takes place from the rack to a basket, which by crane is lowered to the steel floor. The dismantled parts will then be packed into a container as steel scrap.

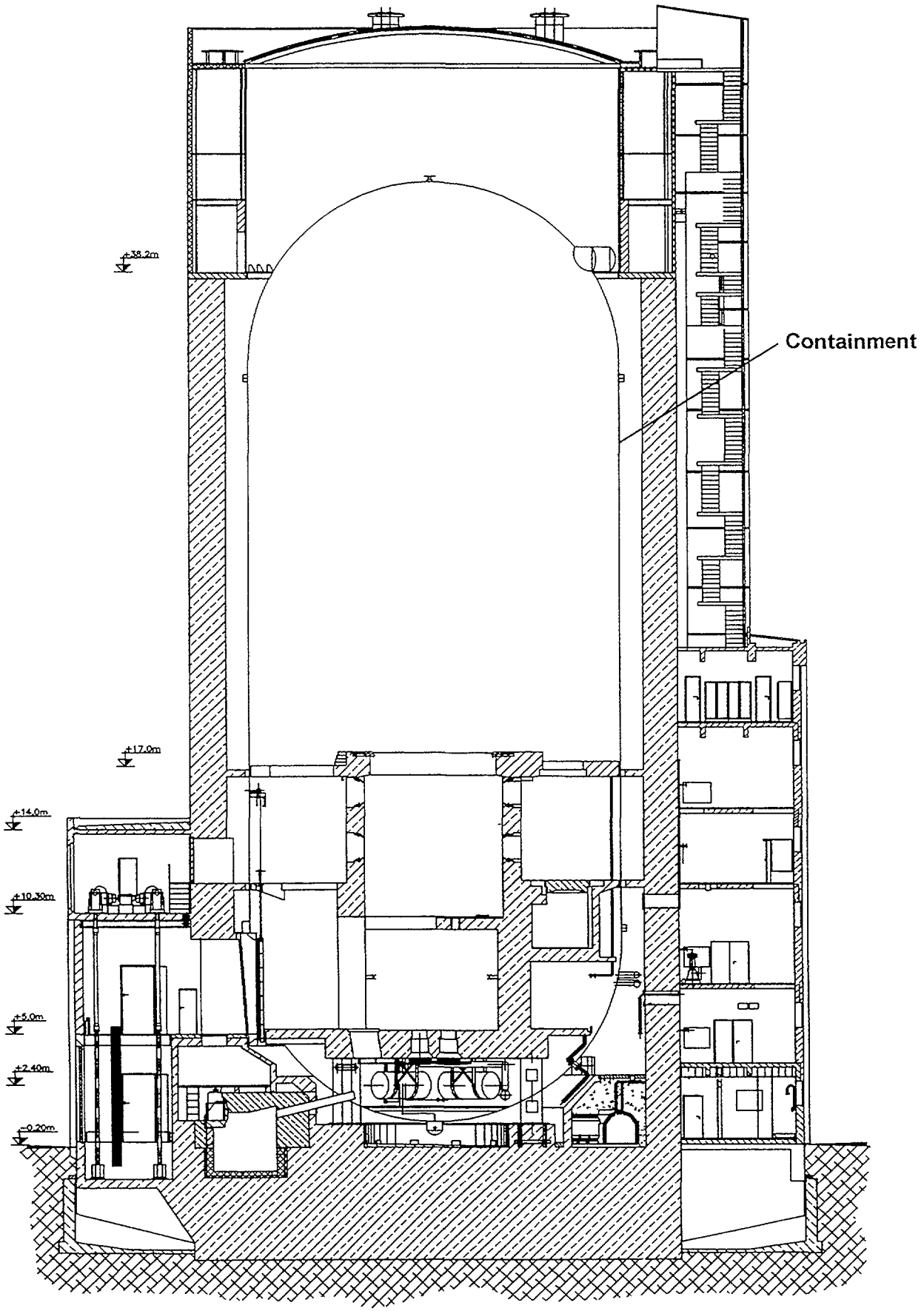

Fig. 9. Status before containment dismantling

During the dismantling of the containment, samples will be taken from the inner side of the biological shield 2 in order to determine cleaning methods for its free release.

4.4.2 Dismantling of remains in the buildings of the former controlled section

All remaining components and facilities which are contaminated or likely to be contaminated (e.g. dowel plates, braces, end plates) and can not be decontaminated together with the building structure will be dismantled and disposed of.

4.4.3 Dismantling of auxiliary units

Auxiliary and ancillary units like

- Liquid waste collecting and discharge facilities
- Vent systems 1 and 2
- Electrical and control systems
- Power supply systems in the controlled areas etc.

will be reduced to the barely necessary extent or be replaced by provisional or smaller new facilities.

The phase of auxiliary and ancillary units dismantling may take place simultaneously to the decontamination and radiation measuring of the building structures.

4.4.4 Decontamination of buildings

Principally, the decontamination of buildings progresses from the higher contaminated areas to the lower contaminated ones, from top to bottom, and from the farest location to the retreat areas of the room access openings. The decontamination normally is followed by the decision measurement and the controlling measurement of the authorities. Decontaminated and measured rooms will be sealed in order to avoid recontamination.

Access and transport routes will always be kept clear of contamination to an extend that a slow contamination build up can be excluded for sure.

4.5 Third dismantling step: Radiation measurements of the buildings, release of the AVR plant from the Nuclear Act (Atomgesetz), demolition of the building structures, and re-cultivating the plant site

After disposal of the decontamination devices (abrasion tools for concrete, washing facilities) and contaminated equipment (mobile filtering stations, vacuum cleaner, decont water collecting facilities etc.), the plant is free of artificial radioactive nuclides generated during the former plant operation. This will be assessed by measurement, documented and confirmed by the authorities. Based on this documentation, the AVR plant will be released from the obligations under atomic law.

After the plant is released from the AtG the demolition of the buildings and the disposal of recyclable materials takes place according to conventional procedures under the conventional regulatory body, e.g. Kreislaufwirtschaftsgesetz, BImSchG.

The dismantling of the AVR will end with the compilation and archiving of the safety documentation and the return of the site to green field.

4.6 Disposal concept

In order to dispose of the waste, different packing variants have been investigated in view of the final repositories ERAM and KONRAD. The calculation of the optimised packing volume resulted in a storage volume of approx. 4 700 m^3 at the KONRAD facility. Part of the drums and bins still need type rating for this purpose. Difficulties arise in particular from the restrictive KONRAD reception conditions for tritium (H-3) and carbon (C-14).

4.7 Time schedule

The time schedule is shown in Fig. 10 from a today's view point. Under the assumptions that the application for the dismantling of the reactor vessels can still take place in 1998 and the permit be granted until the end of 2000, the state of 'Green field' for the AVR plant may be accomplished in 2011.

4.8 Costs

The cost estimates which have been performed during the two pre-engineering phases resulted in approx. 250 Mio DM for the dismantling of the AVR plant. The disposal effort is supposed to be in the same order of magnitude. Thus, for the dismantling of the AVR plant a total cost of approx. 500 Mio DM is expected (without safestore decommissioning).

5 Further proceedings

In July 1997, selected bidders have been invited to tender for the service package of dismantling the AVR plant containing the continued dismantling steps 1 to 3 and partial services of the safestore decommissioning phase 2. Objective of the invitation to tender is to find a qualified general contractor for the engineering and the realisation of the total dismantling project. The closing date for bid acceptance is set for October 1997. AVR is confident that the award to perform the engineering services will take place by beginning of 1998.

During the first project phase, the final concept will be fixed by the general contractor in a modified reference concept. This will be detailed in the subsequent design phase and the documents for the licence application will be generated. The application for the dismantling of the AVR plant along with the safety analysis report (Sicherheitsbericht), the final hazards summary report (Sicherheitsbetrachtung), and the environmental compatibility

AVR DECOMMISSIONING

Safestore decommissioning incl. supplements and Continued dismantling

AVR GmbH

HSz hse-frb3 tg Status 01/09/97 Rev 1

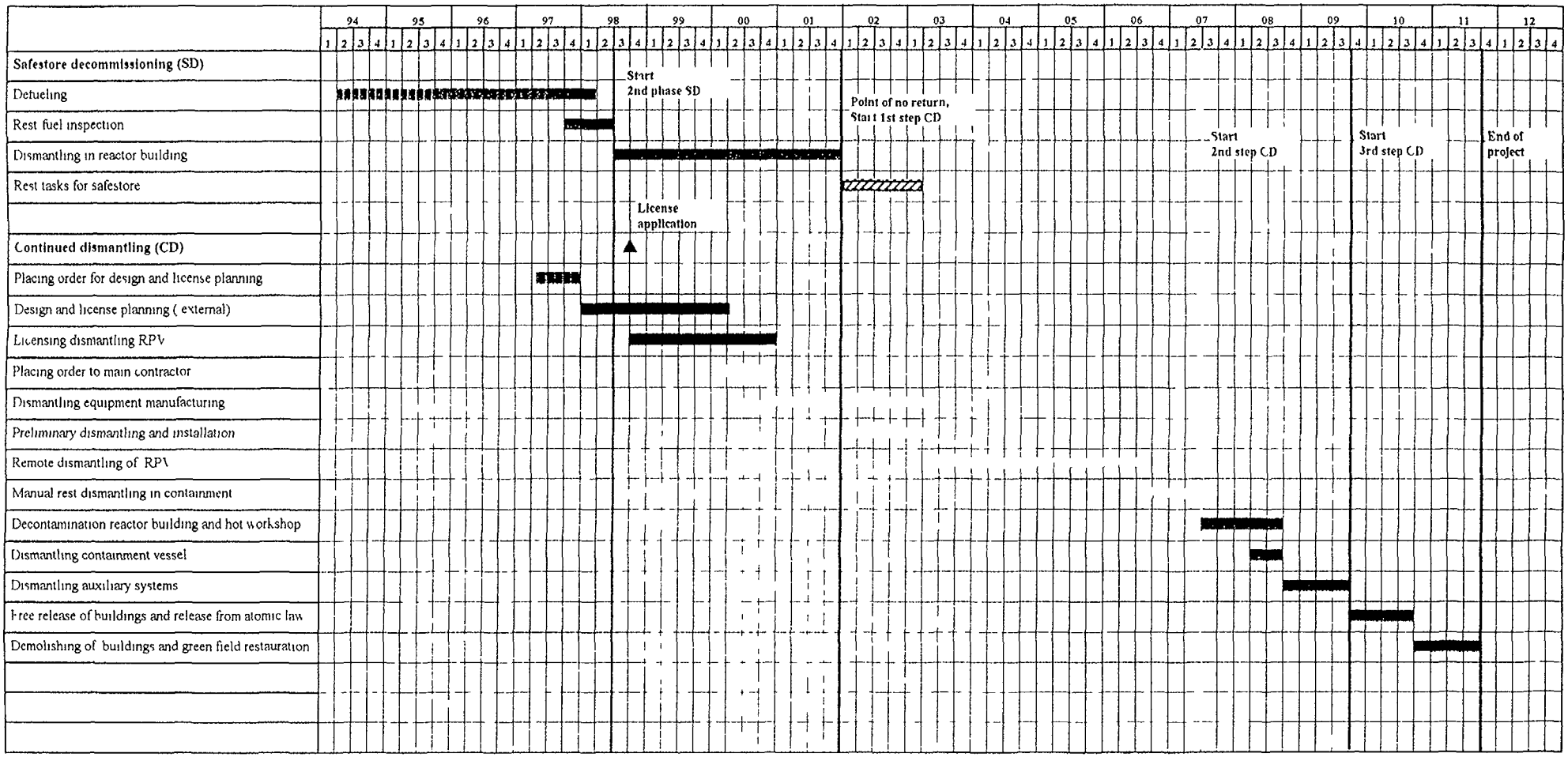

Fig. 10. Decommissioning schedule

report (Bericht zur Umweltverträglichkeit) shall be filed in autumn 1998. The supplementary documents (Erläuterungsberichte) shall be filed in 1999 in order to expect the granting of the licence no later than by the end of 2000.

The second project phase contains essentially the accompaniment of the licensing process and the detailing of the design including preparation of the specifications for the facilities needed to dismantle the reactor vessels.

The third project phase starts with the granting of the approval to dismantle the reactor vessels of the AVR plant and encompasses the production engineering, the preliminary inspection, the production and the procurement of facilities, the testing of the remote controlled devices and the execution of the dismantling measures. The dismantling of the reactor vessels shall in today's view be completed in 2009.

Up to 2001 the already approved dismantling tasks as well as the supplements of the 2nd safestore decommissioning phase, still subject to approval, will be executed. Based on today's time situation, the prerequisites for the dismantling of the reactor vessels with internals will be in line by approx. 2001.

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AVR DECOMMISSIONING, ACHIEVEMENTS AND FUTURE PROGRAMME

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Abstract

Safestore decommissioning of the AVR 15 MWe experimental nuclear power plant with pebble bed high-temperature gas-cooled reactor (HTGR) began in March 1994 with defuelling (phase 1). Beginning of Aug. 1997, defuelling was to 81 % completed. Other achievements: The dismantling in the turbine hall and outside the buildings is nearly terminated, the cooling towers are demolished, and the helium bottle-battery storage and helium compressors were removed from the ring buildings in Dec. 1996. The latter was the first dismantling inside the reactor building and belonged to projects that had been advanced from the 2nd into the 1st phase of Safestore decommissioning because of the delay in defuelling. Furthermore, the licence for a first supplement to Safestore decommissioning was granted in March 1997.

Inside the containment, the removal of shielding material and of insulation material from the secondary circuit components is either already or will shortly be terminated. This will give access for cutting and sealing the 120 steam generator pipe penetrations above the outer reactor vessel.

The scope of Safestore decommissioning, as licensed in March 1994, will be extended by three supplements, comprising mainly the dismantling of (1) the fuel handling system, coolant circulators, and interspace convection pipe, (2) the coolant purification system, and condensation coolers, and (3) the shutdown rod system. The goal is to clear the containment from all auxiliary systems and to seal the outer reactor vessel until the end of 2001.

The final goal of Continued dismantling is the restoration of the green field until 2011. The term indicates the direct transition from the present Safestore decommissioning and a stepwise procedure that can be interrupted after each step and be transferred into a Safestore mode. The decision for Continued dismantling is expected in 1998; a contract for the design and licence planning will be awarded soon.

1. Introduction

The 15 MWe AVR experimental nuclear power plant is one of Germany's oldest nuclear installations; construction began in 1959. Its reactor belongs to the first generation of high temperature gas-cooled reactors (HTGRs) and was among these with its 21 years of operation certainly the most successful. For design and achievements, former publications like /1/ should be referred to. In this report on decommissioning only some key items shall shortly be recalled:

- Core of about 100,000 ball shaped fuel elements (pebble bed) cycled during reactor operation,
- Highest ever reached coolant temperature of 950 °C,
- Indispensable mass test facility for HTGR fuel development,
- First-ever-done experimental simulation of a loss-of-coolant accident /2/.

An overview of the reactor design and the site structure is given in Figures 1 and 2.

The plant was finally shut down end of 1988. A licence for Safestore decommissioning, first applied for in 1986, was granted in March 1994. Since a pebble bed reactor is never defuelled during reactor operation, defuelling is the major concern in Safestore decommissioning, and the whole task was separated in a first phase with defuelling and dismantling outside of the reactor building and a second phase with dismantling and preparations for the later dormancy period inside the reactor building.

The paper looks at the achievements obtained in now three and a half years of decommissioning activities, the future programme of Safestore decommissioning , and gives an outlook on the possible continuation of decommissioning towards the green field. The latter is presented in more detail in an own presentation within this TCM.

2. Overall Progress, Achievements, Highlights

Although defuelling is still not terminated, and the second phase of Safestore decommissioning with major dismantling in the containment could not yet start, the project has not been lacking considerable progress, summarised in the following.

- Since all obstacles and limitations concerning the transfer of the low-enriched part of the AVR fuel to the neighbouring Jülich Research Center could be finally lifted in July 1996 a major progress in defuelling has been achieved. Beginning of August 1997, only 19 % of the fuel was still left in the reactor.

- The dismantling in the turbine hall is nearly and that outside of the buildings is fully terminated.

- The cooling towers are demolished.

- The helium bottle-battery storage and helium compressors were removed from the ring buildings in Dec. 1996. This was the first dismantling inside the reactor building and belonged to the projects that AVR was allowed to advance from the second into the first (defuelling) phase of Safestore decommissioning because of the delays in defuelling.

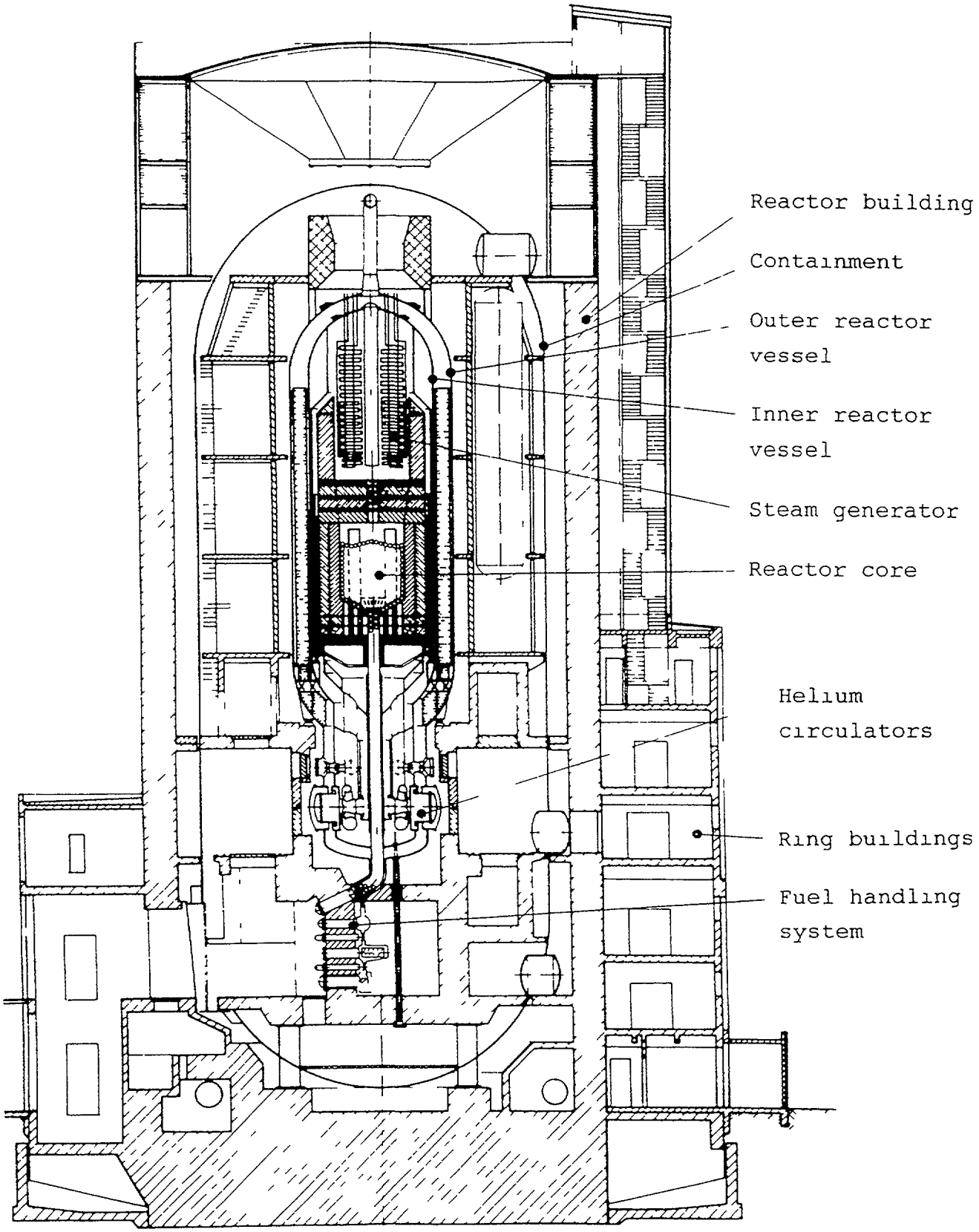

Fig. 1 AVR reactor building

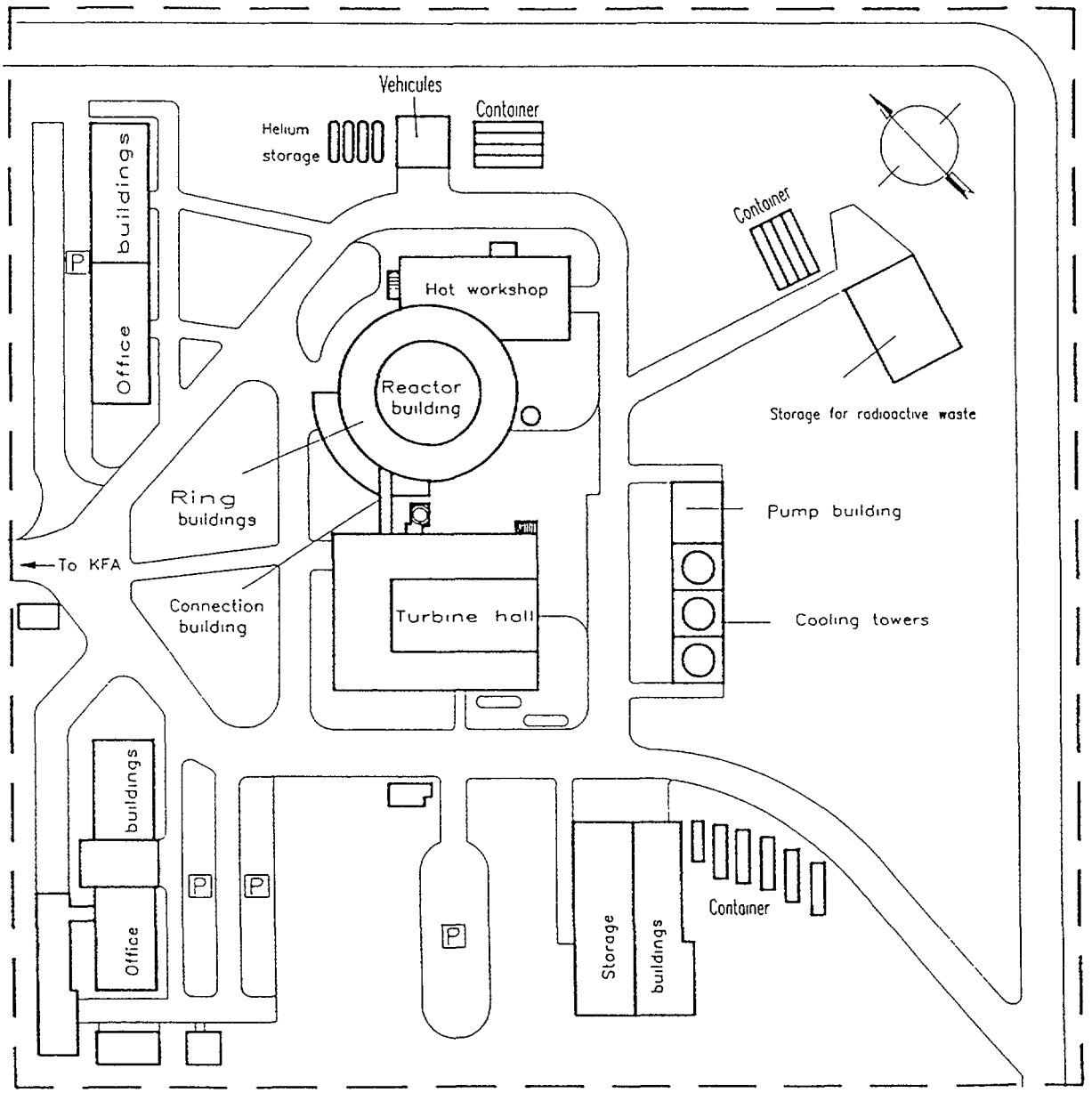

Fig. 2 Plan of AVR site

- Large amounts of shielding material (bricks and lead) with which the top floor of the containment was covered were removed only recently. The removal of all insulation from the secondary circuit components in the containment has begun and is well in progress. These tasks, too, belong to the before mentioned advanced projects.

- A first supplement to Safestore decommissioning was licensed in March 1997 (details below).

- The strategy for a possible continued dismantling (towards green field) has been finally agreed upon between AVR and the Research Center.

3. Progress in Technical Areas

3.1 The defuelling story thus far

In defuelling the about 112,000 pebbles of the AVR two major concerns had to be taken into consideration:

(i) Since the reference fuel of the AVR had always been HEU fuel but the AVR core consisted in the end to about 50 % of LEU fuel, the Research Center needed an extra licence to transfer that fuel through the water basin of their Hot Cells facility on the way to its storage in CASTOR casks.

(ii) Displacement processes of pebbles in the core during defuelling would lead to an increase in the fuel concentration in the core center so that an intermediate increase of reactivity during defuelling could not be ruled out.

To deal with the reactivity concern it was fixed that the course of subcriticality during defuelling had to be closely followed by regular critical measurements so that, if any necessity arose, countermeasures could be taken in time.

The above mentioned licence not obtained in time defuelling started in April 1994 with HEU fuel only. The LEU fuel was charged back to the reactor. The distinction was made by gammaspectrometrical measurement of each pebble, measuring U 232 which is practically only present in the Thorium-containing HEU fuel /3/. The selection quality had to be tested and verified in Nov. 1994, leading to an interruption in defuelling of about 2 months. Within June, July, and August 1995 the share of HEU elements in the discharged pebbles fell from about 50 % to a mere 17 %, and the selective HEU defuelling was stopped after about 35,000 pebbles had been discharged.

After the licence for Hot Cells was obtained, defuelling - and this time HEU and LEU - restarted in March 1996, though at first still under some limitations. Limit values for heavy metals, and Pu 239 was critical here, had to be observed in the Waste Cells of the waste management facilities of the Research Center where the CASTOR casks are loaded. To use up the permitted quantities in a best manner, selective defuelling was continued, with about 10 % of the pebbles (mostly HEU) returned to the reactor. In July 1996, then, the Research Center obtained permission to regard considerably higher limit values in the Waste Cells, and unselective defuelling could finally begin.

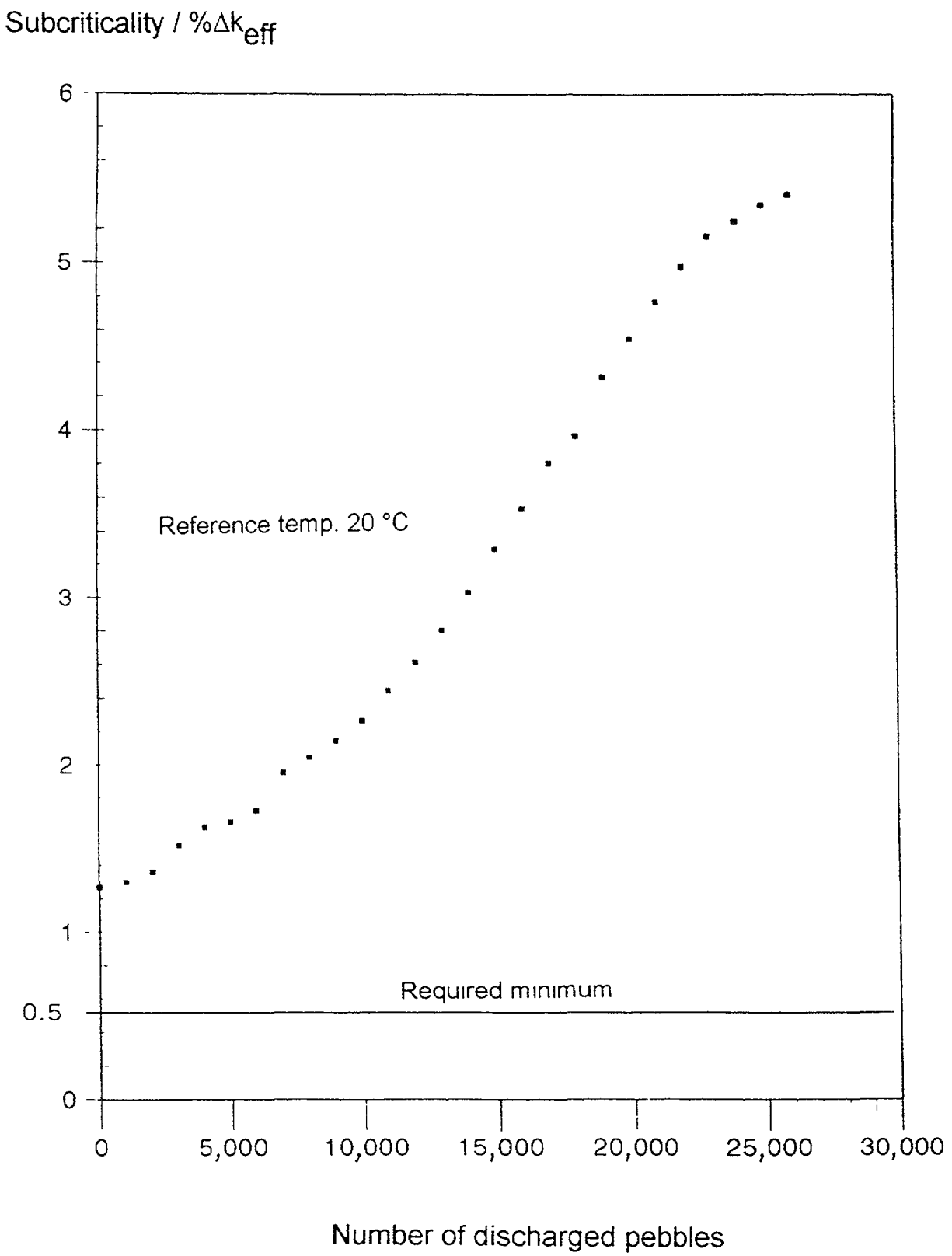

Fig. 3 Measurement of subcriticality during AVR defuelling

Thus, defuelling was no straight path, and neither it was in regard of concern (ii), though that concern, other then (i), fortunately did not cause any delay Fig 3 shows the results of the regular subcriticality measurements (details in /4/) up to a defuelling of 26,000 pebbles. Because of the massive return of pebbles to the reactor, the mentioned displacement processes could not take place, and so the steep rise in subcriticality came as no surprise The return of the LEU elements, however, also led to a certain increase in the fuel density of the inner core, to which the decrease in subcriticality gain in the upper part of the curve in Fig 3 can be attributed Beyond the defuelling of 26,000 critical measurements were no longer possible Regular measurement trials by full withdrawal of the rod bank were, nevertheless, continued, proving that the reactor remained higher subcritical in amount than the full rod bank worth

When the complete, unselective defuelling started, the displacement processes were initiated in full scale for the remaining 63,000 pebbles still in the reactor It was argued then, that the measurement trials were of little use to detect a rapid decrease in subcriticality if a rapid decrease in the rod worth was assumed A different method to determine subcriticality in a very approximate way similar to the critical experiment in loading fresh reactors was turned down since the remaining zero level of the neutron flux instrumentation was regarded as insufficient The way out was a theoretical study taking into account all experimental data gathered so far during defuelling, and also data of the AVR's own critical experiment The result was that regardless of the procedure in further defuelling the reactor would always remain strongly subcritical As a consequence, all measurement requirements were abolished, the neutron flux instrumentation was permanently taken out of service, and the shut-down rods have remained inserted

When defuelling went at its best, 36 fuel cans, or 1800 pebbles, per week could be transported to the Research Center Necessary repairs in parts on the fuel handling system led to several interruptions in defuelling between Feb and May 1997, summing up to about 8 weeks A major setback then, with only 24,000 pebbles left in the reactor, was the break down of the ring channel machinery in June 1997. The ring channel is used as a buffer for empty and filled fuel cans and allows to disconnect can filling and can transportation. This advantage gone, defuelling continued at a much smaller pace, can filling and transportation can by can The repair of the ring channel machinery will take longer, 31 filled fuel cans have to be retrieved first by remote techniques Ways to accelerate defuelling without the ring channel are in examination

3.2 Cutting experience

The cutting of pipes and vessels of the secondary and cooling water circuits in the turbine hall and outside of the buildings was exclusively carried out by sawing This proved to be effective and not too time consuming Since there is no controlled ventilation system in the turbine hall, it is obligatory not to release any radioactive substances at all By sawing inside ventilated plastic sheet housings this could easily be achieved For smaller pipes a reciprocating saw with automatic feed was used that is directly fixed on the pipe Some larger pipes and vessels were cut with a self-advancing machine that uses a circular-saw-blade-like milling tool The machine is held and guided by a chain that surrounds the work piece Most cuts, however, were carried out with hand-held motor-driven backsaws With these, it was possible, e g , to cut through the condensate collecting tank with a diameter of 1 8 m and a wall thickness of 7 mm circumferentially in 5 5 hours

AVR DECOMMISSIONING

Safestore decommissioning incl. supplements and Continued dismantling

AVR GmbH

Status 28/02/97 Rev 0

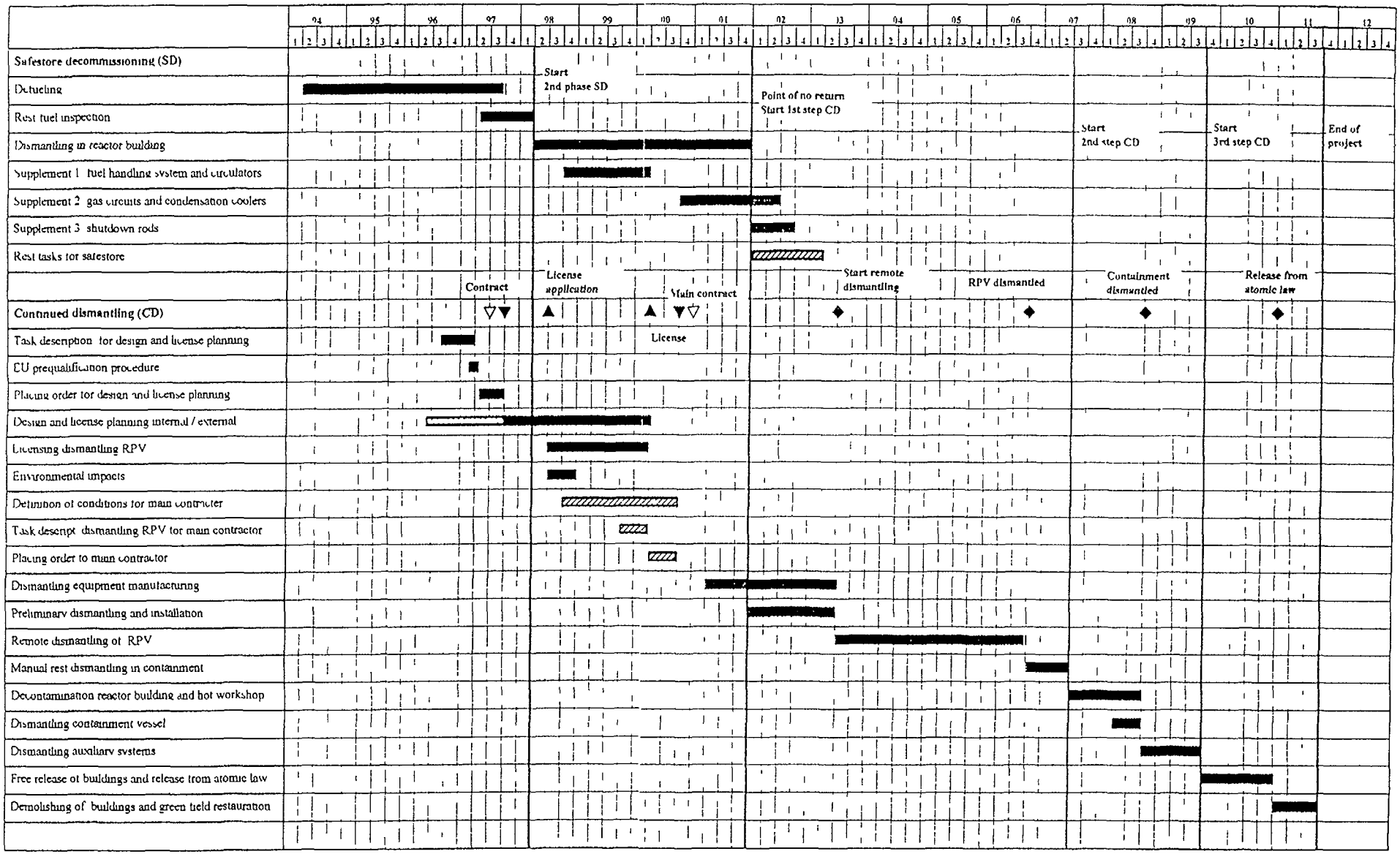

Fig. 4 Project schedule

4. Future Programme

4.1 Near future

The near future - rest of 1997, beginning of 1998 -will see the continuation of Safestore decommissioning. Most important is of course defuelling which despite the above mentioned break down of the ring channel for buffering fuel cans ought to be terminated within 1997. In direct continuation the core cavern will be inspected for rest fuel through a radial borehole drilled at core mid-height and later also all components of the fuel handling system. The boring and inspection equipment has been enabled to take out graphite and carbon brick samples.

Parallel to defuelling the decommissioning activities concentrate on the following tasks:

- Finishing the removal of insulation from the secondary circuit components in the containment.

- Removal of a shielding wall in the top part of the containment.

- Having created the necessary access by the above mentioned removal of insulation and shielding material, cutting of the 120 steam generator tube penetrations through both reactor vessels above the outer vessel and closing the tube ends by welding. The task - referred to as 'hedgehog' - is due to be completed still within 1997.

The remaining task in the turbine hall, the dismantling of turbine and condenser, has got low priority and will be addressed later when suitable from both budget and personnel availability.

4.2 Overall programme Safestore decommissioning

Scope and schedule of Safestore decommissioning, as finally defined in 1996, is represented in Fig. 4. It is characterised by three supplements to the original licence of March 1994. The scope of dismantling inside the containment covered by the original licence is fairly limited, as has repeatedly been reported in the past /5/. The planned supplements, however, will bring about the clearance of the containment from all auxiliary systems and will eventually leave the sealed outer reactor vessel as the only radioactivity containing area. The inclusion of these supplements into Safestore decommissioning, i. e., not deferring the related tasks, is of vital importance for the whole AVR decommissioning project in terms of profiting from own personnel's knowledge and experience and thus reducing dose uptake and costs. It is appropriate, therefore, to regard these supplements in somewhat closer detail in the following.

The supplements concern systems that contained helium and are therefore, no matter if filled with primary coolant or barrier gas helium, extensively contaminated with Sr 90 bound to very volatile graphite dust which is, as has often been reported before, the major point of concern in AVR decommissioning.

4.2.1 First supplement

The first supplement to Safestore decommissioning, licensed in March 1997, comprises the dismantling of the

- fuel handling system,
- coolant circulators, and
- interspace convection pipe.

The **fuel handling system** can be divided into 4 main sections:

- First-components' wall: It contains vertical-in-line all the components below the fuel discharge pipe, from the reducer-wheel that closes that pipe down to the scrap collection bottle.

- Pebble distribution wall: It contains the switch wheel as well as pebble valves in all incoming and off-going pebble pipes. The 5 feeding pipes to the core will be cut and sealed below their penetration through the outer reactor vessel.

- Fresh fuel feed system: It comprises components in- and outside of the containment.

- Spent fuel discharge system: It also comprises components in - and outside of the containment, including the installations in the ring channel for buffering empty and filled fuel cans.

The **coolant circulators**, integrated into the bottom part of the 2 reactor vessels, will be dismantled including their oil lubrication system. The circulators will be removed using existing equipment for removal, shielding and transportation.

The **interspace convection pipe** with the in-built water-operated interspace cooler enabled the natural convection and cooling of the helium in the interspace between the reactor vessels at power operation. The pipe extends from the top of the outer reactor vessel, all the way down the containment wall, spreading up at the bottom, and entering the 4 shut-down rod casings at their lower ends.

4.2.2 Second supplement

The second supplement to Safestore decommissioning is in an advanced planning stadium and the licensing process is about to start. It will address the dismantling of the helium purification system and the condensation coolers.

The term **'helium purification system'** is to understand here in a wider sense since the central part of the system, the adsorption-material-containing vessels (partly deep temperature adsorption), are already covered by the original licence under the task item: removal of operational material. The wider sense comprises here all of the helium systems inside which, in a way, the purification system is central, comprising all pipework, valves (including their control systems), various compressors, a vacuum pump, vessels and filter units. The multitude of valves and many of the smaller components are grouped in a number of steel racks on nearly all floors in the containment. The goal is to remove these racks as

whole or in larger units and further dismantle them at a more appropriate location in order to reduce dose uptake. Since most of the racks contain not only components of one single system it will be a precondition for their dismantling that all the concerned systems are shut-down, switched free (as is usual), and that for those systems that are still needed, like the pressurised air supply, new off-rack installations will have been provided to the necessary extend.

The 3 **condensation coolers** are huge, thick-walled (16 mm) tanks extending over several floors, designed to mitigate accident situations with a rupture of the steam generator at reactor operation by taking up the high-pressure mixture of helium and steam and condensing steam on water then present at the bottom part of the vessels. Since that sort of accident never occurred, the coolers are, in comparison to other vessels, only little contaminated. Besides the coolers, the dismantling comprises their safety valves as well as those of the two reactor vessels.

4.2.3 Third supplement

The third supplement to Safestore decommissioning considers only the removal of the **shut-down rods** including their driving units. There is a whole maintenance procedure with special devices and tools for exchanging rods making use of the Hot Cells' facilities of the Research Center. However, since that procedure is lengthy and has never been fully practised, a new-to-develop, more direct, dismantling-oriented way is supposed to be favourable. Studies are being done.

Anyway, the dismantling of the shut-down rods, as well as that of the condensation coolers, will be taken over into Continued dismantling (next chapter) if that option is chosen.

4.3 Continued dismantling

Strategy

The final goal of 'Continued dismantling' is the restoration of the green field. The term has been chosen to indicate the direct transition from the present Safestore decommissioning and a procedure that can be interrupted after each of a number of dismantling steps and be terminated with and transferred into a new Safestore mode should any obstacle arise, in both financial or organisational terms, to the continuation of the project. According to schedule (Fig. 4), that transition to Continued dismantling, if envisaged, would take place in the beginning of 2002. That date would be the start of the first step of Continued dismantling, addressing remotely controlled the two reactor vessels and their internals. The second step, beginning in 2007, would comprise the decontamination of the reactor building (incl. Hot workshop) and the dismantling of the containment vessel. The third step, starting in late 2009, would concern the free release of the buildings, the release of the site from atomic law, and the demolition of the buildings including the field restoration of the site until late 2011.

Reference Concept

For the remote reactor dismantling the 'in situ' concept has been chosen in which the containment vessel remains intact and the steam generator has to be cut in situ. The concept was given preference to the competing 'extension concept' in which the containment vessel would be opened at the top and largely extended to a veritable dismantling house offering enough space to pull out the steam generator as a whole unit. The key advantage of the in situ concept is that it relies on an existing and accepted boundary which should facilitate and shorten the licensing procedure to a large extend.

The in situ concept has been further detailed and fixed to a reference concept considering on top of the reactor vessels a ventilation tight dismantling area that tightens at its bottom on the cylindrical part of the outer reactor vessel. Furthermore, it has to be designed in a way that it can be moved down on the cylindrical part of the outer vessel according to the dismantling progress. Thus, a certain pre-determination of the overall method to be employed has already been made, and any detailed solution has to be based on this concept.

Design and Licence Planning

A first important step towards Continued dismantling was the decision in 1996 to award a contract for the design and licence planning. The budget for this task has been secured (about 8 mill. DEM). An EU prequalification for bidding was evaluated in July 1997 and the actual bidding process is in an advanced stage. The task list for the contractor has been divided into the following items:

(1) Planning of Continued dismantling
(2) Accompaniment of the licensing process
(3) Execution of Continued dismantling
(4) Execution of a distinguished task from Safestore decommissioning (dismantling the fuel handling system)
(5) Maintenance of the remaining plant

Items (4) and (5) are bound to a transition of AVR personnel to the contractor.

5. Costs

At present, the costs situation of the AVR decommissioning project for both Safestore decommissioning and Continued dismantling can be summarised as follows:

Spent for waiting period 1989 till 1993	c. 120 mill. DEM
Spent for decommissioning from the beginning in March 1994 till the end of 1996	c. 105 mill. DEM
Estimated total for Safestore decommissioning (incl. supplements)	c. 270 mill. DEM
Estimated costs for Continued dismantling	c. 230 mill. DEM
Public funding for AVR project, as of 31 Dec. 1996	c. 670 mill. DEM

6. Conclusions

Although it may not seem so at a first glance, AVR decommissioning is well in progress. Sure, a number of delays caused the defuelling to be a lengthy procedure but it is well advanced and the end is in sight. The task of AVR decommissioning is like the task of writing a text. The full text has not been written yet, but the subject is clear, the text has been structured, and a list of contents been drafted. The way how to proceed is clear. The costs have been estimated without the costs for waste disposal which are still too uncertain to predict. Budget and time requirement for the complete dismantling of the plant will not be exceptional in comparison to other decommissioning projects. The 3 supplements will put Safestore decommissioning on a sounder basis and make use of the own personnel's experience and knowledge in a best manner. And finally, the advantages of a transition from Safestore decommissioning to Continued dismantling are well recognised, and it is quite likely that this path will be followed.

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PERMANENT CESSATION OF TOKAI POWER PLANT'S OPERATION

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Abstract

Tokai power plant (166MWe, Magnox type: GCR) is the first commercial reactor in Japan and has been kept operating stably since its commissioning in July 1966. During this period it has produced electricity of approximately 27.7 billion KWh (as of March 1997) and its stable operation has contributed greatly to the stable supply of electricity in Japan.

Furthermore, technologies in various fields have been developed, demonstrated and accumulated through the construction and operation of Tokai power plant. It also contributes to training for many nuclear engineers, and constructions and operations of nuclear power stations by other Japanese power companies. As a pioneer, it has been achieved to develop and popularize Japanese nuclear power generation.

On the other hand, Tokai power plant has small capacity in its electric power output, even though the size of the reactor and heat exchangers are rather bigger than those of LWR due to the characteristics of GCR. Therefore, the generation cost is higher than the LWR. Since there is no plant whose reactor type is the same as that of Tokai power plant , the costs for maintenance and fuel cycle are relatively higher than that of LWR.

Finally we concluded that the longer we operate it, the less we can take advantage of it economically.

As a result of the evaluation for the future operation of Tokai power plant including the current status for supply of electricity by the Japanese utilities and study of decommissioning by Japanese government, we decided to have a plan of stopping its commercial operation of Tokai power plant in the end of March, 1998, when we completely consume its fuel that we possess.

From now on, we set about performing necessary studies and researches on the field of plant characterization, remote-cutting, waste disposal for carrying out the decommissioning of Tokai power plant safely and economically. We are going to prepare the decommissioning planning for Tokai power plant in a few years based on the guideline recommended by the government and on the situation of establishment of relevant criteria under the consultation and coordination with the government, local communities, utilities and relevant organisations.

1. Introduction

Tokai Power Plant (166MWe, Magnox type GCR) of the Japan Atomic Power Company (JAPC), the first commercial nuclear power plant in Japan, started its operation in 1966. Since then, construction of commercial nuclear power plants has been promoted in Japan. The number of operating nuclear power plants as of August 1997 is 52 and total capacity is 45.1 GWe. Electricity generation by nuclear power amounts to 34.0 % of all commercial electricity generation in fiscal year 1996. Thus, nuclear power is now an essential power source in Japan.

In Japan, where natural resources for energy is not rich, nuclear power is planned to continue to develop in future as well. It is planned to develop up to 70 GWe by the year 2010.

JAPC decided in June 1996 to stop operation of Tokai Power Plant at the end of March 1998, because of economical reasons such as increase of the operating cost and the outlook of the maintenance cost. Tokai Power Plant is the only gas-cooled reactor in Japan. Light water reactors are dominant in nuclear power generation in Japan and have been operated in 13 years on average; see Figure 1. The decommissioning of the light water reactors are not expected for the time being, but the decision on Tokai Power Plant implies that the time will come for the decommissioning of LWRs at any rate.

JAPC has not yet decided the definite programme of decommissioning of Tokai Power Plant and will decide it during the defuelling period. JAPC is now promoting the investigations on decommissioning scenario and technology in accordance with Japan's basic policy on decommissioning.

In the earliest case, defuelling of Tokai Power Plant will be over in the year 2001 and the notification of the decommissioning plan is to be submitted to the Government at this time. The Government and JAPC are developing necessary regulatory arrangements and decommissioning engineering procedures, and they are expected to be completed prior to the notification.

2. Decommissioning Policy in Japan

2.1. Basic Policy for Decommissioning in Japan

The basic policy for decommissioning[1], decided by the Japanese Government (Atomic Energy Commission), is that nuclear reactors should be

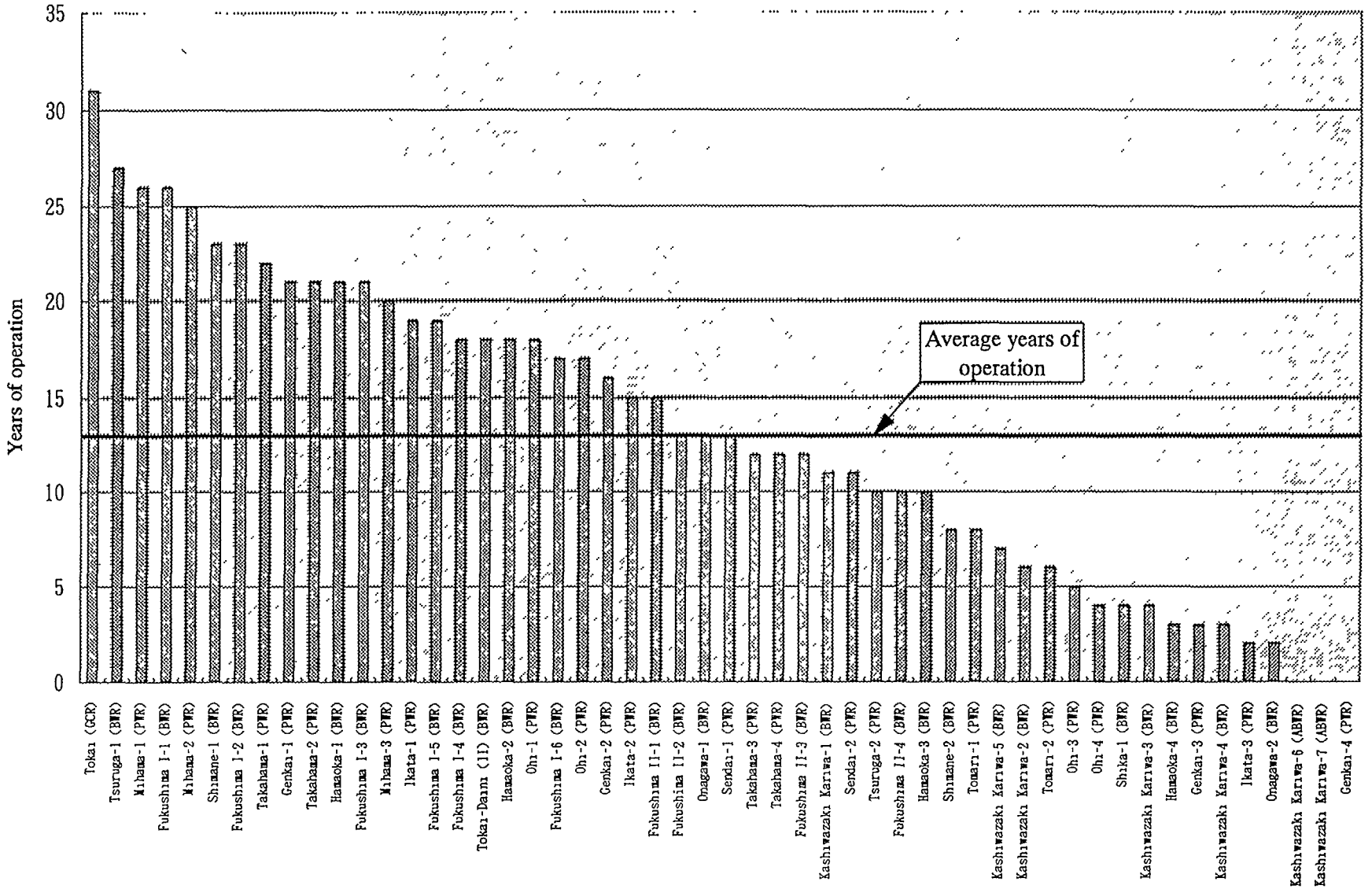

Fig.1. Years of operation of nuclear power plants in Japan [5]

dismantled and removed, in principle, in early phase after termination of operation and the site should be re-used for nuclear power generation, as available areas for sites of nuclear power plants are limited in Japan.

With regard to procedure for decommissioning, the Subcommittee on Nuclear Energy of the Advisory Committee for Energy had recommended (in 1985) the standard process for decommissioning[2], considering the actual Japanese situation. The standard process consists of three stages, that is; system decontamination, safe storage (of about five to ten years) and dismantling.

2.2. Regulatory Issues

In order to perform decommissioning of commercial nuclear power plants, there remains several regulatory issues to be established by the Government and these issues are now being deliberated[3].

(a) Rules for safety affirmation for decommissioning

The regulatory procedures in Japan will be controlled through the notification of decommissioning plans and revision of safety technical specifications. However, it is necessary to define formats and contents of these documents, as there is no experience of actual application for the decommissioning of commercial nuclear power plants. It is also necessary to prepare basis and criteria in order to review an application. The Government is planning to implement a standard format of decommissioning plans and revised technical specifications by the year 2001, when the defuelling of Tokai Power Plant will be finished at the earliest and the decommissioning plan will be submitted.

(b) Issues relating to disposal of decommissioning wastes

In the existing regulatory system, arrangements are not prepared sufficiently for all the low level wastes. Especially, regarding high β/γ low level radioactive wastes, such as reactor internals, (some of them are classified as ILW in some countries), the concept of disposal should be clarified and its regulatory system should be prepared. The Government (Atomic Energy Commission) is now deliberating this issue.

Furthermore, regarding the wastes which are not required to be treated as radioactive wastes, the clearance level is not decided in Japan. The Government (Nuclear Safety Committee) is deliberating to establish a clear criteria, referring to criterion in other countries and the international organisations.

These activities are planned to be completed by the year 2001, when defuelling at Tokai Power Plant will be finished at the earliest (and the notification of its decommissioning plan will be submitted to the Government from JAPC.)

Decommissioning wastes contain a lot of wastes which can be utilised as resources. In order to reduce the influence on the environment, the effective re-use of these wastes is also very important. Various organisations are now performing the research work on this issue.

(c) Reserve fund system for the decommissioning expense

In Japan, the Reserve Fund System for the Decommissioning of Nuclear Power Plants was established in 1988 in order to secure necessary funds for decommissioning, and the fund for each plant has been reserved every year by each utility for the expenses necessary in future.

In this system, the funds for dismantling are included, but disposal costs for decommissioning wastes are not. The method for estimation of disposal cost should be established in accordance with the above establishment of regulatory system for disposal, and the disposal costs are to be reflected to the reserve fund system in future.

3. Tokai Power Plant

3.1. Role of Tokai Power Plant

Tokai Power Plant is the first commercial nuclear power plant in Japan and has kept steady operation since its commissioning in July 1966. It has produced electricity of approximately 27.8×10^9 KWh (as of March 1997) and its steady operation has contributed greatly to the stable supply of electricity in the Tokyo metropolitan area.

TABLE 1. GENERAL DATA OF TOKAI POWER PLANT

Power Output	Thermal	587 MWt
	Electrical	166 MWe
Moderator		Graphite
Coolant		Carbon dioxide gas
Fuel		Natural uranium (MAGNOX can)
Reactor Core	Height	About 7m
	Effective diameter	About 12 m
	Fuel (Initial loading)	186.6ton, 16,348 elements
	Moderator	920ton, 17,912 bricks
Pressure Vessel		
	Material	Carbon steel
	Inner Diameter	About 18 m
	Thickness	About 90 mm
	Weight	About 700 ton
Biological Shield		
	Inner Diameter	About 22 m
	Thickness of Concrete	Max. about 3 m (at upper primary shield floor)
		About 2m (at primary shield wall)
	Total Weight	About 13,000 ton
SRU		
	Number	4
	Total Height	About 25 m
	Inner Diameter	About 6 m
	Weight	750 ton per one SRU
Charge Machine		
	Number	2
	Total Height	About 17 m
	Outside Diameter	2.5 m, at the main pressure vessel
	Weight	600 ton per one charge machine

TABLE 2. DECOMMISSIONING WASTES AND SITUATION OF REGULATORY ARRANGEMENTS

<table>
<tr><th colspan="3" rowspan="2">Classification</th><th rowspan="2">Method of Disposal</th><th colspan="2">Ratio of Wastes from Dismantling</th><th colspan="2">Establishment of Regulatory Systems</th></tr>
<tr><th>1,100MW LWR (500-550 ktons)</th><th>166MW GCR (160 ktons)</th><th>Upper Limit of Activities in Waste</th><th>Technical Standard</th></tr>
<tr><td>I</td><td rowspan="4">LLW</td><td>Hi β γ</td><td>to be established</td><td>less than 0.1%</td><td>2%</td><td>to be established</td><td>to be established</td></tr>
<tr><td rowspan="3">II</td><td>Uniform Solidified</td><td rowspan="3">Concrete Pit (Shallow Land Burial)</td><td rowspan="3">less than 1 %</td><td rowspan="3">8 %</td><td>established</td><td>established</td></tr>
<tr><td>Miscellaneous Solid</td><td rowspan="2">established</td><td>established</td></tr>
<tr><td>Large Scale Metal</td><td>measures for closing holes</td></tr>
<tr><td rowspan="2">III</td><td rowspan="2">ELLW</td><td>Concrete</td><td rowspan="2">Trench (ditto)</td><td rowspan="2">1 - 2 %</td><td rowspan="2">5 %</td><td rowspan="2">to be established</td><td rowspan="2">to be established</td></tr>
<tr><td>Metal</td></tr>
<tr><td rowspan="2">IV</td><td colspan="4">Clearance Level</td><td colspan="2">to be established</td></tr>
<tr><td colspan="2">Non RAW</td><td>as Industrial Waste</td><td>98 - 99 %</td><td>85 %</td><td colspan="2">N/A</td></tr>
</table>

(Remark)

1. In case of LWR.

   ASSUMPTION: ① Power operation period of 40 years. ② Safe storage period of 5 years. ③ Clearance Level based on IAEA TecDoc.[4] ④ Consideration of system loops decontamination.

2. In case of GCR.

   ASSUMPTION: ① Power operation period of 30 years. ② Safe storage period of 5 years. ③ Clearance level based on IAEA TecDoc.[4]

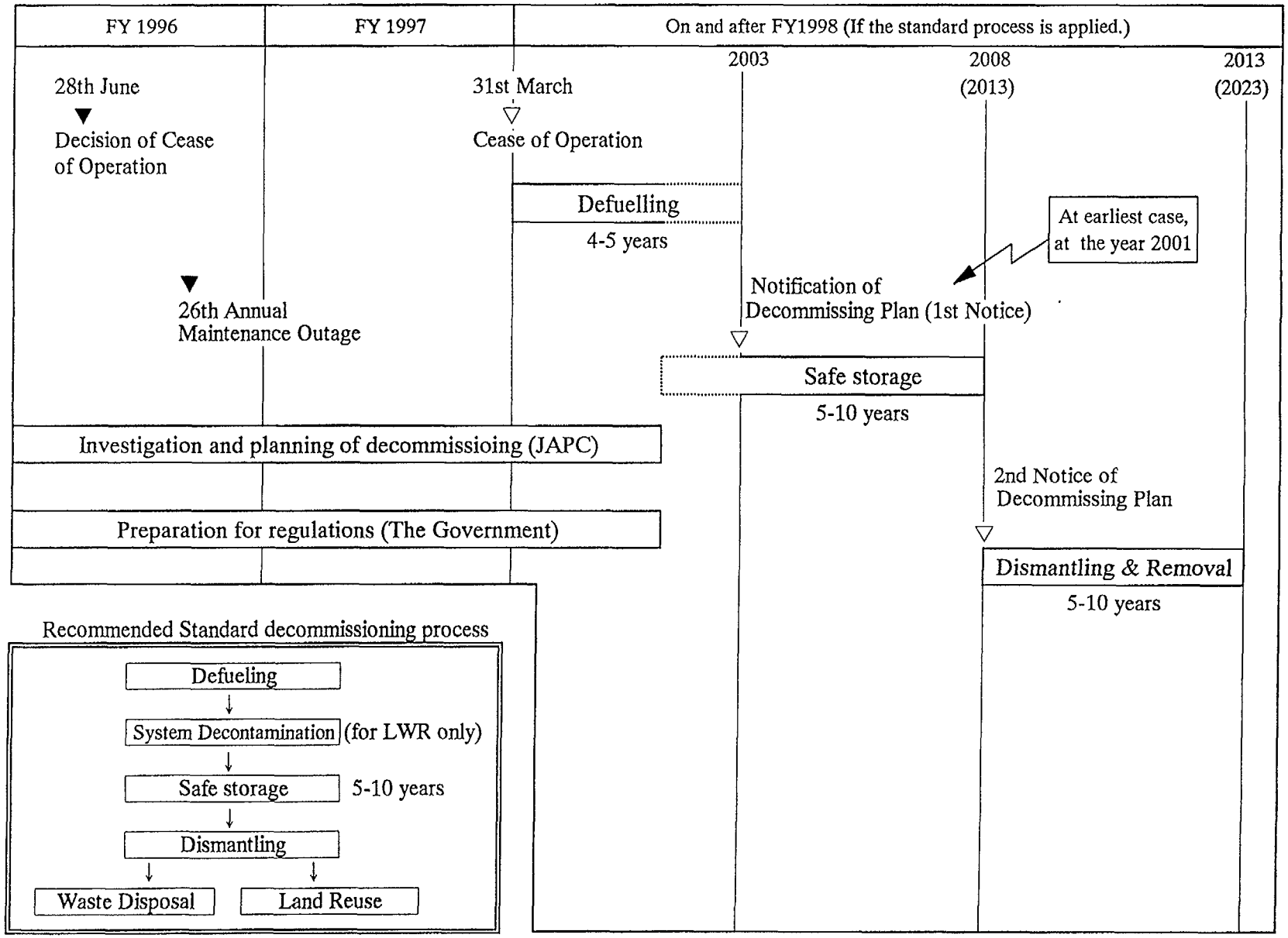

Fig.2. Programme of Decommissioning of Tokai Power Plant (Tentative)

Technologies in various fields has been developed, demonstrated and accumulated through the construction and operation of Tokai Power Plant. It has also contributed to training for many nuclear engineers in other Japanese power companies and industries who have contributed to constructions and operations of nuclear power stations. As a pioneer, Tokai Power Plant has contributed to the development of nuclear power generation in Japan.

3.2. Reason of cessation

Tokai Power Plant has a small capacity in its electric power output, even though the size of the reactor and heat exchangers are rather big, compared with LWRs of the same capacity, due to the characteristics of GCR. Therefore, the generation cost is higher than LWRs. And as there is no plant whose reactor type is the same as that of Tokai Power Plant in Japan, the costs for maintenance and fuel cycle are relatively higher than that of LWRs. It was concluded that the longer we operate it, the less we can take economical advantage of it.

As a result of the evaluation for the future operation of Tokai Power Plant, including the current status for supply of electricity by the other Japanese utilities and the study of decommissioning by Japanese government, we decided to have a plan of stopping its commercial operation of Tokai Power Plant in the end of March 1998, when we completely consume its fuel that we possess.

3.3. Plan after shutdown

After cease of its commercial operation, defuelling will be carried out within four to five years and the fuels will be transported to the reprocessing plant in the UK. The definite plan after defuelling is not decided. (See Figure 2.)

From now on, we set about performing necessary studies and researches on the field of plant characterisation, remote cutting and waste disposal, for carrying out the decommissioning of Tokai Power Plant safely and economically. We are going to prepare the decommissioning plan for Tokai Power Plant in a few years based on the standard process recommended by the government and on the situation of establishment of relevant criteria under the consultation and co-ordination with the government, local communities, utilities and relevant organisations.

We are now preparing the defuelling procedures, and also planning to reduce the frequency and extent of maintenance work and man-power after the cease of operation. In line with these study, we are preparing the drafts of the revised operating rules and revised annual inspection plans.

4. Present Status on Investigation and Research

Decommissioning has already been carrying out in many research facilities in Japan. In many countries, decommissioning of some commercial nuclear power plants have also been carried out. Therefore, technology for executing decommissioning safely can been said to be established.

There remains no significant technical problems with decommissioning which must be solved. In other words, discussion of decommissioning of commercial power plants has already shifted from the phase of the dismantling methods or safety assurance methods to the phase of system engineering such as the proper combination of technologies or how to implement decommissioning economically.

In this respect, it is necessary to develop technologies for more rational and realistic decommissioning. In Japan, NUPEC (Nuclear Power Engineering Corporation) has almost finished development of various basic technologies and now performing technology development aiming for a rational system for the decommissioning of nuclear power plants.

Based on these circumstances, JAPC has been executing the investigations and researches. Among these activities, "Assessment of radioactive inventory" and "Feasibility study of decommissioning method" are described below.

(1) Assessment of radioactive inventory

Characterisation of radioactive inventory of a plant, including the configuration of nuclides of radioactivity and its distribution in the plant, is vitally important not only for actual decommissioning but also in the stage of preparation of regulatory system and decommissioning plan, together with the data base of the plant, such as information on the structures and materials in detail of each piece of equipment and building.

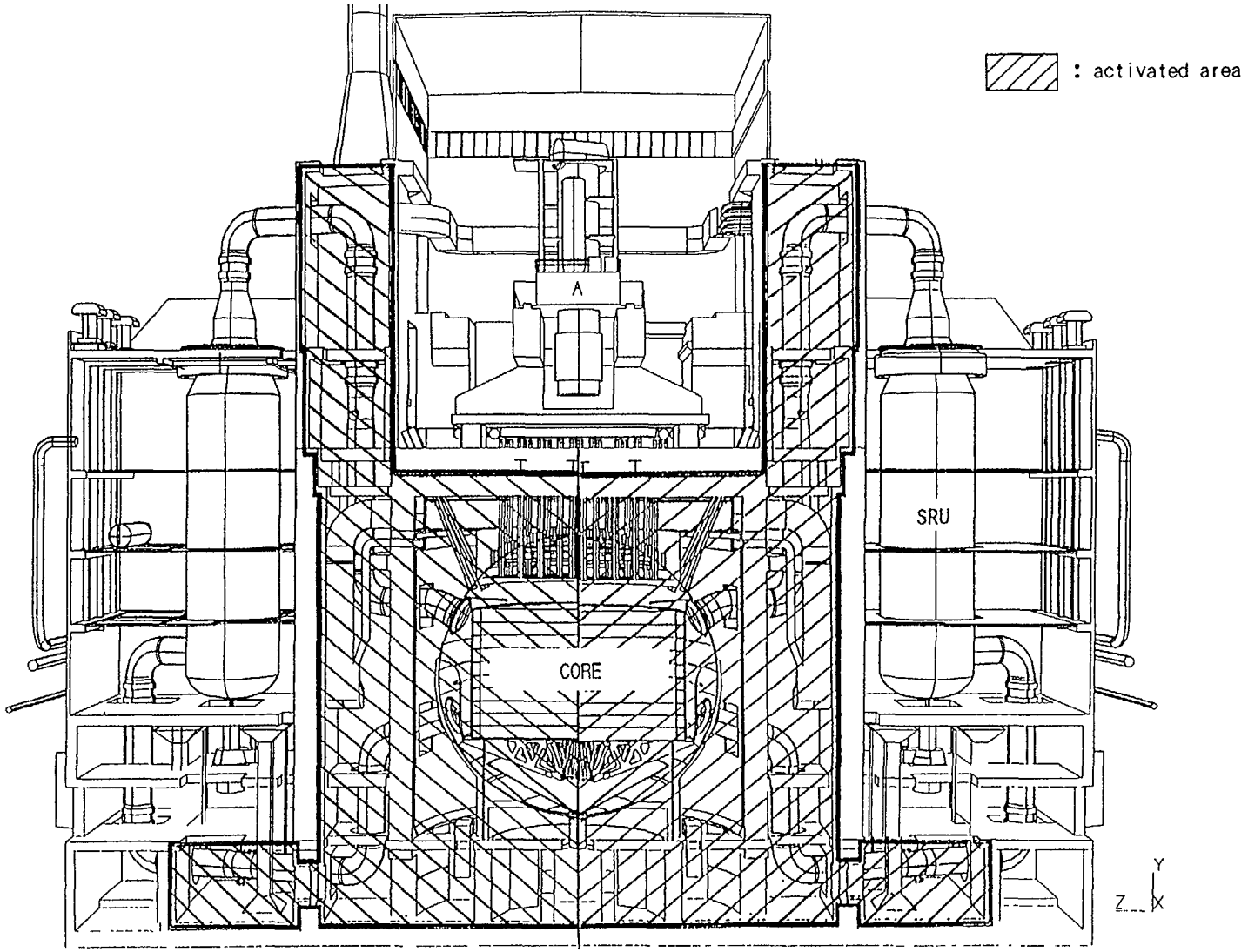

Fig.3 Activated area based on inventory assessment

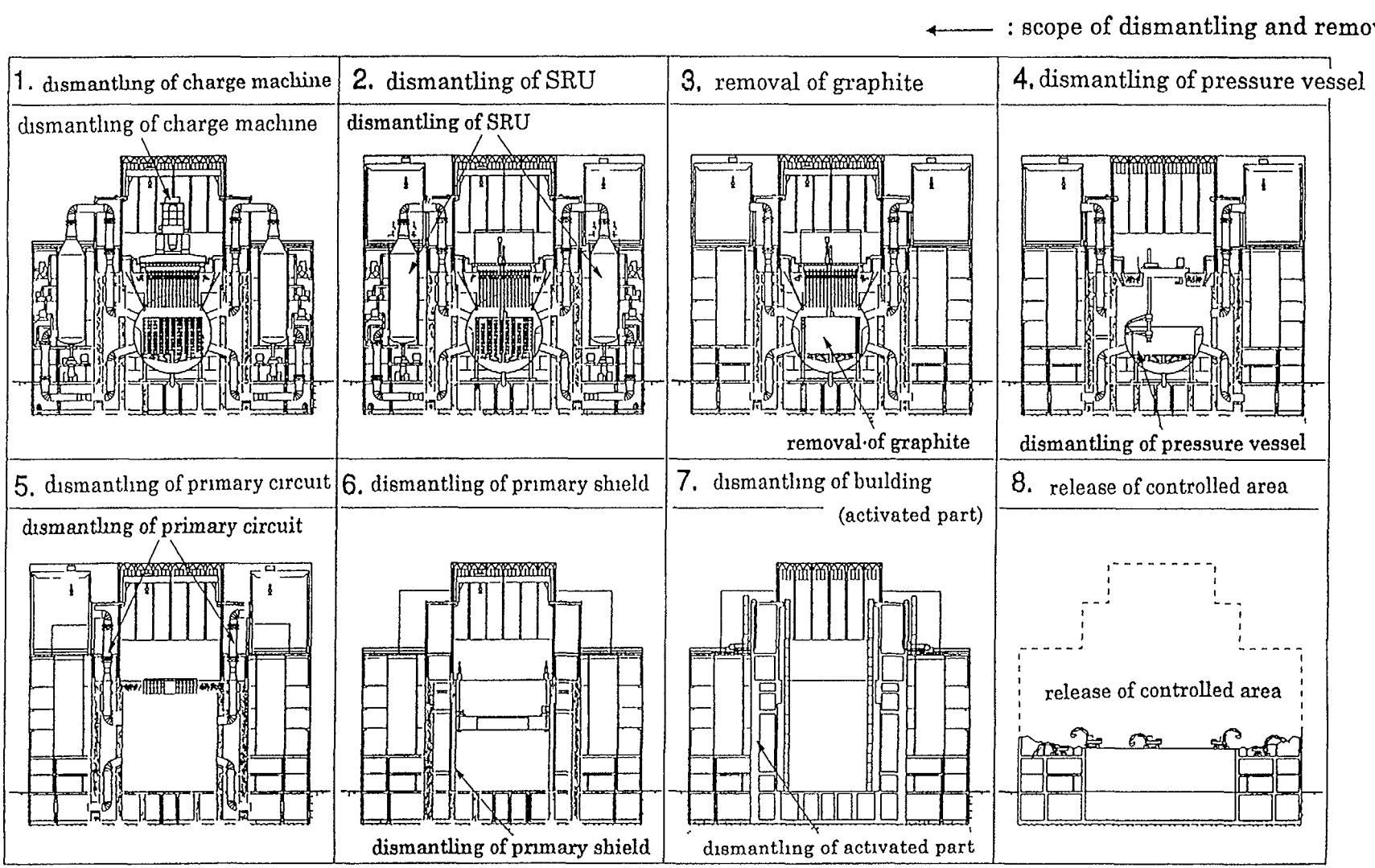

Fig 4 Outline of dismantling process of Tokai Power Plant

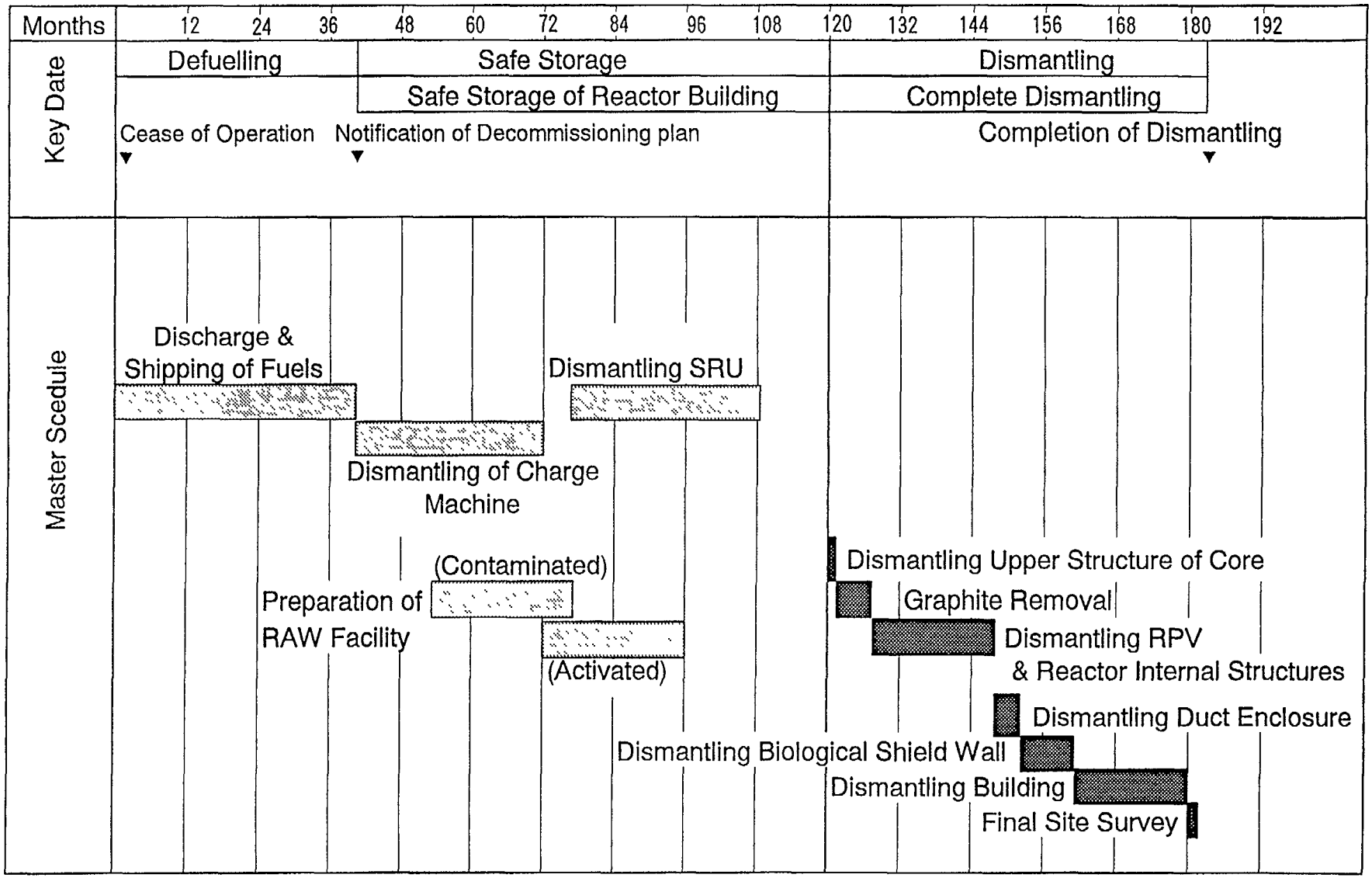

Fig.5 Dismantling programme of Tokai Nuclear Power Plant (for case study)

In Tokai Power Plant, collection of such data has been carried out within the extent that it does not interfere the plant operation, and the data has been accumulated. An example is shown in Figure 3. We think we have to continue these activities further, such as taking the samples which could not be possible during plant operation.

(2) Feasibility study of decommissioning method

The extensive feasibility study on Tokai decommissioning has been carried out. The object of the study is to investigate the methods and procedures which are feasible with more reasonable costs and to find out the points for further cost reductions. The study covers the whole aspects of decommissioning, including the safe storage period and methods, dismantling methods, radioactive waste treatment, packaging of decommissioning wastes, and burial disposal of wastes. From the results of the study, examples of dismantling procedures and programmes are shown in Figure 4 and 5, respectively.

(3) Further R&D

JAPC will focus its efforts on the development of the following technologies from now on:-

① Remote cutting and handling technology (with high autonomy)
② Dismantling method with high efficiency and low cost
③ Decontamination method with less secondary wastes
④ Technology to prevent radioactivity dispersion and technology for collection and treatment of radioactive substances with high efficiency
⑤ Engineering database and engineering system

5. Conclusion

Tokai Power Plant, the first commercial nuclear power plant in Japan, has decided to stop commercial operation in March 1998. However, cease of operation does not mean the immediate start of dismantling. It takes about four to five years for defuelling. Shortening of the period is being investigated. Considering the availability of transportation for the spent fuels to the reprocessing plant in the

UK, it takes at least three years or more. Five to ten years for safe storage is required if it follows the standard process for decommissioning in Japan. As a result, the period of about ten years or more will be necessary before the actual start of dismantling of major facilities. It means that planning for decommissioning should consider technological level and social situation in the future of ten years or more.

Tokai Power Plant will have an important role in Japan to demonstrate that the decommissioning of commercial nuclear power plants, including LWRs, can be executed safely and economically. Therefore, on the planning for its decommissioning, we have to take consideration of the application for LWRs, which are now dominant in the commercial nuclear power generation in Japan.

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THE DECOMMISSIONING OF COMMERCIAL MAGNOX GAS COOLED REACTOR POWER STATIONS IN THE UNITED KINGDOM

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Abstract

There are nine commercial Magnox gas-cooled reactor power stations in the United Kingdom. Three of these stations have been shutdown and are being decommissioning, and plans have also been prepared for the eventual decommissioning of the remaining operational stations. The preferred strategy for the decommissioning of the Magnox power stations has been identified as 'Safestore' in which the decommissioning activities are carried out in a number of steps separated by quiescent periods of care and maintenance. The final clearance of the site could be deferred for up to 135 years following station shutdown so as to obtain maximum benefit from radioactive decay.

The first step in the decommissioning strategy is to defuel the reactors and transport all spent and new fuel off the site. This work has been completed at all three shutdown stations. Decommissioning work is continuing on the three sites and has involved activities such as dismantling, decontamination, recycling and disposal of some plant and structures, and the preparation of others for retention on the site for a period of care and maintenance. Significant experience has been gained in the practical application of decommissioning, with successful technologies and processes being identified for a wide range of activities. For example, large and small metallic and concrete structures, some with complex geometries, have been successfully decontaminated. Also, the reactors have been prepared for a long period of care and maintenance, with instrumentation and sampling systems having been installed to monitor their continuing integrity. All of this work has been done under careful safety, technical, and financial control.

1. INTRODUCTION

There are nine commercial gas-cooled reactor power stations of the Magnox type in the United Kingdom. These are owned and operated by the recently established (1996), sole remaining public sector electricity utility, Magnox Electric. Each of these stations consists of twin gas-cooled, graphite moderated reactor units which use Magnox clad natural uranium fuel. The stations were commissioned between 1962 and 1971 and six of them are still operational. The other three stations have been shutdown and are in the process of being decommissioned. These decommissioning stations are Berkeley in England which was shutdown in 1989, Hunterston A in Scotland which was shutdown in 1990 and Trawsfynydd in Wales which was shutdown in 1993.

Magnox Electric, and its predecessor companies, have undertaken extensive studies over many years to develop detailed strategies, plans, processes and costings for the decommissioning of all of its stations. These plans are now being implemented at the three shutdown stations and are available for implementation in the future when the other stations reach the end of their operating lives.

2. DECOMMISSIONING STRATEGY

A wide range of decommissioning strategies have been considered for the Magnox stations ranging from dismantling the whole station immediately following station shutdown through to not clearing the site but burying the main radioactive parts such as the reactors in-situ.

Work originally undertaken in the 1980s identified that there were potential technical and economic benefits in deferring the dismantling of parts of the stations, and that this was particularly marked for the reactors. A typical radioactive dose decay curve over time following station shutdown for a Magnox reactor internals is shown in Fig 1. The rate of dose decay is initially dominated by the short half-life radionuclide Cobalt-60 and hence it continues to fall rapidly by orders of magnitude until a level is reached at which the radiation dose becomes dominated by longer lived radionuclides such as Niobium-94 and Silver-108m. The rate of dose decay over time then begins to level off and, after about 135 years following reactor shutdown, there is no further significant reduction in dose rate over time. Although the dose rate within the reactors is too high to allow any internal man-access for some decades following station shutdown, by the time the dose decay curve has levelled off significant man-access is allowable.

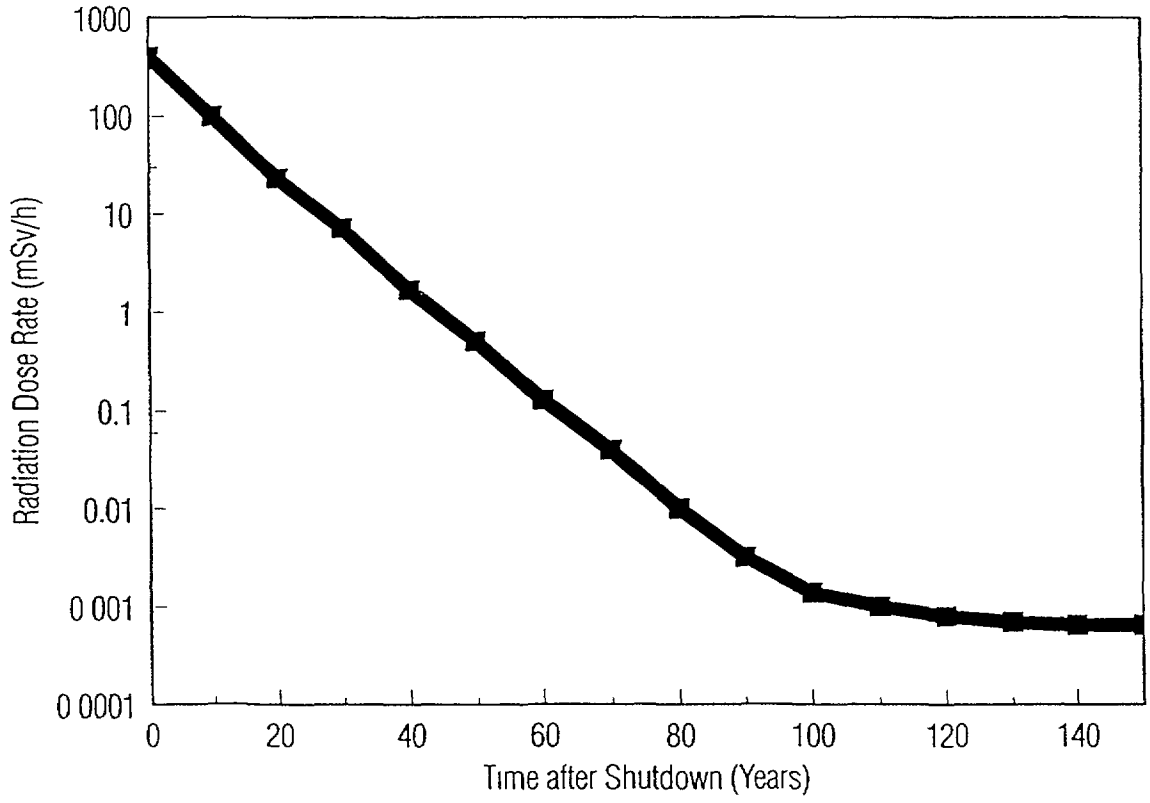

FIG 1 Reduction in dose rate with time inside a typical Magnox reactor

If reactor dismantling was to be undertaken in the early years following station shutdown it would have to be done fully remotely using sophisticated machinery, and would generate large quantities of radioactive waste and involve the handling of highly radioactive materials. Deferring the dismantling of the reactors by up to 135 years to allow the substantial radioactive decay to occur would mean that 'manual' dismantling using simple technology would be possible. Also, radioactive waste quantities would be much lower as would be the radiation dose rates on the materials needing to be handled.

The shape of the dose decay curve is typical for all reactor types but, for the Magnox gas-cooled graphite moderated reactors, the dose rate at which the curve levels off is much lower than for water reactors for which it is 2 to 3 orders of magnitude higher. This means that there are not the same benefits in deferring the dismantling of water reactors as there clearly are for the Magnox reactors. This may not be a significant disadvantage for water reactors because of their small size and ease of dismantling but the benefits that accrue for the Magnox reactors are particularly marked because of the large size and complexity of the reactor structures. Magnox steel reactor pressure vessels are typically spherical of 20m diameter and weigh 5,000te.

Extensive studies have been performed to determine the most appropriate strategy for the decommissioning of the Magnox reactor power stations. These have taken account of the benefits associated with radioactivity decay, as indicated above, as well as a wide range of other technical, safety, environmental and economic factors. Various strategy options have been considered and compared in a rigorous and systematic way using multi-attribute decision analysis techniques. These studies concluded in 1990 that the 'Safestore' strategy was the preferred option for the decommissioning of the UK's commercial gas cooled reactors, including the Magnox reactor power stations (Ref 1). The final study leading to this decision was independently reviewed, confirming the outcome, and in 1995 a UK Government policy review identified Safestore as a potentially feasible and acceptable strategy for decommissioning nuclear power stations (Ref 2).

The Safestore strategy can be described as follows in terms of the three standard stages of decommissioning:-

Stage 1: Defuelling
Care & Maintenance Preparations
Care & Maintenance 1

Stage 2: Safestore Construction
Care & Maintenance 2

Stage 3: Site Clearance

The Safestore strategy has not been rigorously defined in detail but is flexible in application so that it can accommodate changes in circumstances or reflect specific situations at individual stations. A key feature of the Safestore strategy is to defer the dismantling of radioactive plant and structures where there are demonstrable overall benefits in doing this, subject of course to the essential requirement that the safety of the workforce and public is maintained at all times. This strategy enables the hazards associated with the station to be systematically and progressively reduced.

The first step in the Safestore strategy is to defuel the reactors and transport the fuel off-site for reprocessing, which removes about 99.9% of the radioactivity from the site. This is followed by a period of preparatory work during which the majority of the inactive and some of the radioactive plant and buildings will be removed, and the remaining materials, plant and buildings prepared for a period of care and maintenance. After this period, which could typically last until 35 years following station shutdown, it is assumed that it will be necessary to upgrade the buildings on the site to enable them to remain in a safe, secure and weatherproof condition until final site clearance which could beneficially be deferred until up to about 135 years after station shutdown. It is therefore proposed, as the first step in Stage 2, to convert the buildings into what has been termed 'safestore structures', eg by recladding them in very durable materials. The second period of care and maintenance will then eventually lead to the final site clearance stage when the site will be returned to a 'green-field' status.

The Safestore strategy as described above indicates that there will be two periods of care and maintenance prior to final site clearance and this is how it is presently being applied at Berkeley and Hunterston A. However, the built-in flexibility means that if circumstances are appropriate it is possible to move directly from defuelling through to the Stage 2 activities and only implement the second period of care & maintenance. This version of the Safestore strategy is being applied to Trawsfynydd.

3. ARRANGEMENTS FOR DECOMMISSIONING

All activities on a UK nuclear power station are subject to the Conditions stated in the Nuclear Site Licence and are subject to monitoring and scrutiny by independent regulators). These Conditions are standard for all nuclear sites and apply equally to operational and decommissioning. There is therefore no requirement for a re-licensing stage when a station ceases operation and starts decommissioning. However, the change in status of the site does involve new activities, procedures and arrangements and this does necessitate the preparation, submission, approval and demonstration of various aspects of these changes. All of these processes are subject to the Company's own internal scrutiny and approval arrangements but the more significant aspects require submission to and acceptance by the regulators. These include changes to the overall site safety case, eg for defuelling, and to the site management arrangements. All decommissioning activities are subject to an appropriate level of safety assessment.

Berkeley was the first full scale commercial nuclear power station to be shutdown in the UK and therefore this was the test-bed for developing in detail, applying and demonstrating suitable arrangements for progressing decommissioning and satisfying the regulatory requirements. The lessons learnt from this period have been applied to the decommissioning of Hunterston A and Trawsfynydd Power Stations which closed at a later date. These arrangements are sufficiently flexible to take account of the changing nature of the site, the reducing staffing levels and the associated safety issues as decommissioning progresses. The arrangements and regulatory considerations need to recognise the significantly reduced safety issues and hazards associated with a decommissioning station as compared to an operating station. This reduction occurs after shutdown when the reactor is held firmly sub-critical and not at temperature or pressure, and is most marked following the completion of defuelling and the removal of all fuel from the site.

Financial arrangements for decommissioning is another key issue. The UK practice is to make appropriate financial provision during the operating lifetime of stations for their eventual decommissioning, taking due account of when decommissioning activities are expected to occur by applying cost discounting. Detailed technical and economic studies have been performed to derive decommissioning cost estimates for provisioning purposes, and allowances for uncertainties and risks have been included. To support actual decommissioning work on stations, financial control and monitoring processes have been implemented and these have enabled, for example, actual expenditure to be checked against that predicted thus allowing improvements to be made in future expenditure estimates.

Decommissioning of a nuclear power station is a major project and therefore to manage all the work, including the safety and regulatory issues, requires the application of the full range of project management systems and disciplines. Magnox Electric performs the overall Project Manager role for the decommissioning on the sites and, as appropriate, lets contracts for services or decommissioning activities. The Company, as the holder of the Nuclear Site Licence is responsible for and must remain in control of all safety issues.

4. DEFUELLING

Defuelling of the reactors is the first decommissioning activity that is undertaken and it has been completed at each of the three shutdown stations. Magnox reactors contain typically about 30,000 to 40,000 individual Magnox fuel elements each about 60cm long. Magnox reactors are designed for on-load refuelling and at the standard rate of refuelling it would take about 5 years to defuel the reactors. However it was recognised that this process could be speeded up as there was no longer a requirement to add new fuel alongside fuel removal, and operations could be performed with the reactors depressurised and with an air rather than carbon dioxide atmosphere. These revised fuel removal processes necessitated some modifications to the reactor fuelling machinery.

Defuelling of the reactors requires a new safety case to be prepared that addresses relevant fault conditions and, for example, confirms the acceptability of removing fuel which introduces empty fuel channels and could reduce the cooling flow to the remaining fuel. Fuel was removed systematically, normally starting at the peripheral channels and working towards the centre of the reactor. Prior to the start of defuelling all the control rods were lowered into the reactors and electrical supplies to them isolated so that no more than one rod could be raised at any one time.

The rate at which fuel could be removed from the site was not controlled by how quickly the fuel could be removed from the reactors but by the rate at which fuel transport flasks could be prepared for dispatch. One of the limiting factors in the transport safety case, and hence the number of fuel elements that can be put into a fuel flask, is the residual heat load of the elements. During the defuelling period the heat load reduces sufficiently to allow an increased loading within the flasks by up about 20%. Revised transport safety cases were therefore prepared for irradiated fuel from defuelling which assisted in reducing the overall duration.

Berkeley was the first station to be defuelled and, with the improvements that were identified and introduced in that period, it was completed within 3 years of station shutdown, ahead of the originally predicted programme and about 30% below the identified budget. The experience gained and lessons learnt at Berkeley were applied to later defuelling activities thus enabling, for example, defuelling at Trawsfynydd to be completed in 2 years.

One issue associated with defuelling that was recognised as important prior to the start of defuelling at Berkeley, and which has been borne out by experience, is the need to apply quality control procedures to verify the removal of all fuel from the reactors, and the station. This is particularly relevant to Magnox reactors which contain a large number of fuel channels and fuel elements, each of which are removed and handled separately. Special procedures were adopted and video recordings of fuel handling operations made and reviewed. Human factor assessments were performed to check where errors could occur and to predict their probabilities and significance. Although there was a strong emphasis and regulator interest in verifying complete fuel removal, safety assessments have shown that the consequences of some fuel remaining in the reactors is not significant.

5. PREPARATIONS FOR CARE & MAINTENANCE

Following the completion of defuelling on the three shutdown stations, decommissioning work has continued in preparation for a quiescent period of care and maintenance in accordance with the Safestore strategy. The extent of work undertaken and planned at each site does vary and reflects the time period since station shutdown and individual circumstances at each site. Progress at Berkeley has been the most extensive as can be seen by comparing Fig 2 which shows the station prior to decommissioning and Fig 3 which is a more recent photograph. The work undertaken at Berkeley is more extensive than the minimum level possible under the Safestore strategy but this has been due to a number of reasons. For example, the station layout was unique with external primary circuit ducting and boilers, the commercial climate has been favourable and, as it was the first commercial nuclear power station to be shutdown, there were perceived to be benefits in demonstrating that decommissioning could be progressed without difficulty.

Some of the decommissioning work done at Berkeley is being repeated at Hunterston A and Trawsfynydd although progress is not as advanced. At Trawsfynydd it is intended to moved directly to Stage 2 of the Safestore strategy without applying the intervening Care & Maintenance 1 step. This reflects its unique remote, inland location in a National Park and a local public consultation exercise that was undertaken which highlighted the importance of maintaining local employment and reducing the site's visual impact. It is therefore planned to significantly reduce the height of the reactor buildings, which will involve significant active plant dismantling, and to reclad and improve the appearance of the residual buildings. The extent of work is indicated by comparing Fig 4 which shows the station prior to shutdown and Fig 5 which is an artist's impression of the final safestore structures.

Some of the main post-defuelling decommissioning activities that have been undertaken on the shutdown sites are described below.

5.1 Dismantling

The scale of the Magnox reactor primary circuit components, such as gas ducts and boilers, is large and they are lightly contaminated internally (predominantly by Cobalt-60) although some sections have also been activated by neutron irradiation. At Berkeley the high level top gas ducts had to be removed from their external position where they would have been exposed to weather induced degradation if dismantling had been deferred. This required the ducts to separated from the primary circuit and then lifted and lowered to ground level in large 35m long 100te sections (Fig 6) prior to being cut up into smaller pieces for longer term on-site storage, or decontamination and disposal as inactive scrap metal. For similar reasons it was also found necessary to dismantle the 16 buildings containing the boilers and

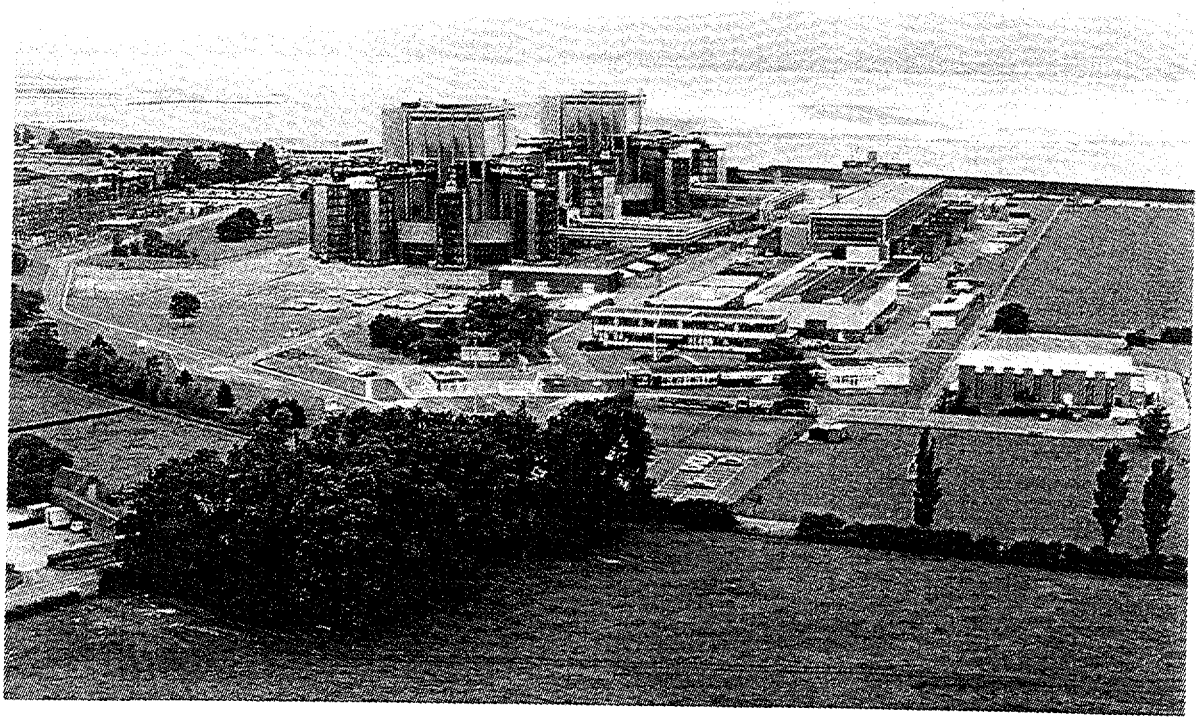

FIG. 2. Berkeley Power Station before the start of decommissioning

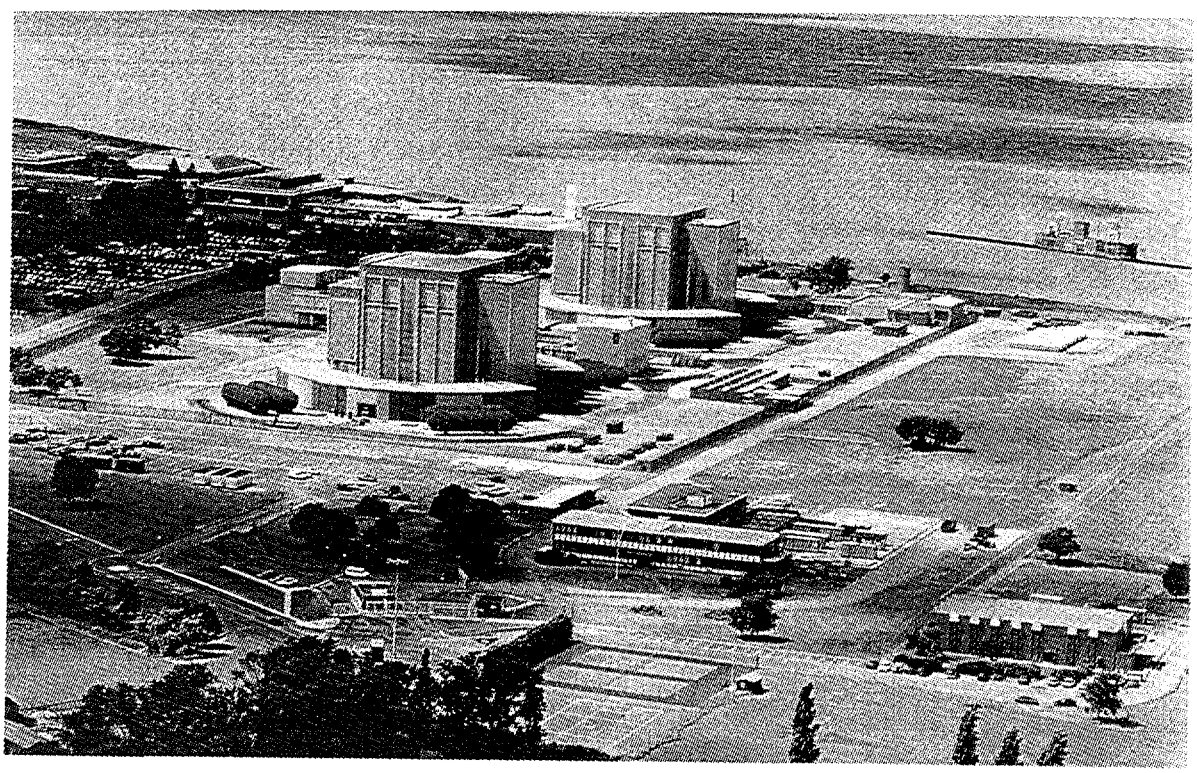

FIG. 3. Berkeley Power Station following decommissioning works (1996)

FIG. 4. Trawsfynydd Power Station before the start of decommissioning

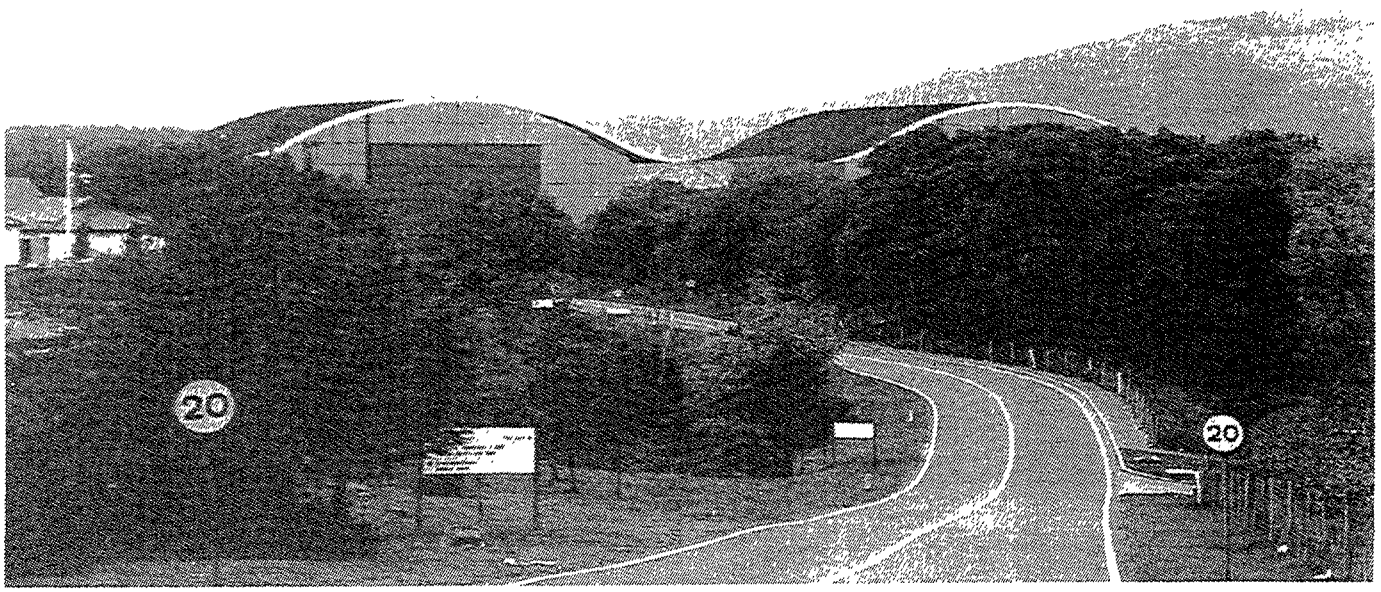

FIG. 5. Trawsfynydd Power Station following safestore construction (artist's impression)

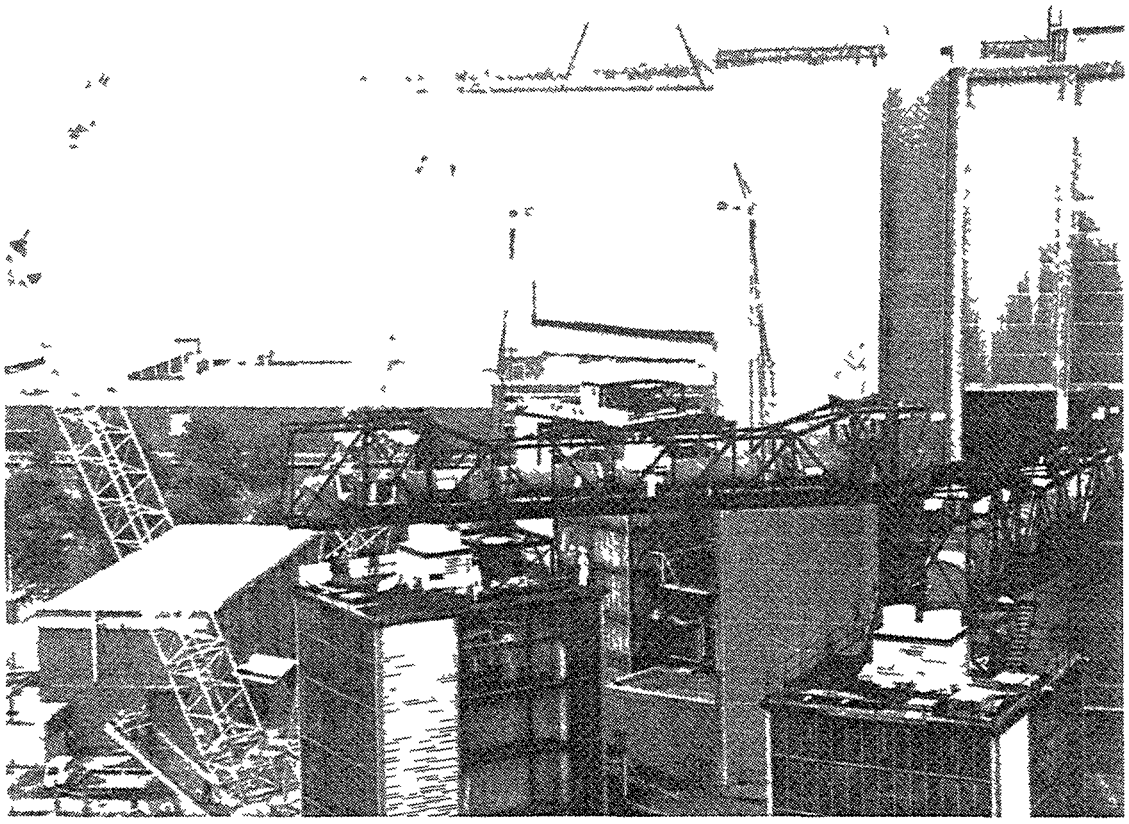

FIG 6 Berkeley Power Station top gas duct removal

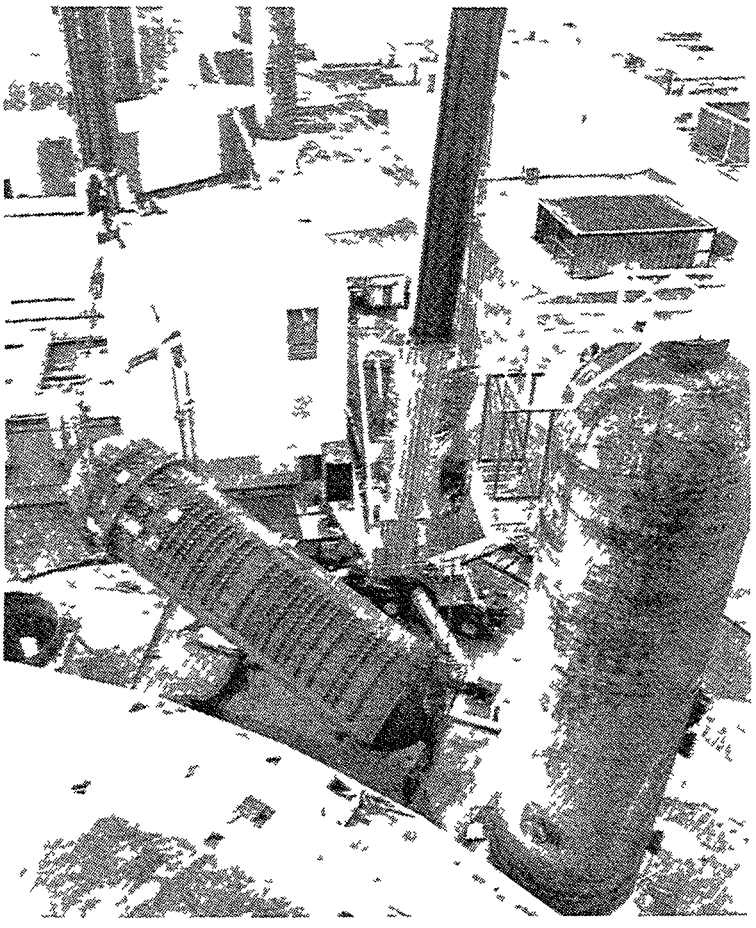

FIG 7 Berkeley Power Station boiler lowering to the ground

then, to improve the visual appearance of the site, the boilers which weigh about 300te each were lowered to a horizontal position at ground level (Fig 7). Dismantling has therefore involved heavy lifting and handling issues as well as the more usual radioactive aspects.

Other primary circuit related components such as fuelling machinery and reactor gas circulators and plant have been dismantled. This has involved the removal, handling and treatment of some unusual constituents such as bitumen in semi-liquid form and high density polythene chippings that were used for neutron shielding purposes. The gas circulator plant also contains large quantities of oil, typically $50m^3$ per reactor, which is lightly contaminated by tritium. Treatment options being considered are incineration and continued storage to achieve natural decay to inactive levels.

As well as plant associated with reactor gas circuits which are generally contaminated by activation products such as Cobalt-60, there is also plant associated with liquid systems contaminated by fission products such as Caesium-137. Plant associated with the fuel cooling ponds falls into this category. A large proportion of it has been submerged below the pond water level during the station operational period but, when stations are being decommissioned, it is intended that the ponds will be drained and it is therefore necessary to remove such plant. This work has been undertaken at the shutdown stations and has required more extensive contamination control provisions than for the dismantling of the reactor gas circuit plant.

It has been found that the dismantling of radioactive plant that has been done can be performed using standard available industrial techniques and that, apart from the application of normal radiological control provisions, special techniques and technology do not usually need to be developed, ie the 'simple-is-best' approach can be applied.

The majority of the plant, materials and buildings that are to be dismantled in this period are not radioactive. Conventional dismantling techniques can therefore be used but their application needs to be controlled and carefully assessed before implementation to ensure there are no consequent radiological implications, eg as a result of the work being done in a radiation area or because of unexpected radioactive contamination being found. The removal of inactive plant and materials can also involve non-nuclear hazards that need precautions to be taken. For example, a major task on the stations has been to remove thermal insulation materials that have been predominantly asbestos based.

5.2 Decontamination

In order to reduce the quantities of radioactive waste needing to be disposed, decontamination has been applied where applicable and beneficial overall. One example has been the gas ducts which are generally simple shaped surfaces and are not heavily contaminated. It has proved possible to decontaminate them using simple manually applied cleaning, abrasive or jetting techniques. However, although this removes all surface contamination, it has been found that tritium is dispersed throughout the body of the material which initially prevented the material being free-released. A technique was therefore developed whereby the material is heated in a furnace to drive out the tritium which can then be released to atmosphere. Following the de-tritiation process the material can be released for uncontrolled recycling in the scrap metal market.

Although the boilers contain the same contaminants as the gas ducts they have much more complex geometries, eg finned tubes, and hence simple manual decontamination techniques cannot be used. To check the feasibility of boiler decontamination, and to compare

it with the alternative options of longer term storage to allow natural radioactive decay or disposal as active waste, a trial decontamination of one boiler at Berkeley is being progressed. This is using chemical techniques and is now achieving decontamination to a degree which, following de-tritiation, will allow free release of the material as inactive.

The other main area where decontamination is being applied is to contaminated building surfaces, and most notably the walls and floors of the fuel cooling ponds. The ponds are simple, single-contained painted concrete structures and fission product contamination from the spent fuel elements has penetrated the surfaces to a depth of some centimetres. To prevent the spread of contamination and airborne release once the ponds are drained it has been necessary to apply surface removal decontamination techniques. The bulk of the contamination is in a surface layer and ultra-high pressure water jetting techniques have been used successfully to remove this. Once this layer is removed, which significantly reduces background radiation levels, it is possible to perform more detailed and accurate surveys to determine the residual depth of contamination that requires removal. Final decontamination of these surfaces is being achieved by techniques such as concrete planing.

5.3 Disposal

The materials resulting from dismantling and decontamination activities need to be released or disposed of appropriately. The options that are available are, in order of preference: re-use, recycling and disposal. A number of plant items such as inactive tanks, transformers and instrumentation have been transferred for re-use at other sites. A large proportion of the dismantled materials is inactive and can be placed in the recycling market. Over 20,000te of such materials have been recycled from Berkeley and this has included ferrous and non-ferrous metals, cables, cast iron and glass. Concrete and building rubble has been used as infill material on the site. It has been necessary to dispose of some materials and, at Berkeley, this has included asbestos which has been sent to a special licensed disposal site and low level radioactive waste, typically 300te per year, which has been sent to the Drigg repository.

A very important feature in the disposal process is the monitoring of materials to determine and confirm radioactivity levels, particularly where material is to be free-released as non-radioactive. Detailed procedures have been developed, agreed with the appropriate independent regulators, implemented and demonstrated. These include detailed monitoring of individual components and bulk monitoring, and confirmatory checks using gamma spectroscopy equipment and detailed radiochemical analysis.

5.4 Long Term Safe Enclosure

It is proposed, under the Safestore decommissioning strategy, that some buildings and structures could be retained on the site for up to about 135 years after station shutdown. It is therefore necessary to ensure that these facilities can endure this long period in a safe, secure and weatherproof condition. The plant and structures, eg the reactors, that it is intended to retain are of substantial and robust construction and the bulk of the remaining radioactivity is within activated materials and hence physically locked in and not mobile. Therefore the main requirement is to provide conditions that will minimise degradation mechanisms affecting these materials.

A key protective measure is to retain buildings in a weatherproof condition and, ideally, this should result in only minimal maintenance and repair being required throughout the care

and maintenance periods. The buildings that are being retained are therefore being refurbish and upgraded, primarily by repairing or replacing roofs and by re-cladding with durable materials. Attention is also being paid to minimising ground-water ingress into buildings and providing appropriate drainage arrangements. Detailed baseline civil surveys have been performed on all retained buildings to identify the present condition of all key structural features, eg the extent and type of any existing concrete degradation. This information is being used to establish repair programmes and long term monitoring requirements.

The main degradation process that needs to be considered for the retained radioactive plant is metallic corrosion, eg of the steel reactor vessels. Detailed studies have been undertaken to determine the potential degree of corrosion, to identify the atmospheric conditions that apply, and to measure and monitor the actual corrosion rates that are being experienced at various positions. A range of corrosion mechanisms or drivers have been considered. For example, there was originally some suggestion that nitric acid may be formed within the vessels as a result of defuelling in an air atmosphere but detailed investigations have now shown that this is not a problem.

Reactor vessels have been placed in their long term storage state which has involved closing or sealing all penetrations other than for a single vent line to the external atmosphere. This provides predictable aerobic conditions within the vessel and the rate of air interchange between the vessels and the external atmosphere is very low. Instrumentation connected to data loggers has been placed within and external to the reactor vessels to measure atmospheric conditions, eg temperature and humidity. Corrosion samples and probes have also been introduced and are inspected routinely. The monitoring that has been done to date indicates that actual corrosion rates are very low, eg $<1\mu m.y^{-1}$.

Studies have also been undertaken on the potential degradation mechanisms that might apply over the long term to the graphite moderator within the reactor vessels. These indicate that there are no significant problems and this is reported in a separate paper (Ref 3).

5.5 Accumulated Operational Wastes

The UK practice has been to store on the power station site, pending the availability of a waste repository, certain intermediate level wastes that have been produced during the operational period of the station. These include sludges and ion exchange materials resulting from the treatment of effluents and fuel pond water, and solid wastes including those materials removed from spent fuel elements prior to transport off site and other miscellaneous contaminated and activated materials. These wastes have been stored in a variety of concrete vaults and metal tanks and part of the decommissioning process is to determine what actions need to be taken on these wastes and to progress the necessary work.

Some of the stored wastes such as activated materials are very stable and held within very robust structures, eg the concrete reactor biological shields. These wastes could therefore be retained on site until the reactors themselves are dismantled. Other wastes are potentially more mobile and some of the storage facilities do not have the same long term integrity. It is therefore necessary to consider each situation on a case-by-case basis to determine the most appropriate management strategy. As a result of such assessments it has been decided that a significant proportion of the total wastes on the three shutdown sites will now be retrieved, treated and solidified, eg in cement, and the resulting packages then stored on the sites until a disposal repository is available. These waste management activities are a major part of the overall work that is being progressed on the sites and will utilise technology and processes that have been developed by Magnox Electric and its predecessors over a number of years.

5.6 Miscellaneous

It has been recognised that decommissioning sites can be a useful source of information that is relevant to those stations that are still operational. For example, one of the key issues relevant to the safety cases for the continued operation of the steel reactor pressure vessel Magnox stations is the effects of neutron irradiation induced embrittlement. The reactor vessels at Trawsfynydd have been identified as a source of representative and highly irradiated material that can be sampled and subjected to detailed analysis to determine the actual rather than just the predicted effects of embrittlement. This has required special remotely operated equipment to be developed that is capable of reaching the base of one of the reactor vessels, which is not readily accessible and is in a high radiation area, and that can then remove and retrieve steel samples for laboratory analysis. This equipment is now being deployed successfully.

6. CONCLUSION

Magnox Electric is progressing the decommissioning of three of its commercial Magnox gas-cooled reactor power stations, and has prepared plans for the eventual decommissioning of its six other stations that are still operational. The decommissioning strategy that is being employed is termed Safestore and involves the deferral of some decommissioning activities, such as reactor dismantling, for potentially up to about 135 years after station shutdown. Under this strategy, substantial decommissioning work is still undertaken in the years following shutdown. This starts with reactor defuelling, which has been completed at all three stations, and is followed by a range of dismantling, decontamination, preservation and waste management activities to prepare the site for a quiescent period of care and maintenance. Significant progress has been made with this work on the three shutdown sites. This has involved the successful development and application of appropriate safety, environmental, technical, financial and project management arrangements and processes. It has also been demonstrated that full scale commercial gas-cooled reactor power stations can be decommissioned using available and simple technologies.

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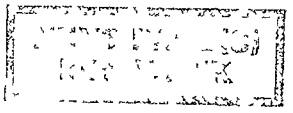

PROGRESS IN THE DEVELOPMENT OF TOOLING AND DISMANTLING METHODOLOGIES FOR THE WINDSCALE ADVANCED GAS COOLED REACTOR (WAGR)

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Abstract

Decommissioning of the Windscale Advanced Gas-Cooled Reactor (WAGR) is a major UK reactor decommissioning project co-funded by the UK Government, the European Commission and Magnox Electric WAGR was a CO_2 cooled, graphite moderated reactor which served as a test bed for the development of Advanced Gas-Cooled Reactor technology in the UK It operated from 1963 until shutdown in 1981

AEA Technology plc are currently the Managing Agents on behalf of UKAEA for the WAGR decommissioning project and are responsible for the co-ordination of the project up to the point when the contents of the reactor core and associated radioactive materials are removed and either disposed of or packaged for disposal at some time in the future

Decommissioning has progressed to the point where the reactor has been dismantled down to the level of the hot gas collection manifold with the removal of the top biological shield, the refuelling standpipes and the top section of the reactor pressure vessel The 4 heat exchangers have also been removed and committed to shallow land burial

This paper describes the work carried out by AEA Technology under separate contracts to UKAEA in developing some of the equipment and deployment methods for the next phase of active operations required in preparation for the dismantling of the core structure Most recent work has concentrated on the development of specialist tooling for removal of items of operational waste stored within the reactor core, equipment for cutting and removal of the highly radioactive stainless steel 'loop' pressure tubes, diamond wire cutting equipment for sectioning large diameter pipework, and equipment for dismantling the reactor neutron shield

The paper emphasises the process of adaptation and extension of existing technologies for cost-effective application in the decommissioning environment, the need for adequate forward planning of decommissioning methodologies together with large-scale 'mock-up' testing of equipment to ensure confidence during the active work phase

1.0 Introduction

Decommissioning of the Windscale Advanced Gas-Cooled Reactor (WAGR) is a major UK reactor decommissioning project co-funded by the UK Government, the European Commission and Magnox Electric WAGR was a CO_2 cooled, graphite moderated reactor which served as a test bed for the development of advanced gas-cooled reactor technology in the UK (Figure 1) It operated from 1963 until shutdown in 1981

Following shutdown the decision was taken to dismantle the reactor. This work has progressed as follows:

- The turbine hall and ancillary equipment have been cleared
- All fuel has been removed and sent for storage prior to reprocessing Fuel-related components eg neutron shield plugs, have been conditioned for future disposal and stored in the reactor core. These components have been termed 'operational waste'
- The reactor pressure vessel and containment have been isolated from the rest of the primary circuit and a ventilation system installed into the reactor vault and pressure vessel
- A waste packaging route for both Intermediate Level Waste (ILW) and Low Level Waste (LLW) has been constructed utilising access through one of the four heat exchanger bioshields This route connects to a custom-built waste grouting facility
- The top reactor bioshield, reactor refuelling machine, top of the pressure vessel and refuelling standpipes have been size reduced and disposed of as LLW to the LLW disposal site at Drigg
- The reactor structure above the top of the hot gas manifold or 'hotbox' has been removed and size reduced.
- A Remote Dismantling Machine (RDM) has been installed over the reactor.
- The four heat exchangers have been prepared, lifted clear of WAGR and committed to shallow land burial at Drigg.

The next phase of operational work is to remove and dispose of the remaining reactor internals and the pressure vessel

Initially this will involve a combination of semi remote and remote operations to prepare the reactor for fully remote operations using the RDM

This paper describes the work carried out by AEA Technology under separate contracts to UKAEA in developing some of the equipment and deployment methods for the next phase of active operations required in preparation for the dismantling of the reactor core

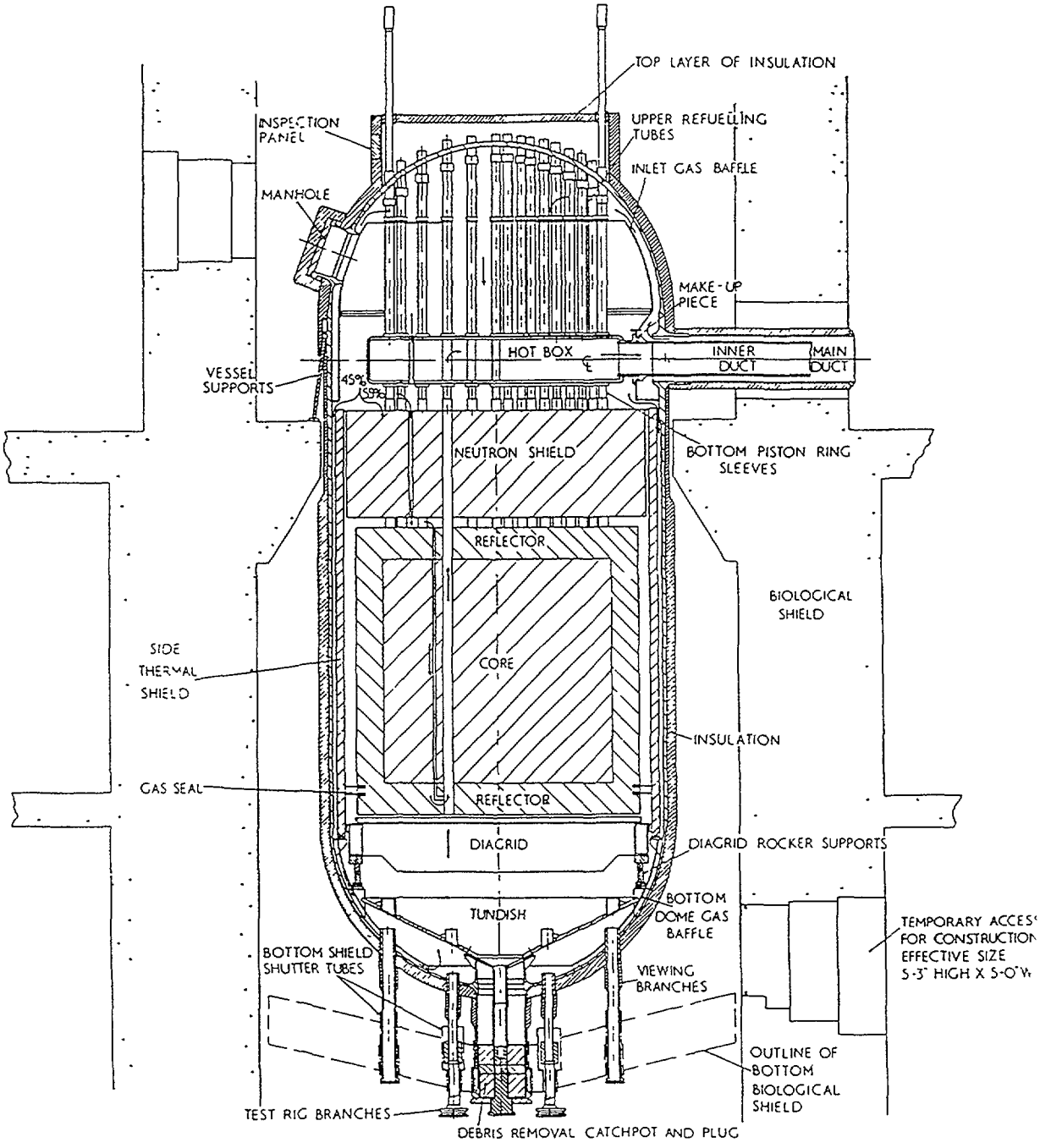

FIGURE 1 WAGR SECTIONAL ELEVATION

structure Most recent work has concentrated on the development of specialist tooling for

- Sectioning large diameter pipework using a diamond wire cutting tool
- Equipment for the removal of items of operational waste stored in the reactor core
- Customising standard equipment for size reducing and removing the hot gas manifold (the "Hotbox")
- Equipment for the removal and cutting of the highly activated stainless steel 'loop' pressure tubes
- Equipment for dismantling and removal of the complex steel and graphite structures which form the reactor's inner and outer neutron shields

The following sections underline the need for adequate testing of equipment before deployment in the remote environment and, in particular, the use of large scale mock-ups to simulate in-reactor operations This strategy has proven to be highly effective in increasing confidence in systems particularly in the development of dismantling methodologies to reduce operational timescales and operator dose uptake Lessons learnt have been valuable in directing future work

20 Development of Tooling and Dismantling Methodologies

21 PRELIMINARY OPERATIONS

In order to simplify subsequent remote dismantling operations, using the RDM and dismantling manipulator, a series of operations using manual or semi remote tooling have been performed on WAGR These operations are known as the 'preliminary operations'

The most significant of these operations was the severing of the four mild steel ducts which run from the hotbox to the heat exchangers Each duct consists of a 685 mm diameter cylindrical section, with a 25 mm wall thickness, lined with a 25 mm depth of stainless steel foil insulation As a result of burst fuel experiments, on completion of reactor operations, the insides of the ducts has also become heavily contaminated with Cs-137

One method of cutting the ducts would have been to use a thermal technique such as oxy-fuel gas cutting or plasma arc, but this would have been difficult to accomplish in the confines of the reactor It would also have generated significant secondary waste in the form of dross and aerosol and caused the caesium in the ducts to volatalise To overcome this, equipment which uses a diamond impregnated wire was developed The system is capable of semi-remote operation such that, for the duration of each cut, there is no requirement for the operator to be positioned in the radiation field around the reactor

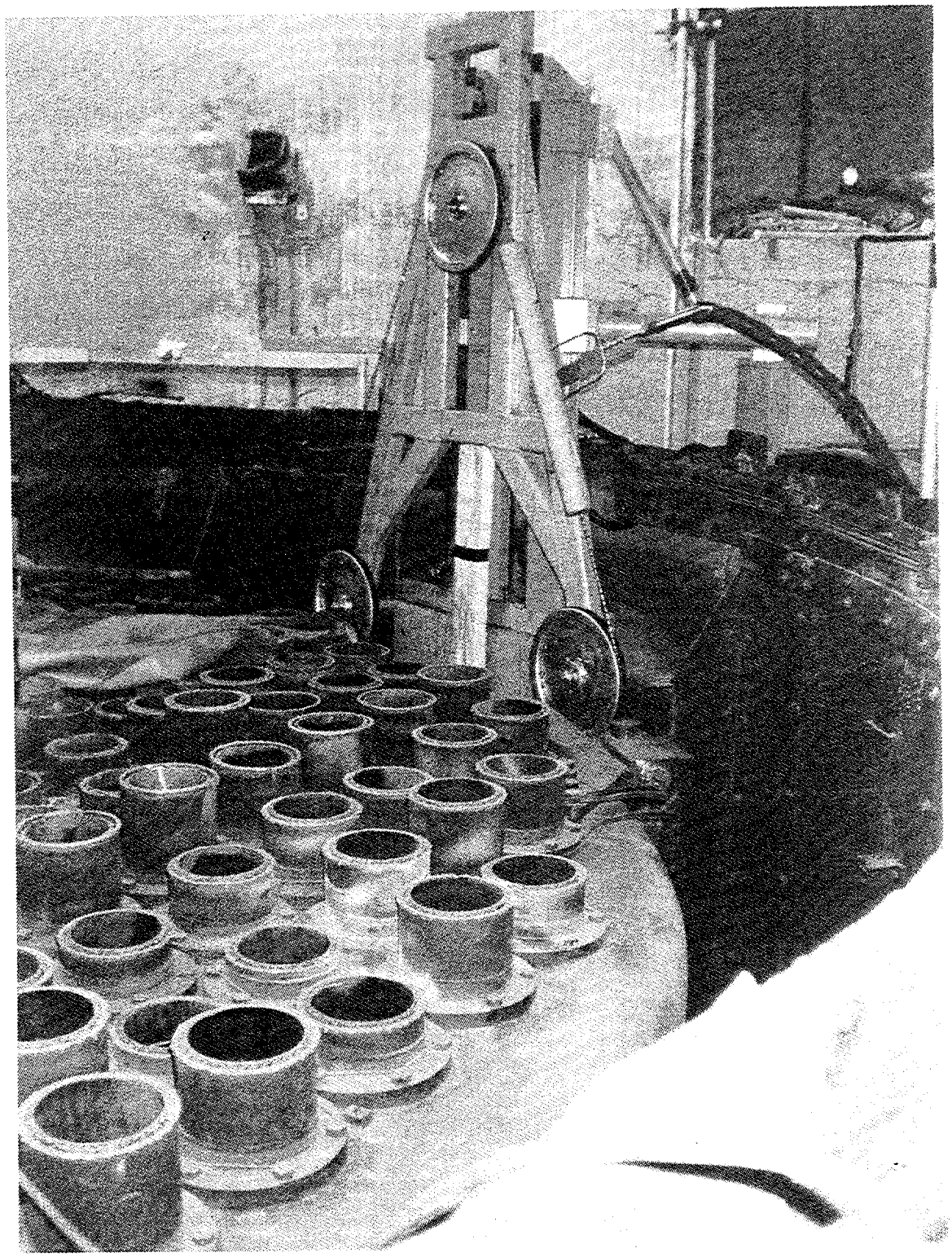

FIGURE 2 CUTTING OF THE CO-AXIAL DUCTS CONNECTED TO THE WAGR HOTBOX USING A DIAMOND WIRE SAW

The equipment consists of (Figure 2)

- A frame which is clamped to the outer coaxial duct attached to the pressure vessel wall
- A hydraulic, motor-driven diamond cable to cut through the duct
- A hydraulic ram to feed the diamond cable onto the duct
- A liquid nitrogen cooling system to prevent the wire from overheating during cutting operations

Cutting operations in the reactor using this equipment have now been successfully completed, with cutting of individual ducts taking approximately four hours with minimal wire wear Higher cutting speeds were achieved but these resulted in damage to the diamond beads

2 2 OPERATIONAL WASTE REMOVAL

Operational waste is the term applied to all ancillary equipment sited in the fuel channels in WAGR, which could be handled using the original fuel handling equipment During the initial stages of decommissioning, after Stage 1 defuelling, all the operational waste was removed, size reduced where necessary, and the ILW items returned to the reactor fuel channels for storage Examples of operational waste are neutron shield plugs, auto control rods and airestor mechanisms

To recover the operational waste from the reactor a custom built grab has been designed This grab has been based on the original refuelling machine handling equipment to minimise development costs

The grab will be deployed from the RDM, 3 te slewing beam hoist via a device which prevents the hoist cables splaying out and keeps them within the confines of the fuel channel

Extensive trials using this equipment have been carried out in AEA Technology's test facilities to assess the reliability, safety features and to optimise waste recovery and handling procedures This grab has also been deployed in the reactor to recover waste items for inspection purposes (Figure 3)

2 3 DISMANTLING OF THE HOT GAS MANIFOLD

The hot gas manifold or 'hotbox' (Figure 2 and 4) is a large flat cylindrical vessel situated near the top of the WAGR pressure vessel It is approximately 5 m in diameter and 1 m high and has a wall thickness varying between 25 mm and 32 mm It is penetrated by the 247 refuelling channels and 6 loop tube channels and is a complex structure of mild and stainless steel It is also insulated internally with stainless steel foil known as 'Refrasil' Its purpose was to distribute the hot coolant gas emerging from the reactor fuel channels to the four heat exchangers

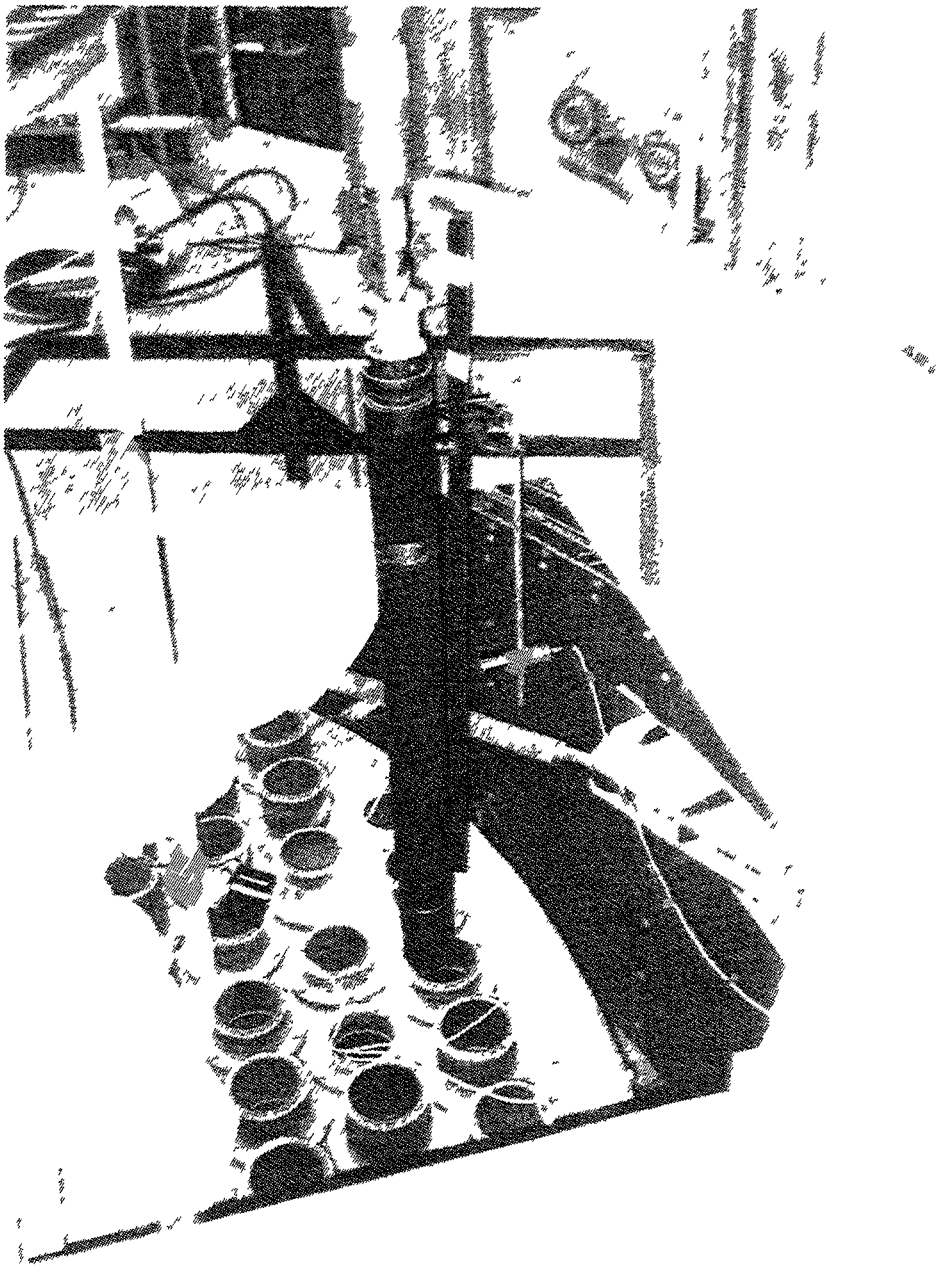

FIGURE 3 OPERATIONAL WASTE INSPECTION

FIGURE 4 HOTBOX SIDE WALL

The WAGR hotbox is an interesting item with regard to its radioactive inventory. It sits above the top core reflector and neutron shield, and, as such, has received little neutron activation. It is, however, the first permanent reactor component in which the coolant gas came into contact with after passing over the operating fuel pins. It is hence the first site for deposition of fission products which were from time to time released from failed fuel pins.

The initial strategy for size reducing the hotbox was to use oxy-propane powder injection cutting, but following trials it was assessed that the quantity of dross produced, and the release of contamination, would have an adverse effect on the subsequent dismantling of the neutron shield. Also the fume generated would spread contamination throughout the reactor which would result in an increased dose uptake to operators performing operational tasks and maintenance operations.

Therefore the proposed method for dismantling the hotbox involves a combination of remote and semi remote operations and aims to minimise the effects on subsequent dismantling operations.

The current strategy for dismantling the hotbox divides the structure into several areas:

- Severing the structure from the neutron shield.
- Top plate dismantling.
- Cutting and removal of the box internals.
- Side wall cutting and removal.
- Bottom plate dismantling.
- Removal of the burst cartridge detection pipework.

For these operations it is proposed to use a combination of a diamond tipped saw (Figure 5), grinders, a small hydraulic shear and a controlled plasma arc cutting system.

Some of this equipment will be deployed by personnel setting up the tools and then withdrawing from the area to operate it semi remotely. Other pieces of equipment will be deployed and operated using a combination of the RDM, manipulator and 3 te slewing beam hoist.

This proposed strategy of combining remote and semi-remote operations offers the most efficient and cost effective solution for dismantling the hotbox whilst minimising the impact on subsequent dismantling tasks.

Trials to optimise cutting parameters and equipment performance and to minimise operator dose uptake are currently in progress.

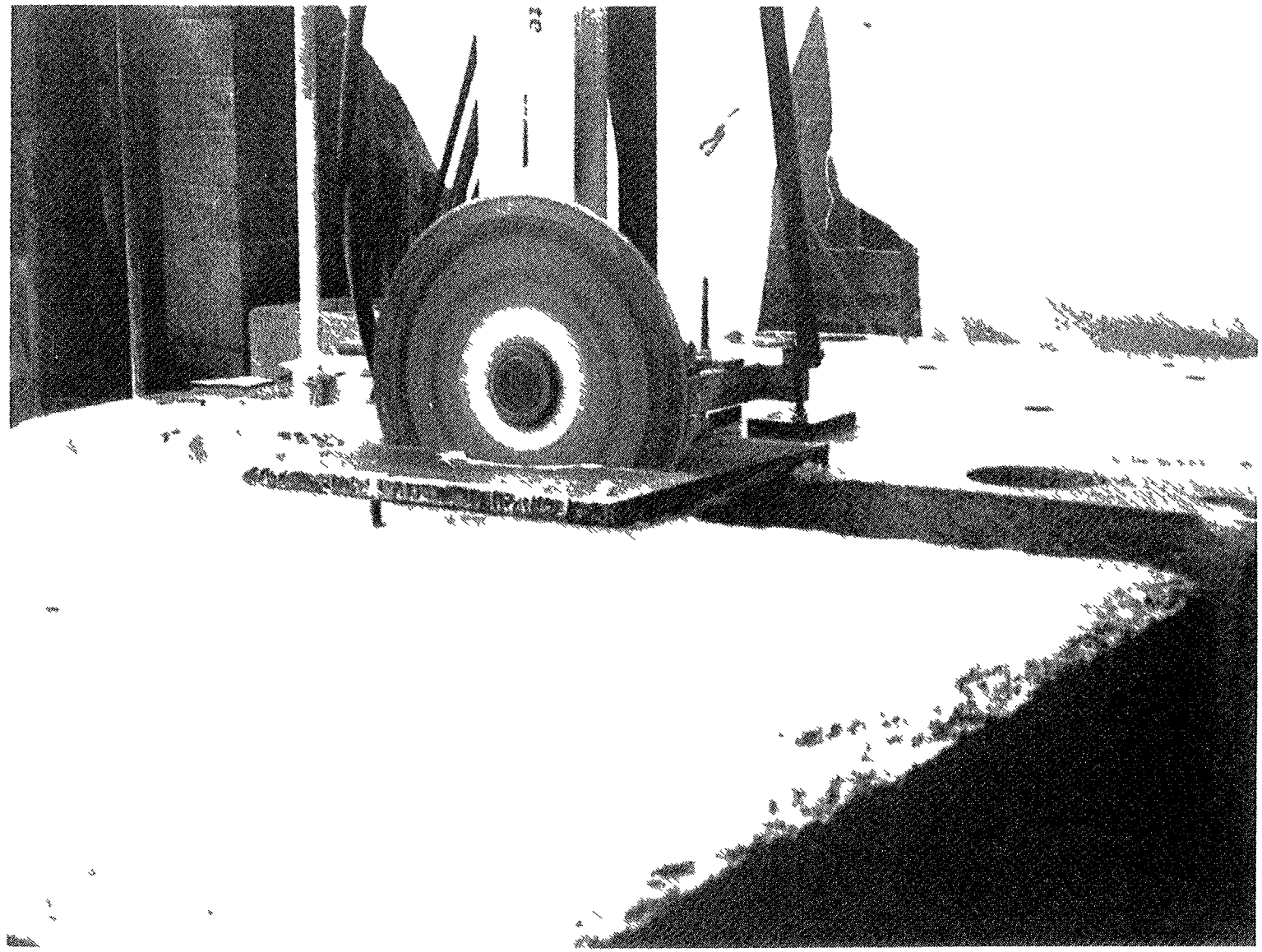

FIGURE 5 DIAMOND TIPPED SAW

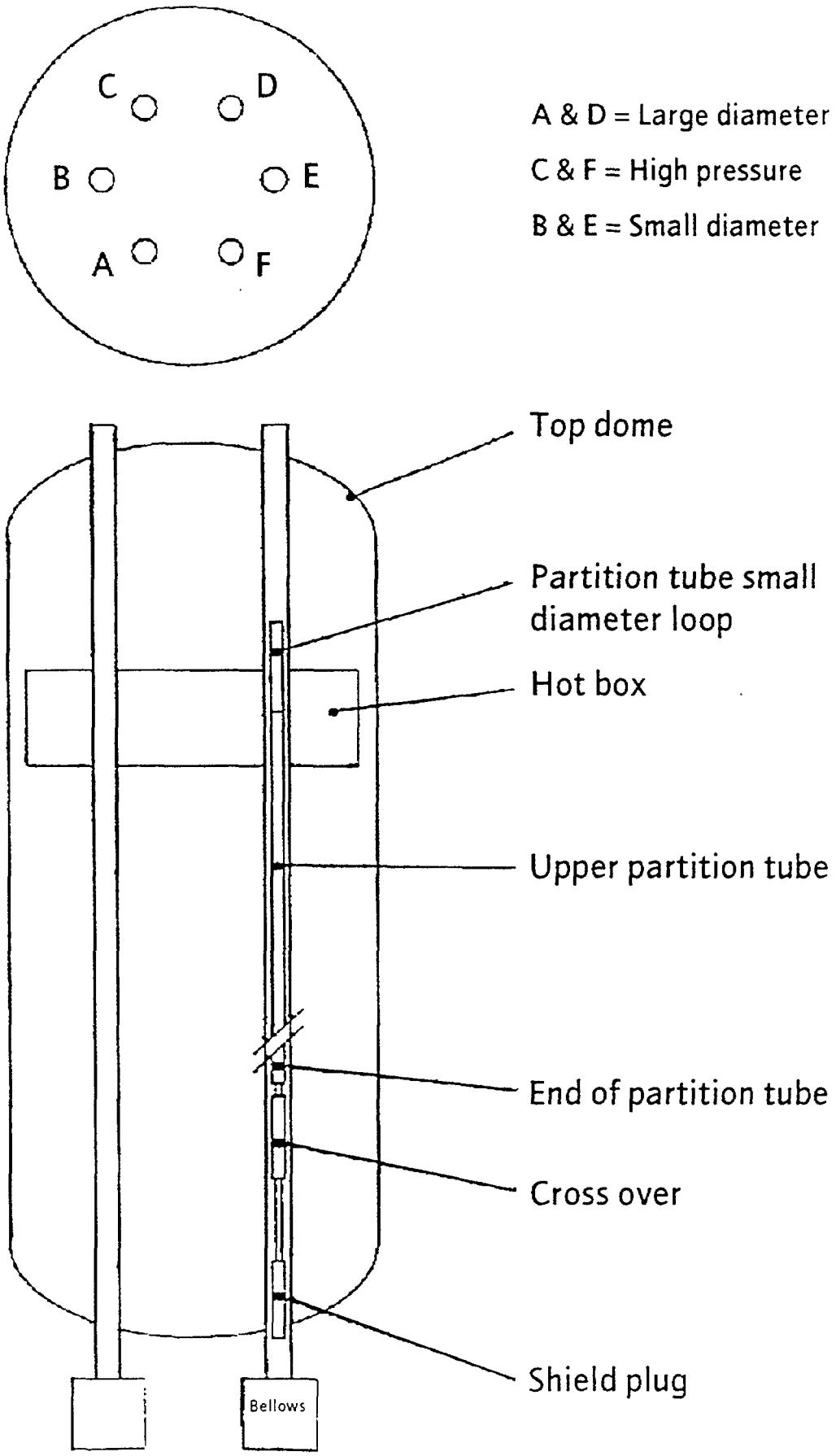

FIGURE 6 POSITION OF LOOP CHANNELS IN REACTOR

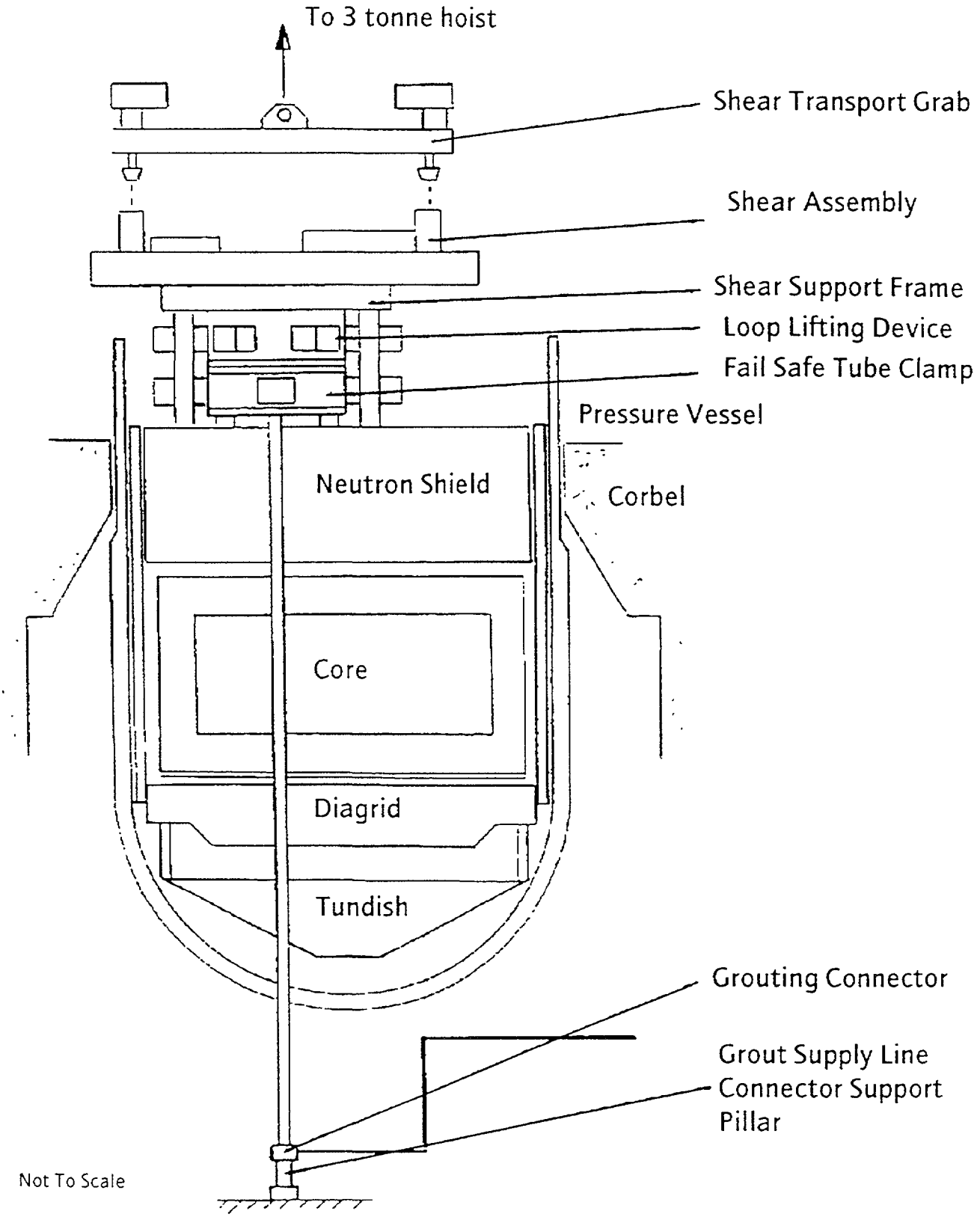

FIGURE 7 LOOP SHEAR SYSTEM

2.4 LOOP TUBE DISMANTLING

Six 'loop' pressure tubes were inserted into the core of WAGR to enable fuel performance experiments to be conducted at the full coolant pressure of the Civil Advanced Gas-Cooled Reactor (CAGR), 600 psi. The loops are constructed in stainless steel and are now highly activated accounting for around 25% of the reactor inventory. The complex design and necessity to avoid spreading highly activated secondary wastes within the reactor structure has necessitated the development of a 'cold', swarfless cutting method to be developed.

A number of different ways were originally considered for removing the loops. The initial methods were to use the reactor refuelling machine or a heavily shielded facility constructed on the reactor cap. The loops (Figure 6) are so constructed that they could be drawn into such a facility and then size reduced ready for disposal. For various reasons, including the lack of a suitable repository for the cut tubes, a decision was taken to delay loop removal until after the RDM had been installed. As a consequence the loops must now be removed using remotely-deployed tooling mounted on the reactor internals.

A removal methodology has now been formulated, which is to raise each loop in its entirety into a purpose-built cutting system mounted on the reactor neutron shield (Figure 7) and then to handle the cut sections using grabs attached to the RDM, 3 te hoist. To undertake this work the equipment below was identified:

- An external tube grab to transport the cut sections of loop from the reactor to the waste packaging route. The grab will be deployed from the RDM hoist.

- An internal tube grab to initially raise the loop section into a stand-alone lifting system mounted on the reactor internals.

- A lifting system which will lift the tubes out of the reactor in discrete lifts and cannot fail in such a way that a tube remains jammed in the device, or that a tube is accidentally released. This system releases the RDM hoist to allow transfer of the cut tube sections.

- A system for cutting the loop tubes, which is capable of cutting the tube while producing minimal secondary waste and reaction forces (figure 8).

A custom-built cutting and handling system which meets the above requirement has the following key features.

The internal and external tube grabs are direct adaptations of the expanding mandrel and scissor-type, plate grabs used in previous decommissioning campaigns. The lifting equipment uses 'lazy cams' to grip the tubes since these provide both a degree of compliance in operation and are fail safe.

The cutting system has been the most difficult system to develop due to the requirement for reliability, minimal maintenance and control of secondary waste production. Extensive trials have been carried out on plasma cutting systems, swaging

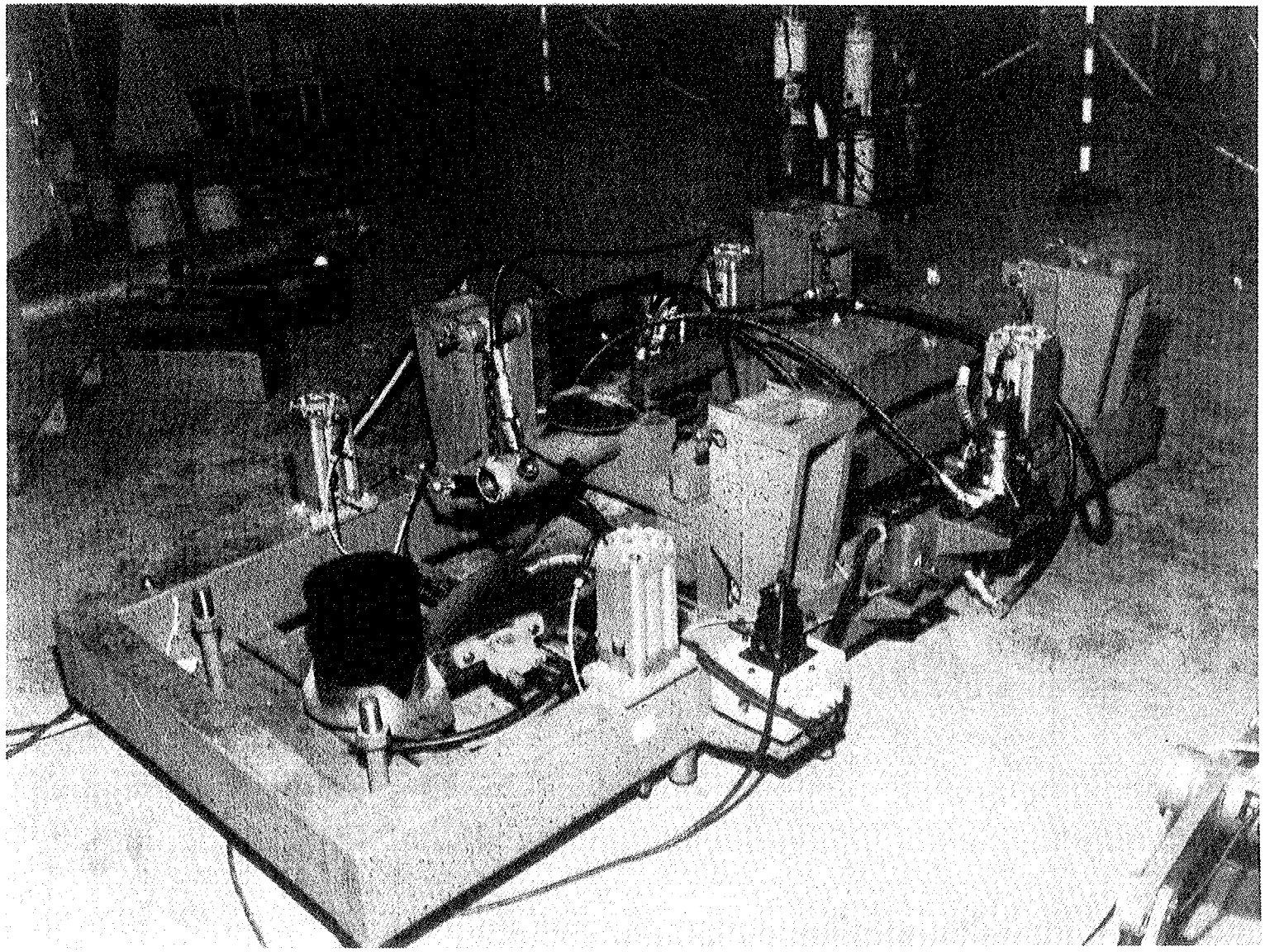

FIGURE 8 LOOP TUBE SHEAR

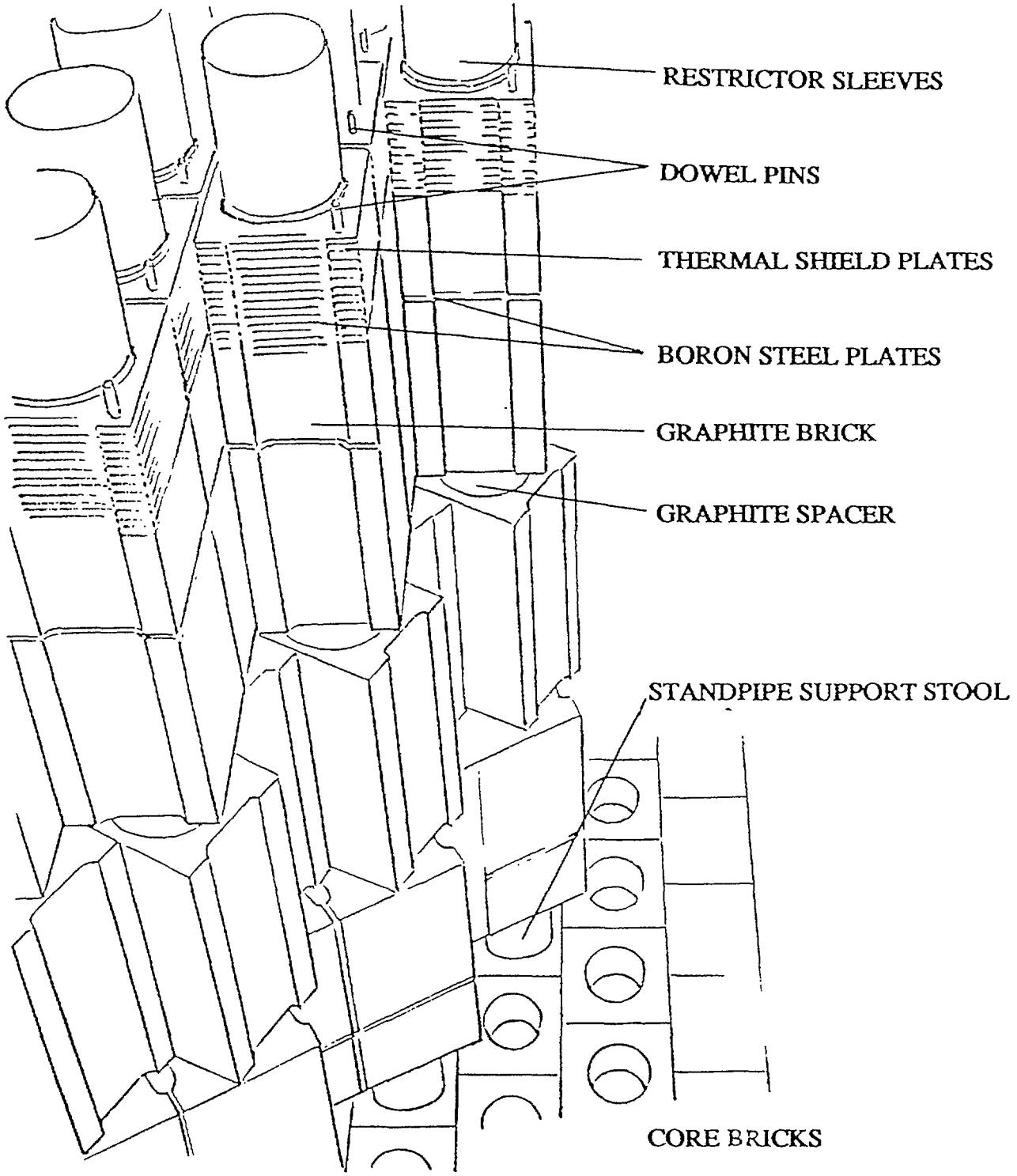

FIGURE 9 INNER NEUTRON SHIELD

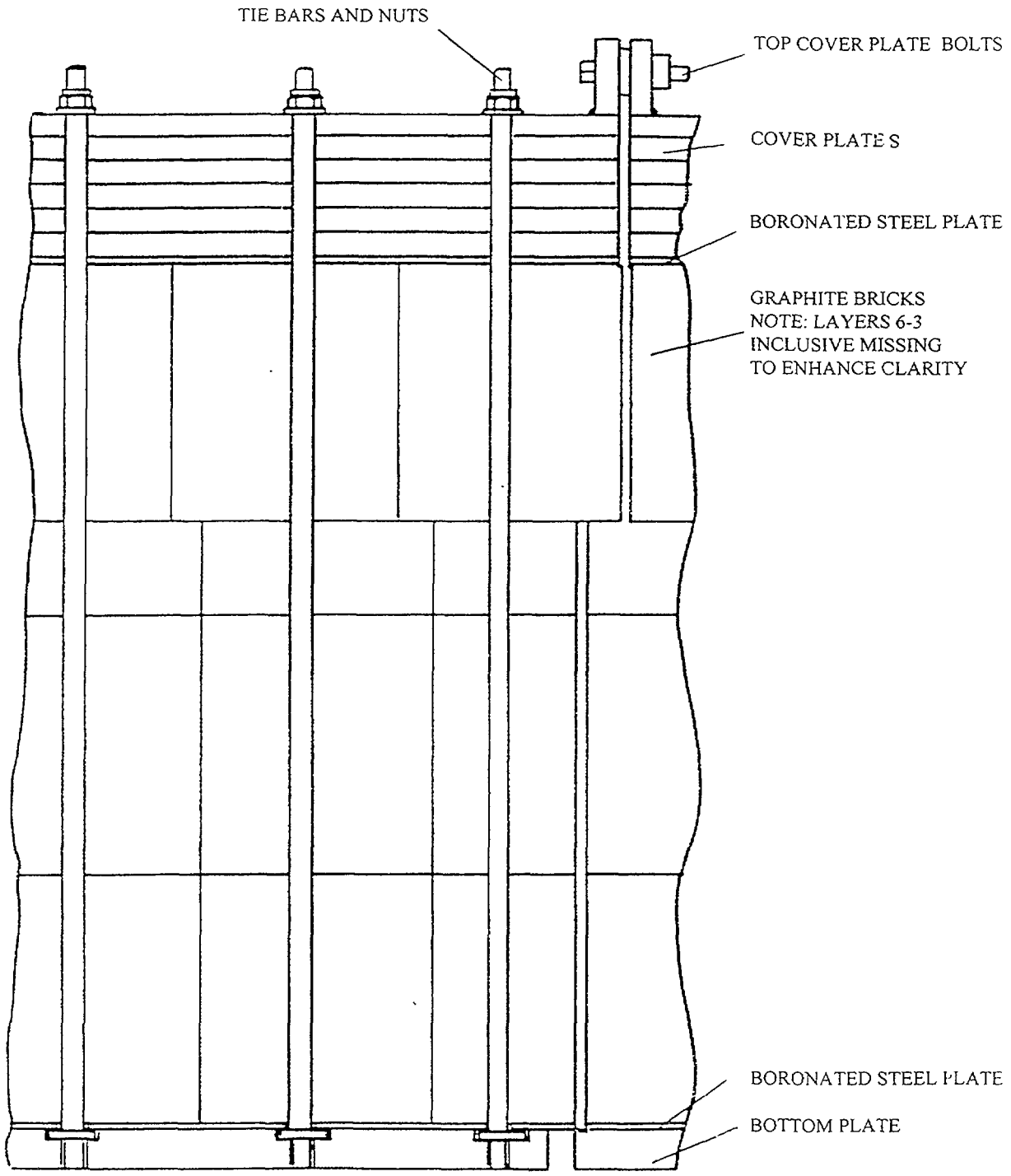

FIGURE 10 OUTER NEUTRON SHIELD

systems and systems for hydraulically shearing the tubes. Of the systems tried only plasma cutting and hydraulic shearing proved reliable at cutting the tube sections

Plasma cutting was eventually rejected on the grounds that the amount of secondary waste produced as fume could recategorise other reactor components from LLW to ILW by deposition of particulates generated during thermal cutting of the highly activated loops Hydraulic shearing could cut the tubes but generated large sideways forces and massive distortion of the tube end section, causing problems with grab placement. Experience suggested that filling of the tubes with grout prior to cutting could minimise end section deformation and reduce overall cutting forces This was found to be the case and is now the adopted approach.

2 5 NEUTRON SHIELD DISMANTLING

To reduce the radiation exposure of operators working on the reactor 'pile cap' and to minimise neutron activation of the reactor components above the reactor core, a neutron shield was installed in WAGR. The neutron shield was installed directly above the core reflector and is divided into two distinct regions known as the inner and outer neutron shield (Figures 9 and 10) Due to the operation of the reactor, both structures have been subjected to a neutron flux and hence trace elements within the materials of construction have been activated

The inner neutron shield is a construction of graphite and steel components with refuelling standpipes running through it Each standpipe has, at the top, the restrictor sleeve components which connect to the hotbox and, at the bottom, the stools which connect it to the reactor core Between these two ends and arranged on the standpipe are three graphite bricks and two graphite spacers, brick layers 1 and 3 having intersecting boron steel plates The top layer (3) also has a layer of 12 x ½" mild steel thermal shield and 1 x ¼" boron steel plates doweled into the top of each brick

The outer neutron shield graphite is made up of seven layers of graphite bricks Layer 7 is the top layer, layer 1 the bottom layer nearest the core. There are 72 bricks in each layer, giving a total of 504 bricks for the assembly. The bricks are located on the 25 4 mm (1") diameter tie bars which pass through holes within the bricks

Layers 7 and 6 bricks are slightly large and overlap those of the inner neutron shield to prevent neutron streaming This means that layers 7 and 6 of the outer neutron shield require removal prior to the removal of layers 2 and 1 of the inner neutron shield The bricks, when assembled, have a minimum of 1 3 mm (0 05") clearance between adjacent bricks in the same layer

The dismantling plan for the inner and outer neutron shield can be sub-divided into 31 tasks and will follow on from dismantling the hotbox Dismantling of the inner and outer neutron shield will be a top down, layer-by-layer approach

To dismantle the neutron shield graphite bricks a number of purpose-built grabs have been developed (Figures 11 and 12) These grabs consist of vacuum grabs which can pick up on a flat surface of a brick Ball grabs which use the internal bore of a brick to lift from There are also a series of grabs which have extended arms which can locate under the base of the bricks to achieve the required lift To ease the removal of the

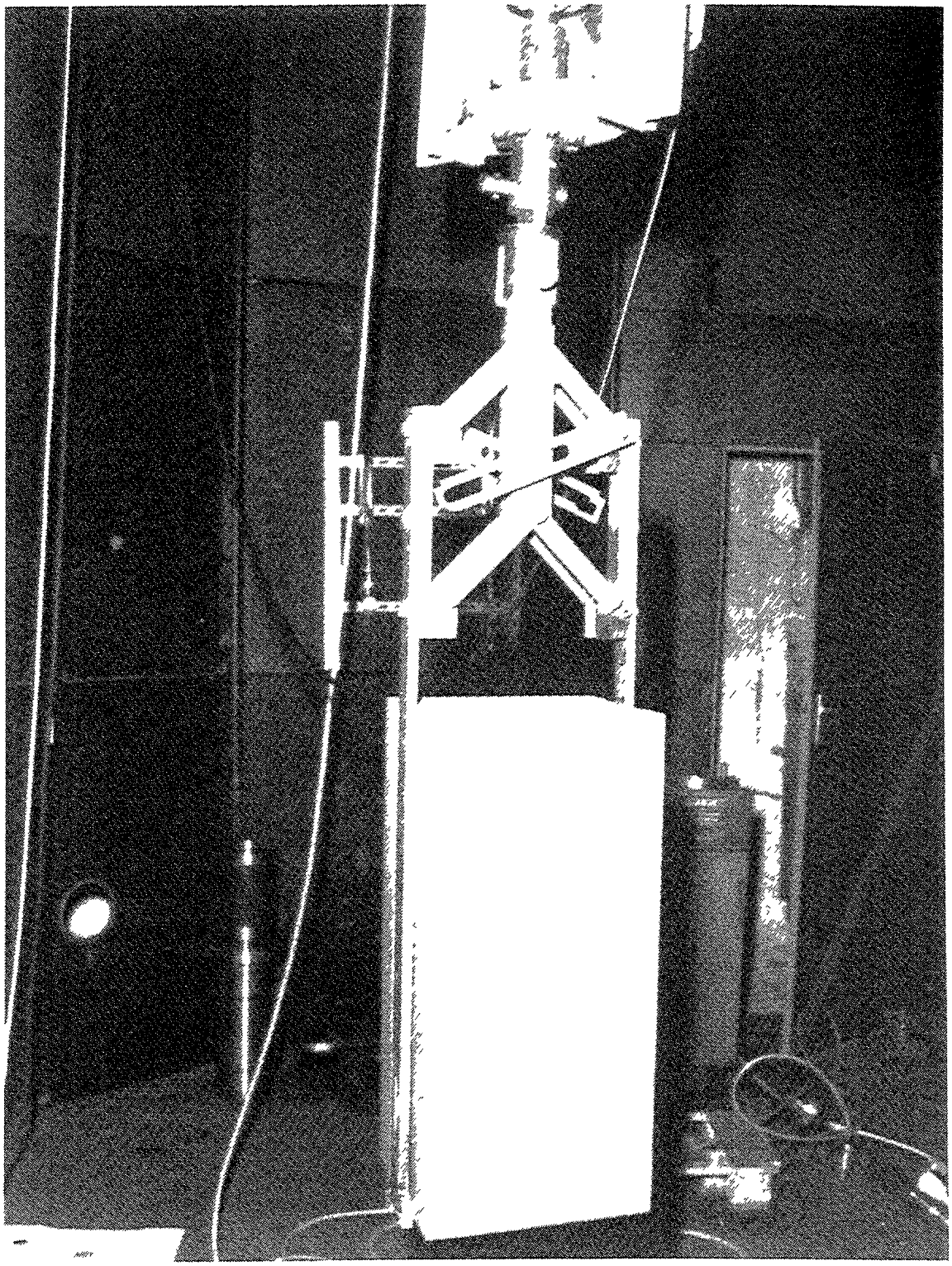

FIGURE 11 GRAPHITE BRICK GRAB

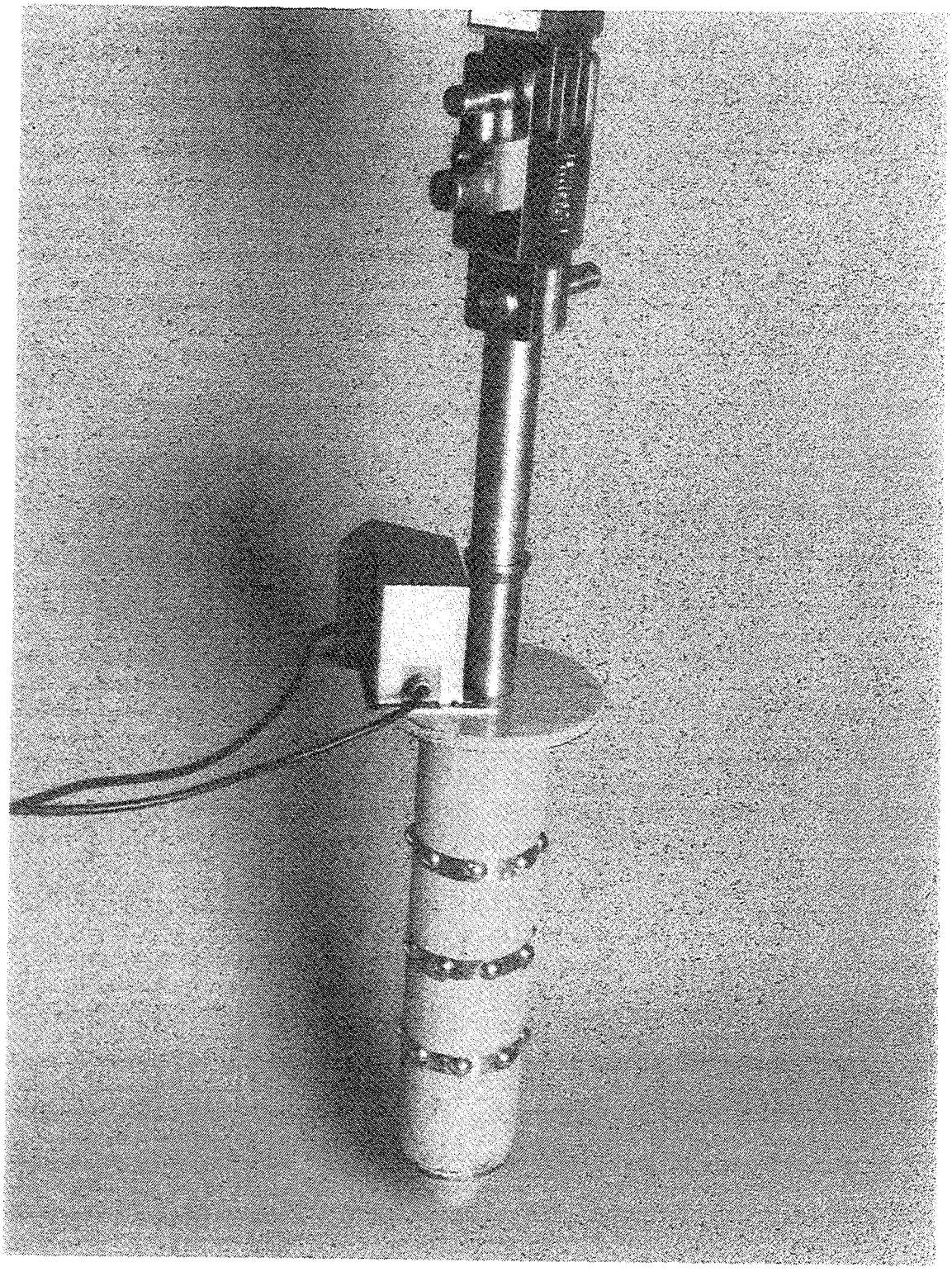

FIGURE 12 GRAPHITE BRICK BALL GRAB

inner neutron shield bricks, staged cutting of the standpipes will be carried out. Similarly, the tie bar s on the outer neutron shield will be cut at stages to ease the removal of graphite bricks in the outer neutron shield. To perform the standpipe cutting an internal plasma cutter has been developed which has an integral grab to remove the waste item after it has been cut. The tie bars will be cut using a grinder deployed by the RDM manipulator.

Using a combination of these grabs and cutting tools, in conjunction with purpose-built pallets for stacking the waste items, the neutron shield can be effectively dismantled.

3.0 Conclusions and Lessons Learned

The conclusions that can be drawn from this work are:

- Modification and adaption of existing technology, where possible, gives the most cost effective solution to some dismantling tasks.

- Detailed risk and safety assessments of the proposed dismantling strategies allow the high risk areas to be reduced or avoided resulting in greater confidence of the final solution.

- 'As-built' information should be obtained and surveys of the area should be carried out wherever possible. This will help minimise modification to equipment. Where accurate information cannot be obtained, a degree of flexibility should be incorporated into the design of dismantling equipment.

- Manual and semi-remote dismantling are quicker and generally cheaper than fully-remote operations and offer a high degree of control over the dismantling task. Where radiation fields permit, these options should be used in preference to remote ones providing ALARP criteria are met.

- The use of a large scale mock-ups for both equipment development and operator training is essential to the success of a decommissioning project. These mock-ups must closely mimic the actual working environment. Although detailed mock-ups are expensive to produce in the first instance, they generally will result in an overall cost saving and in increased confidence in the equipment and methodologies being proposed.

ACKNOWLEDGEMENTS

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DECOMMISSIONING OF NPP A1 — HWGCR TYPE

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Abstract

Prototype nuclear power plant A-1 located at Jaslovské Bohunice, was a HWGCR with channel type reactor KS 150 (refuelling during operation) and capacity of 143 MWe. Single unit has been constructed with reactor hall building containing reactor vessel, heavy water system and equipments for spent fuel handling. Another compartment of main building contents coolant system piping, six steam generators and six turbocompressors, the turbine hall was equipped by three turbines. Unit also shares liquid radwaste treatment and storage buildings and ventilation systems including chimney. It started operation in 1972 and was shutdown in 1977 after primary coolant system integrity accident. In 1979 a final decision was made to decommission this plant.

The absence of waste treatment technologies and repository and not sufficient storage capacity affected the planning and realization of decommissioning for NPP A-1. The decommissioning policy for the first stage is for lack of regulations based on "case by case" strategy. For these reasons and for not existence of Decommissioning Fond until 1995 the preferred decommissioning option is based on differed decommissioning with safe enclosure of confinement.

Transfer of undamaged spent fuel cooled in organic coolant to Russia was finished in 1990. It was necessary to develop new technology for the damaged fuel preparation for transport.

The barriers check - up and dismantling of secondary circuit and cooling towers was performed during 1989/90.

The complex plan for the first phase of A-1 decommissioning - the status with treated operational radwaste, removed contamination and restored treated waste and spent fuel (in case of interruption of transfer to Russia) was developed in 1993 - 1994. Under this project bituminization of all liquid operational radwaste (concentrates) was performed during 1995/96, vitrification of inorganic spent fuel coolant started at 1996, decontamination of spent fuel pool coolant occurs (under AEA Technology support) in 1997 as well as preparation for bituminization of organic spent fuel coolant. The new equipment for spent fuel handling including new storage (semi dry) for spent fuel was projected and should be built up in 1997.

The decontamination and dismantling of auxiliary equipments (radwaste tanks, evaporation plant and original solid storage) should start after the commissioning of conditioning centre and bituminization plant with new evaporation plant in 1998 and finish at 2000. The decontamination and dismantling of original spent fuel storage should finish at 2007/8. Supporting activities to these works started at 1994/95.

Introduction

Prototype nuclear power plant A-1 located at Jaslovské Bohunice, was a HWGCR with channel type reactor KS 150 (refuelling during operation) and capacity of 143 MWe. Single unit has been constructed with reactor hall building containing reactor vessel, heavy water system and equipments for spent fuel handling. Another compartment of main building contents coolant system piping, six steam generators and six turbocompressors, the turbine hall was equipped by three turbines (Fig. 1). Unit also shares liquid radwaste treatment and storage buildings and ventilation systems including chimney. It started operation in 1972 and was shutdown in 1977 after primary coolant system integrity accident. In 1979 a final decision was made to decommission this plant.

Accident Description [1]

The first accident (failure of the closing mechanism of technological channel) happened on January 5, 1976. Fresh fuel assembly (together with the technological plug) ejected to the reactor hall. Coolant (carbon dioxide) flowed out of the reactor short time until the refuelling machine was reconnected with open technological channel and stopped coolant leakage. NPP A1 recommenced the operation after channel repair and inspection in September 1976.

During refuelling , the insufficiently transmissive fuel assembly was charged into the reactor core February 22, 1977. Local overheating of fuel, technological channel and heavy water circuit tube caused to the loss of barriers integrity between heavy water moderator and fuel with cooling gas. The cladding and steam generator tube corrosion under water saturated by carbon dioxide occurred and resulted to the contamination of primary and secondary circuit.

Decommissioning Strategy

The absence of waste treatment technologies and repository and not sufficient storage capacity affected the planning and realization of decommissioning for NPP A-1. The decommissioning policy for the first stage was for lack of regulations based on "case by case" strategy. For these reasons and for not existence of Decommissioning Fond until 1995 the preferred decommissioning option is based on differed decommissioning with safe enclosure of confinement. The updated conceptual plan for A1 decommissioning including will be issued at the end of this year.

The complex plan [2] for the first phase of A-1 decommissioning called "safety status"- the status with removed spent fuel, treated operational radwaste, removed contamination and restored treated waste and spent fuel (in case of interruption of transfer to Russia) was developed in 1993 - 1995. The complex plan for the first phase of A-1 decommissioning contents also the conceptual plan for handling with spent fuel in case of transport to Russia interruption till the year 2038 and decontamination and dismantling of contaminated hot chamber till 2027.

The NRA SR policy on decommissioning will be codified in the Act on Peaceful Uses of Nuclear Energy (Atomic Energy Act) what is currently under preparation. In the relevant part of this Act the decommissioning is defined as safe removal of nuclear facilities from service and reduction of residual radioactivity to a level that permits release of property for another nuclear facility or unrestricted site release and termination of licence.

First Phase of A-1 Decommissioning

After the decision to decommission A1, the main activity focused to the spent fuel transfer to Russia. The transfer of all spent fuel what was possible to manipulate (440 fuel assemblies) was finished in 1990.

The barriers check - up and dismantling of secondary circuit and cooling towers was performed during 1989/90.

Paralelly, the technologies for radwaste treatment and conditioning were developed. Bituminization of all liquid operational radwaste (concentrates) was performed during 1995/96, vitrification of inorganic spent fuel coolant started at 1996, decontamination of spent fuel pond coolant started (under AEA Technology support) in 1996 as well as preparation for bituminization of organic spent fuel coolant.

The decontamination and dismantling of auxiliary equipments (radwaste tanks, the part of the evaporation plant and original solid storage) should start after the commissioning of conditioning centre and bituminization plant with new evaporation plant in 1998 and finish at 2000. The decontamination and dismantling of original spent fuel storage should finish at 2007/8. Supporting activities to these works started at 1994/95.

The Radioactive Inventory

The remaining spent fuel represents the main part of radioactive inventory at NPPA1. Because the fuel cladding was damaged , several events during spent fuel handling caused to the contamination of pond cooling water, reactor hall etc. by spent fuel coolant. Radwaste from spend fuel cooling represents therefore very significant part of inventory (Fig. 2). The increasing of total radwaste activity as a consequence of accidents was not so significant in comparison with the increasing of radwaste volume because of low contaminated secondary circuit (Fig. 3).

The remaining spent fuel

Although during the accident in 1976 some assemblies were overheated [3] and during the accident in 1977some of them were partially influenced because of cladding corrosion due to its contact with heavy water saturated by carbon dioxide, the most of fuel assemblies after their cooling in Dowtherm (biphenyl and biphenyloxid mixture) was transferred to the Russia.

The source of problem of very bad cladding status of all remaining fuel (132 assemblies) is their corrosion during the original way of spent fuel cooling in water solution

1. reactor KS 150
2. steam generator (6 units)
3. turbocompressor (6 units)
4,5) turbine set (3 units)
6) main condenser
7) cooling towers
8)circulation pumps
9)steam cooler
10, 11)condenser with cooler
12-14) condensate pumps, heating
15) vacuum degasificator
16,17) circulation pumps
18-20)feedwater pumps and tank

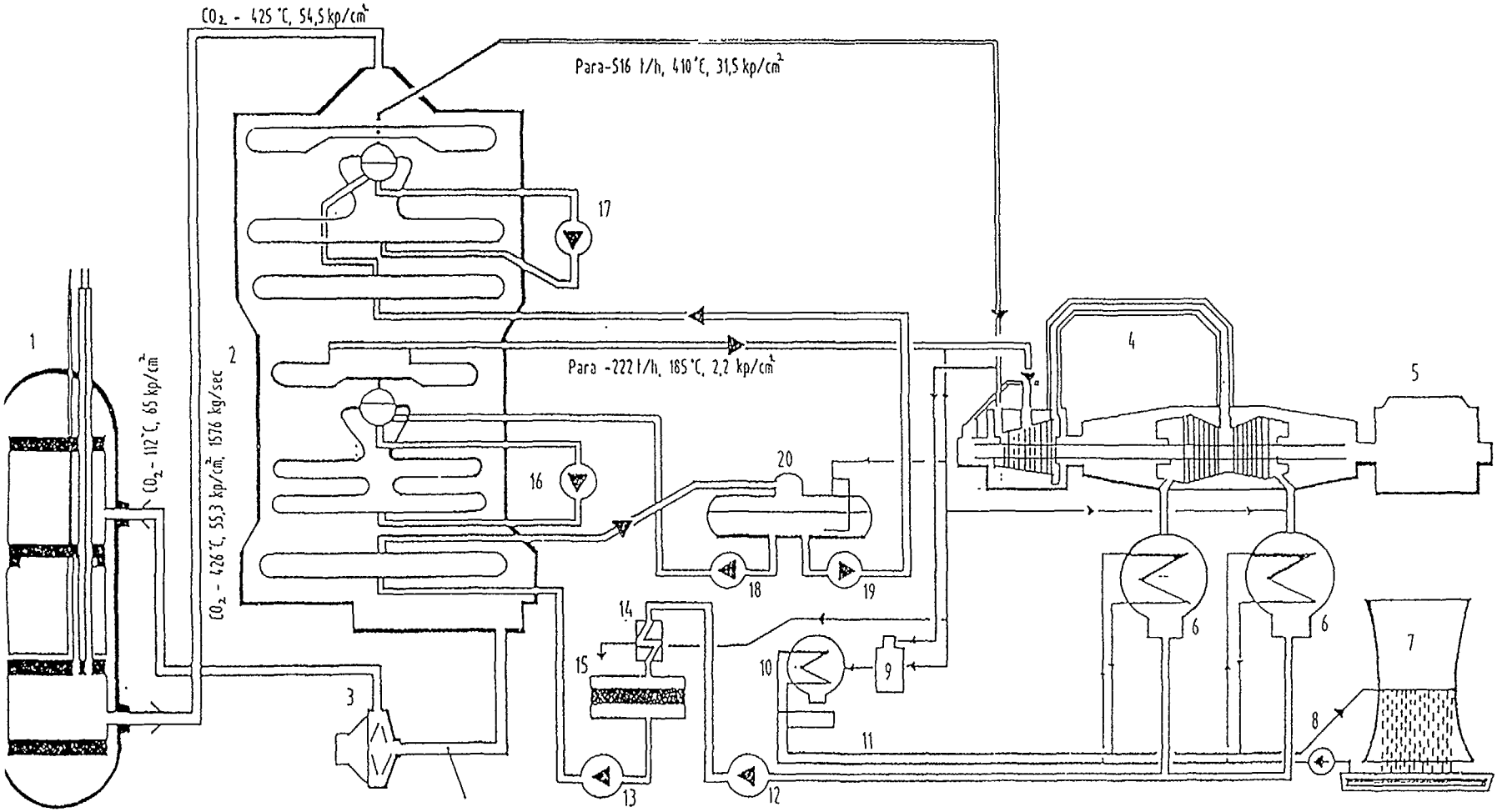

FIG. 1. NPP A1 primary and secondary circuits.

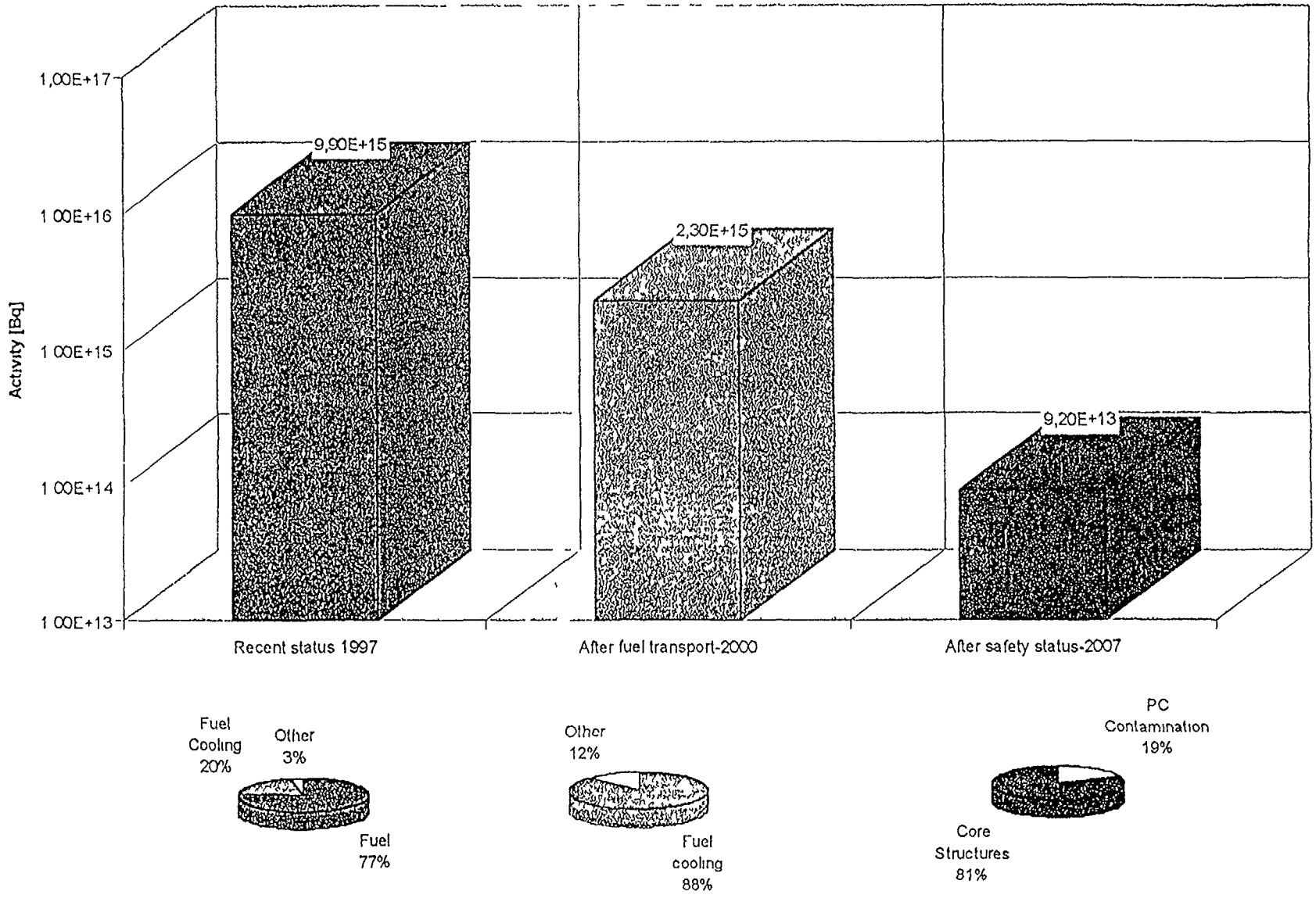

FIG 2 Radioactive inventory during decommissioning phases [5]

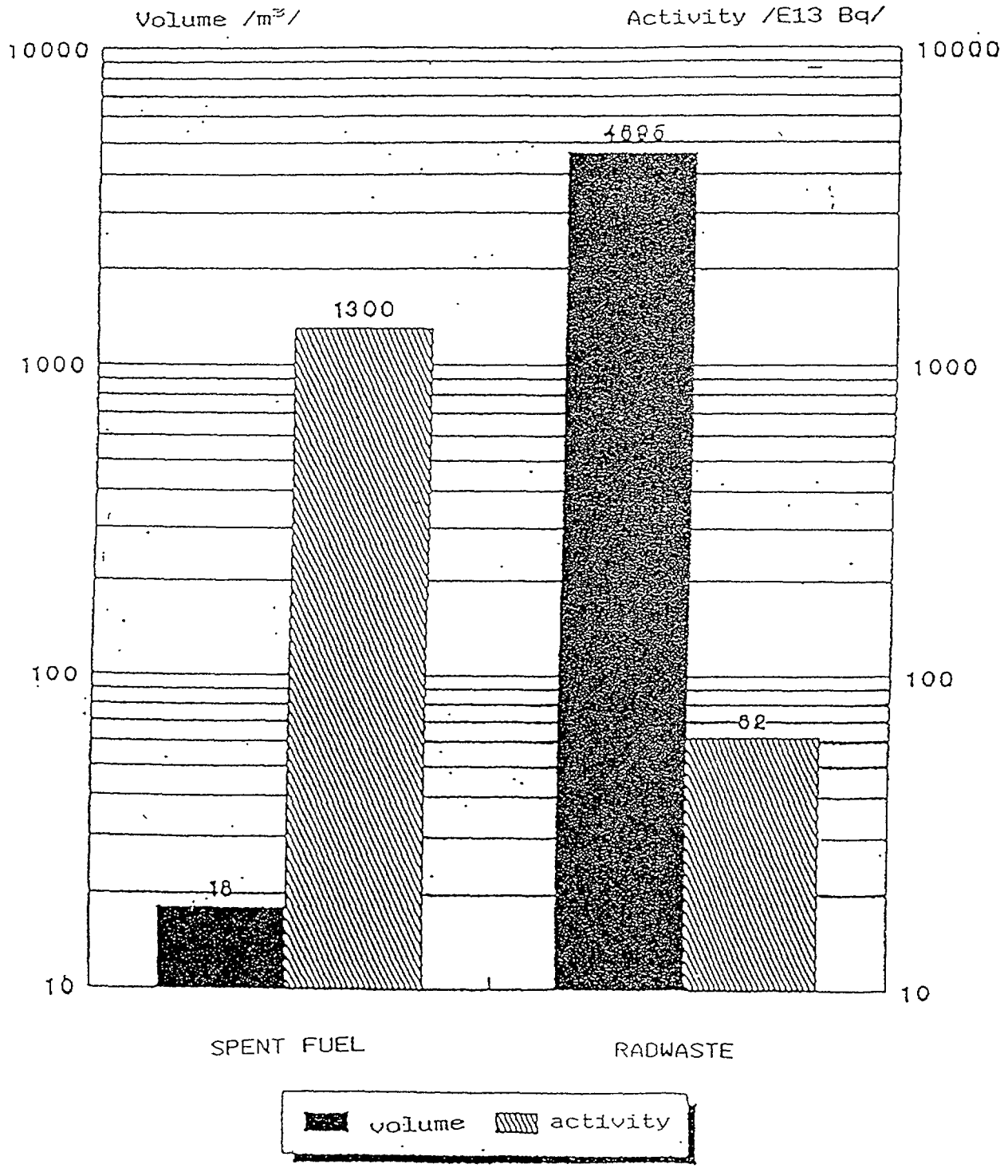

FIG. 3. Diagram of raw volumes and activities arising from the NPP A-1 operation and decommissioning [1].

of potassium bichromate (chrompik) saturated in starting cooling period with carbon dioxide (short term storage). Especially in the case of fuel with cladding damaged during operation in reactor due to higher temperatures and burn-up [4] the cladding is so significantly damaged, that is not possible to manipulate with fuel and to take it off from the cooling containers.

In 1974/5 the coolant was changed, and all fuel assemblies stored/ cooled in Dowtherm were transferred to Russia.

The new technology for spent fuel preparation for transfer represents the drainage of original inorganic coolant with high content of cladding corrosion and fission products from cooling container with fuel assembly and repack fuel assembly and shorted original container to the new hermetic container. The development of this technology started at 1988 and the equipment was in the stage of active tests in 1991.

The accident leading to contamination of reactor hall during these tests occurred, so only 4 fuel assemblies were prepared for transport and were later transferred to Russia. The decontamination of reactor hall was provided in 1993 - 1996 under technical support of AEA Technology, the transfer of fuel was interrupted.

Paralelly to the reactor hall decontamination the improved equipment for damaged spent fuel handling and preparation for transfer based on gravitational drainage was developed with safety features including fire protection (argon atmosphere) . The facility is now under active test, the first transport performed last month.

The new storage (semi dry) for spent fuel was projected and should be built up at the end of 1997. The remaining spent fuel will be replaced from the wet pond with expired time of operation to this storage during the first half of the next year.

Radwaste from the spent fuel cooling

The main portion of such type activity represents chrompik in canisters with spent fuel, what will be drained during fuel preparation to transport to special tank and later vitrified.

Approximately 10 % of the chrompik activity represents water of spend fuel pond. Decontamination of this coolant (2-3 orders activity decreasing) is based on Caesium removal on sorption columns. The cleaning of water will be finished after removal of fuel, chrompik and dowtherm, water will be evaporated and bitumenized.

The chrompik from canisters (the fuel was transfer to Russia in late eighties, activity is significantly lower) will be replaced to the new tanks and later vitrified or cemented.

Dowtherm will be bitumenized and/or incinerated, bitumenization with concentrates from WWER 440s will start this year.

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3) Czechoslovak- Soviet Report, NPPA1 Operation during 1.10.1974 - 30.9.1976, Jaslovské Bohunice 1976
4) HREX, Conference on the result of experiment with A1 fuel assemblies, Piešťany June 1-2, 1976
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VANDELLOS 1 NPP. DISMANTLING AT THE LEVEL 1

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Abstract

Because of the fire in a main turbogenerator in October 1989, the Spanish Ministry of Industry ordered the definitive shutdown of Vandellos 1 NPP.

The tasks allowed to the owner in the Ministerial Order were.

- the reactor defuelling,
- the operation radwaste conditioning.

The size of the reactor core needed to prepare an adequate defuelling plan in order to prevent the potential reactivity oscillations and ensure the refrigeration of the nuclear fuel remaining in the core.

The operation radwastes were divided in four types, according to the conditioning method:

- the low level solid radwaste,
- the irradiated metallic materials,
- the resins and zeolites used for decontaminating the liquid effluents,
- the radwaste stored in three graphite silos.

The low level solid radwastes were stored during operation in drums of 220 litres. Recently they were compacted at a pressure of 40 tones before to be shipped to the ENRESA disposal.

The irradiated metallic materials are, essentially, some parts of the refuelling machine.

For desactivating the liquid effluents, Vandellos 1 used both organic resins and zeolites. The presence of zeolites helps the cementation, but its rough surface makes difficult to flow in the pipes of the cementation plant. 35 m^3 of this mixture have been conditioned into 670 drums of 220 litres.

Vandellos 1 has three silos designed to store the graphite sleeves (reactor fuel support). In the silo number 1 some other radwastes were stored, as low level solid radwastes and two fuel elements. An international request for tenders was made in order to undertake the extraction and conditioning all these radwastes. The project was awarded to the Spanish/French Consortium EQUIPOS NUCLEARES-FRAMATOME.

The achievement of the graphite silos project needed to design specific devices for separating irradiated wires from graphite, and searching and extracting two fuel elements jumbled up with the graphite sleeves.

The spent fuel ponds have been emptied and its internals confined

The radiological protection during dismantling activities took care of α contamination.

The plant safety is always surveyed by the Regulatory Authority (CSN) which required to revise the Technical Specifications several times, according to the nuclear evolution of the site.

1. Introduction

Vandellos 1 is a nuclear power plant owned by HISPANO-FRANCESA DE ENERGIA NUCLEAR, S. A. (HIFRENSA), whose shareholders are :

- Fuerzas Eléctricas de Cataluña 29%
- Electricité de France 25%
- Empresa Nacional Hidroeléctrica del Ribagorzana 23%
- Iberdrola 23%

The construction of the plant started in June 1967. The technology belongs to the European model of natural uranium-graphite-gas cooled reactors, specifically based on a joint project between Electricité de France (EDF) and the Commissariat à l'Energie Atomique (CEA). Vandellos 1 is a replica of the GCR Saint-Laurent-des-Eaux NPPs in France.

The first criticality took place in February 1972, and the two main turbogenerators of 250 MW each were connected to the grid in May 1972.

Because of a fire in the main turbogenerator nº 2, in October 1989, the Spanish Ministry of Industry ordered the definitive shutdown of the station.

The total electrical production was 55,647 GWh, which means a load factor of 72.3 %. The reactor availability factor reached 92.2 %.

2. The dismantling process in Spain

In Spain, the three dismantling phases defined by the IAEA are shared as follows:

– **LEVEL 1**: under the responsibility of the owner.

Duration: The time needed for defuelling the reactor and conditioning the radwaste produced during the operational period.
Funds: The funds are provided by the owner.

– **LEVEL 2**: under the responsibility of ENRESA[1].

Duration: The time elapsed for achieving the decommissioning and dismantling plan submitted to the Spanish Ministry of Industry.
Funds: Tax in the electrical invoice to consumers, collected by ENRESA.

[1]ENRESA (acronym of Empresa Nacional de Residuos, S.A) was created in 1984 to manage all radwaste produced in Spain.

- **WAITING PERIOD**: the nuclear site is under responsibility and surveillance of ENRESA.

  Duration: The length of time is 25 - 30 years. During this period, the disqualified area of the site is returned to the owner, which can use it for any purpose.
  Funds: The same as at level 2.

- **LEVEL 3**: Under the responsibility of ENRESA.

  Funds: The same as at level 2.

Concerning level 1, the tasks allotted to the owner HIFRENSA in a Ministerial Order of 31 July 1990 were:

- to unload the reactor core and ship the spent fuel off the site,
- to condition of the radioactive waste generated in operation.

3. Unloading of the reactor core

The size of the reactor core (a cylindrical pile of graphite of 15.73 metres of diameter and 10.20 metres high, with 3072 loadable channels) needed to prepare an adequate defuelling plan in order to prevent the potential reactivity oscillations and ensure the refrigeration of the nuclear fuel remaining in the core.

It is important to note that the unexpected definitive shutdown didn't allow (the plant) to manage the last fuel load. So, the unloading was undertaken with a very large spectrum of burn-up. Two days before the shutdown, the monthly refuelling campaign has just finished.

The elaboration of the reactor defuelling plan required an analysis of:

- the cooling of the remaining loaded channels,
- the reactor reactivity evolution,
- the pond cooling capacity.

As a first approach, and according to the criteria adopted in the Saint-Laurent-des-Eaux A1 NPP, the following points were considered:

(1) First and foremost, the reactor channels had to be defuelled in an increasing order of cooling flow rate (in the Saint-Laurent-des-Eaux NPP, where the unloading of the reactor core took place very soon after the shutdown, the cooling of the channels remaining loaded was the first priority).
(2) To maintain the triangular symmetry during the defuelling, due to the position of the three sets of neutron chambers around the core. (This criterion had been respected throughout the operation for the refuelling).
(3) To avoid large variations in the average of the nuclear fuel burn-up remaining in the reactor, in order to minimize the reactivity changes, and to seek to keep the reactivity at the lowest possible value.

Two years after the reactor shutdown, the priorities for the defuelling plan were revised. It was more important to avoid the increase of the potential reactivity, since the residual power was a secondary factor.

In order to accelerate the decrease of the potential reactivity, it was necessary to unload the intermediate reactor zone.

The new order of priorities was:

(1) To defuel 12 "supercells" from the intermediate zone.
(2) To maintain the triangular symmetry.
(3) To defuel the reactor in increasing order of the channel cooling flow rate.
(4) To defuel the channels with lower burn-up.

The plan issued from the new priorities was applied in June 1992, 15 months after the start of the reactor defuelling.

The unloading of the reactor started in April 1991 and ended in October 1994. The unloading program was adapted to the availability of the reprocessing plant. So, there was no problems in ponds, neither in the dissolution of magnesium (cladding) nor in the cooling capacity.

4. Conditioning of the radwaste generated in operation

VANDELLÒS 1 NPP
RADWASTE PRODUCTION (LIFETIME: 55 647 157 Mwh)

LOW LEVEL SOLID RADWASTE	– Compactible: 545 drums in operation + 275 in level 1 – Non-compactible: 139 drums in operation + 32 in level 1 – Filters: 20 drums in operation
RESINS + ZEOLITES	– Volume produced: 34.1 m^3. Cemented in 670 drums of 220 l
GRAPHITE SLEEVES	– Crushed graphite: 1000 tons in 240 containers of 6.5 m^3 – Support wires: 2 tons in 74 containers of 0.35 m^3
IRRADIATED MATERIALS	– Various 1.71 m^3 4.44 E+12 Bq (at 31.12.93) – Absorbers 3.62 tons in 19 containers of 0.35 m^3
LIQUID RADIOACTIVE RELEASES	– Global except ^3H: 81.11 Ci in operation + 2.70 Ci in level 1 – ^3H: 3780.00 Ci in operation +504.86 Ci in level 1
GAS RADIOACTIVE RELEASES	– Noble gases: 3.54 E+14 Bq (mainly ^{41}A) in operation 2.00 E+12 Bq (mainly ^{85}Kr) in level 1 – Halogens + Particles (T>8 days): 2.87 E+9 Bq in operation 1.01 E+7 Bq in level 1 – ^3H: 3.30 E+11 Bq in operation
NUCLEAR FUEL REPROCESSING RADWASTE	Reprocessing still in progress

The operating radwastes were divided in four types, according to the conditioning method:

- low level solid radwaste,
- irradiated metallic materials,
- resins and zeolites used for decontaminating the liquid effluents,
- radwaste stored in three graphite silos.

4.1. Low level solid radwaste

During operation, the low level solid radwaste were stored in drums of 220 l.

In 1977 and in 1983, Vandellos 1 was authorized to put the low level solid radwaste into the silo 1. A compacting plant equipped with a press of 40,000 kg of capacity was installed in the site in 1990. The compacting ratio is about 1/3. The conditioned drums are shipped off to radwaste disposal site of ENRESA.

The total production of low level solid radwaste is estimated at:

- Compactable: 545 drums in operation + 275 in level 1
- Non-compactable: 139 drums in operation + 32 in level 1
- Filters: 20 drums in operation

Non-compactable drums hold radwaste like contaminated scrap iron or of similar stiffness. In the drums with filters are immobilized with cement used filters from the contaminated liquid circuits.

4.2. Irradiated metallic materials

The irradiated metallic materials are, essentially, some parts of the refuelling machine. No specific conditioning has been required. They have been handed over to ENRESA 1.71 m^3 of irradiated materials with a total activity of 4.44×10^{12} Bq at 31.12.93.

Furthermore, the neutron absorbers from the reactor core, stored in the silos with the graphite sleeves (see 4.4), are also irradiated metallic materials, but they have been managed in the graphite project.

4.3. Resins and zeolites

For decontaminating the liquid effluents, Vandellos 1 used both organic resins and zeolites. In the conditioning, the presence of zeolites helps the cementation, but its rough surface makes flow difficult in the pipes of the cementation plant. 35 m^3 of this mixture has been conditioned into 670 drums of 220 litres and shipped off to radwaste disposal site of ENRESA.

4.4. Graphite and diverse radwaste

Vandellos 1 has three silos (8.7 m high, 7.2 m wide, 24 m long and 0.75 to 1 m thick concrete walls) designed to store the graphite sleeves (reactor fuel support). In the silo number 1 some other radwastes were stored, as low level solid radwastes and two fuel elements.

VANDELLÒS 1 NPP
RADWASTE INVENTORY OF THE GRAPHITE SILOS

Type of radwaste	SILO 1	SILO 2	SILO 3
GRAPHITE SLEEVES	**36123** 195.1 tons	**107450** 580.2 tons	**43778** 236.4 tons
SOLID GRAPHITE CYLINDERS	**4834** 50.7 tons	**60** 0.7 tons	
ABSORBERS (metallic elements)	**210** 1.5 tons	**282** 2.1 tons	
METALLIC REACTOR BASKETS	**3** 0.048 tons	**1** 0.016 tons	
COMPACTABLE RADWASTE (equivalent drums)	**891** 58.8 tons		
METALLIC DRUMS	**38** 2.5 tons		
FUEL ELEMENTS	**2** 0.02 tons		

A Resolution of the General Direction of Energy dated 22nd March 1991 required HIFRENSA to undertake the extraction and conditioning of the waste stored in the silos. HIFRENSA argued that the graphite sleeves had to be considered as structural waste, but the government resolution established that they had to be treated as operating waste.

HIFRENSA made an international request for tenders, in order to accomplish a fixed price project, according a set of technical specifications. The project was awarded to the Spanish/French Consortium EQUIPOS NUCLEARES-FRAMATOME.

The owner keeps the nuclear responsibility of the project with regard to the Nuclear Safety Council (CSN) and also maintains the relationships with this regulatory organism.

The achievement of the graphite silos project needed to design specific devices for separating irradiated wires from graphite, and searching and extracting two fuel elements jumbled up with the graphite sleeves.

In general terms, the objectives of the graphite project has been to perform the removal, sorting, preliminary packaging and temporary storage of radwaste, while considering:

- the applicable regulations,
- ALARA exposure and radiation protection standards,
- releases limits,
- the aim of achieving the smallest possible number of preliminary packages.
- keeping secondary radwaste down to a minimum.

The project has been developed as a plant modification, and to obtain the authorization to operate the documents submitted to the assessment of the Regulatory body were:

- a technical description of facilities,
- a risk analysis,
- a radiation protection plan,
- a quality assurance programme
- an operating organization.

Three items were deeply assessed by the designer and by the Regulatory Body:

- the ventilation system for ensuring the confinement,
- the radiological measures for environmental control,
- the fire protection system.

Other special radiological measures were implemented to control some specific activities. In particular, during the search and handling of the fuel elements there were:

- Continuous α- monitor of silo air with a removable filter.
- Complementary silo air sampling.
- Continuous β-monitor in working areas.

The sampling filters were measured in the laboratory by γ-spectrometry (searching ^{241}Am, ^{137}Cs and ^{60}Co) and α/β counter (gas counter).

The comprehensive control of personnel has led to systematic nose wipe test, if working in an atmosphere with significant α o β contamination, even with breathing protection.

A mobile containment has been disposed for the extraction of radwaste. This facility protects a pair of holes of the silo against the open air. A telemanipulator arm is fitted into one of those holes and loads a basket fitted in the other one. When full, the basket is hoisted and introduced into a transfer cask.

The cask is transferred to the hot cell in the prepackaging workshop and unloaded. The hot cell is equipped with a set of devices to allow a selection of radwaste and to direct them towards the different conditioning ways.

The graphite sleeves, the most common radwaste, are transferred to a crushing machine with a high intensity magnetic separator. A magnet deflects the stainless steel wires towards a shielded container with a thickness of 180 mm of iron plus at least 60 mm of lead, at least. The graphite falls down to a cubic steel container of 8 mm thick.

The searching and extracting of the fuel elements needed to set up several systems of detection, in order to reduce the possibility of crushing a piece of uranium.

These systems were:

- visual, through a camera,
- Ge-detector, searching the ^{137}Cs inside the silo,
- two NaI-detectors controlling each loaded transfer cask.

– a simulation of the silo operation,
– a test of the magnetism of absorbers caught by the telemanipulator arm. (Absorbers and fuel elements had the same appearance).

The first fuel element was found in good condition and extracted with the current jaw of the arm. The second one was found in a very damaged condition and it had to design a special tool for taking it out.

As densities of graphite and uranium compounds are very distant, a test with a vacuum machine, carried out with small lead bullets and clean graphite, showed the impossibility to extract uranium by this method. As neither uranium nor graphite are magnetic materials, it was not possible to extract them selectively by a magnet.

Therefore, a cryogenic sticking device was designed, and the problem was safely solved. The scrap uranium was taken out without contaminating neither the surrounding graphite nor the silo's atmosphere.

5. Dismantling of the spent fuel ponds

Due to the water of the spent fuel ponds is a shielding and not operating radwaste, to empty them had to be a task of ENRESA in the level 2 phase of the dismantling. Nevertheless, in order to take advantage of operating personnel and the availability of the equipment, HIFRENSA agreed to accept the responsibility of undertaking this activity previously to the hand over.

The spent fuel ponds have been emptied, their internals dismantled and confined and the liner cleaned up, according to a project elaborated by ENRESA and abstracted from the general level 2 dismantling project.

The work started in January 1996 and ended in May 1997. This large extended period is due to the stoppages for improving the hall ventilation and work procedures. The lessons learned from the α contamination incident in Saint-Laurent-des-Eaux were considered in Vandellos 1 from the beginning.

The ponds were the temporary storage of the spent fuel. The separation of the graphite sleeve from the uranium cartridge took place under the water. The sleeves were transferred and stored in the graphite silos. The uranium cartridges, after a cooling period, were shipped off to the reprocessing plant.

The spent fuel ponds of Vandellos 1 are a set of four ponds, separated by hatches, in which there were the internals required to separate sleeves from the uranium cartridge and to store them. A pipe with a lift connected the ponds with a dry hot cell. The total volume of water was 1315 m^3.

The dismantling of the ponds consisted in:

– dismantling of aerial parts not contaminated and storing in an appropriate area,
– emptying of water throughout resins,
– dismantling, decontaminating, cutting and packaging of internals.
– decontamination of the liner, at least up to 4 Bq/cm^2 for $\beta+\gamma$ and 0.4 Bq/cm^2 for α,

- storage of cut materials and packaging inside an empty pond,
- confinement of stored materials with sealed concrete slabs of 15 cm thick.
- deep final cleanup of the hall and collection of all secondary radwaste produced.

6. Evolution of the Technical Specifications

The safety of the plant is always surveyed by the Regulatory Authority, which required several revisions of the Technical Specifications, according to the nuclear evolution of the site.

While the residual power was < 1 MWth, the reactor core air cooled and most fuel was in the core, the requirements were:

- 1 fan in operation + another on standby
- 2 air-conditioning units in operation
- 1 external power supply
- 2 auxiliary boilers
- 2 auxiliary turbo-generators
- 1 amid 4 heat exchangers on standby
- 1 amid 4 turbo-blowers on standby

With a residual power < 30 kWth and less than 50 % of nuclear fuel into the reactor core:

- 1 fan in operation + another on standby
- 2 air-conditioning units in operation
- 1 external power supply
- 1 diesel
- 1 heating boiler

Finally, when the reactor core was unloaded:

- no air circulation
- 1 auxiliary fan to maintain a depression (>2 mbar) inside the reactor containment

7. Radiological health

The radiological protection during dismantling activities took care of alpha contamination. A significant increase of hall contamination, at the end of February 1996, during the dismantling of the ponds, led to control personnel, in spite the breathing protection.

VANDELLÒS 1 NPP. GRAPHITE SILO PROJECT. TOTAL DOSES

	Staff	Contractors	Total
Number of workers	**36**	**101**	**137**
Total doses (mSv·P)	**138.14**	**632.32**	
Max. individual dose (mSv)	**10.92**	**53.26**	

VANDELLÒS 1 NPP. GRAPHITE SILO PROJECT. TOTAL DOSES

	Operation	Maintenance	Total
Number of workers	72	65	137
Total doses (mSv·P)	336.06	434.40	770.46
Max. individual dose (mSv)	14.50	53.26	

VANDELLÒS 1 NPP. DISMANTLING OF THE PONDS. TOTAL DOSES

	Staff	Contractors	Total
Number of workers	4	39	43
Total doses (mSv·P)	6.25	248.27	254.52
Max. individual dose (mSv)	2.74	30.19	

VANDELLÒS 1 NPP. DISMANTLING OF THE PONDS. INTERNAL DOSE CONTROL (INCLUDED ABOVE)

	Staff	Contractors	Total
Number of workers	2	13	15
Total doses (mSv·P)		132	132
Max. individual dose (mSv)		20	

Invited Paper

FORT ST. VRAIN DECOMMISSIONING PROJECT

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Abstract

Public Service Company of Colorado (PSCo), owner of the Fort St. Vrain nuclear generating station, achieved its final decommissioning goal on August 5, 1997 when the Nuclear Regulatory Commission terminated the Part 50 reactor license. PSCo pioneered and completed the world's first successful decommissioning of a commercial nuclear power plant after many years of operation. In August 1989, PSCo decided to permanently shutdown the reactor and proceed with its decommissioning. The decision to proceed with early dismantlement as the appropriate decommissioning method proved wise for all stakeholders – present and future – by mitigating potential environmental impacts and reducing financial risks to company shareholders, customers, employees, neighboring communities and regulators. We believe that PSCo's decommissioning process set an exemplary standard for the world's nuclear industry and provided leadership, innovation, advancement and distinguished contributions to other decommissioning efforts throughout the world.

1. INTRODUCTION

The Fort St. Vrain (FSV) nuclear generating station (Fig. 1) in the United States had been in commercial operation for over a decade prior to its permanent shutdown in 1989. This 330 MWe high temperature gas cooled reactor (HTGR) station was subsequently defuelled and decommissioned with the plant site released for unrestricted use in August 1997. The decommissioning method selected was "early plant dismantlement" rather then the 60 year safe storage option. The decommissioning programme was accomplished within cost and schedule goals while achieving an outstanding personnel safety record and with radiological exposures well below the original estimate.

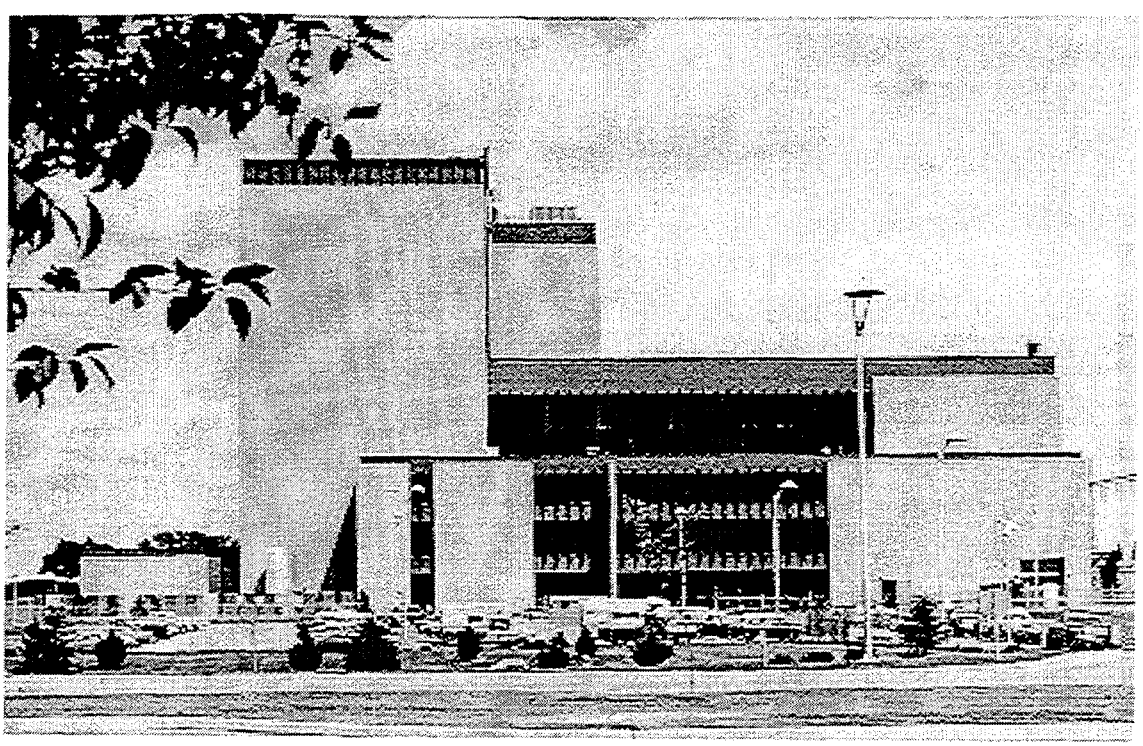
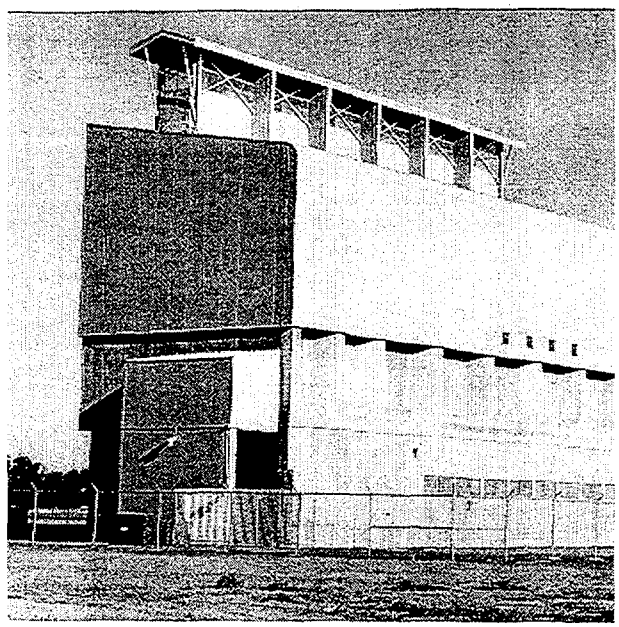


FIG. 1. Fort St. Vrain Nuclear Generation Station.

2. DEFUELLING

The first step in the decommissioning process required disposition of the spent nuclear fuel. PSCo and the U. S. Department of Energy (DOE) had an agreement to ship the fuel to a location in Idaho. The Governor of Idaho objected to this arrangement, thus leaving PSCo with no choice but to place the plant's fuel in an Independent Spent Fuel Storage Installation (ISFSI). This passively cooled, stand-alone facility was licensed by the NRC per 10CFR Part 72 independent from the power reactor license (Fig. 2a). The ISFSI was designed by GEC Alsthom Engineering Systems LTD. The hexagonal graphite fuel elements (31" tall and 14" across the flats) are stored vertically in steel canisters-six fuel elements per canister. Each of the six vaults in the modular dry vault storage system contains 45 storage locations. Each storage location is closed by a removable shield plug allowing for easy access to load and eventually unload the ISFSI. The modular dry vault storage system is cooled by natural circulation. Cool air is drawn in from the outside, passes through each vault, is warmed and rises through the chimney structure for discharge into the environment (Fig. 2b). Since the air is never in contact with the fuel only the outside of the storage containers the air remains free of any contamination. This simple design assisted the company in defueling the reactor to the ISFSI over a six month time frame which was approximately 10 weeks ahead of schedule.



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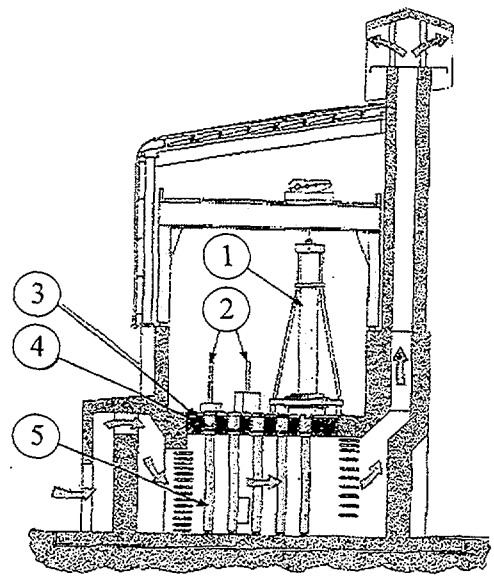


b - ISFSI Schematic

1 - Fuel Handling Machine
2 - Shield Plug Handling Devices
3 - Charge Face Structure
4 - Shield Plugs
5 - Fuel Storage Containers

FIG. 2. Independent Spent Fuel Storage Installation (ISFSI).

3. PRIMARY SYSTEM COMPONENT REMOVAL

Following defueling, the next challenge entailed removing the radioactive components from the Prestressed Concrete Reactor Vessel (PCRV) (Fig. 3), which contained more than 95 percent of the radioactivity at FSV. To accomplish this task, the decommissioning team flooded the PCRV with water to shield the workers from radioactivity. Using two circulating loops of 500 gallons per minute each and a side stream demineralizer, the water was filtered and processed to ensure water cleanliness and clarity.

Next, the 1,320 ton, 15-foot-thick reinforced concrete top head was removed to provide access to the internal PCRV cavity. This step was accomplished by using diamond-wire cutting cables and cutting the top head concrete into 12 pie-shaped wedges. Each of these 110-ton wedges were radioactive due to neutron activation and read approximately 1.5 rem (15mSv) per hour at the bottom of the wedge (Fig. 4). When removed from the PCRV the wedges were placed in a large segmenting tent, cut into three pieces, placed in special steel cans and shipped as low-level waste to Richland, Washington. The nearly one-inch-thick PCRV liner was then cut using oxilance cutting tools, removed and also shipped as low-level waste. At the completion of the top head removal effort, which took approximately nine months, the upper plenum of the reactor was open and PCRV internals were accessible.

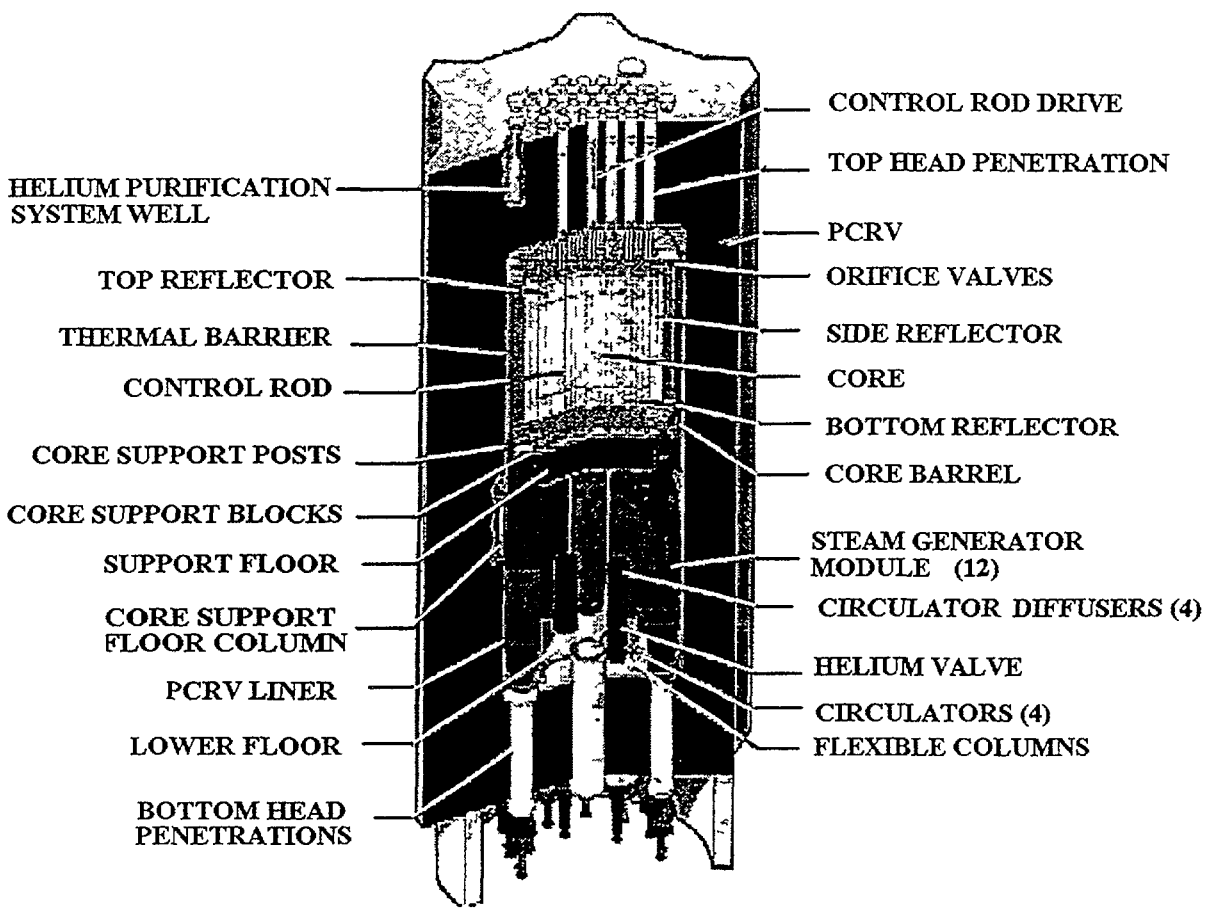


FIG 3 Prestressed Concrete Reactor Vessel (PCRV)

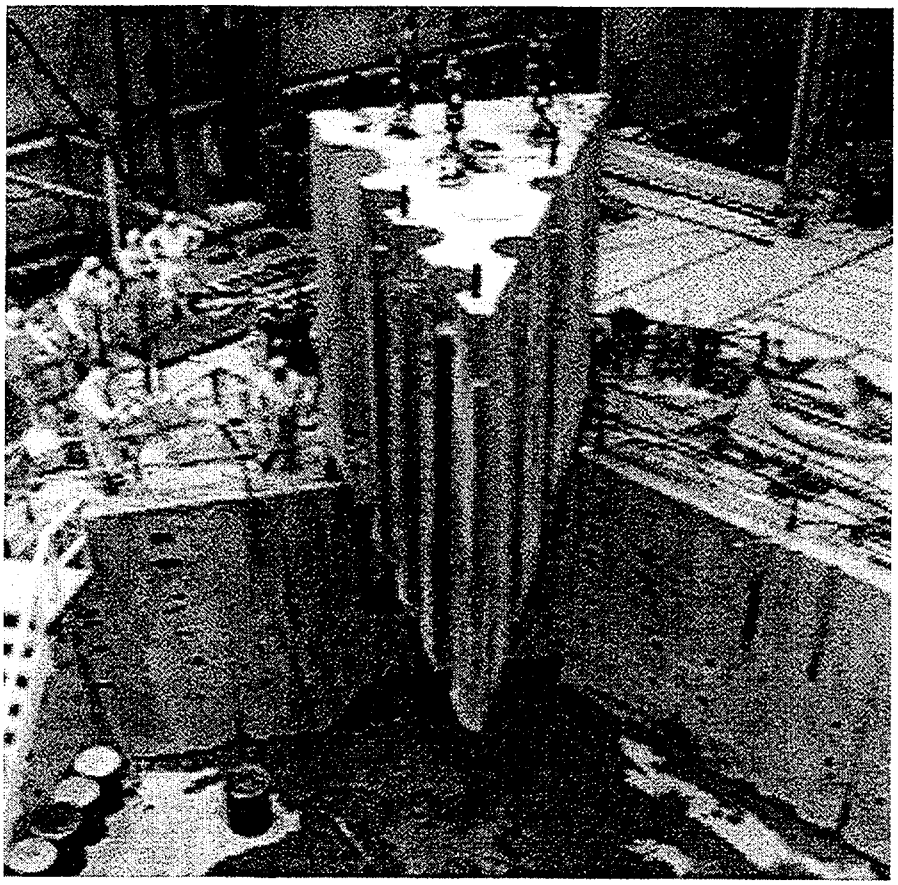


FIG. 4. PCRV Top Head Segment Removal.

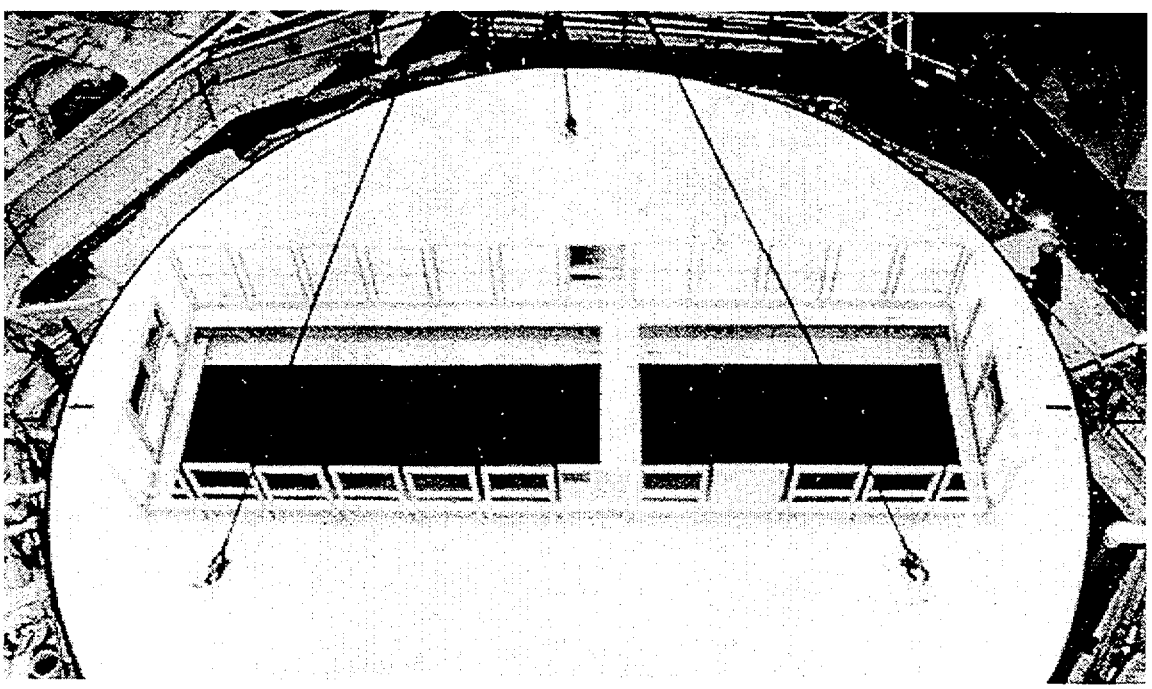
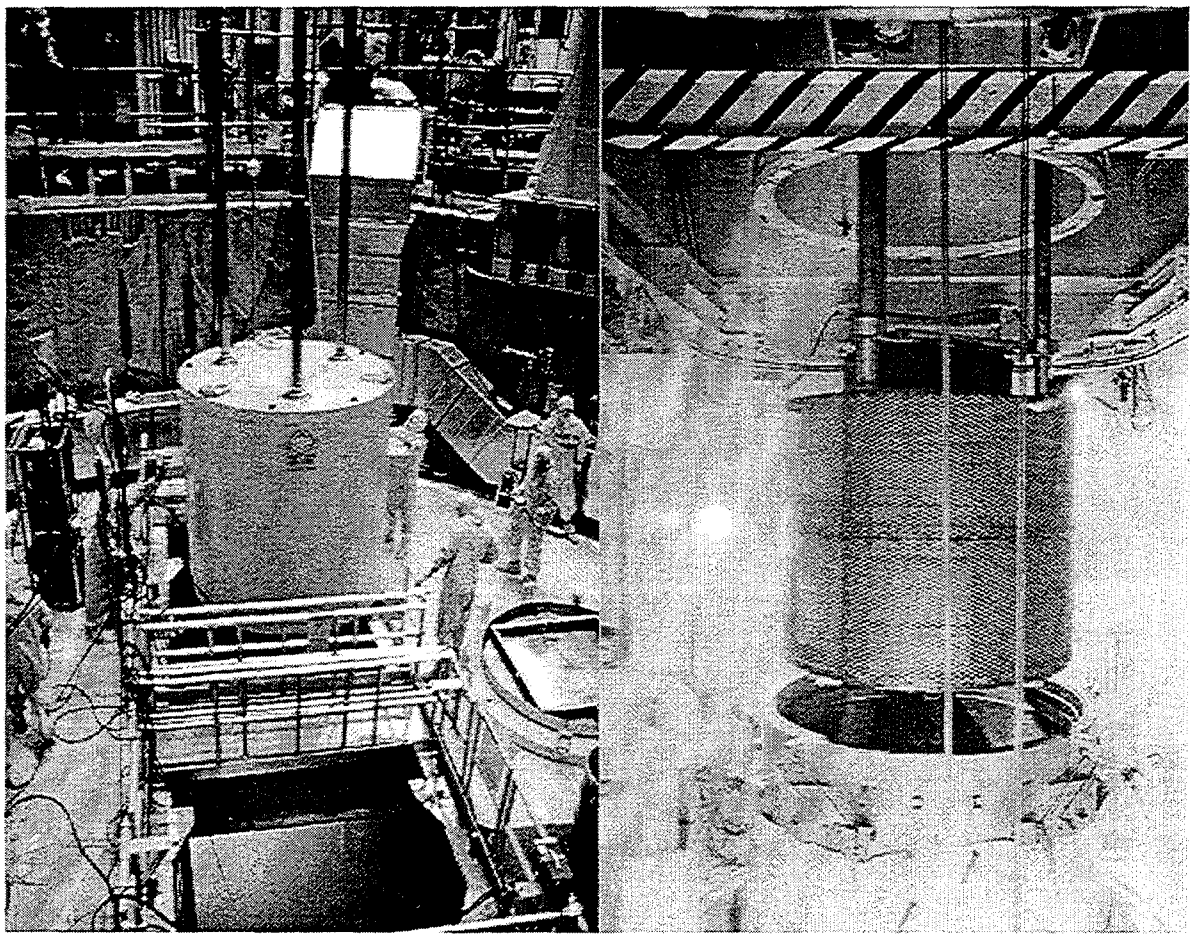


FIG. 5. Component Work Platform.

A. rotating work platform (Fig. 5) was then installed on the PCRV. Operating from this platform, the FSV team removed more than 5,000 graphite components from the upper plenum. These components, some of which read as high as 300 rem (3 Sv) per hour, were removed and placed into a transfer basket that had been lowered into the water. The basket was then drawn into a lead shield bell (Fig. 6a) and was subsequently taken to a hot cell. There the basket was lowered into a shipping cask for shipment as low-level waste (Fig. 6b).

Finally, to expose the lower plenum, the core support floor had to be raised and removed. During construction of the plant the core support floor (CSF) as installed weighed 270-ton. The CSF is a five-foot-thick, 31-foot-diameter, concrete structure encased in a carbon steel liner. Since the CSF was radioactive, steel shielding plates were positioned on top of the floor prior to its removal with a hydraulic jacking system (Fig. 7). Shield water in the PCRV protected underwater divers who entered into the steam generator ducts that went through the core support floor. Once inside these ducts the divers had to cut their way out to access the underside of the CSF. Once under the CSF the divers were used to sever all connections so it could be raised and removed. The CSF, due to the added weight of the shield plates, the attachments on the underside that had been cut and any entrained water



a b

FIG. 6. Component Shield Bell (a) / Basket Arrangement (b).

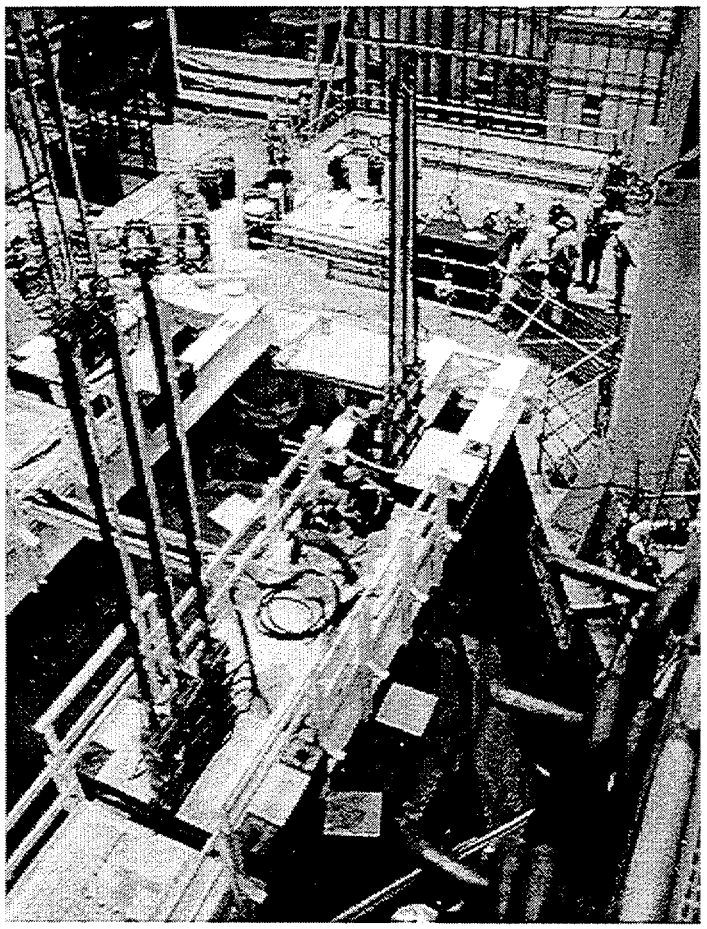


FIG. 7. Core Support Floor Hydraulic Jacking System.

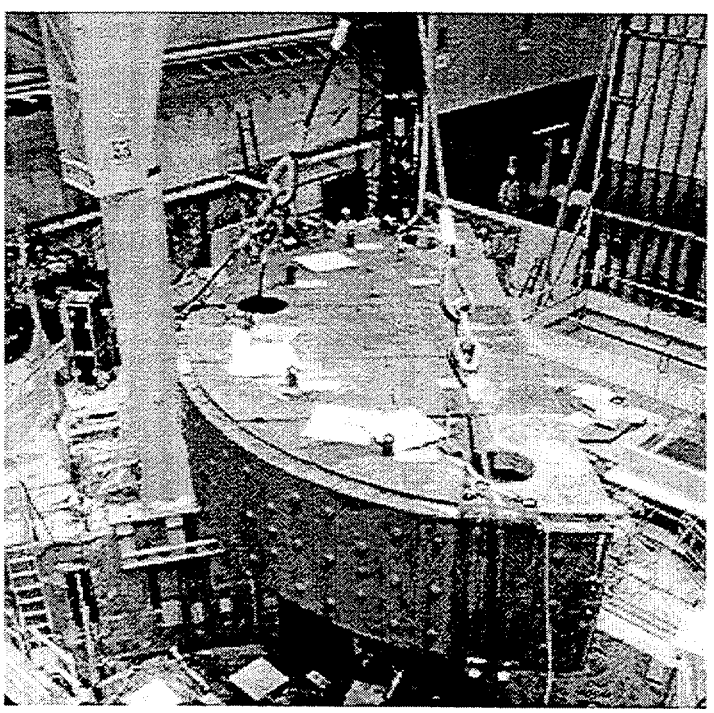


FIG. 8. Core Support Floor

weighed 345 tons during its removal. Again using diamond-wire cutting technology, the CSF was sectioned (Fig. 8), removed from the building and shipped off site as low-level radioactive waste. Once the CSF was extracted, all components within the PCRV were removed, including 12 steam generators and four helium circulators. The photo (Fig. 9) shows a steam generator being lifted out of the PCRV. Then, the activated concrete ringing the inside of the PCRV beltline and lower plenum areas was removed using diamond-wire cutting technology. The upper beltline concrete sidewall blocks were approximately eight-feet wide, 30 inches thick and 42 feet long. The lower plenum concrete blocks were approximately eight feet wide, 27 inches thick and 26 feet long (Fig. 10).

Steps also were required to radioactively decontaminate the entire PCRV cavity (75 feet high by 31 feet in diameter), the reactor building and the support buildings to meet the final acceptance criteria. Decontamination was also required on plant piping and the balance of plant systems and equipment. Depending on their levels of contamination, these systems either were cleaned and left in place or removed for disposal as low-level radioactive waste. During this complex dismantlement, decontamination and system-removal process, 511 shipments containing 71,412 curies of low-level waste and weighing approximately 15 million pounds were made without incident to the low-level radioactive waste burial site. This effort was required to meet the NRC's release criteria of 5 microrem (.05 microsievert) per hour exposure rate above background 1 meter from previously activated surfaces and components, and less than 5000 disintegrations per minute per 100 cm^2 (0.75 becquerel per cm^2) for previously contaminated surfaces and components. The dose from all sources of residual activation had to be less than 10 millirem per year (100 microsievert per year) based on an occupancy factor of 2080 hours per year.

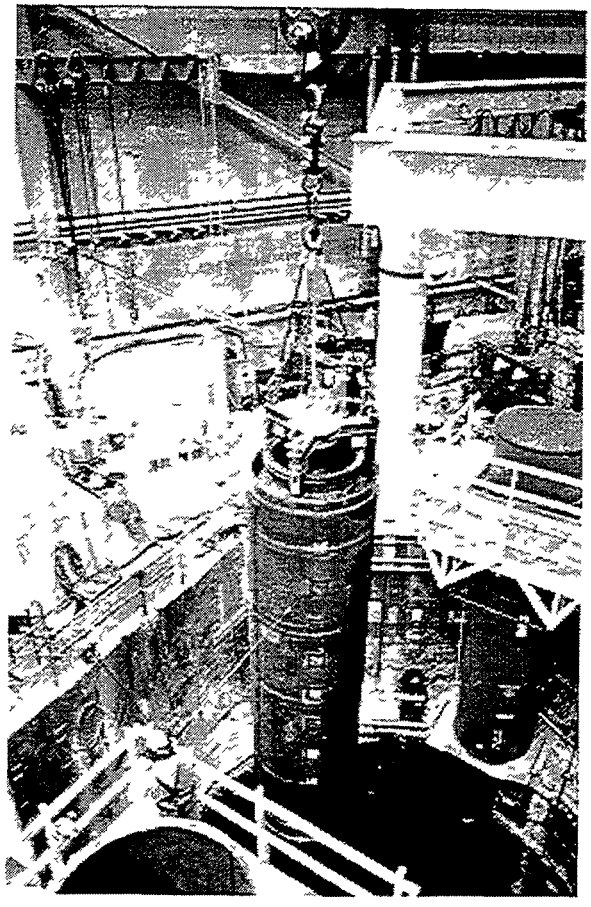


FIG. 9. Steam Generator.

4. FINAL SURVEY

The final radiological survey process began in late 1994 and continued throughout 1995 and into 1996. The final survey plan document, which the NRC must approve, took more than six person-years to complete. The objective of the survey was to allow for unrestricted release of FSV from the NRC license. This survey consisted of characterization, final survey, investigation, and remediation measurements which accounted for the more than 400,000 physical measurements taken throughout the facility. This effort took more than 900 person months over a period of one and a half years to complete. The final survey areas for the entire FSV site were divided in to 10 survey groups. Each area was evaluated to determine its classification as unaffected, suspect affected or non-suspect affected. These classifications determined the survey methodology required for each area. By the end of the survey over 300 areas had been surveyed. To prove the accuracy of the final survey results, the company contracted for an independent verification survey (see Fig. 11 as an example of a survey grid). Also, the NRC conducted its own verification survey. Two specific issues that had to be addressed were hard to detect nuclides and background determination. Hard to detect nuclides identified at FSV were tritium and iron 55. Since these two nuclides can not be easily measured as part of a general survey site specific release criteria were determined for FSV.

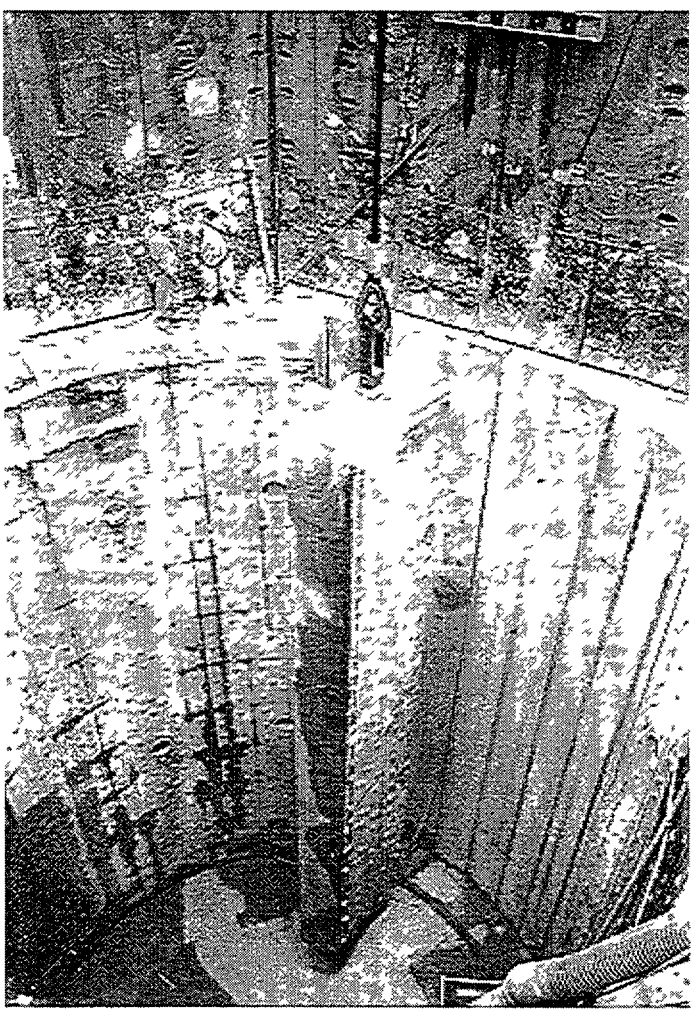


FIG. 10. Lower Plenum Concrete Sections.

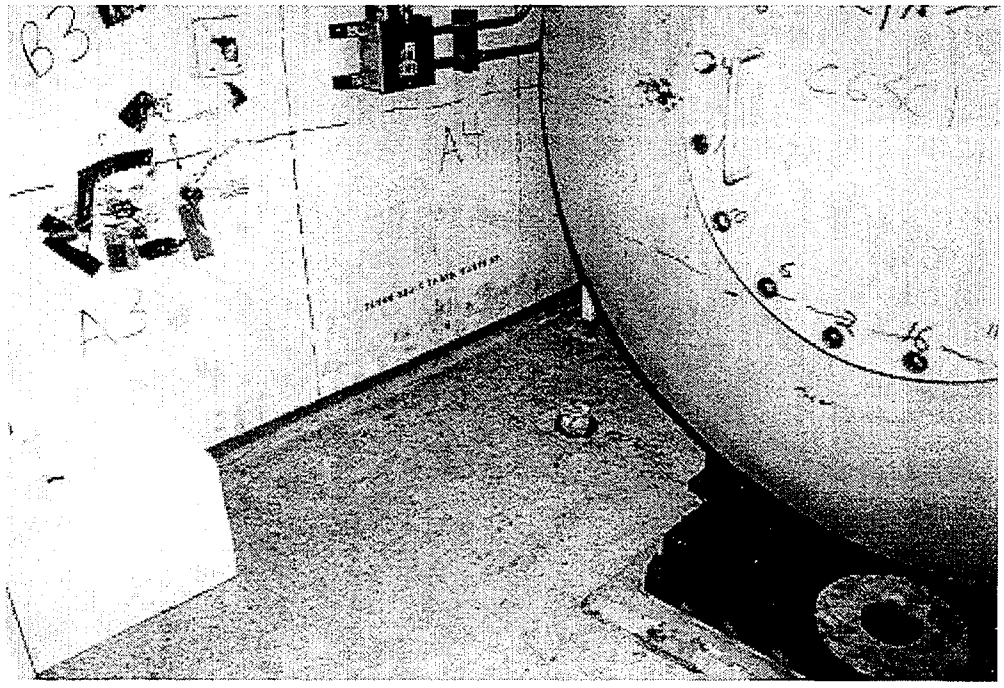


FIG. 11. Grid for Final Survey.

These release criteria were lower than the regulatory numbers to include the effects of the hard to detect nuclide contribution. The background determination was important as well because the release criteria was 0.05 micro Sv per hour above background. Background measurements both onsite and offsite varied between 0.02 micro Sv and 0.35 micro Sv per hour. Permission was obtained from the NRC to use gamma spectroscopy to directly measure exposure rate from licensed material in selected areas. This massive effort cost approximately $20 million and produced a report that covers over 11 feet shelf space to document the measurements and results.

Above all, a decommissioning work environment must be safe. Because of significantly higher radiation levels than encountered during normal plant operations, there was no room for complacency. PSCo is extremely proud of the project's safety record. During the four-year decommissioning period, and despite the fact that personnel spent 340 percent more time in the radiologically controlled areas than originally forecast, the project experienced a total radiation exposure of only 380 person REM (3.80 person sievert). This number, approximately 12 percent under the original radiation exposure estimate, is roughly equivalent to the expected person-REM exposure during one year of operation for a light water reactor. In addition, the FSV personnel contamination rates were only 54 percent and 24 percent of the contamination rates for typical pressurized water reactor and boiling water reactor outages, respectively. Moreover, the project maintained a low (including all subcontractors) lost-workday incident rate of 0.70 per 200,000 person-hours. This rate, when compared to the construction industry average incident rate of 5.5, is exemplary.

ON THE ISSUES OF FUEL STORAGE AND DECOMMISSIONING OF THE HTR-10 TEST REACTOR

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Abstract

The HTR-10 is a helium cooled, graphite moderated test reactor using spherical fuel elements with coated fuel particles. The rated thermal power output is 10 MW. Average burn-up of the spent fuel is designed at 80,000 MWd/tU. Discharged spent fuel elements are stored in lead-steel containers. These spent fuel containers will be stored in a concrete compartment designed in the reactor building with the capacity of storing all the spent fuels of the reactor life span.

Decommissioning of the HTR-10 is subject to the Chinese national standards and codes for research and test reactors. China's national nuclear safety authority has issued the following documents which are applicable to the decommissioning work of HTR-10: Nuclear Safety Regulation HAF 1000-1 *Safety Regulations on the Design of Research Reactors*, Nuclear Safety Regulation HAF 1000-2 *Safety Regulations on the Operation of Research Reactors*, Regulatory Guide HAF 1004 *Decommissioning of Research Reactors and Critical Facilities*.

The paper discusses the following aspects concerning decommissioning the HTR-10: decommissioning considerations in the design and operational stage, facility characteristics and site features, preliminary activities of decommissioning. The key radioactive materials of the HTR-10 are spent fuel elements, core internals, pressure vessels and steam generators, components of fuel discharge system and helium purification system as well as some other auxiliary systems. Site of the HTR-10 is Institute of Nuclear Energy Technology, which is a nuclear research center and should exist for a long term. Designed plant life of HTR-10 is 20 years. Technical approaches of decommissioning the HTR-10 depends on a number of aspects, but mainly on the national strategies.

1. The HTR-10 test reactor

The HTR-10 reactor is a major project in the energy sector of China's national high technology programme. The reactor is a helium-cooled, graphite moderated test unit with a thermal output of 10 MW. It is one of the advanced reactors to be developed for the next century, when China's economy is expected to continually grow rapidly and the country's nuclear programme is planned to be greatly expanded.

The HTR-10 project is to be carried out in two phases. In the first phase, the reactor will be operated with a coolant outlet temperature of 700°C. It is coupled with a steam generator providing steam for a steam turbine cycle which works on electricity/heat co-generation basis. In the second phase, it is planned to raise the reactor coolant outlet temperature to 900°C. A gas turbine cycle will be coupled to the reactor in addition to the steam turbine cycle.

The key design parameters of HTR-10 are listed in Table I, and a cross-section view of the HTR-10 primary circuit is presented in Figure 1.

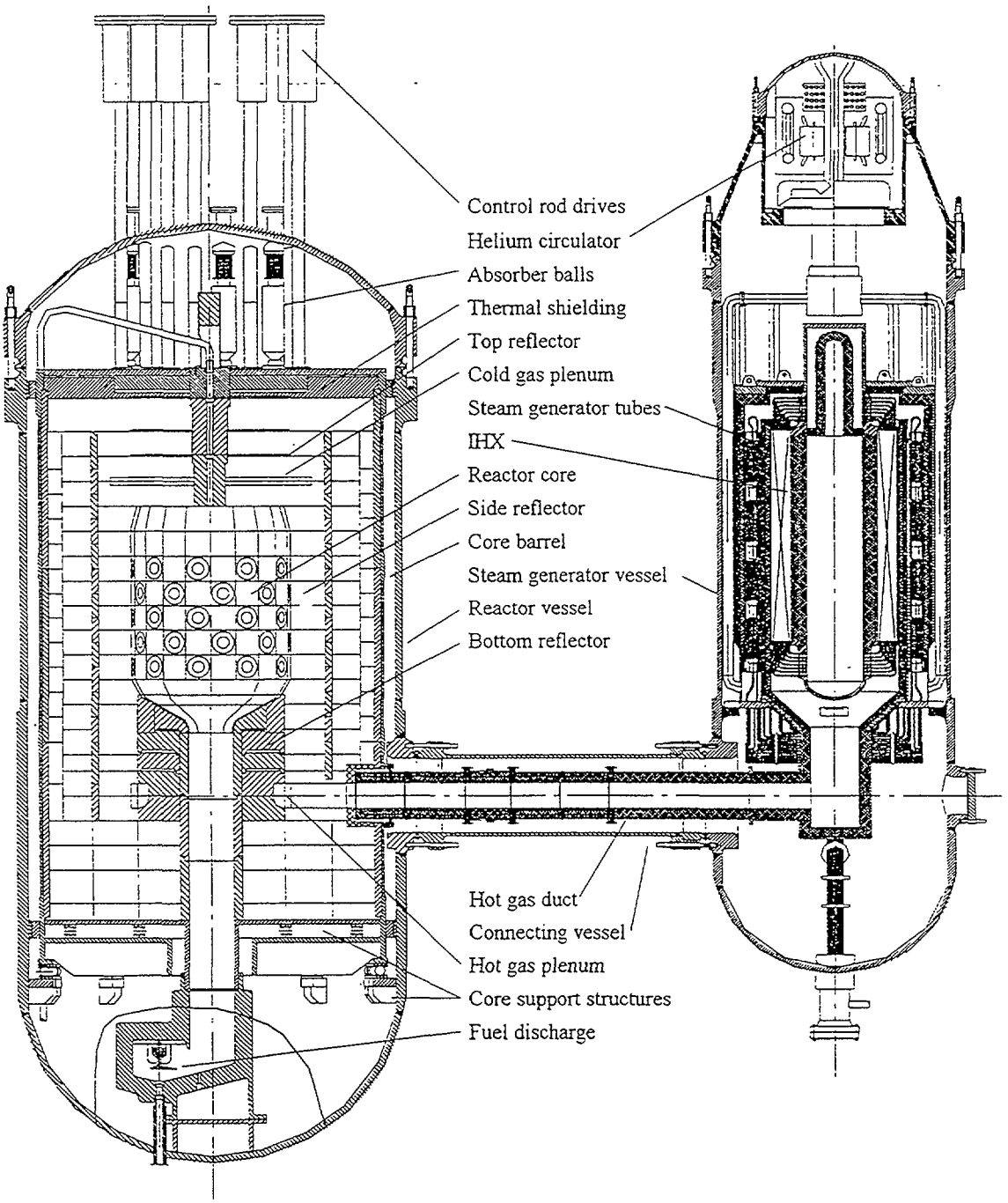


Figure 1 The HTR-10 Reactor and Steam Generator

The HTR-10 test reactor is now being constructed. First criticality is scheduled for 1999. When the test unit has been erected, following aims are expected to be met: acquiring know-how in the design, construction and operation of HTGRs; establishing an irradiation and experimental facility; demonstrating the inherent safety features of modular HTGR; testing electricity/heat co-generation and gas turbine technologies; carrying out R&D work on high temperature process heat application.

Table I Key Design Parameters of HTR-10

Reactor thermal power	10 MW
Primary helium pressure	3.0 MPa
Reactor core diameter	180 cm
Equivalent core height	197 cm
Average helium temperature at reactor outlet	700°C
Average helium temperature at reactor inlet	250°C
Helium mass flow rate at full power	4.3 kg/s
Main steam pressure at steam generator outlet	4.0 MPa
Main steam temperature at steam generator outlet	440°C
Feed water temperature	104°C
Main steam flow rate	3.47 kg/s
Number of control rods in side reflector	10
Number of absorber ball units in side reflector	7
Nuclear fuel / enrichment of fresh fuel elements	UO_2 / 17%
Form of fuel elements / diameter	spherical / 6 cm
Heavy metal loading per fuel element	5 g
Number of fuel elements in equilibrium core	27,000
Average discharge burn-up	80,000 MWd/tU
Fuel loading mode	multi-pass

2. Applicable national regulations

Decommissioning of the HTR-10 is subject to the Chinese national standards and codes for research and test reactors. China's national nuclear safety authority has issued the following documents which are applicable to the decommissioning work of HTR-10: Nuclear Safety Regulation HAF 1000-1 *Safety Regulations on the Design of Research Reactors*, Nuclear Safety Regulation HAF 1000-2 *Safety Regulations on the Operation of Research Reactors*, Regulatory Guide HAF 1004 *Decommissioning of Research Reactors and Critical Facilities*.

In these documents, concerning the decommissioning work, administrative procedures are described, responsibilities of the reactor owner are formulated, principal requirements and possible decommissioning approaches are described, the coverage and execution of the decommissioning programme are described.

3. Site features

The site of HTR-10 is located in the Institute of Nuclear Energy Technology (INET). INET is a national nuclear research center, founded in the early 1960's. It has about 620 employees and more than 20 divisions and laboratories of various disciplines, most of which are related to the research and development of nuclear technology. INET itself is located in the northwestern suburb of Beijing city, about 40 km from the city center.

On the area of INET, there exist already two reactors. One twin-core swimming pool reactor, with a total power of 3.8 MWt, was erected in 1964 and is still in service. The purpose of this reactor is to perform shielding and irradiation tests In 1989, a 5 MWt water cooled reactor was erected in INET, mainly for the demonstration of district heating with nuclear energy. This reactor now is also used for performing tests on nuclear desalination and refrigeration.

The HTR-10 will be the third nuclear reactor facility in INET. Besides these reactors, the institute has a few laboratories which have to do with radioactive sources or other radioactive materials.

4. Facility characteristics

4.1 Reactor core internals

The reactor core internals consist of ceramic structures (graphite and carbon) and metallic structures.

The graphite internal structures, including top, side and bottom reflectors, are large graphite bricks which form a cavity for the pebble bed of fuel elements. All graphite reflector bricks are segmental. The top reflector is composed of three layers of graphite bricks. The side and bottom reflectors are composed of thirteen and five layers of graphite bricks, respectively. There are twenty segmental graphite bricks in each layer of the top and side reflector, while in each layer of the bottom reflector there are ten graphite bricks. In each layer of side graphite reflector there are twenty cold helium channels with a diameter of 80 mm, ten control rod guide holes with a diameter of 130 mm, seven slotted holes with the dimension of 160 x 60 mm for the small absorber ball system and three holes for irradiation and neutron sources purposes. The radial thickness of graphite bricks of the side reflector is 77.5 cm. The overall weight of the graphite internals is about 65 t.

The graphite reflectors are surrounded by boronated carbon bricks. These carbon bricks provide thermal and neutron shielding to the metallic internal structures and the reactor pressure vessel. The total weight of carbon bricks is about 46.5 t.

The metallic internal structures are composed of the core vessel, upper and lower support plates, holddown plates and a positioning plate. They provide support and housing to the ceramic internal structures and also transmit the mechanical loads coming from the fuel pebble bed and the ceramic internals to the reactor pressure vessel. 15CrMoR is selected as the material for most metallic core internal structures. The total weight of the metallic core internals is about 72.5 t.

4.2 Reactor pressure vessel

The reactor pressure vessel (RPV) consists of the upper closure head, the cylindrical vessel and the lower closure head. In terms of form, both closure heads are sections of semi-spheres. The lower head is welded to the cylindrical part, while the upper head is connected to the cylindrical vessel through flanges. The cylindrical part has two sections with the lower section thicker than the upper one. The plate and forging materials for the RPV are respectively SA516-70 and 15MnNi. On the upper head, control rod drives and absorber ball releasing mechanisms are mounted through tube nozzles. The RPV is supported to the reactor cavity concrete through four brackets.

The cylindrical vessel has an inner diameter of 4.2 m, thicknesses of the two cylindrical sections are respectively 80 mm and 120 mm. The overall height of the RPV is 12.58 m. The total weight of the RPV is approximately 167 t. At the end of the reactor life time, the fast neutron fluence of the RPV is estimated to be 1.26×10^{17} / cm^2.

4.3 Steam generator and its pressure vessel

The steam generator mainly consists of the steam generator pressure vessel (SGPV), heat exchanging tube bundle modules and internals. Thirty helical tube bundle modules are arranged in a circle between two insulation barrels inside the SGPV. Inside the inner barrel, the installation of an intermediate heat exchanger is planned for the second phase of the project. Each tube module consists of a helical heat exchanging tube, a central pipe, an outer case, fix and support structures as well as leak preventers. The tube dimension varies at different tube sections to improve hydrodynamic stability of steam/water two phase flow. The tube material is $2^1/_4$Cr1Mo. The total heat transfer area is about 55 m^2. Total weight of the tubes is about 2 t. The weight of the steam generator internals are 16 t.

The SGPV contains the steam generator, the intermediate heat exchanger and the helium blower. The same materials are used for the SGPV as for the RPV. The SGPV has an inner diameter of 2.5m and is approximately 11.37m in height. The weight of the SGPV is about 84.6 t. The SGPV and the RPV are connected by a so-called connecting vessel (CV), in which the hot gas duct is designed. The CV and the hot gas duct form the primary gas passages between the reactor and the steam generator.

4.4 Fuel handling system

The spherical fuel elements are successively fed to and removed from the reactor core by the fuel handling system. After passing through the core, the fuel elements are removed from the fuel element discharge tube via a pulse pneumatic single-exit gate, which is designed inside the reactor pressure vessel. With the aid of the separator, fuel elements which are not geometrically satisfactory drop into a fragments container, while the intact fuel elements are transported to the elevator set. From there, the fuel elements are either discharged to spent fuel containers when they have reached the discharge burn-up, or returned to the reactor core when they have not reached the discharge burn-up.

4.5 Helium purification system

The function of the helium purification system is to continuously remove, from the primary helium, dusts and particles, gaseous impurities including fission products. There is one purification train, which mainly consists of a filter, a molecular sieve trap, a copper oxide fixed bed, a cold charcoal trap and a compressor. The purification train must be regenerated after about 2,000 operation hours.

Dusts and particles are absorbed in the filter. Hydrogen and carbon monoxide are respectively converted to water vapor and carbon dioxide in the copper oxide fixed bed. The resultant water vapor and carbon dioxide are absorbed by the molecular sieve trap. Nobel gases, methane, oxygen and nitrogen are absorbed by the cold charcoal trap.

5. Fuel storage

The 6 cm spherical fuel elements of HTR-10 are made of TRISO type coated particles (CP) and graphite matrix. One CP consists of a UO_2 kernel with a diameter of 0.5 mm, which is successively coated with layers of low density pyrolytical carbon, inner high density isotropic pyrolytical carbon, silicon carbide and outer high density isotropic pyrolytical carbon, with thicknesses of respectively 90, 40, 35 and 40 μm. About 8,300 coated particles are dispersed in the graphite matrix, which is 5 cm in diameter, to form the fuel zone of a fuel

element. This fuel zone is surrounded by a 0.5 cm thick fuel-free graphite shell. As has been stated, each fresh fuel element contains 5 g uranium with an ^{235}U enrichment of 17%.

The fuel elements are designed to pass through the reactor core five times in average to reach an average burn-up of 80,000 MWd/tU after about 1080 effective full power days. According to calculation, the discharged spent fuel elements have a burn-up distribution between 74,300 and 87,100 MWd/tU.

The spent fuel elements are discharged into lead-steel containers. Each container is designed to receive 2,000 fuel elements. These spent fuel containers are stored in a concrete compartment inside the reactor building and are cooled by natural convection of the air. The compartment is big enough to store all the spent fuel elements of HTR-10 which is designed for 20 years operation on an average load factor of 50%. The storage of the spent fuel in the HTR-10 reactor building is surely the temporary storage. The issues of treatment, intermediate and final storage of the spent fuel elements remains to be studied.

6. Decommissioning considerations

Regulations require that the issues of decommissioning a reactor be considered already in its design and operation stages. The HTR-10 test reactor adopts the modular high temperature reactor design. It uses steel pressure vessels and, the RPV and SGPV are arranged in a "side-by-side" manner. They are separately housed in two concrete cavities. The spherical fuel elements can be easily removed from the reactor core into containers without opening the RPV. These factors, among other design features, make the decommissioning of HTR-10 relatively easier.

Important to decommissioning is that the design documents and related operational records should be well archived. Also important is that in the monitoring and surveillance programmes, aspects of decommissioning are also to be considered.

The following radioactive materials are relevant to the decommissioning work of HTR-10:

- Metallic materials: the reactor pressure vessel and the metallic core internals, the steam generator and its pressure vessel, the components of the fuel handling system and the reactor shutdown systems, some components and pipes of some auxiliary systems, mainly of the helium purification system and its regeneration system.

- Graphite and carbon bricks of the core internals including dusts.

- Filters and other absorbing components in helium purification system and its regeneration system as well as ventilation system.

- Routine liquid and solid radioactive materials which are produced during reactor normal operation.

Quantity of these materials and their radioactivity shall be evaluated before decommissioning starts based on the real reactor operation history.

The HTR-10 test reactor is designed for 20 year's operation. Presumably, the execution of its decommissioning is a matter around 2020. Till then, much development is expected in the national nuclear strategies, including the fuel cycle/waste disposal strategies. INET will also experience much development. The decommissioning programme of HTR-10 will be much dependent on the status of these developments then.

Independent from the above strategical development, following activities are believed to be performed when HTR-10 is to be put out of service.

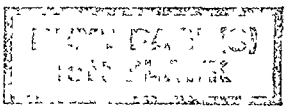
- After sufficient cooling time, the fuel elements in the reactor core are to be removed into the storage containers. All the spent fuel storage containers are supposed to be transported to places, designated by the nation, for further treatment or storage.

- Removable radioactive materials, activated or contaminated, treated when necessary, are to be removed from the HTR-10 site and transported to the nation's regional waste repositories.

- For the remaining components and structures, decontamination are to be made as reasonably much as possible.

- Radioactive place or components are to be sealed or blocked, the facility is under appropriate monitoring and surveillance.

After the above activities, the facility is in safe conditions. Entrance to some places in the facility remains controlled. Further decommissioning activities would be much dependent on the situations then.



FUEL STORAGE, STATUS AND PROGRAMMES

UNLOADING OF THE REACTOR CORE AND SPENT FUEL MANAGEMENT OF THTR 300

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Abstract

Following granting of License 7/12a on October 22, 1993 and preparatory work, unloading of the THTR pebble bed reactor core was initiated on December 7, 1993.

Achieving the state 'plant free of nuclear material' was one prerequisite for implementation of further preparatory activities to establish safe enclosure. To reach this target, it was necessary to remove approx. 670,000 operating elements (approx. 84% of which were fuel elements).

Basically, unloading of the core was implemented in the same way as removal of the operating elements during duty operation, however, process engineering modifications to the charging system were required due to replacement of the primary gas helium with nitrogen and air and reduced temperature and pressure as compared to duty operation.
During unloading operation, the operating elements were sorted by means of the burn-up measuring system and were transferred into operating element containers (steel cans), 2,100 elements per container.

Insertion of absorber rods and addition of unirradiated absorber elements ensured clearly subcritical conditions at any moment during unloading of the core, which was confirmed by the measured values of neutron flux density.

The residual inventory of fissile material remaining in the reactor pressure vessel after completion of core unloading activities by December 1994 is 0.976 kg and is thus significantly lower than the required value of 2.5 kg.

Due to the limited storage capacities of the plant, it was necessary to ship the fuel element containers simultaneously with core unloading. In a remote-controlled process, the fuel element containers were transferred from the spent fuel store to a shielded loading station, loaded into one transport and storage cask of the CASTOR THTR/AVR-type each, which was then sealed with the primary lid. Following leak testing and definitive sealing by staff working on a working platform outside of the loading station, the transport and storage casks were transferred to six-axle purpose-designed railway wagons and shipped to the Ahaus fuel element interim storage facility (BZA). By April 1995, a total number of approx. 620,000 fuel elements had been transported from THTR to BZA in 57 shipments, on general 6 transport and storage casks on 2 railway wagons per shipment.

Due to actual burn-up of the THTR fuel elements falling below the design values (mean burn-up per fuel element container max. 85,000 MWd/t HM) and the long cooling-down period, dose rates on the casks were very low. Neutron dose rate measurements taken on a loaded transport and storage cask showed results of < 1 µSv/h at the cask surface.

After loading the cask on the transport wagon a gamma dose rate of 1 - 2 µSv/h at the closed transport hood and of 0.5 µSv/h in a distance of 2 m from the transport wagon was measured.

The dose load received by the personnel was very low during the complete cask handling. The evaluation of the official dosemeters did not either show any of relevant exposure the employees (0.0 mSv/month effective dose).

1. INTRODUCTION

The THTR 300 prototype nuclear power plant in Hamm (Westphalia) with a graphite-moderated and helium-cooled reactor was shut-down for a scheduled revision after an operation time equivalent to 423 days of full-load operation. About one year later, the decision on decommissioning was taken by the federal and state authorities and the shareholders of HKG, the plant operator.

Following granting of License 7/12a on October 22, 1993 and preparatory work, unloading of the THTR pebble bed reactor core was initiated on December 7, 1993.

Achieving the state 'plant free of nuclear fuel' was one prerequisite for implementation of further preparatory activities to establish safe enclosure. To reach this target, it was necessary to remove approx. 670,000 operating elements (approx. 84% of which were fuel elements) from the reactor core (Figure 1), and to ship the fuel elements to the fuel element interim storage facility in Ahaus (BZA).

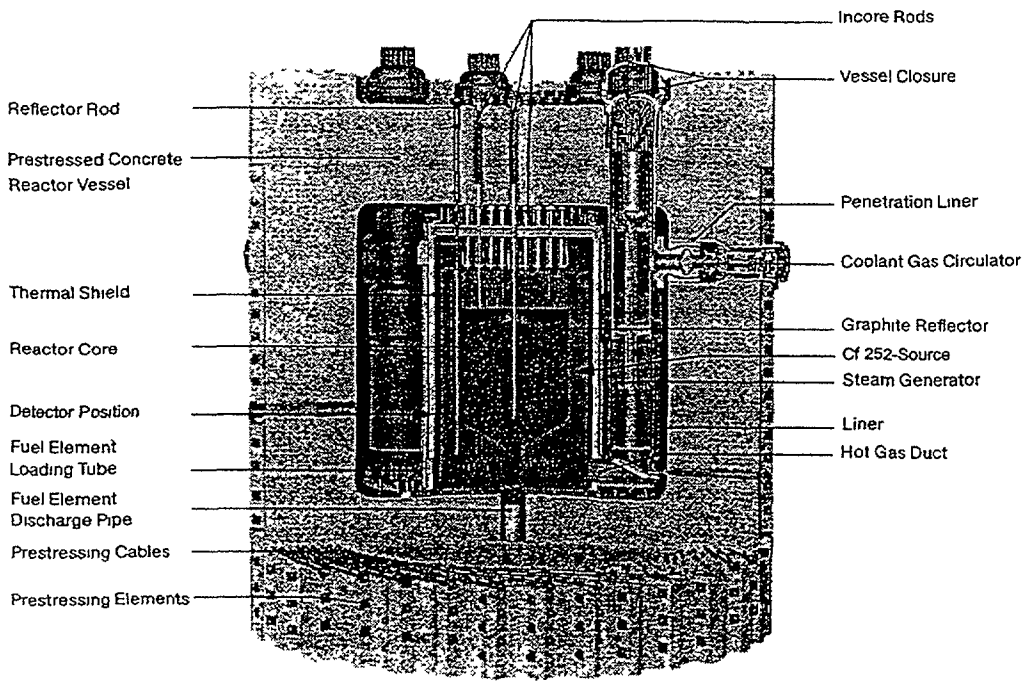


FIG. 1. Reactor vessel of THTR 300 with internals (model)

2. UNLOADING OF THE REACTOR CORE

The reactor core of the THTR 300 consists of a loose bed of spherical elements. At the beginning of unloading operation, the core contained approx. 563,000 fuel elements, 76,000 graphite elements and 31,000 absorber elements. These so-called 'operating elements' are spherical elements with a diameter of 60 mm and consist exclusively or in the main of graphite. Unirradiated fuel elements of the THTR contain approx. 1 g of highly enriched uranium (93% U 235) and approx. 10 g of thorium; the absorber elements and graphite elements used do not contain fuel.

Figure 2 shows diagrammatically the charging system. During duty operation of the plant (September 1985 to September 1988), it was used for continuous charging of the reactor with fuel elements. During this period, the fuel elements were recirculated several times and damaged elements sorted out by the damaged spheres separator.

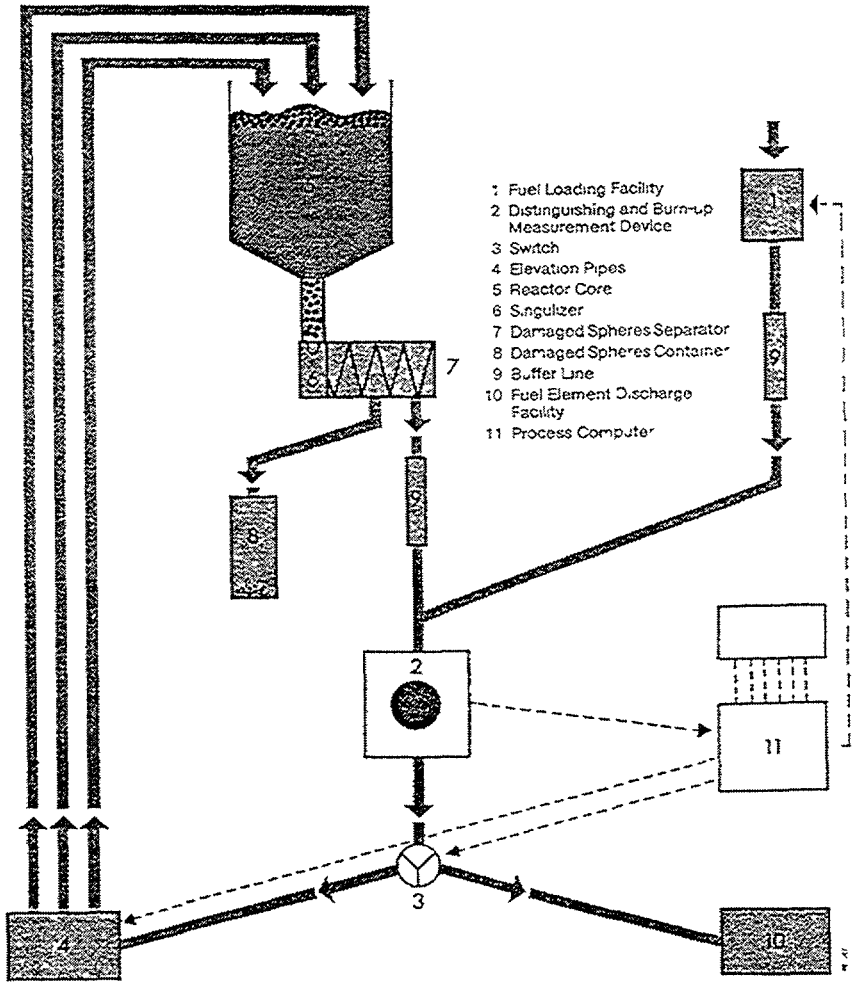


FIG. 2. THTR fuel circulating system

Basically, unloading of the core was implemented in the same way as removal of the operating elements during duty operation; however, process engineering modifications to the charging system were required due to replacement of the primary gas helium with nitrogen and air and reduced temperature and pressure as compared to duty operation.

During unloading operation (December 1993 – October 1994), the operating elements were sorted by means of the burnup measuring system (conisting of a graphite-moderated reactor with a thermal output of 500 W and an evaluating process control computer) and transferred into operating element containers (steel cans), 2,100 elements per container. Balancing of the removed fuel elements was carried out by means of the process control computer of the charging system and, independently from it, by means of the pebble counters of the charging and outward transfer installations.

From the beginning, core unloading was organized as a three-shift operation with seven working days per week. Due to the fact that only a few interruptions due to malfunction occurred, requiring only short periods for repair, and a well-trained staff was on duty, it was possible to remove an average of approx. 2,500 operating elements per day.

Figure 3 shows the distribution of graphite and absorber elements and of the burnup of fuel elements during unloading operation. Unloading steps, each one corresponding to unloading of 2,100 fuel elements, are plotted as abscissa. Burnup is stated in 'fima' (fissions per initial metal atom).

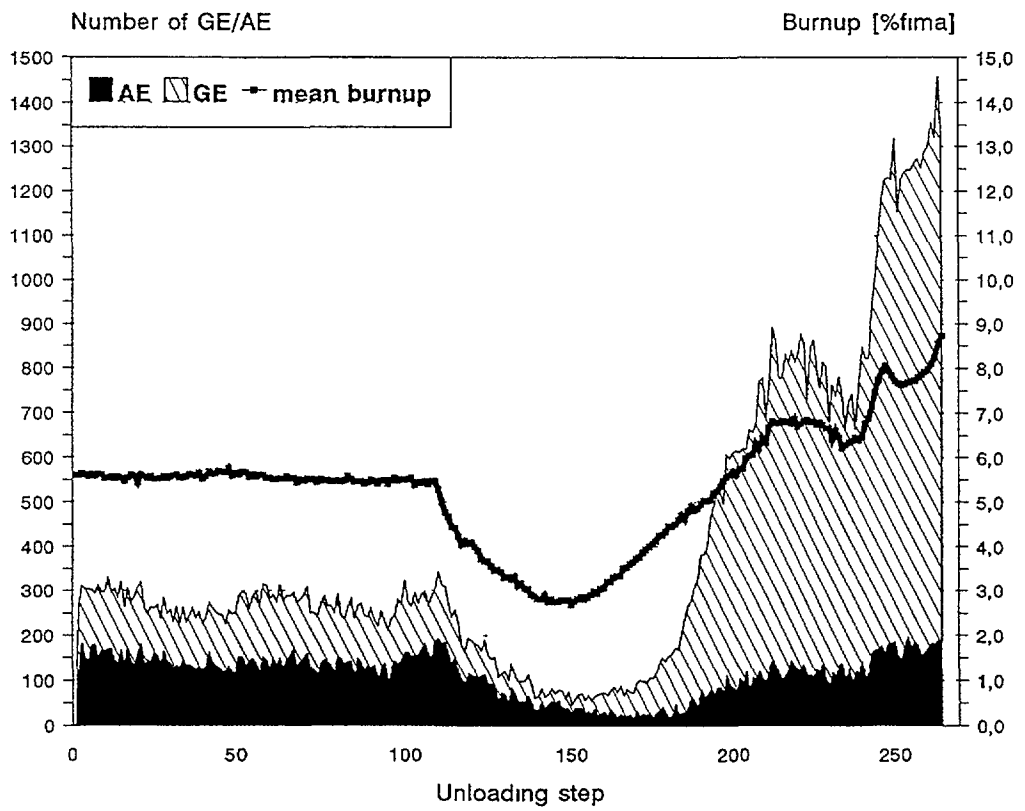


FIG 3. Number of removed graphite and absorber elements per charge of a fuel element container and mean burnup of the fuel elements.

The diagram reflects the sequence of removal of elements from certain core sectors. The minimum in the area of unloading step 150 is due to removal of fuel elements from the surface of the outer core (low irradiation of fuel elements), the maximum at the end of unloading operation is due to unloading of graphite elements and intensely irradiated fuel elements from the bottom edge of the core. The determined relative shares of operating element types and their variations during unloading time correspond well with the results of model experiments.

Fully inserted absorber rods and addition of a total of approx. 4,200 unirradiated absorber elements at certain unloading steps ensured clearly subcritical conditions at any moment during unloading of the core, which was confirmed by the measured values of neutron flux density.

The development of neutron flux densities during the unloading period is shown in Figure 4. The decrease corresponds to the radioactive decay of the neutron source (Cf 252-source). When the core surface comes closer to the position of the neutron source, the decrease accelerates due to influences of geometry. Finally, only the neutron flux density caused directly by the source remains.

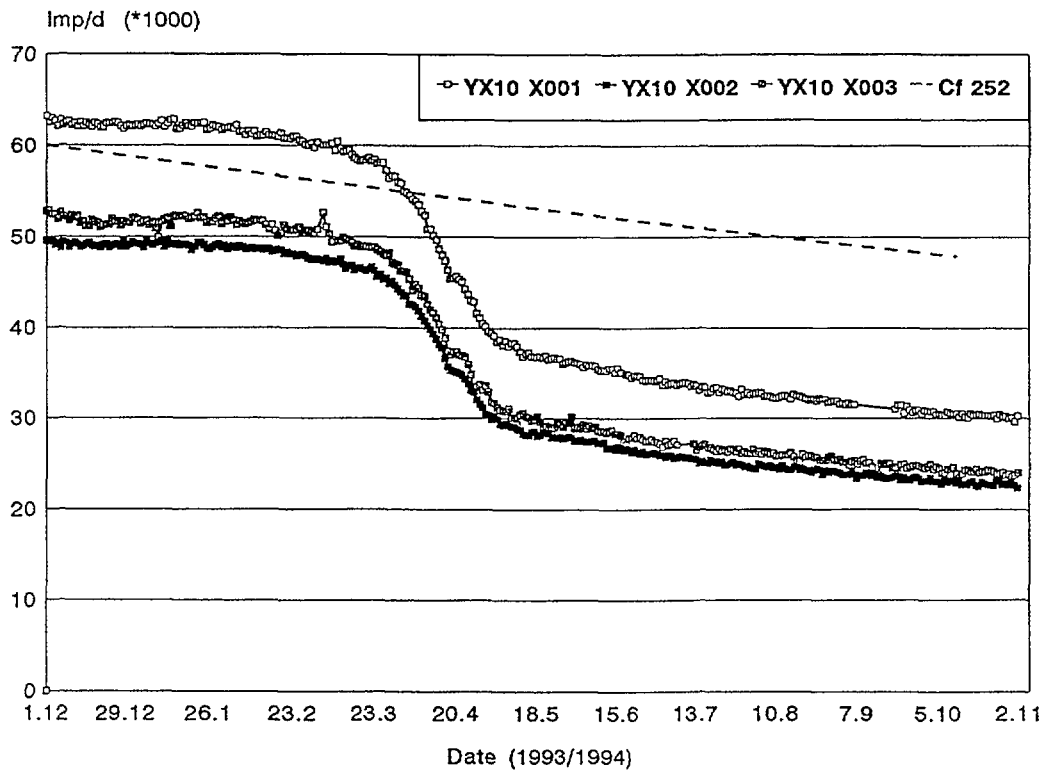


FIG. 4. Neutron flux measurement with three detectors of the YX10 system. (as reference: decay of Cf 252, normalized by arbitration)

Images supplied by a video camera that had been brought into the reactor core from time to time showed that the gradient of the funnel during reactor core discharge was within expectations.

During the final inspection, some operating elements were removed from the lower part of the operating element discharge tube and pushed into the containers provided for damaged fuel elements. Altogether 14 containers for damaged operating elements were filled during the period from the start of operation of the plant until the end of unloading operation.

After the first amendment to license 7/12a had been issued on February 2, 1995, fuel elements that might have been filled into 20 containers during the first year of reactor operation (1985/1986), containing possibly a mix of different types of operating elements, were sorted out and filled into the containers with damaged fuel elements.

The residual inventory of fissile material remaining in the reactor pressure vessel after completion of core unloading activities by December 1994 is 0.976 kg (equivalent to 2,198 irradiated fuel elements) and is thus significantly lower than the required value of 2.5 kg.

3. SPENT FUEL MANAGEMENT OF THTR-300

3.1 Process of the outward transfer of THTR fuel elements

Due to the limited storage capacities of the plant, it was necessary to ship the fuel element containers simultaneously with core unloading. Prior to transport, the fuel element containers had to be transferred into transport and storage casks of the CASTOR THTR/AVR type.

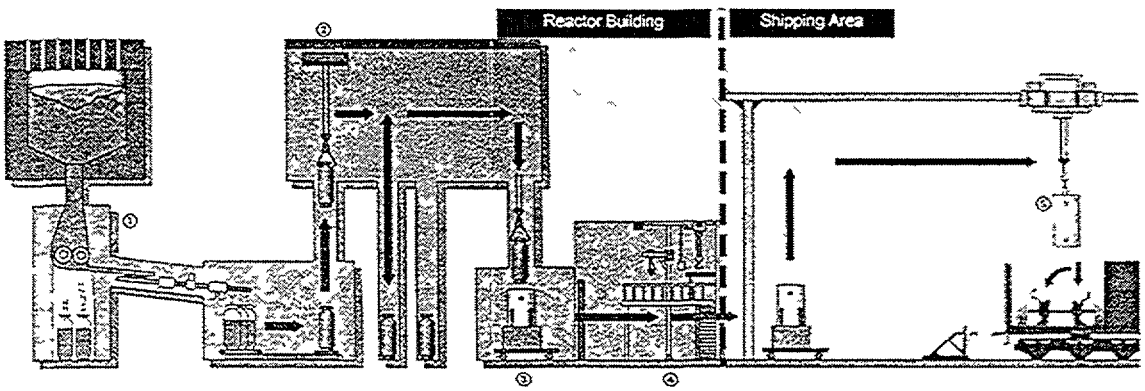


FIG 5 Process of the outward transfer of the THTR fuel elements
1 Fuel element outlet 2. Crane of operating element store 3. Loading station
4 Working platform 5. Transport wagon

Figure 5 gives a schematic illustration of the process of outward transfer of the THTR fuel elements.

For outward transfer, the transport and storage cask CASTOR THTR/AVR had to be prepared for loading. The cask was opened except for the primary lid and transferred into the loading station. Due to the high dose rate during loading, the shielding gate of the loading station was then closed. By means of the manipulator, the primary lid was removed and the operating element container inserted into the opened transport and storage cask through a ceiling hatch of the loading station from the internal store for operating elements above the loading station. After re-inserting the primary lid into the transport and storage cask and screwing it into place initially by the manipulator, the loading station was opened and after radiological measurements, the cask was transferred from the loading station to the working platform. Here, screwing of the primary lid was completed and the leak-tightness of this first cask barrier (leakage rate $< 1 \cdot 10^{-7}$ mbar $\cdot l \cdot s^{-1}$) proved.

After positioning and screwing of the secondary lid, again the leak-tightness of this second barrier was tested. For transport, the secondary lid was additionally provided with an electronic transport seal. Finally, the protective plate required for interim storage was fitted prior to loading the transport and storage cask onto the transport wagon.

For the shipment of the transport and storage casks to an interim storage facility, HKG has four special six-axle railway wagons at its disposal, each capable of transporting three casks.

Due to HKG's very tight schedule for decommissioning, processing of the transport and storage casks was implemented from the beginning in a multiple-shift operation. Through introduction of 3-shift operation and 6 days working week and through additional optimizing measures during transport and cask handling, a weekly processing rate of max. 11 CASTOR casks was reached.

By April 1995, a total number of approx. 620,000 spent fuel elements had been transported in 305 CASTOR casks from THTR to BZA in 57 shipments, usually six transport and storage casks on 2 railway wagons per shipment.

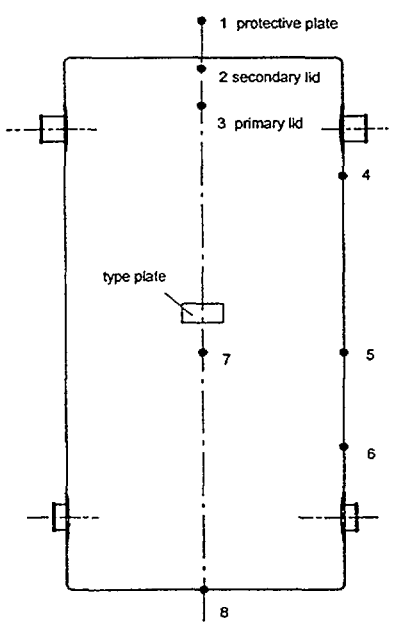
3.2 Exposure of the operating personnel to radiation during cask processing

According to the originally planned burnup and cooling time of the irradiated THTR fuel elements to be stored in the casks (mean/max. burnup 11.4% / 15% fima, 200 days minimum cooling time) a surface dose rate of max. 100 µSv/h (from gamma and neutron radiation) at a 37 cm shielding thickness of the cask material GGG-40 cm had been established in the supply specification.

Due to the real burnup history of the irradiated THTR fuel elements (reduced burnup and longer cooling time prior to storage in the transport and storage casks; max. burnup per fuel element container was approx. 8.8 % fima or 85,000 MW·d/t HM), the dose rate was reduced by about one decimal exponent to below 10 µSv/h. At a measured maximum surface dose rate of a loaded unshielded fuel element container of 10,000 mSv/h, this results in a weakening of the radioactive radiation by a factor of approx. 10^6.

With max. 100 W, the decay heat of the charged fuel element containers also was significantly lower than the design parameters.

Figure 6 shows the typical gamma dose rates measured on the loaded transport and storage cask at various points of a CASTOR cask filled with high-burnup fuel elements.



Measuring Point	Dose Rate
1	0.3 µSv/h
2	1.5 µSv/h
3*	24.0 µSv/h
(open screw holes)	(300 µSv/h)
4	0.4 µSv/h
5	1.8 µSv/h
6	0.6 µSv/h
7	2.0 µSv/h
8	3.0 µ Sv/h

* only accessible during preparation on the working platform

FIG. 6. Ambient dose rates on transport and storage casks

Measurements of the neutron dose rate at the loaded transport and storage cask after inserting the primary and secondary lid, as well as at the protective plate and at the cask body showed the dose rate to fall below 1 µSv/h.

After loading the cask on the transport wagon, a gamma dose rate of 1 - 2 µSv/h at the closed transport hood and of 0.5 µSv/h in a 2-m distance from the transport wagon was measured.

The total-body doses received by the personnel were monitored with operator-owned digital dosimeters as well as with official dosimeters. The evaluation of the official dosimeters did not either show any measured values at any employees. Measuring results are listed in Table 1.

Table 1. Total-body doses received during cask handling

Year	Number of Casks	Collective Dose [1]			
		Partial-Body Dose [2]		Total-Body Dose [3]	
		Operator's Measurements	Official Measurements	Operator's Measurements	Official Measurements
1992	14	-	-	0,10 mSv	< 0,0 mSv
1993	6	-	-	0,05 mSv	< 0,0 mSv
1994	278	-	38 mSv	0,99 mSv	< 0,0 mSv
1995	7	-	0 mSv	0,01 mSv	< 0,0 mSv

[1] Processing personnel : 10-20 persons
[2] Annual dose limit: 500 mSv/person (dose received during inserting the screws into the screw holes in the primary lid)
[3] Annual dose limit: 50 mSv/person

A comparison of these measuring results with the total-body dose rate limits from Annex X, Table X1, column 2 of the Radiation Protection Act shows that the measured values fall by several decimal exponents below the limits per person laid down in there. This statement is also valid for the partial-body doses that the employees received during inserting the screws into the still open screw holes. Here, monitoring was effected by means of finger badge dosimeters.

4. SUMMARY

The results described above show that based on a good preparation of activities and with a well-trained staff it was possible to carry out at the same time both unloading of a reactor core and shipping of the removed fuel elements to the interim storage site within a short period of time and with very low dose rates.

R&D ON INTERMEDIATE STORAGE AND FINAL DISPOSAL OF SPENT HTR FUEL

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Abstract

The back-end of the fuel cycle concept for spent High-Temperature Reactor fuel elements in Germany is based upon intermediate storage in shielded casks in a surface facility followed by direct disposal in a deep repository. Two storage facilities are in operation, whereas disposal in a salt dome repository is being designed. R&D results obtained so far support the chosen concept and underline the special safety features of the fuel elements, i.e. the coated particle fuel stabilised in a graphite matrix, which is extremely resistant against all conceivable attacks during storage and disposal.

1 Introduction

Based upon a former development at the Forschungszentrum Jülich (FZ Jülich, Research Centre Jülich), two high-temperature gas cooled and graphite moderated reactors (HTR) had been operated in Germany: a) the 15 MWe AVR reactor from 1967 until 1988 in Jülich, and b) the 300 MWe Thorium High Temperature Reactor (THTR 300) from 1985 until 1988 in Hamm-Uentrop. The status of their decommissioning has been reported in Session I of this conference.

Both reactors have in total produced about 1 Million of spent fuel elements during their operating time. The typical fuel element is a tennis-ball sized sphere from graphite, containing up to twenty thousand pinhead-sized fuel particles containing oxide or carbide fuel each. The particles are surrounded by a high-porosity buffer layer to limit the internal pressure from swelling and gas production, and coated with a high-density pyrocarbon layer (BISO) or with a combination of two pyrocarbon layers with a silicon carbide layer in between (TRISO) to retain radionuclides (see FIG. 1).

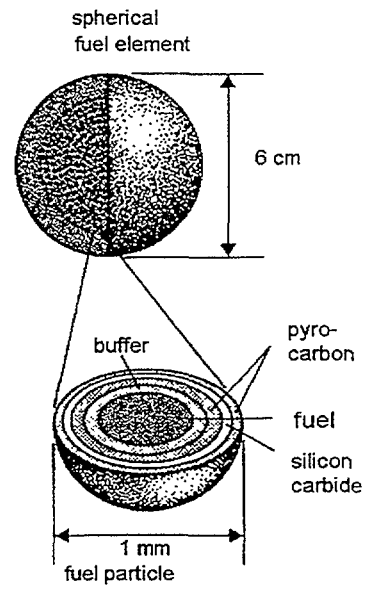

FIG. 1: Sectional view of a HTR fuel element

The spent fuel management concept for HTR in Germany is based upon intermediate storage followed by disposal in a deep rock salt repository without reprocessing.

Techniques for the intermediate dry storage in CASTOR-type transport/storage casks are available and practised with AVR fuel in Jülich as well as THTR fuel in Ahaus. Experiences are reported in Session II of this conference.

For disposal, emplacement in horizontal drifts using shielded casks, or in deep vertical boreholes using simple packaging were chosen to be the most promising concepts /1/.

This paper summarises the results obtained so far, as well as R&D still do be done to support intermediate storage and final disposal of spent HTR fuel. It supplements previous reports, e.g. /2/ through /13/.

2 R&D work on intermediate storage

Principal goals of the activities on interim storage at FZ Jülich are to demonstrate the safety of dry storage and to provide data for the licensing of corresponding commercial storage facilities. Complementing the storage of AVR fuel, a research program was initiated to measure the release of gaseous radioactivity under realistic storage conditions. FIG. 2 shows the equipment in the hot cell of

the AVR dry storage facility to measure the built up of the ^{3}H, ^{85}Kr and $^{14}C(CO_2)$ concentration in the gas atmosphere of a AVR dry storage can, filled with 950 fuel elements of the type GO[a] and connected to a gas sampling circuit The storage temperature was 30 °C max and the burn-up of the fuel 13-16 % FIMA The specially designed dry storage can could be connected by remote handling to flexible tubes inside the hot cell The build-up of the concentration was measured over several years FIG 3 - 6 show the trend of gaseous release

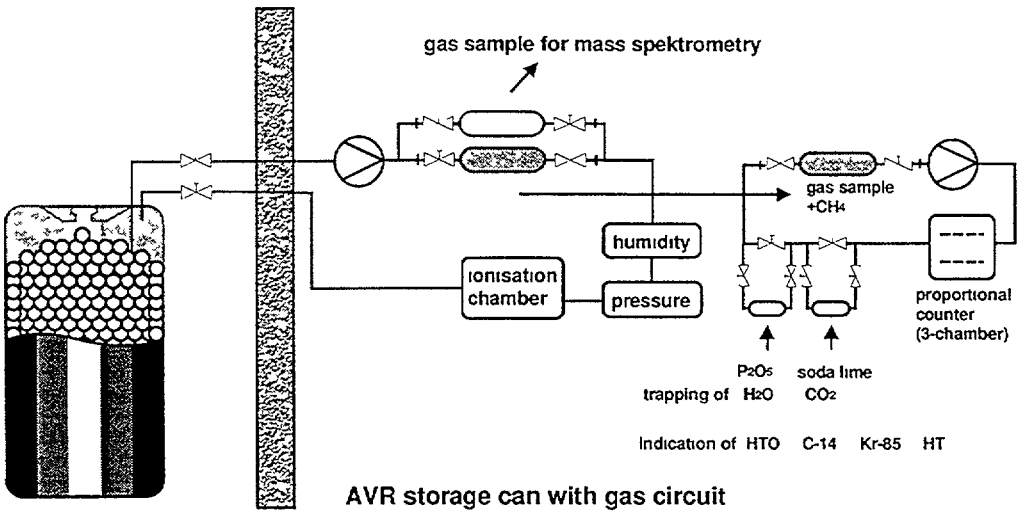

FIG 2 Release of gaseous activity during the storage of AVR fuel elements

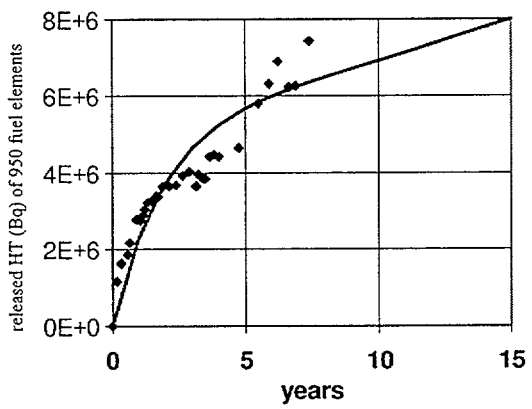

FIG 3 Release of ^{3}H

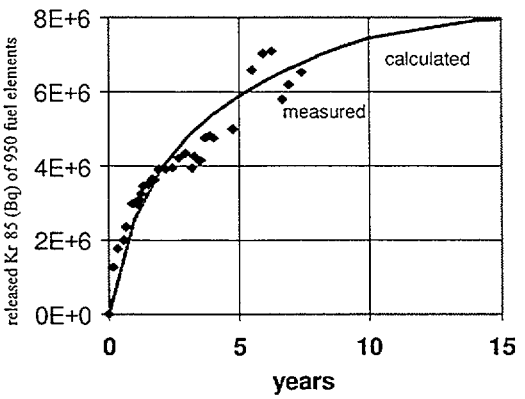

FIG 4 Release of ^{85}Kr

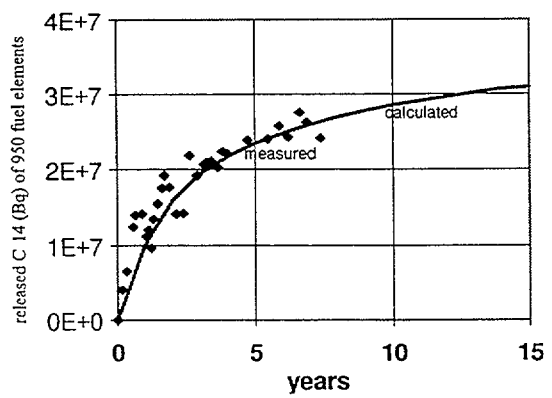

FIG 5 Release of ^{14}C

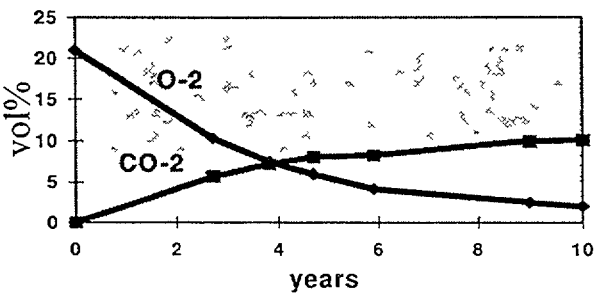

FIG 6 Gas composition within a storage can

[a] Compressed graphite matrix, oxide fuel

The ^{3}H inventory of a fuel element is mainly generated in the graphite matrix by ^{3}He and ^{6}Li impurities. The release is controlled by absorption and desorption processes at the grain boundaries of the graphite and diffusion in the grains. The complete ^{3}H inventory of a dry storage can amounts to about $2*10^{12}$ Bq.

The ^{14}C inventory is mainly generated by (n-p)-processes of ^{14}N impurities of the graphite. The release of ^{14}C during storage is initiated by corrosion processes of the carbon by contact with air and gamma radiation. CO_2 will be generated. The inventory of 950 fuel elements amounts up to $7*10^{9}$ Bq. Only 1% can be released, until the oxygen content of a storage can is consumed.

The ^{85}Kr inventory is generated by fission and amounts to $1*10^{13}$ Bq in a storage can. The release mainly depends on the number of defect particles. The specification of the fuel elements permits a defect rate of $3*10^{-4}$, that means a possible release of $3*10^{9}$ Bq. The extrapolation of the measurements shows a maximum amount of $1*10^{7}$ Bq.

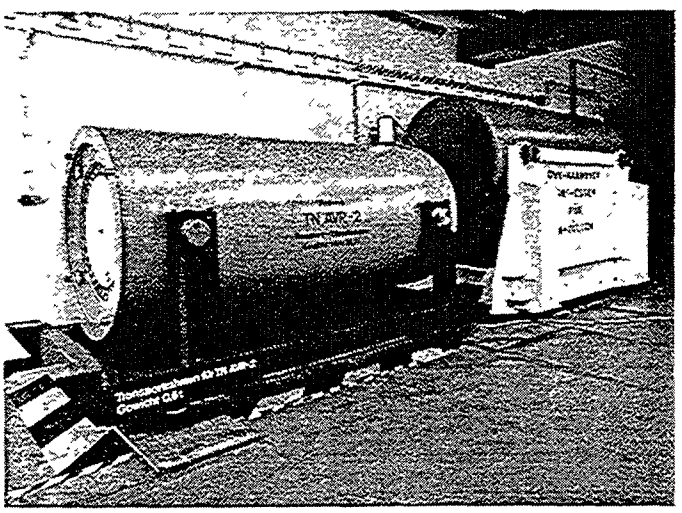

FIG. 7: 2 prototype storage casks, each loaded with 1900 fuel elements (= 2 AVR-cans)
burn-up: 12 - 16% fima

Further measurements were performed at the two prototype transport and storage casks (FIG. 7). FIG. 8 and FIG. 9 shows the distribution of gamma- and neutron dose rate at the surface. Each of the casks was loaded with two dry storage cans, filled with 950 fuel elements of the type GO and GK[b] with high enriched fuel $(U,Th)O_2$ each. The burn-up of the fuel amounted to 13 - 16 % FIMA. The measurements were done after different decay times. One year after discharge (1990) the main part of the dose rate at the surface of the 30 cm thick steel wall of the cask was generated by ^{144}Ce-^{144}Pr with the high gamma energy of 1,5 MeV. Fuel elements with lower burn-up (12%FIMA in the outer cans) produced due to the higher content of ^{235}U more short-lived fission products like ^{144}Ce. After several years of decay time, ^{137}Cs is the dominating dose rate source caused by the higher burn-up (16%

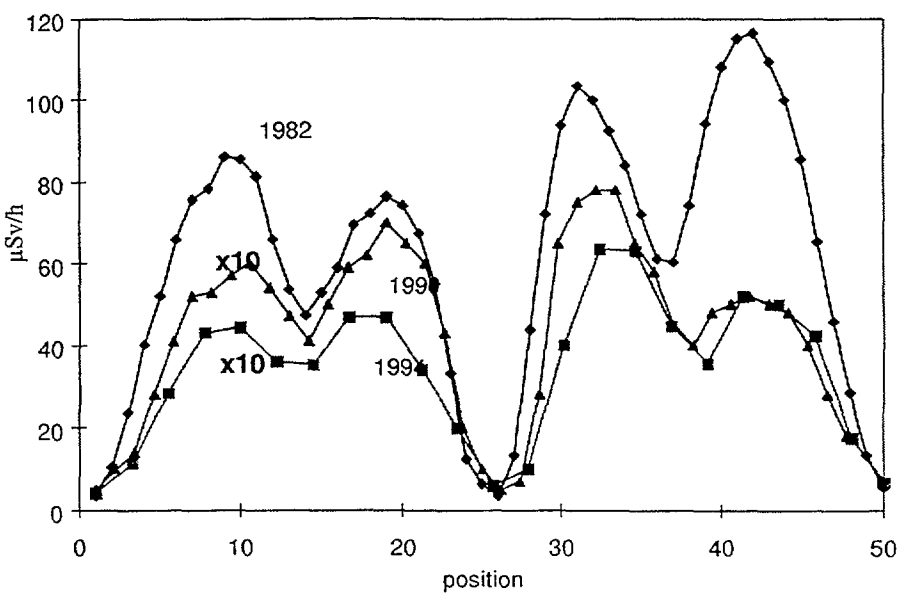

FIG. 8: Gamma dose rate of AVR storage casks

[b] compressed graphite matrix, carbide fuel

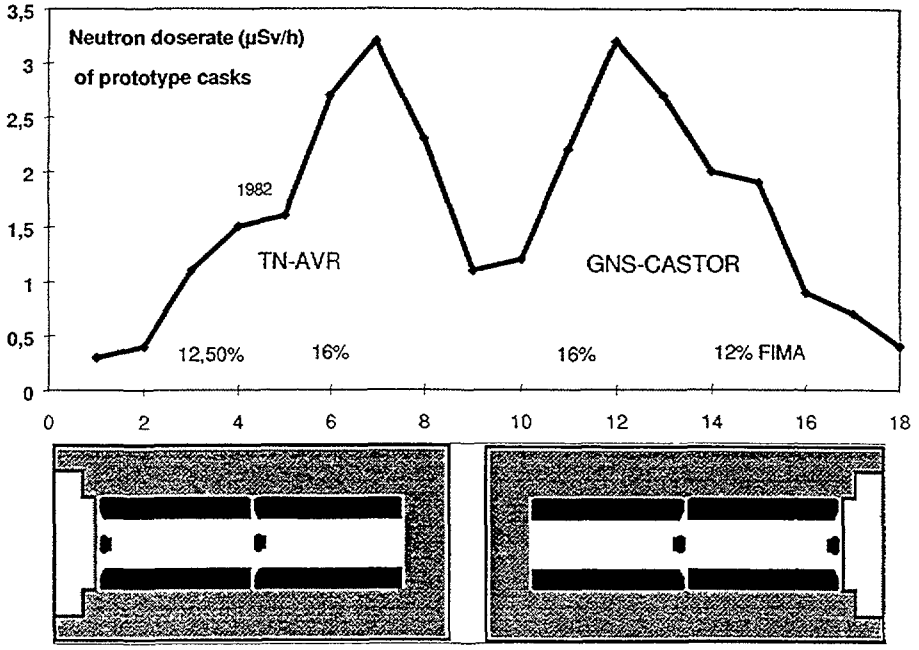

FIG. 9: Neutron dose rate of AVR storage casks

FIMA) of the inner positioned storage cans (FIG. 8).

The neutron irradiation at the surface is essentially caused by the alpha decay of ^{238}Pu. Due to the high portion of ^{232}Th in the fuel (Th/U=5), a low concentration of ^{238}Pu is breeded and accordingly a low level of neutron irradiation is measured (FIG. 9). The measurements were performed with the neutron REM counter of FAG. The distribution of the neutron dose rate shows very high values in the position of the two cans with the higher burn-up. This effect is caused by non-linear breeding of ^{238}Pu with increasing burn-up.

3 R&D work on final disposal

The fuel elements of the HTR are proposed for final disposal without reprocessing. The aim of the disposal is the protection of individuals and the environment from the hazards of nuclear waste. Because of the long half-life of some radionuclides this protection must be guaranteed for time periods of hundreds or even thousands of years. One possibility to reach the protection aim is the disposal in a stable salt dome as geologic host formation. For this final disposal it must be distinguished between the normal disposal conditions and the hypothetical, but not absolutely impossible accident scenario of an water ingress into the repository.

3.1 Behaviour under normal disposal conditions

Under normal disposal conditions the repository will be dry. The rock salt will become plastic under the pressure of the overlaying strata and will thus creep into cavities not completely filled with waste. This will result in a high mechanical load on the fuel elements which may be crushed up if no countermeasures are taken. Appropriate countermeasures like e.g. grouting with cement have been developed and successfully tested /3/.To investigate the stability of the fuel elements a remote press with a maximum pressing power of about 30 MPa was installed in the hot cell facility „GHZ" of the FZ Jülich. The equipment consisted of a cylindrical steel vessel and the pressure piston. The vessel was swept with an inert gas and connected to a gas measuring circuit (see FIG. 10).

The release of ^{85}Kr is an indication for the particle failure, because this fission gas is closed up in the coated particles. This ^{85}Kr release can easily be monitored by means of an ionisation chamber. The fuel elements are crushed and densified until the cylindrical space of the pressure vessel is completely filled. The tests showed that particle failure depends on the coating design and the burn-up. However, the failure fraction observed remained below 1% (see FIG. 11) /4/. Coated particle

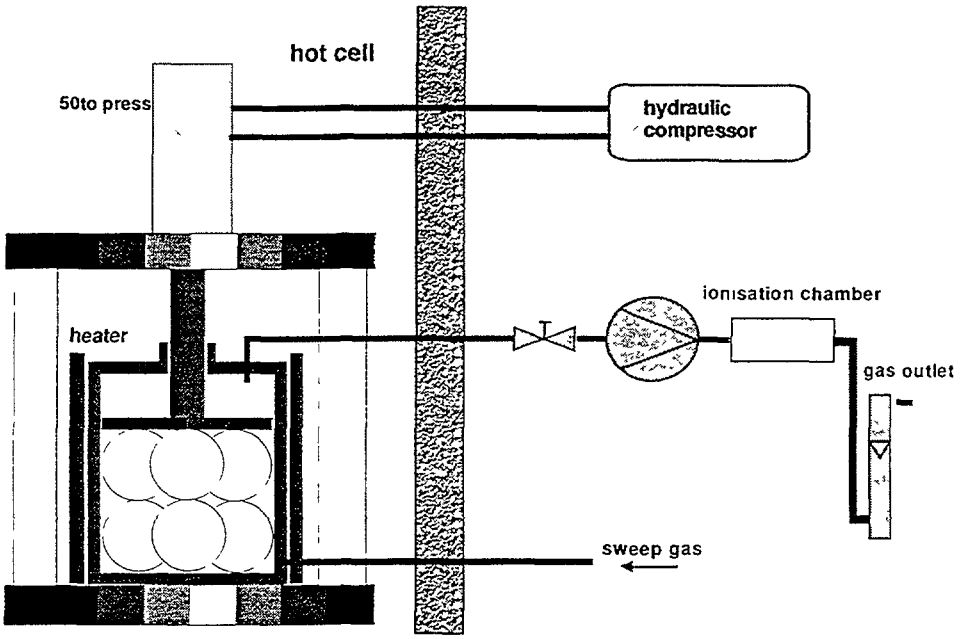

FIG. 10: 50to press to crush up to 9 spherical fuel elements (simulation of a mechanical pressure of 30 MPa)

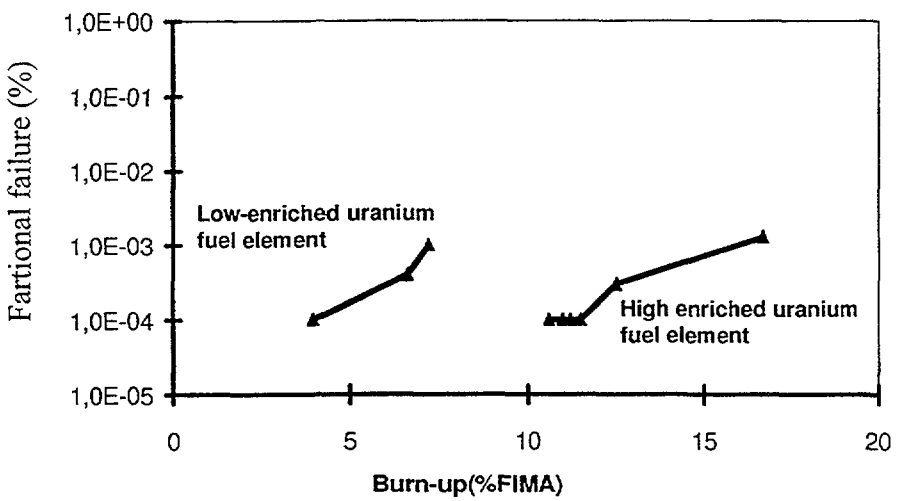

FIG. 11: Fractional failure of coated particles after crushing the matrix

failures may largely influence the long-term safety of a repository as far as the containment of long-lived radionuclides is concerned.

3.2 Behaviour under accidental conditions under the aspect of long-term safety

No radionuclide release can take place in a dry repository. Only if the radioactive waste gets into contact with the ground water radionuclides can be transported to the environment. Salt domes are geologically stable formations, which have been sealed from the ground water for more than 10^6 years. Therefore these formations are considered as ideal for final disposal.

However, in different accident scenarios it is assumed, that ground water may penetrate into the storage field through little crevices in the anhydride layers, which may be part of the salt dome. This water will form saturated, high corrosive salt brines and after corrosion of the storage casks the brines will interact with the fuel elements. A large number of experiments to study the behaviour of HTR fuel elements in such salt brines were performed at the FZ Jülich starting in the late 70^{th} /5, 6, 7/. A short review of the obtained results is summarised in this chapter.

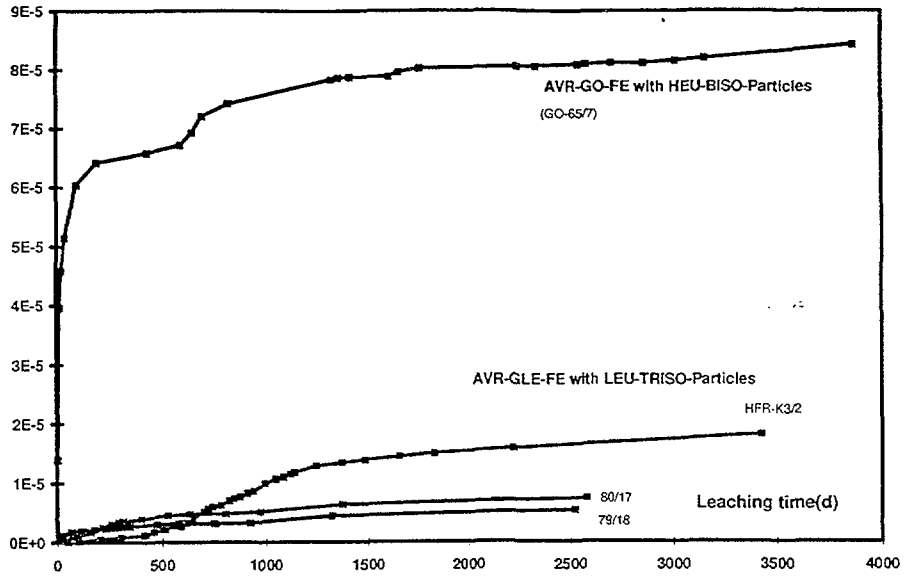

FIG. 12: Fractional release of ^{137}Cs inventory in Q-Brine at 90 °C/13 MPa

Different types of AVR fuel elements[c] were exposed to Q-brine[d] /8/at temperatures between 50 and 90°C and pressures up to 13 MPa. The fractional release of ^{137}Cs is shown in FIG. 12. The release from fuel elements with BISO particles is less than 10^{-4}, related to the inventory of one coated particle respectively the inventory generated by the U-contamination of the matrix graphite. After 200-300 days the release rate decreases to a stable equilibrium controlled by the diffusion of ^{137}Cs from the pyrocarbon kernel into the matrix. Calculations based on a diffusion/adsorption model with the GETTER code match the experimental results very well. For the long time calculations the release of ^{137}Cs can be described as a instantaneous release of about 20% of the matrix inventory, which is related to the amount of kernels with a defect in the coating, and a slow release of the remaining ^{137}Cs in the graphite particles over a time period of several 100 years /4, 6, 9/.

The comparable low fractional release of the fuel elements with TRISO-coated particles is caused by a corresponding lower U-contamination of the graphite matrix during fuel manufacturing, and therefore the long time behaviour of the fuel elements with TRISO particles is even more convenient.

For radionuclides other than cesium the leaching rates are presumed to be lower because of their lower mobility. The following experiment has proved this for the long-term relevant nuclides ^{99}Tc and ^{237}Np /10/. A bore in a graphite sphere had been filled with 400 µl of a radionuclide solution. Then the hole was sealed with a screw and the sphere were immersed in Q-brine or water at 90°C and a pressure of 15MPa. Table 1 shows the breakthrough time of different nuclides. Although no breakthrough had occurred for technetium after 400 d, a permeability of technetium can not be ruled out with regard to the long time scale of a final repository; moreover the breakthrough can be forced by a cyclic change of the pressure from 100 kPa to 15 MPa.

TABLE 1. BREAKTHROUGH TIME OF DIFFERENT RADIONUCLIDES

	Leachant		
Nuclide	Water (d)	Q-Brine (d)	Q-Brine pressure cycled (d)
^{99}Tc	90	> 400	< 4
^{137}Cs	20	20	10
^{237}Np	150	150	30

[c] high and low enriched U, TRISO and BISO coated particles

[d] Q-Brine is a high concentrated salt brine with $MgCl_2$ as main component.

The radionuclide release depends not only on the diffusion of the nuclides through the graphite matrix. The basic process is the dissolution of the radionuclides in the fuel kernels, which differs for the fuel matrices UO_2 or $(Th,U)O_2$. A release from intact coated particles didn't occur, but the fuel elements contain between 10^{-4} and 10^{-5} defect particles from the production process. This rate increases by a factor of approximately ten for highly irradiated material. Therefore, the source term for radionuclide release is mainly influenced by the number of broken coated particles. To investigate the behaviour of the irradiated fuel, coated particles were collected from irradiated, electrochemically disintegrated fuel elements. The particles were carefully point-loaded until the coating cracked. The single kernels were leached with Q-brine in air at 20°C and at 90°C, respectively, and 100kPa or 13MPa, respectively. The following Fig. 13 and FIG. 14 show the release rates of different radionuclides from the two fuel matrices /7/.

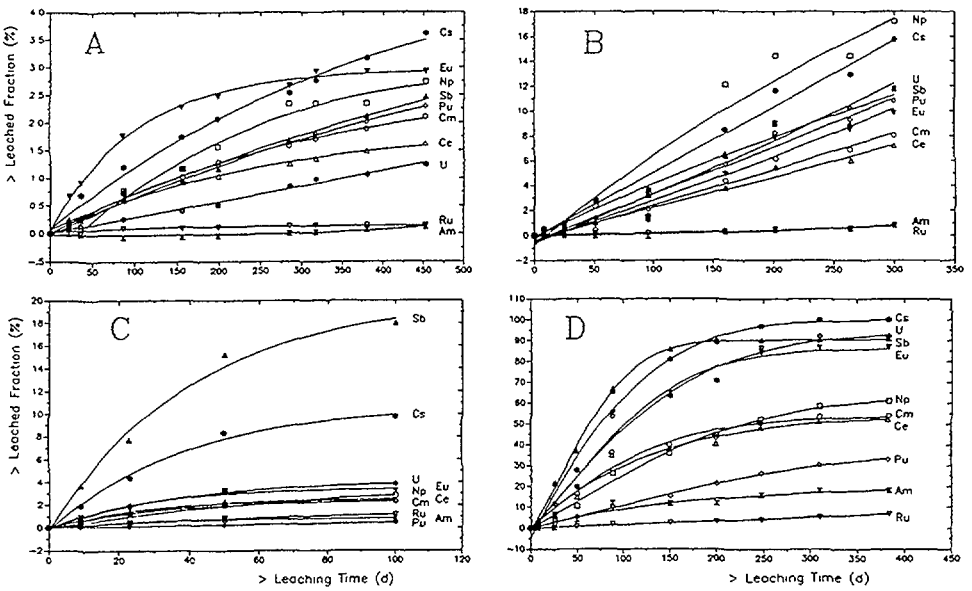

FIG. 13: Radionuclide leaching from UO_2 kernel with Q-Brine
A: 20°C/100 kPa B: 20°C/13 MPa C: 90°C/100 kPa D: 90°C/13 MPa

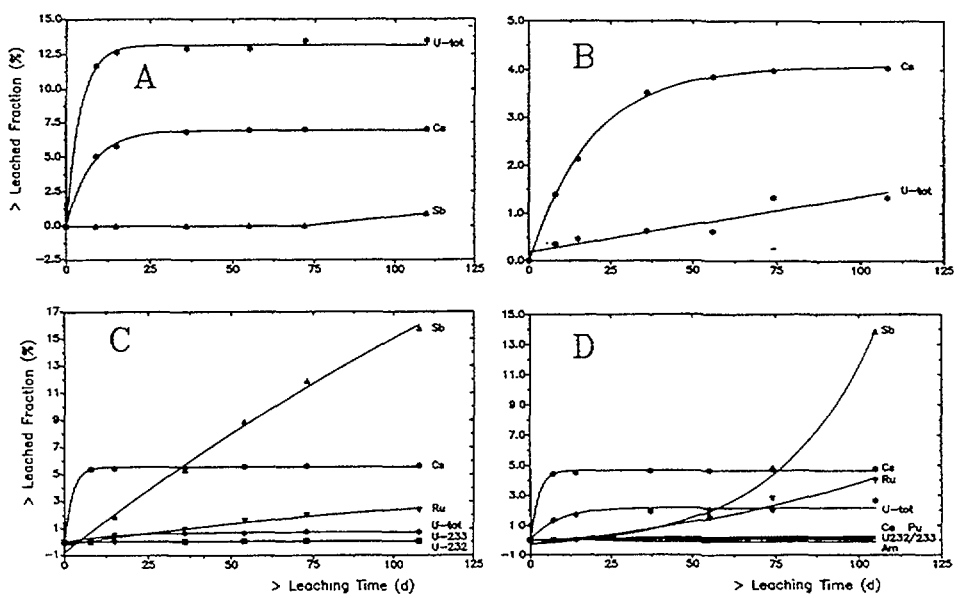

FIG. 14: Radionuclide leaching from $(Th,U)O_2$ kernel with Q-Brine
A: 20°C/100 kPa B: 20°C/13 MPa C: 90°C/100 kPa D: 90°C/13 MPa

The high leaching rates from the UO_2 fuel cannot be explained by the metal oxide's solubility in water only. The main is very probably the combination of weakening of the UO_2 during irradiation by fission product build-up, mainly at the grain boundaries, and the very aggressive salt brines. In contrast to the UO_2 the ThO_2 withstands such an attack /7/.

In conclusion the amount of leachable radionuclides is largely dependant upon the uranium contamination of the fuel element during manufacturing and the failure of the particle coating. Therefore the corrosion of the coating is a central problem for the long-term safety. The corrosive attack of the coating by Q-brine was investigated by the following experiment /4/. The top of an unirradiated particle was cut off and then fixed with the intersection towards an irradiated particle in order to produce a typical radiation field. The space between these particles was filled with Q-brine (FIG. 15). After a leaching time of about 2 years the state of the coatings at the intersection has been investigated by optical and electron microscopy. No indication of a corrosive attack could be detected, as it is shown in FIG. 16. However, with respect to the time scale of final disposal this result does not rule out any corrosive process. Therefore further investigations are necessary to describe the long-term behaviour of the coating under disposal relevant conditions.

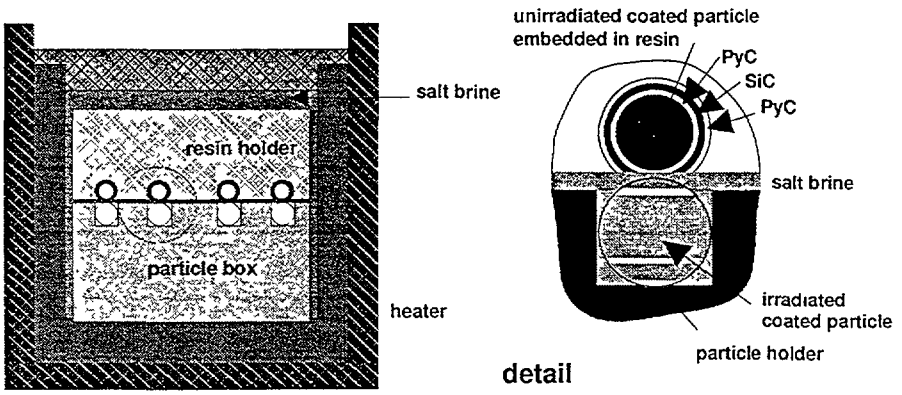

FIG. 15: Leaching of a polished particle with realistic radiation effects

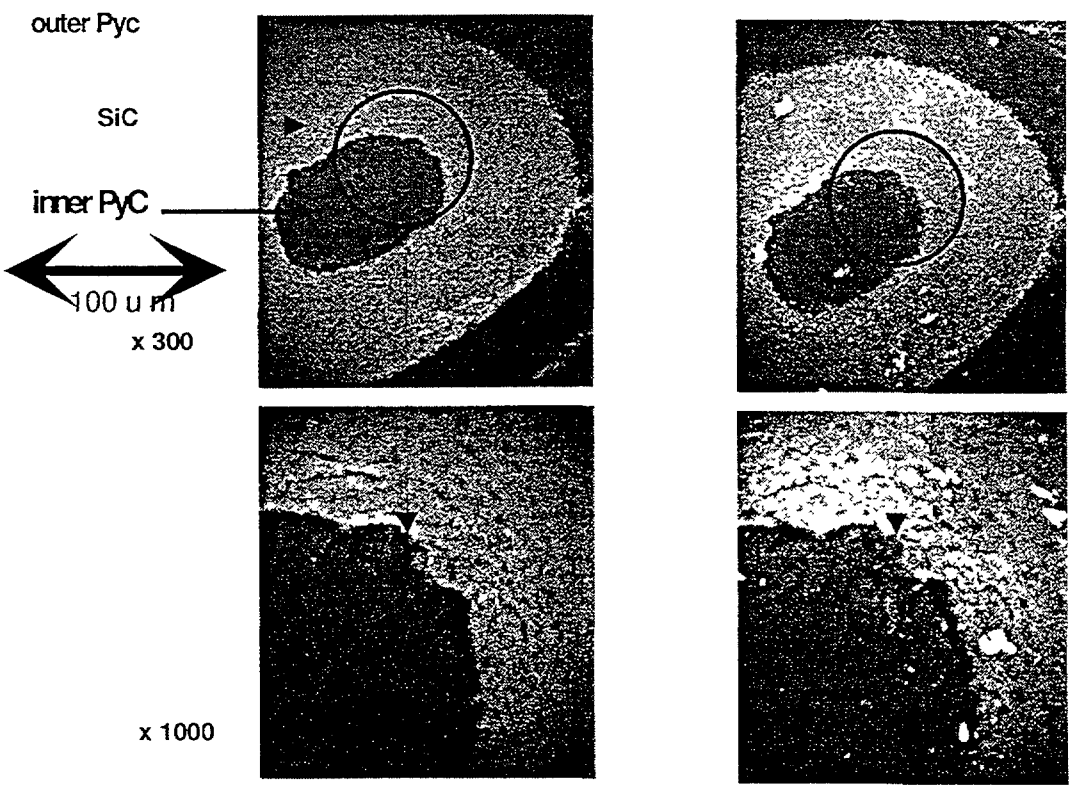

FIG. 16: SEM micrographs of a polised TRISO particle at the beginning (left) of leaching and after 2 years (right).

An additional problem of final disposal is the production of hydrogen by either radiolysis of water or corrosion of metals. This hydrogen may increase the release of radionuclides to the environment by pressing contaminated brine out of the repository. A set of experiments concerning the formation of hydrogen by radiolysis has recently been finished /11, 12, 13/.

Different irradiated fuel elements were exposed to Q-brine under argon or air atmosphere in a spherical autoclave (FIG. 17). The gap between autoclave wall and fuel element had a thickness of 1 or 2 mm, respectively. The experiments were performed at 22 or 55°C brine temperature. A gas plenum with a pressure gauge for continuos measuring was located above the autoclave. Gas samples were taken with an attachable gas sampling tube to analyse the gas composition by gas chromatography and radio gas chromatography.

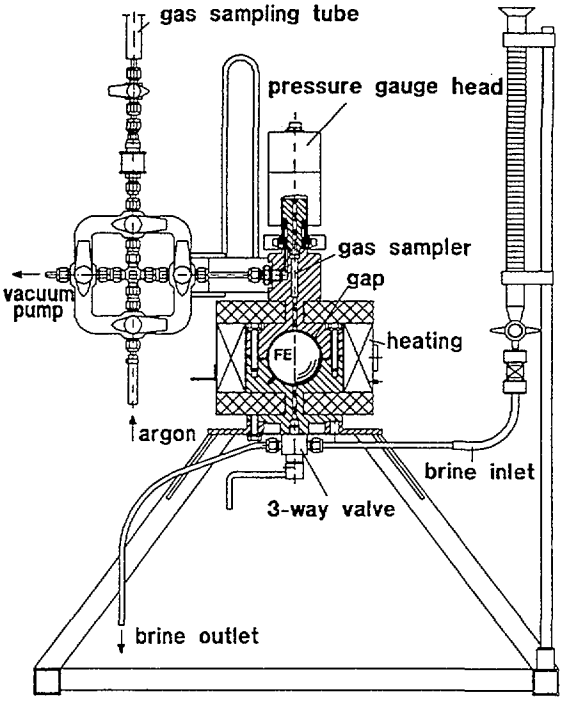

FIG. 17: Spherical autoclave

These experiments had the additional aim to determine the release of ^{14}C in gaseous form of $^{14}CO_2$ and solved in the brine. The ^{14}C is mainly formed by an (n,p) reaction of the nitrogen impurities in the cooling gas helium. The fresh ^{14}C is absorbed at the graphite matrix and due to the high temperature mounted into the crystal structure of the graphite. This radionuclide is important for the long-term safety because of its long half-life together with its different chemical behaviour in comparison to the other, mostly cationic radionuclides. Moreover, it acts as an indicator for the corrosion of the graphite matrix. Therefore brine in- and outlet were attached to the autoclave. The brine was replaced in the same time intervals as the gas samples had been taken.

The following diagram (FIG. 18) shows the pressure build-up for the different experimental conditions.

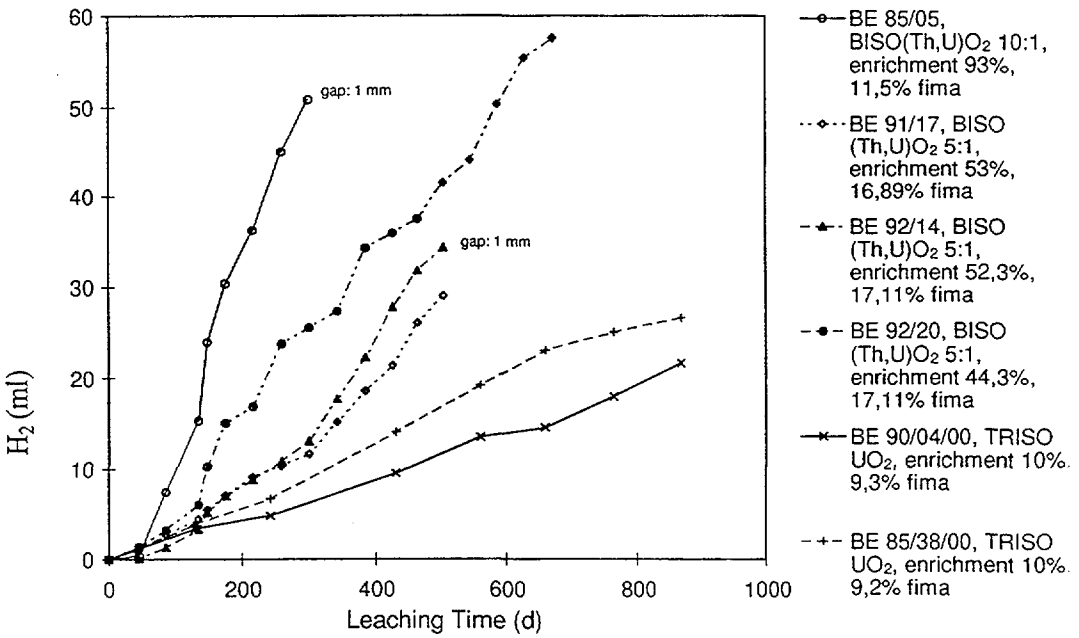

FIG. 18: Hydrogen build-up

Gas-chromatographic analyses proved, that the raising pressure was caused by hydrogen. Helium and tritiated hydrogen could be detected in trace amounts only, ^{85}Kr and ^{14}C were never found in the gas plenum.

The formation of hydrogen was not significantly influenced by the thickness of the brine layer. This led to the theory that the water radiolysis mainly depends on degradation on the surface and in the pores of the graphite matrix by α- and β-radiation rather than γ-radiolysis in the gap. The next step will now be the development of a model to prove this theory by calculation.

^{14}C was not found in the gas phase but dissolved in the brine. However, the main part of the dissolved ^{14}C occurs not as inorganic compound. The organically bound ^{14}C is considered as C_1-compounds, but this could not be proved because of the low amount. Fig. 20 and Fig. 19 show the fractional release of inorganic and organic ^{14}C for the different fuel element types.

However the total release is very low and the leaching rate decreases to nearly zero with time. Due to the rather short half-life of about 5000 years, ^{14}C may therefore be no major problem for long-term safety of a well-designed deep repository.

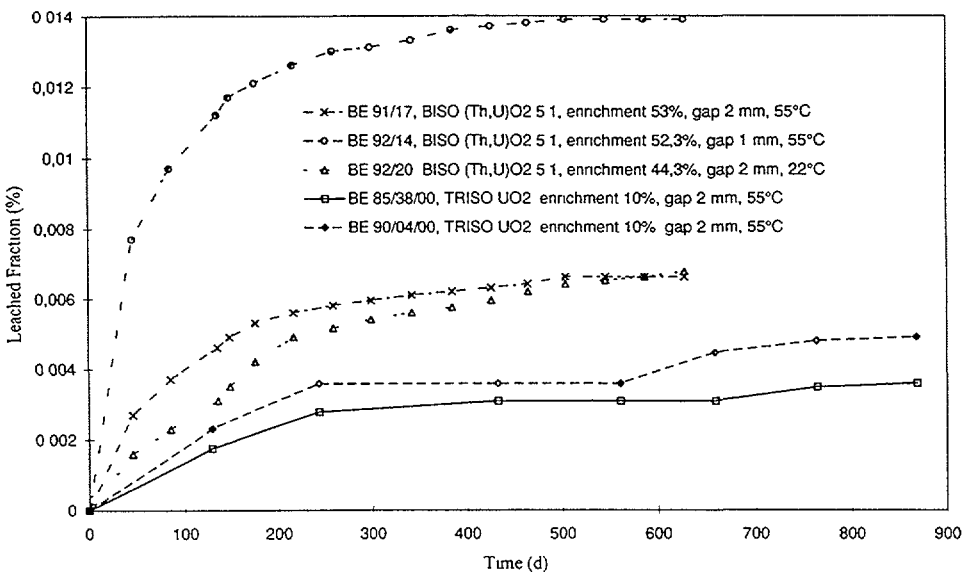

Fig 19: Fractional release of organic ^{14}C

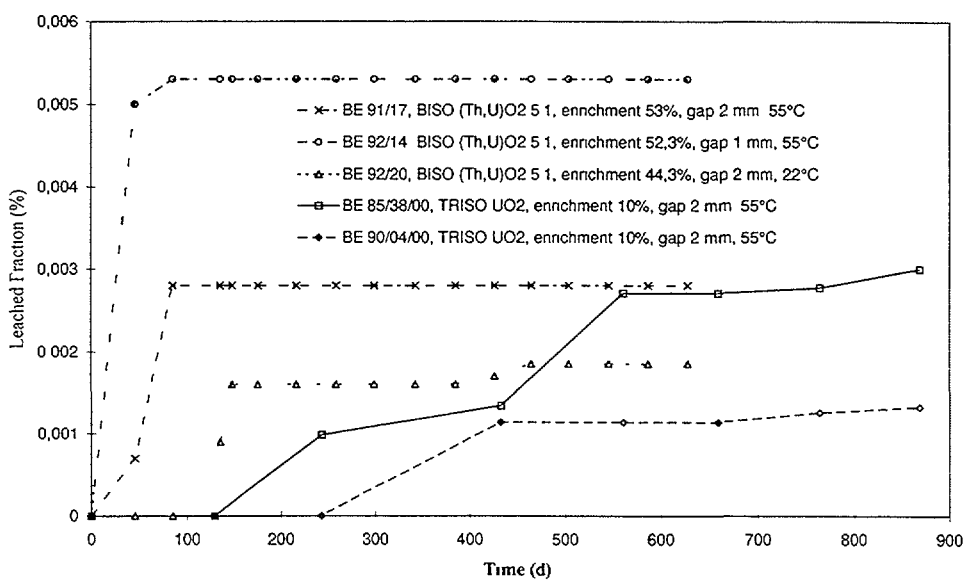

Fig. 20: Fractional release of inorganic ^{14}C

4 Conclusions

The back-end of the fuel cycle concept for spent High-Temperature Reactor fuel elements in Germany is based upon intermediate dry storage in shielded casks in a surface facility followed by direct disposal in a deep salt repository. Depending on the disposal technique a simple conditioning may be carried out prior to disposal.

Long-term experiments have proven, that HTR fuel elements can safely be stored in dry casks. Only trace amounts of volatile or aerosol-bound radionuclides were found to be released during storage, which represent no risk for the public or the environment. Two facilities to store spent HTR fuel in dry CASTOR-type casks are being operated in Jülich and Ahaus.

Disposal concepts assume the emplacement of spent HTR fuel elements in thick-walled casks in horizontal drifts, or in thin-walled containers in boreholes. In both cases, the ceramic fuel element itself represents the main technical barrier against the long-term release of radionuclides, if the waste disposed off comes into contact with water at all. Leaching experiments have proven that only extremely low amounts of radionuclides are released from the graphite matrix. The release from the coated particles is extremely low and results mainly from defect coatings. Defects in the coating are known to be very low and depend upon type of coating and burn-up. Hence, the fuel elements are well-suited for disposal in a salt repository.

However, there are some questions left: How stable is the particle coating against mechanical and chemical impacts in the long-term run? Does a changing pressure during disposal have a major influence on the release of radionuclides from the graphite matrix? These questions should be further investigated in order to optimise the concept in terms of safety and economics.

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EXPERIENCE WITH THE INTERIM STORAGE OF SPENT HTR FUEL ELEMENTS AND A VIEW TO NECESSARY MEASURES FOR FINAL DISPOSAL

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Abstract

In the Federal Republic of Germany the AVR pilot high-temperature reactor was operated successfully for more than 20 years and the THTR prototype high-temperature reactor for more than three years The reactors were shut down for decommissioning at the end of 1988 and the discharge of core inventories and packaging of the fuel, together with the temporarily stored fuel, for long-term interim storage in appropriate casks and facilities was started in 1992 and finished in 1995 for the THTR and began in 1994 for the AVR and will be completed at the beginning of 1998.
With a view to the long-term interim storage and final disposal of spent HTR fuel from both reactors many experiments have been carried out to characterize the spent fuel and to learn about its behaviour and during the operating period of the AVR reactor much experience has been gathered by remote handling, shipping and temporarily storing fuel packages at different appropriate facilities of the Forschungszentrum Jülich GmbH (FZJ). Furthermore, after starting the discharge of the AVR core more than 200 so-called AVR dry storage canisters (AVR-TLK), each containing 950 spent fuel elements have been reloaded from an AVR single shipping cask into CASTOR THTR / AVR shipping and storage casks in the hot cell facility, which is one part of the waste treatment and storage building of FZJ, and currently about 100 CASTOR casks, each containing in all 1900 fuel elements, have been prepared and stored in the AVR interim storage facility (AVR-BL), as another part of this building.

1. INTRODUCTION

In the Federal Republic of Germany the AVR pilot high-temperature reactor was operated successfully for more than 20 years and the THTR prototype high-temperature reactor for three years. During operation they were charged with several types of spherical graphite fuel elements, containing different U/Th mixtures such as coated HEU or LEU fuel particle dispersions. About 300,000 AVR and 620,000 THTR fuel elements were irradiated during the operating times. THTR spent fuel was temporarily stored on site and AVR spent fuel was temporarily stored at different hot cell and pool facilities of the Forschungszentrum Jülich GmbH (FZJ).

During the long operating period of the AVR reactor a lot of R&D work was carried out by FZJ to characterize the different types of spent fuel elements for developing interim storage and final disposal concepts /1, 2, 3/ and as part of this work much experience has been gathered by using spent fuel elements for experimental set-ups and by handling, shipping and temporarily storing fuel packages at different appropriate facilities of FZJ.

At the end of 1988 the reactors were shut down for decommissioning and discharge of core inventories and packaging of the fuel, together with the temporarily stored fuel for long-term interim storage in appropriate casks and facilties was started in 1992 and finished in 1995 for the THTR and began in 1994 for the AVR and will be completed at the beginning of 1998.

At the Ahaus facility, 305 casks, each loaded with canisters containing 2100 spent THTR fuel elements have been managed and stored by the Brennelement-Zwischenlager Ahaus GmbH. At the Jülich facility currently 100 CASTOR casks, each loaded with two AVR-TLK containing in all 1900 spent AVR fuel elements, have been prepared and stored by FZJ.

2. OVERVIEW OF SPENT AVR FUEL MANAGEMENT

By the end of 1988 about 190,000 spent fuel elements had been discharged during the reactor operating period and were packaged and shipped to FZJ. After granting the licences according to the Atomic Energy Act for discharging the core inventory for decommissioning by AVR GmbH and for

handling and long-term interim storage by FZJ, work began in August 1993 on managing fuel from core discharging by means of so-called AVR cans (AVR-K), each containing 50 fuel elements, and from different FZJ facilities for fuel reloading from AVR-K into AVR-TLK as well as charging of CASTOR casks for interim storage in the AVR interim storage facility (AVR-BL).

At that time about 84,000 fuel elements packaged and sealed in AVR-K were stored in the water cooling facilities of the Hot Cell (HZ) and the Research Reactor (FR) Departments, about 106,000 HEU fuel elements enclosed in AVR-TLK were stored in the LZ storage cell of the AZ hot cell facilities which is one part of the waste treatment and storage building of the Decontamination Department (DE), and about 110,000 fuel elements were still in the AVR reactor core. The paths of the AVR fuel elements from the reactor core to the AVR-BL are shown in *Figure 1.*

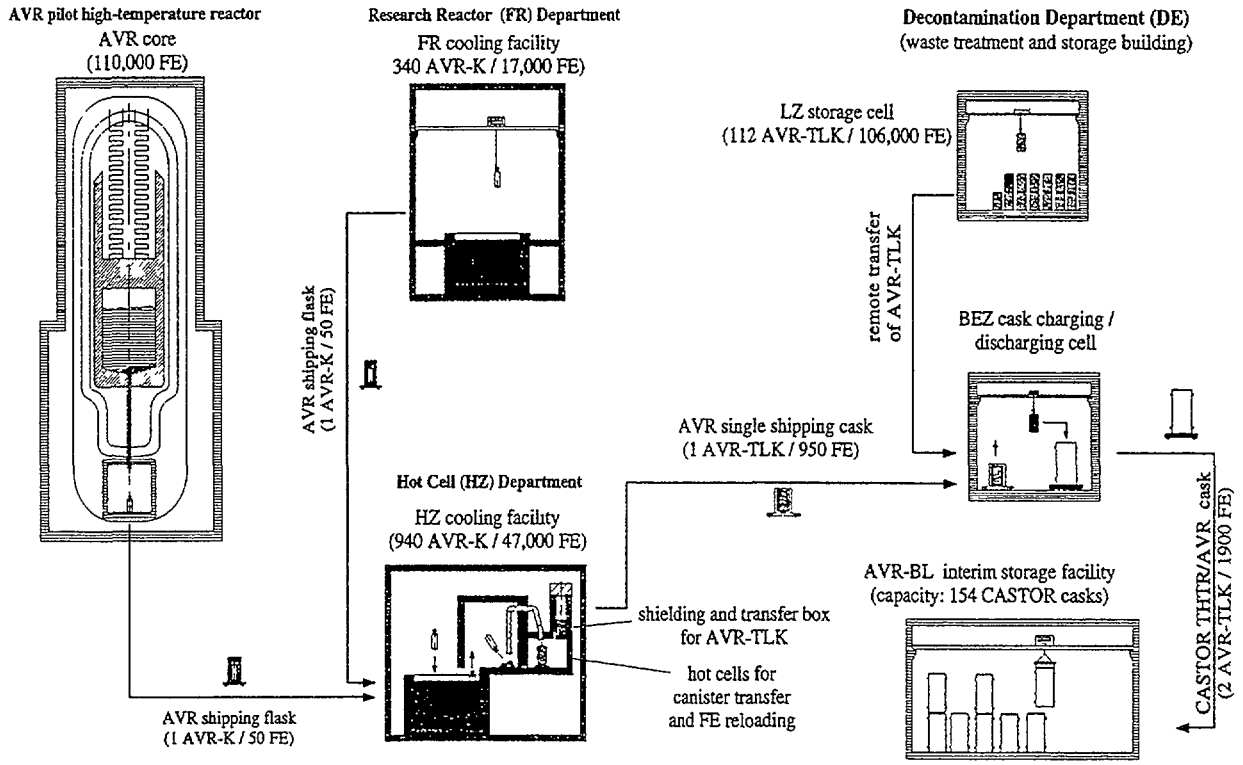

FIG. 1: Paths of the AVR fuel elements from the AVR reactor to the AVR interim store (fuel element inventories(FE) in the different facilities given by the end of 1993)

From August 1993 up to the present time about 91,600 fuel elements have been discharged from the reactor core, enclosed and shipped by means of AVR-K to HZ and reloaded there into AVR-TLK. About 65 AVR-TLK have been removed from the LZ and about 50,000 fuel elements enclosed in AVR-K have been removed from the above-mentioned water cooling facilities and reloaded into AVR-TLK so that in all 100 CASTOR THTR/AVR casks have been prepared and stored in the AVR-BL.

The delay in AVR core discharging in comparison to initial planing is caused by disturbances and failures of components from the different facilities and the equipment necessary for discharging the fuel and handling and reloading fuel packages and additionally, due to of problems, which arose at the beginning of 1995 with the liscensing procedures for LEU fuel handling and reloading in the HZ Hot Cell Department as well as handling LEU fuel packages in the DE Decontamination Department.

3. AZ HOT CELL FACILITY FOR HANDLING FUEL CANISTERS AND SHIPPING BAY FOR PREPARING AND ASSEMBLING CASTOR THTR / AVR CASKS

3.1 Preparing of CASTOR casks for charging

Preparing of CASTOR casks for charging is carried out in the shipping bay, which is part of the hot cell facility (AZ) and which covers the hot cells. Apart from a 50 Mg bridge crane for handling heavy

loads, whose range of operation covers the whole shipping bay area, a 5 Mg crane is installed above the the so-called mounting area for handling CASTOR lids (*FIG 2*).

Preparation of the sealing systems of casks, i.e. visual inspection, cleaning and if necessary, manual refurbishing of sealing groove surfaces of lids, sealing surfaces of casks as well as of the metallic gaskets, is carried out by means of the lid tilting device and the assembly station, which are installed in the mounting area. Positioning of casks onto the flat-bed cargo trailer, which is part of the assembly station, is carried out by means of the 50 Mg bridge crane.
After inspection and refurbishing work the metallic gaskets are fixed in the sealing grooves, the cask is closed with the primary lid and shipped into the BEZ cask charging / discharging cell.

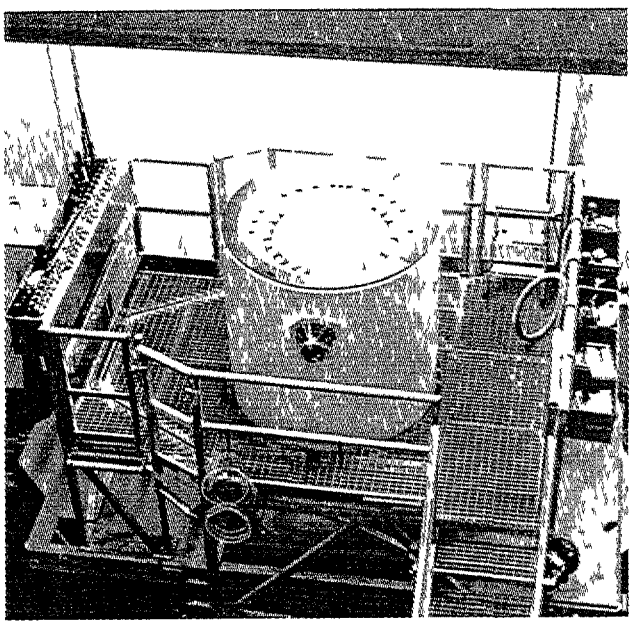

FIG 2 View onto the assembly station with a CASTOR cask positioned on the flat-bed cargo trailer

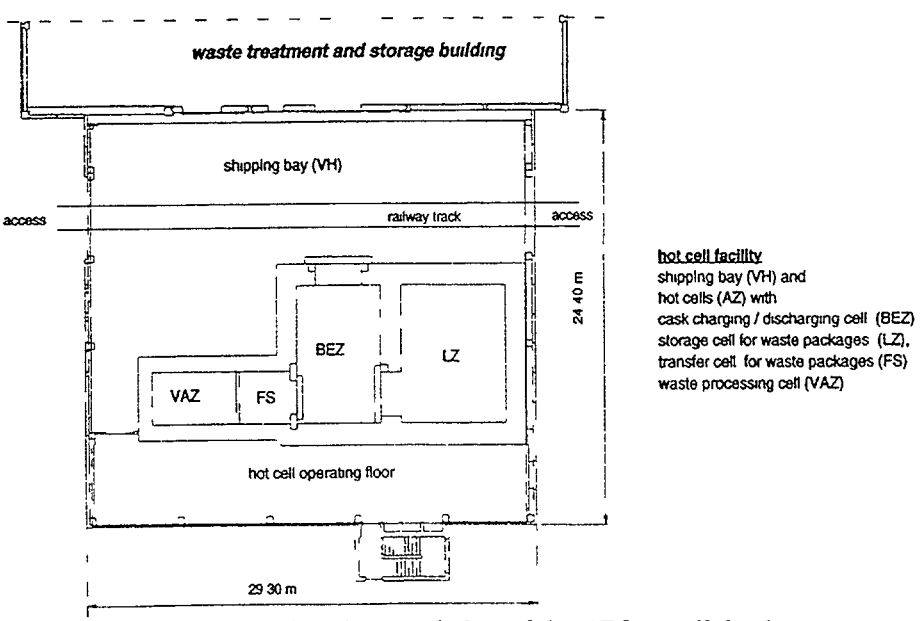

FIG 3 Simplified ground plan of the AZ hot cell facility

3.2 Charging of CASTOR casks in the BEZ hot cell

Before remote charging of two AVR-TLK and closing of a CASTOR THTR/AVR shipping and storage cask, AVR-TLK must be shipped and discharged from the AVR single shipping cask or transferred from the LZ storage cell and lowered into lay-down positions in the BEZ cask charging / discharging cell of the AZ hot cell facility, which is accessible from the shipping hall by means of the so-called BEZ shielding gate (*FIG. 3*) . For remote handling of waste drums and heavy loads of up to 4 Mg a power manipulator with drum tongs and a hook is installed in the BEZ (*FIG. 4*).

For handling of AVR-TLK a special pintle grapple and for handling the CASTOR primary lid a coupling link can be attached to the hook. Furthermore, the manipulator is equipped with a laser positioning system for accurate lifting and lowering of the CASTOR primary lid.

During the entire handling, charging and closing procedure the CASTOR cask remains on the flat-bed cargo trailer, which is equipped with a removable scaffold framing the cask and enabling access to the top of the cask. After the primary lid is in place, the low radiation level allows opening of the BEZ shielding gate for radiation protection measures and for preliminary tightening of primary lid's screws and for shipping the cask back to the assembly station in the shipping bay

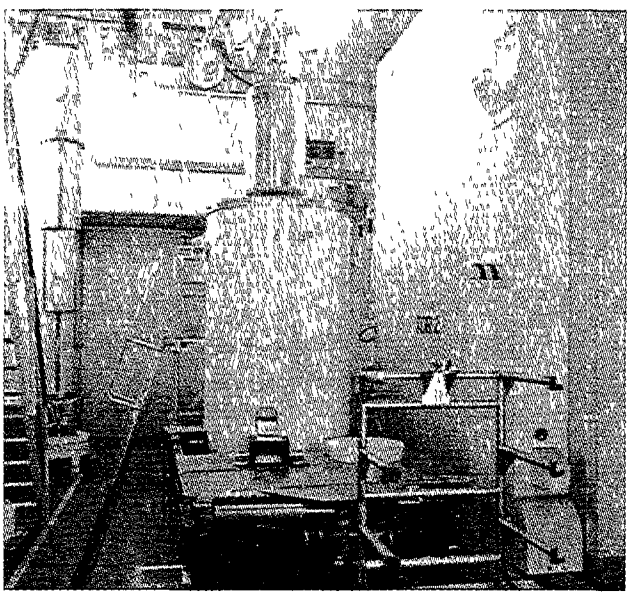

FIG. 4: Remote charging of a CASTOR cask in the BEZ cask charging / discharging cell

3.3 Assembling and leak testing procedures

After the CASTOR cask has been transferred back to the assembly station the assembling and leak testing procedures are as follows:

- tightening of the primary lid's screws
- evacuation of the cask and replacement of the withdrawn gas by an Ar /He-gas mixture
- He leak testing of the primary sealing system
- inserting the secondary lid
- tightening of the secondary lid's screws
- evacuation of the space between the lids and pressurizing the space with He gas
- He leak testing of the secondary sealing system
- mounting and He leak testing of a pressure gauge
- covering the top with the protective lid
- mounting the VACOSS seal on the protective lid
- removal of the scaffold and shipment to the AVR-BL interim storage facility

4. THE AVR-BL INTERIM STORAGE HALL

The AVR-BL interim storage facility has been built and licensed according to the Atomic Energy Act (§6 AtG) for the interim storage of spent AVR fuel elements which have been irradiated during the operating period of the AVR pilot reactor and which have to be enclosed in CASTOR THTR/AVR shipping and storage casks. Lay-out of the storage area will serve for the interim storage of 154 casks, which are stacked alternately on one and two levels (FIG. 5 and 6)

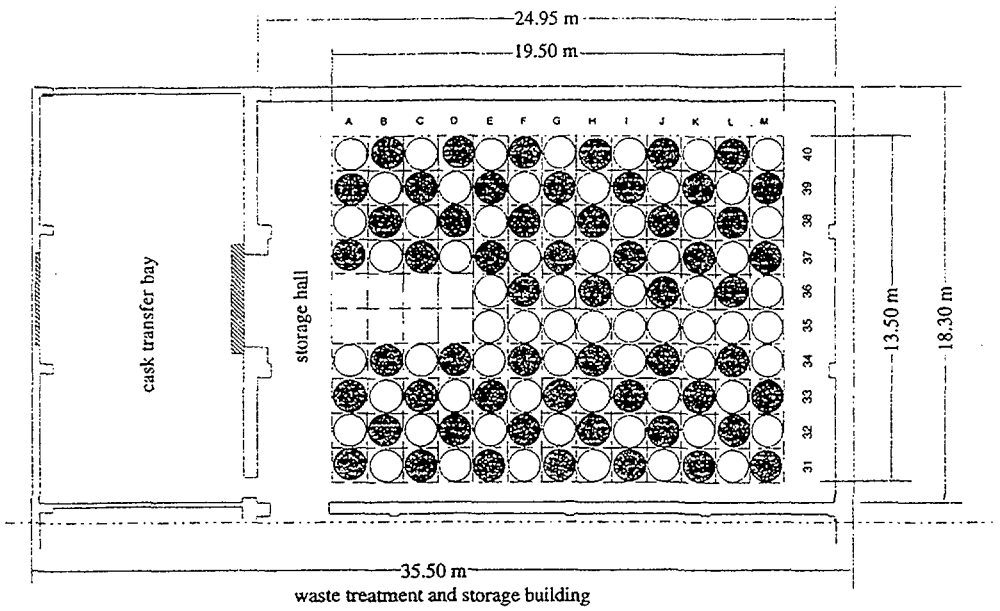

FIG .5: Scheme of the arrangement of CASTOR casks in the AVR-BL interim storage facility as another part of the waste treatment and storage building

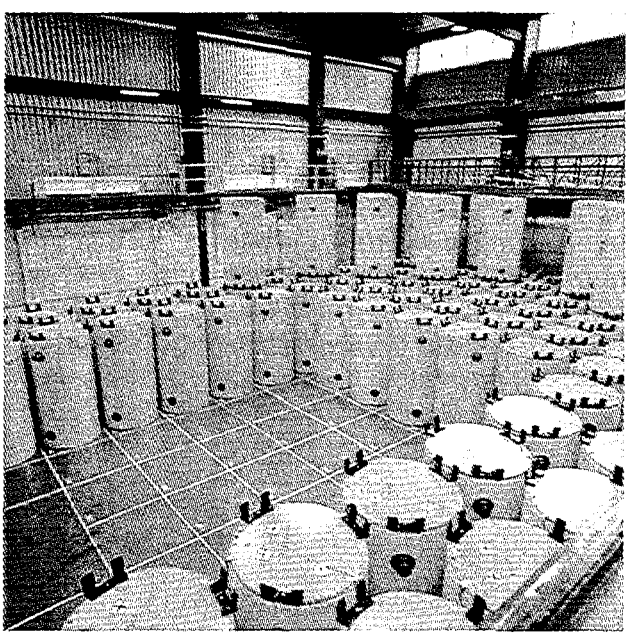

FIG. 6: View into the AVR-BL storage hall

5. SAFETY CONCEPT FOR INTERIM STORAGE

The safety concept for the interim storage of AVR spent fuel elements is based, in particular, on design requirements for CASTOR THTR/AVR shipping and storage casks as a tight enclosure so that any undue release of radionuclides is excluded both in normal operation and under conceivable accident conditions. According to design, the sealing function of both lid sealing systems is monitored during storage so that any deterioration or failure of a sealing barrier is detected and repair measures for restoring the two-barrier system can be carried out in the AZ hot cell facilities.

Within the Atomic Energy Act licensing procedures (§6 AtG) for the AVR and the BZA interim storage facility a leakage rate of $L \leq 10^{-7}$ mbar × l / s for each lid sealing sytem of the CASTOR casks is required.

Apart from the results obtained on the basis of long-term experiments with comparable lid sealing set-ups by FZJ /4, 5/ and other institutions /6, 7, 8/, which confirm the good long-term behaviour of such sealing systems with respect to the design requirements, confirmation of the required and specified tight enclosure of the spent fuel canisters has also been provided by the experience gathered from loading and preparing more than 400 CASTOR THTR/AVR transport and storage casks at the Jülich and Ahaus sites.

6. VIEW TO NECESSARY MEASURES FOR FINAL DISPOSAL

According to the plans of the Bundesamt für Strahlenschutz (BfS), solid and solidified radioactive waste forms, but in particular those with marked decay heat generation shall be disposed of in a final repository in a salt dome formation /9/. Heat-generating waste includes spent HTR fuel elements which are not to be reprocessed.

By the end of 1992 R&D work in establishing a final disposal concept for HTR fuel was focused on small 400-l fuel packages to be emplaced in 300-m deep boreholes in the final repository still to be constructed and then ultimately confined. Most of the work was discontinued at the end of 1992 /10, 11,12/.

The BfS subsequently gave preference to a final disposal concept for HTR fuel oriented along the lines of the direct disposal concept for irradiated LWR fuel elements, which is based on packaging the fuel in so-called POLLUX shipping and final disposal casks /13/.

Due to the fact that the design features, radioactive inventories and long-term behaviour of LWR fuel is completely different from those of HTR fuel further studies on the suitability of the CASTOR THTR/AVR cask to establish an adequate final disposal concept for HTR fuel have been currently initiated and will be carried out by Forschungszentrum Jülich GmbH in cooperation with Gesellschaft für Nuklear-Behälter mbH (GNB).

7. CONCLUSIONS

At the present time handling, reloading and packaging of fuel elements from AVR reactor core discharge, in addition to handling and reloading of some thousand fuel cans (AVR-K) and some hundred fuel canisters (AVR-TLK) in different facilities of FZJ as preparatory steps for charging CASTOR casks, has been conducted in a safe manner according to the requirements of the Radiation Protection Ordinance (StrlSchV).

Extensive cold testing of the equipment, training of the personnel responsible for charging and gas-tight closure of the CASTOR casks before starting hot operation and feedback of experience in addition the experience gathered by charging 100 CASTOR casks, has led to safe routine handling without the occurence of non-normal events.

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CASTOR THTR TRANSPORT/STORAGE CASKS

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Abstract

For the management of spent fuel from nuclear power plants, two possibilities are available in Germany. One possibility is the reprocessing of the spent fuel and the realization of a socalled closed nuclear fuel cycle, the other is the direct disposal after a period of interim storage, without reprocessing. For the German GCR plants "THTR 300" and "AVR", only the way of direct disposal is available to date for managing the spent fuel (pebble-bed fuel). For the period of interim storage, dry storage in casks was selected.

Development

In order to insure the prompt availability of such casks for interim storage, the development of the CASTOR THTR/AVR (see Fig. 1) began in 1982. The design of this cask was based on experience gained in the application of ductile cast iron (GGG 40) for the manufacture of CASTOR transport and storage casks for radioactive materials.

Within the scope of development and licensing, a drop test of a prototypecask without shock absorbers onto a yielding foundation was performed, in addition to the usual analyses - required by the IAEA-Regulations - necessary to obtain transport license. The reason for this additional drop test was that the casks are not stored standing alone, but with two casks one stacked on top of the other (see Fig. 2). This saves half the number of storage positions in the storage facility.

All analyses and test results have shown that the CASTOR THTR/AVR goes far beyond simply meeting the requirements placed on it.

On the basis of the safety analysis report, and the tests performed the transport license was issued in 1987. The storage license for BZA was issued in 1992 and for AVR in 1993.

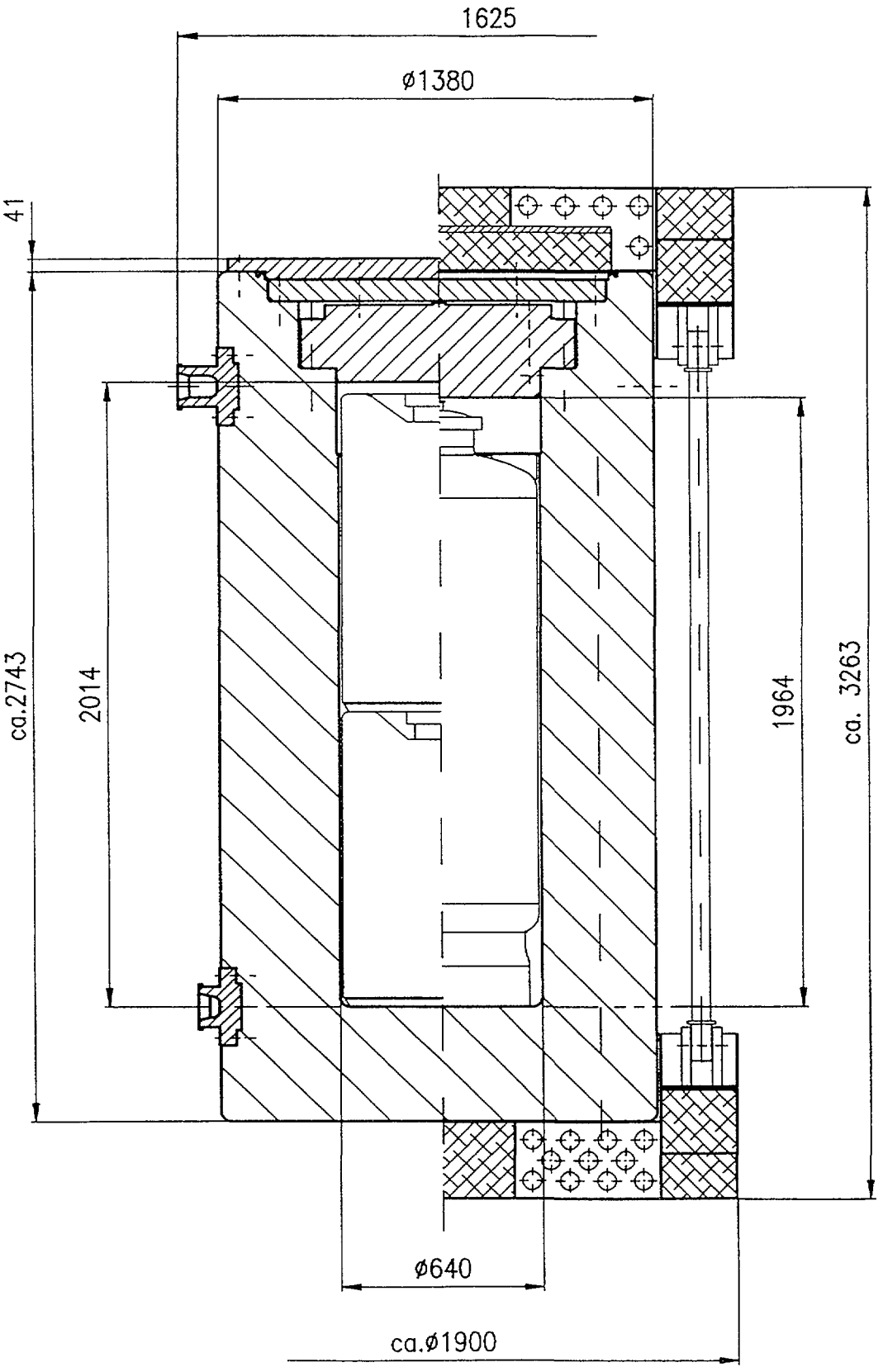

FIG. 1. CASTOR THTR/AVR

FIG. 2. BZA-Interim storage area for CASTOR THTR/AVR

CASTOR - Cask

The CASTOR THTR/AVR cask consists of a thick-walled cylindrical body which is closed with two lids, the primary and secondary lid, as well as a protection plate. For handling, the cask is equipped with two trunnions respectively at the top and bottom ends of the cask.

The primary lid is made of forged carbon steel (TStE 355) and is bolted with 28 bolts to the cask body. The primary lid is sealed off with a metal and an elastomer gasket. In the primary lid, one penetration is located for performing the necessary leak-tightness tests. This orifice is closed with a small lid (flange) and a metal gasket.

The secondary lid is made of carbon steel plate (St 52-3) and is also bolted with 28 bolts to the cask body. The secondary lid is sealed with a metal and an elastomer gasket. There are two penetrationes in the secondary lid. One is used for leak testing

and setting the monitoring-pressure between primary and secondary lid. It is closed by a small lid (flange) with a metal gasket as well. The other is used for installation of the pressure monitoring system.

The protection plate is also made of carbon steel plate (St52-3) and is fastened over the primary and secondary lids with 20 bolts. It serves to protect the lid system from dust, moisture and mechanical influences, for example during handling.

The trunnions are made of forged carbon steel (TStE-355) and are connected in pairs at the top and bottom ends of the cask with 12 bolts each.

The outer surfaces of the cask, as well as primary and secondary lid are provided with a multi-coated decontaminable paint. The inner cavity is protected with a zinc-silicate coating. The trunnions are protected with a zinc-coating.

The outer dimensions are :

length :	2743 mm
diameter :	1380 mm

The inner dimensions are :

length :	1964 mm
diameter :	640 mm

The overall weight is approx. 28 t (incl. shock absorbers)

Cask Contents

The loading of the cask is performed differently in the two GCR-plants. For the THTR 300, the cask is loaded with one fuel canister containing approx. 2100 fuel elements. For the AVR, the cask is loaded with 2 fuel canisters, which together accommodate approx. 1900 fuel elements.

Fabrication

Fabrication of the cask began in 1987. From 1987 to 1996 a total of 463 casks of the type CASTOR THTR/AVR were made, with 150 casks being manufactured in the years 1991 and 1992 respectively.

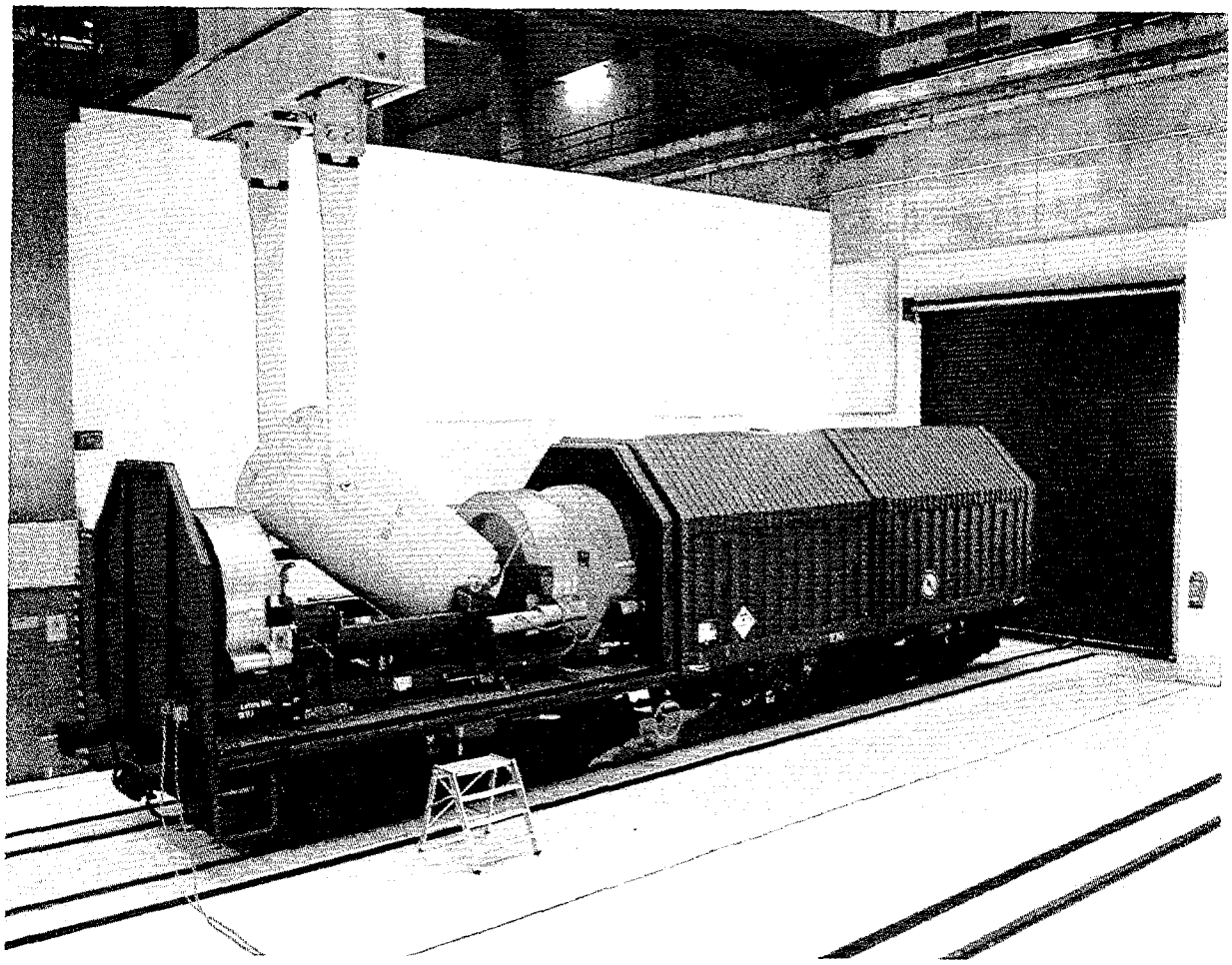

FIG. 3. Transportsystem of CASTOR THTR/AVR

Transport / Intermediate Storage

For the transport of the casks, both from the manufacturing plant to the THTR 300 as well as from there to the interim storage facility in Ahaus (BZA) special rail wagons were developed and built (see Fig. 3). Three of these transport units were finished in 1988 and commissioned. Each unit can accommodate 3 CASTOR THTR/AVR casks in special transport frames with the shock absorbers, which are integrated into the structure of the wagons. For reduction of the dose exposure of personnel during loading and unloading of the wagon, all handling operations would be done by remote control.

During the time from June 1992 to April 1995, a total of 305 CASTOR THTR casks were transported in 57 shipments to BZA and placed into interim storage. The casks contain all the fuel elements which were used in the THTR 300.

For the interim storage of the AVR fuel elements, a facility was built at the site in Jülich. In the time from August 1993 to date, 96 CASTOR THTR/AVR casks have been put into storage there.

In both storage facilities, the leak-tightness of the casks is permanently monitored by a pressure monitoring system in order to guarantee safety througout the period of interim storage.
This system monitors the pressure (approx. 6 bar) between the primary and secondary lid. A loss of pressure down to a pre-set level (approx. 3 bar) is indicated by an alarm.

Finale Storage

On the basis of the current status of planning at the BfS, a final disposal of spent fuel assemblies in a salt dome can be assumed.

The casks to be used for this purpose must withstand the rock pressure of the salt dome during its operation and remain leak-tight during this time.

Estimations of the pressure resistance of the CASTOR THTR/AVR-cask body and of the lid system have shown that these casks are able to withstand the rock pressure to be expected in the final storage site without failure.

The leak-tightness is guaranteed by the existing sealing system provided that no alkaline salt solution reaches the sealing system. In order to ensure this safely, it is possible to close the CASTOR THTR/AVR in the top area with a lid welded to the cask body, so that it is leak tight against alkaline solutions.

The feasibility of such a welded connection in a cask body of ductile cast iron has been confirmed by preliminary tests within the scope of the POLLUX-final-disposal-project. Within the scope of these tests, a cask lid of ductile cast iron was manually welded to a cask body.

The analyses aimed at ensuring this concept will begin shortly.

STUDY ON STORAGE AND REPROCESSING CONCEPT OF THE HIGH TEMPERATURE ENGINEERING TEST REACTOR (HTTR) FUEL

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Abstract

The Japan Atomic Energy Research Institute (JAERI) is proceeding with the construction of High Temperature Engineering Test Reactor (HTTR), which is the first High Temperature Gas-cooled Reactor in Japan. The first criticality of the HTTR will be attained in the end of 1997. A fuel assembly of the HTTR is so-called pin-in-block type, which consists of fuel rods and a hexagonal graphite block. A fuel rod contains fuel compacts in which coated particles are dispersed. The coated fuel particle consists of a microsphere of low enriched UO_2 with the TRISO coating. All fuel assemblies are discharged from the core after about three years operation. The spent fuels are transferred to air-cooled spent fuel storage racks after two years cooling in the fuel storage pool in the reactor building. According to basic Japanese concept, all spent fuel shall be reprocessed, although the coated fuel particle is considered to be able to retain fission products during long period. Then a calculation study has been carried out to investigate the fission products release behavior from the HTTR fuel during long-period repository and/or disposal. In addition, some head-end reprocessing techniques have been investigated to apply the conventional Purex process to the HTTR fuel. JAERI studied graphite-CO_2 reaction and jet grind method as a head-end reprocessing technique. In fuel fabrication process, burn-crush-leach method is employed to recover uranium from the fuel compacts. This experience is supplying useful data to investigate the head-end reprocessing method. This report describes the fuel storage system of the HTTR and present status of the storage and reprocessing study in JAERI.

I. INTRODUCTION

The Japan Atomic Energy Research Institute (JAERI) is proceeding with the construction of High Temperature Engineering Test Reactor (HTTR), which is the first High Temperature Gas-cooled Reactor (HTGR) in Japan. The first criticality of the HTTR will be attained in the end of 1997. The HTTR has been so designed as to be an engineering test reactor which aims to establish and upgrade the technological basis for advanced HTGRs and to conduct various irradiation tests for innovative high temperature basic researches and various modes of operation and test for advanced HTGRs [1].

The HTTR plant, composed of a reactor building, a spent fuel storage building, a machinery building and so on, is constructed in the Oarai Research Establishment of the JAERI. The reactor building is centered in the HTTR plant. The main reactor facilities of the HTTR such as a reactor pressure vessel, a primary cooling system, a reactor containment vessel and a refueling machine are housed in the reactor building as illustrated in ***Fig. 1***. The reactor pressure vessel is 13.2 m high and 5.5 m in diameter, and contains the core of 30 MWt. The main cooling system is composed of a primary cooling system, a secondary helium cooling system and a pressurized water cooling system. The primary cooling system has two heat exchangers, an intermediate heat exchanger and a primary pressurized water cooler, in parallel. The major specifications of the HTTR are summarized in ***Table 1***.

According to basic Japanese concept, all spent fuel shall be reprocessed, although the coated fuel particle is considered to be able to retain fission products during long period. Then a calculation study has been carried out to investigate the fission products release behavior from the HTTR fuel during long-period repository and/or disposal. In addition, some head-end reprocessing techniques have also been investigated to apply the conventional Purex process to the HTTR fuel [2]. This report describes the fuel storage system of the HTTR and present status of the storage and reprocessing study in JAERI.

II. FUEL STORAGE SYSTEM OF THE HTTR

2.1 Description of the HTTR Fuel

A fuel assembly of the HTTR is composed of fuel rods and a hexagonal graphite block as shown in ***Fig. 2***. Each coated fuel particle is a microsphere of low enriched UO_2 (about 6 % on the average) with TRISO coatings. The TRISO coatings consist of a low-density, porous pyrolytic

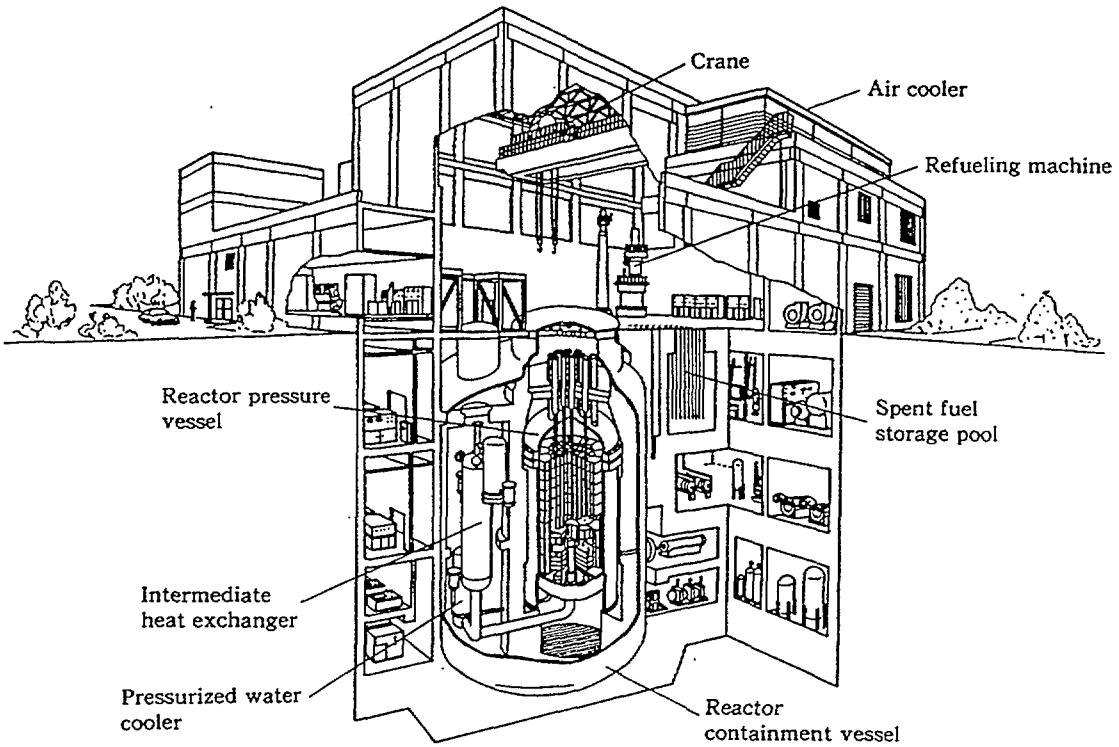

Fig. 1 Reactor building of the HTTR.

TABLE 1. MAJOR SPECIFICATION OF HTTR

Thermal power	30 MW
Outlet coolant temperature	850/950 °C
Inlet coolant temperature	395 °C
Fuel	Low-enriched UO_2
Fuel element type	Prismatic block
Direction of coolant-flow	Downward flow
Pressure vessel	Steel
Number of main cooling loop	1
Heat removal	Intermediate heat exchanger Pressurized water cooler
Primary coolant pressure	4 MPa
Containment type	Steel containment
Plant lifetime	20 years

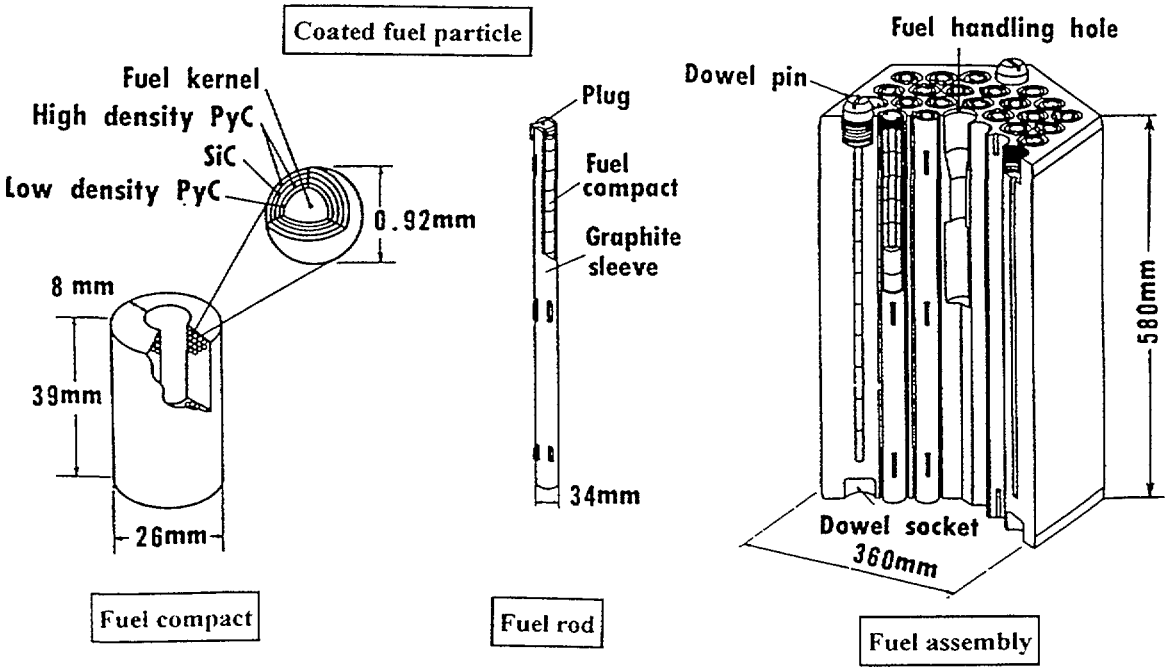

Fig. 2 Fuel assembly of the HTTR.

TABLE 2. SPECIFICATIONS OF THE HTTR FUEL

Fuel kernel	
Material	UO_2
Diameter	600 μm
Density	95 %T.D.
Coating layers	
1st layer material	Low density PyC
1st layer thickness	60 μm
2nd layer material	High density PyC
2nd layer thickness	30 μm
3rd layer material	SiC
3rd layer thickness	25 μm
4th layer material	High density PyC
4th layer thickness	45 μm

Fuel compact	
Materials	cfp, binder, graphite
Packing fraction	30 vol%
Outer/inner diameter	26/10 mm
Height	39 mm
Graphite sleeve	
Material	Graphite
O.D./thickness	34/3.75 mm
Length	580 mm
Fuel block	
Material	Graphite
Width of across flats	360 mm
Height	580 mm

carbon (PyC) buffer layer adjacent to the spherical fuel kernel, followed by an isotropic PyC layer, a SiC layer and a final (outer) PyC layer. The coated fuel particles are incorporated into a graphite matrix to form a fuel compact. The fuel rod, which is composed of the fuel compacts and a graphite sleeve, is introduced into a vertical bore hole of the hexagonal graphite block. The specifications of the coated fuel particle, the fuel compact, the graphite sleeve and the fuel block of the first-loading-fuel of the HTTR are listed in ***Table 2***. Helium gas flows downwards through an annular gap between the vertical hole and the fuel rod in order to remove heat produced by fission.

2.2 Fuel Storage System for the HTTR Fuel

Fuel treatment flow of the HTTR facility is shown in ***Fig. 3***. All fuel assemblies are discharged from the core after about three years operation (660 EFPD (Effective Full Power Days)). The maximum burnup is designed to be 33 GWd/t as a block average value [3]. Fuel handling and storage systems in the HTTR building are shown in ***Fig. 4***. The fuel handling machine moves new fuel assemblies from the new fuel storage cell to the reactor before operation. After about 3 years operation, the fuel assemblies are transferred to the spent fuel storage pool in the reactor building by the fuel handling machine.

The spent fuel storage system in the reactor building consists of a spent fuel storage pool, pool water cooling and purification system, etc. The spent fuel storage pool in the reactor building, which is fabricated of ferroconcrete and has sufficient shielding for personnel, includes 63 storage racks as shown in ***Fig. 5***. It can store spent fuel assemblies of about two core inventories. The spent fuel storage pool is lined inside the pool with stainless steel to prevent pool water leakage. When pool water leaks, the leakage can be detected by monitoring the water from the leakage check ditch which is located within the lining. The storage rack forms a vessel of a vertical cylinder with a shielding plug, and has sufficient distance to the adjacent storage racks to keep subcriticality even if the inside of the storage rack is filled with water. The pool water cooling and purification system removes the decay heat from the spent fuel assemblies by cooler. The maximum decay heat from the two core inventories of spent fuel assemblies is evaluated to be 55 kW and the cooler is designed to keep the water temperature below 65 °C.

The spent fuels will be transferred to air-cooled spent fuel storage racks in the spent fuel storage building after more than two years cooling in the fuel storage pool of the reactor building. The air-cooled spent fuel storage pool will store the spent fuel assemblies of about ten core inventories.

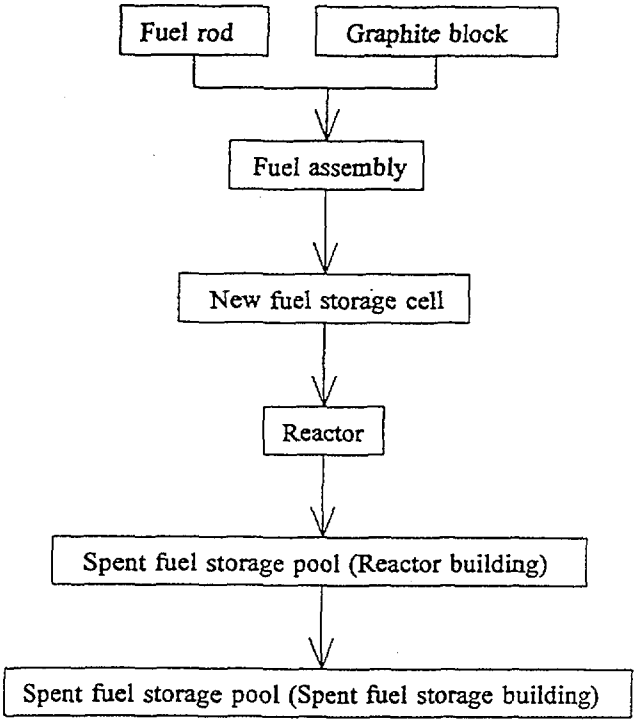

Fig. 3 Fuel treatment flow in the HTTR facility.

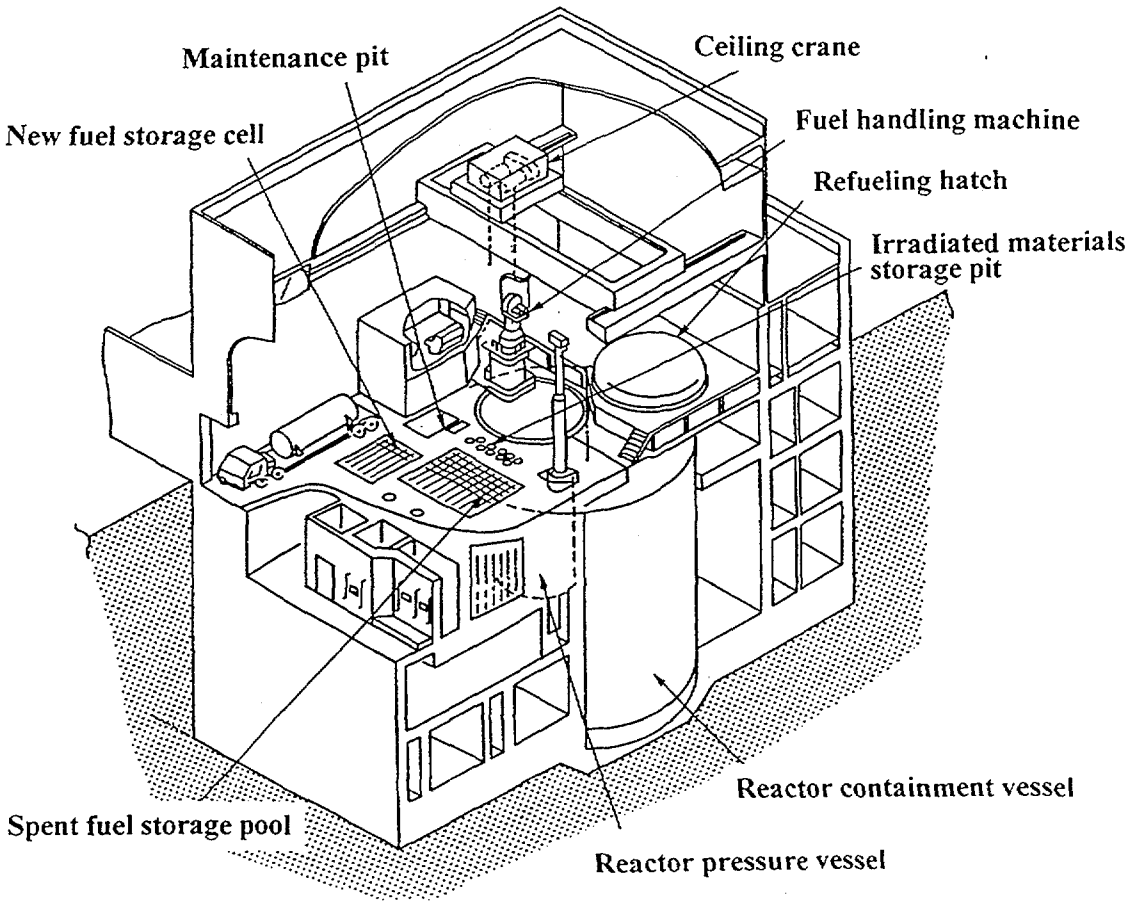

Fig. 4 Fuel handling and storage systems in the HTTR.

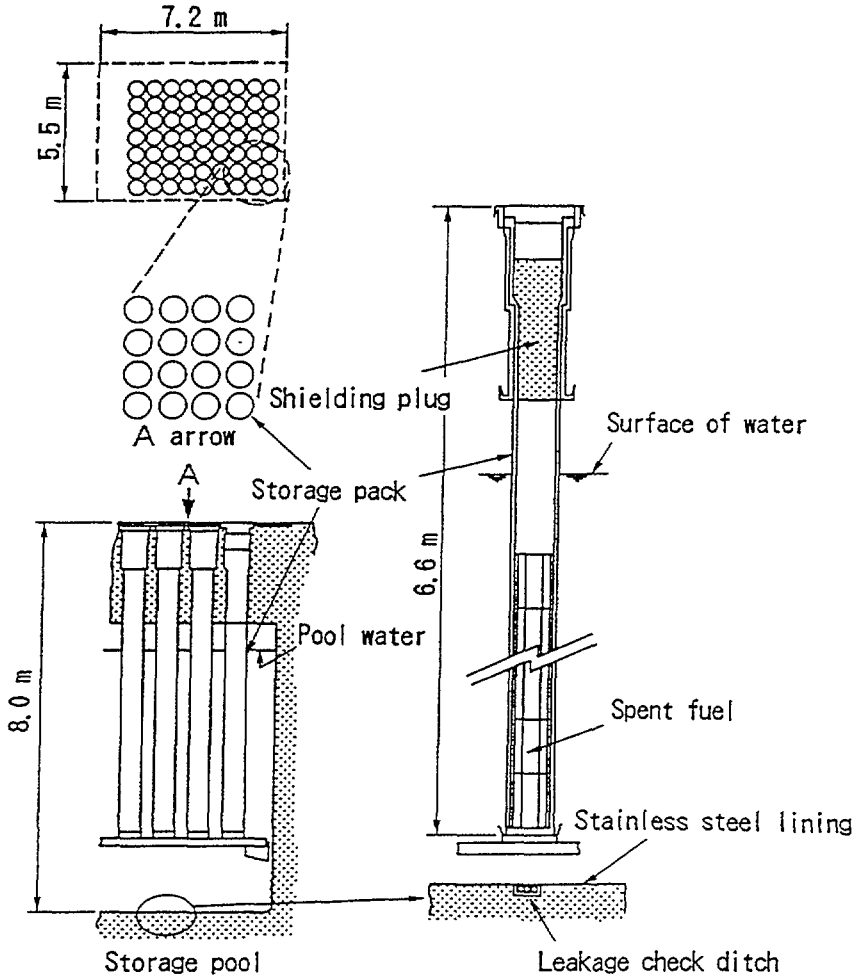

Fig. 5 Spent fuel storage system in the HTTR reactor building.

III. OPTIONS FOR TREATING HTTR FUEL

There are three options for spent fuel treatment, namely, (1) reprocessing, (2) long-period repository and reprocessing (intermediate storage) and (3) direct disposal [4]. According to basic Japanese concept, all spent fuels including the HTTR fuel shall finally be reprocessed. On the other hand, the HTTR fuel consists of small coated fuel particles of UO_2 with PyC and SiC layers, which serve as pressure vessels to contain fission products and fuel. Then, technically, the HTTR fuel should be treated by options (2) long-period repository and reprocessing or (3) direct disposal.

TABLE 3. QUANTITIES OF RADIOACTIVE NUCLIDES IN A HTTR FUEL ASSEMBLY

	Quantity (Bq/Fuel assembly)
Actinides (and daughters)	8.1×10^{15}
Fission products	3.5×10^{16}
Carbon-14	2.1×10^{9}
Tritium	8.0×10^{7}

A calculation study has been carried out to investigate the fission products release behavior from the HTTR fuel during (2) intermediate storage or (3) disposal, and the results are summarized in Section 3.1. For option (1), some head-end reprocessing techniques have been investigated to apply the conventional Purex process to the HTTR fuel. Section 3.2 describes review of head-end reprocessing technique in JAERI and experiences obtained in uranium recovery process of the HTTR fuel fabrication.

3.1 Feasibility study of long period repository and direct disposal

It is considered that the coated fuel particle acts as a miniature containment vessel during long period repository or direct disposal. However, there are few quantitative investigations to evaluate the fission gas retentiveness in the coated fuel particle. Then, a preliminary calculation study is carried out to evaluate the release fraction of fission gases from the coated fuel particle during long-period repository or disposal. The evaluated fission product inventory in a HTTR fuel assembly is shown in ***Table 3***. Considering 5 or more years repository, release fractions of long-lived fission gases, Kr-85 (half life=10.7 years) and I-129 (half life=1.6×10^{7} years), are important.

The release fractions from an intact particle and a through-coatings failed particle are calculated. The through-coatings failed particle is modeled as a bare kernel and the release fraction is given by the expression [5]:

$$F_k = 1 - 6\sum_{n=1}^{\infty} \frac{\exp(-n^2\pi^2 D' t)}{n^2\pi^2},$$

where F_k is the release fraction from the fuel kernel, and D' the reduced diffusion coefficient in the fuel kernel. The intact particle is modeled as an one-layer coating particle and the release fraction is calculated by the following equations [5].

$$F=1-6(1+\gamma)\sum_{n=1}^{\infty}\frac{\exp(-n^2\pi^2D't)}{n^2\pi^2((\gamma-\frac{x_n^2}{3\gamma})\frac{\sin x_n}{x_n}+\cos x_n)}-3\sum_{m=0}^{\infty}T_m(\frac{1}{\alpha^2y_m^2}-\frac{\cot\alpha y_m}{\alpha y_m})\exp(-y_m^2D^*t),$$

$$\tan y_m=\frac{3\gamma y_m}{y_m^2-3\gamma^2},\qquad y_m>0,$$

$$T_m=\frac{1+\gamma}{\cos y_m}\frac{1}{\frac{y_m^2}{6\gamma}+\frac{1}{2}+\gamma+\frac{3\gamma^2+4.5\gamma^3}{y_m^2-3\gamma^2}},$$

$$D^*=\frac{D_c}{(R_o-R_i)^2},$$

$$\gamma=\frac{R_o-R_i}{R_i}.$$

where $\alpha=(D^*/D')^{1/2}$, $x_n=n\pi/\alpha$, D_c is the diffusion coefficient in the coating layer, R_o and R_i are the outer and inner radii of the coating layer, respectively.

In the calculation, it is assumed that UO_2 kernel itself and the most retentive layer of SiC coating determine the release behavior of fission gases from the intact particle. The diffusion coefficients, (m^2/s), of $5.0\times10^{-6}\exp(-3.8\times10^4/T)$ for the UO_2 kernel [6] and $1.7\exp(-7.5\times10^4/T)$ for the SiC layer [7] are employed in the calculation (Here, T=temperature in Kelvin). The fuel temperature is conservatively assumed to be 600 ° C during repository. The result is shown in ***Fig. 6***. Krypton and iodine are completely retained in the intact particle during 10^{15} years. Form the through-coatings failed particle, which is modeled as a bare kernel, remarkable release starts beyond 1000 years.

3.2 Reprocessing procedures

(1) Review of head-end reprocessing technique in JAERI [2]

JAERI's study was focused on applicability of graphite-CO_2 reaction technique and jet grind method to head-end reprocessing. The block diagram of the JAERI's head-end reprocessing is illustrated in ***Fig. 7***. The graphite-CO_2 reaction technique was developed to reduce CO gas release in the burning process of the fuel compacts and fuel kernel. The jet grind method was investigated to reduce maintenance works of roll-gap clearance of the roll grinder, which was used to remove the SiC layer from burned SiC-particles. The feasibility of these methods was approved through laboratory scale experiments and the research and development work was terminated. Pilot scale tests will be carried out to confirm applicability of these methods to the HTTR spent fuel in future.

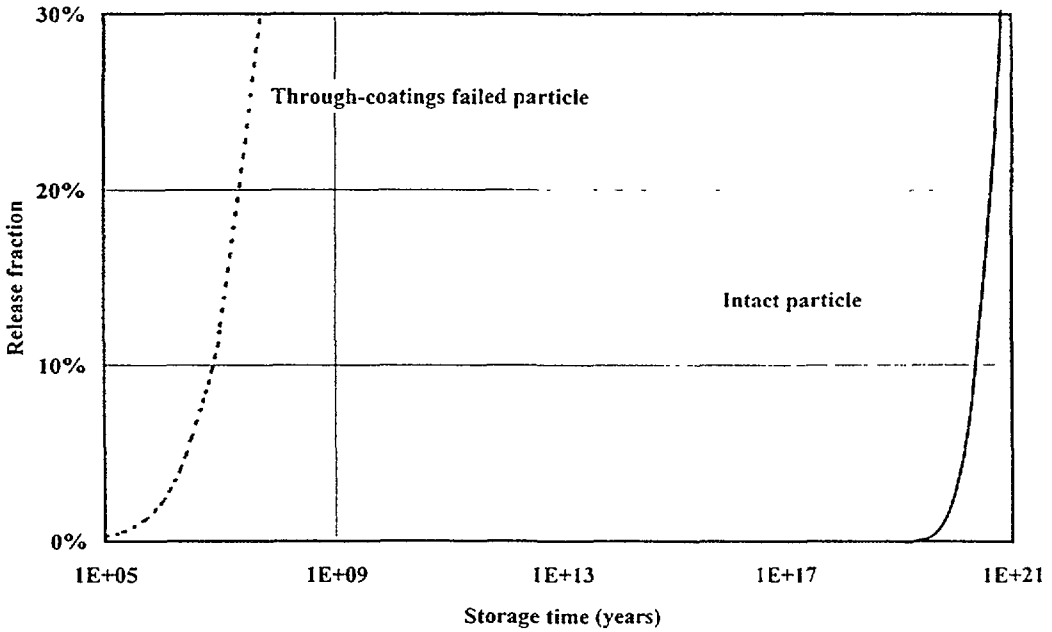

Fig. 6 Calculated result of release fraction of krypton and iodine from coated fuel particles.

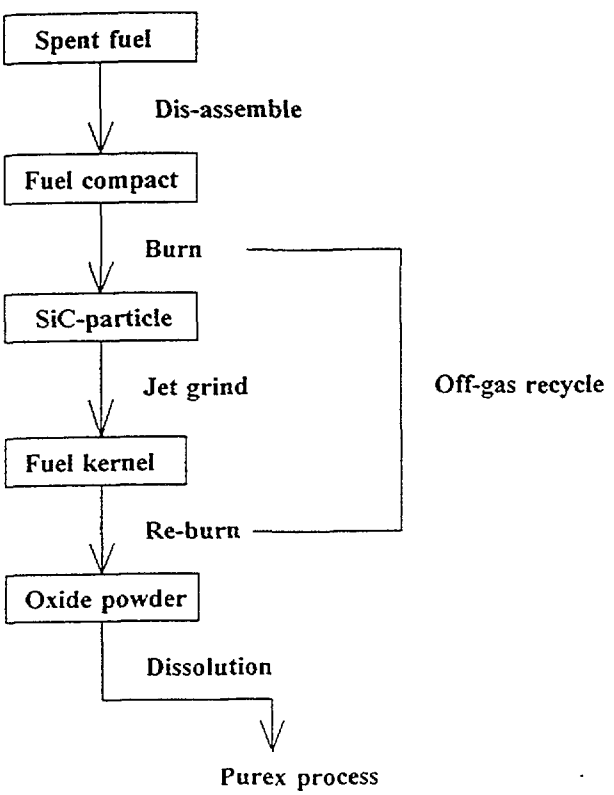

Fig. 7 Block diagram of JAERI head-end reprocessing.

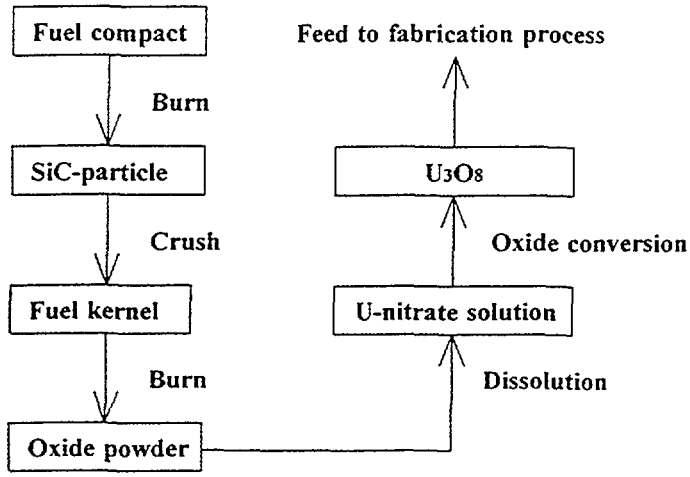

Fig. 8 Block diagram of uranium recovery in fuel fabrication process.

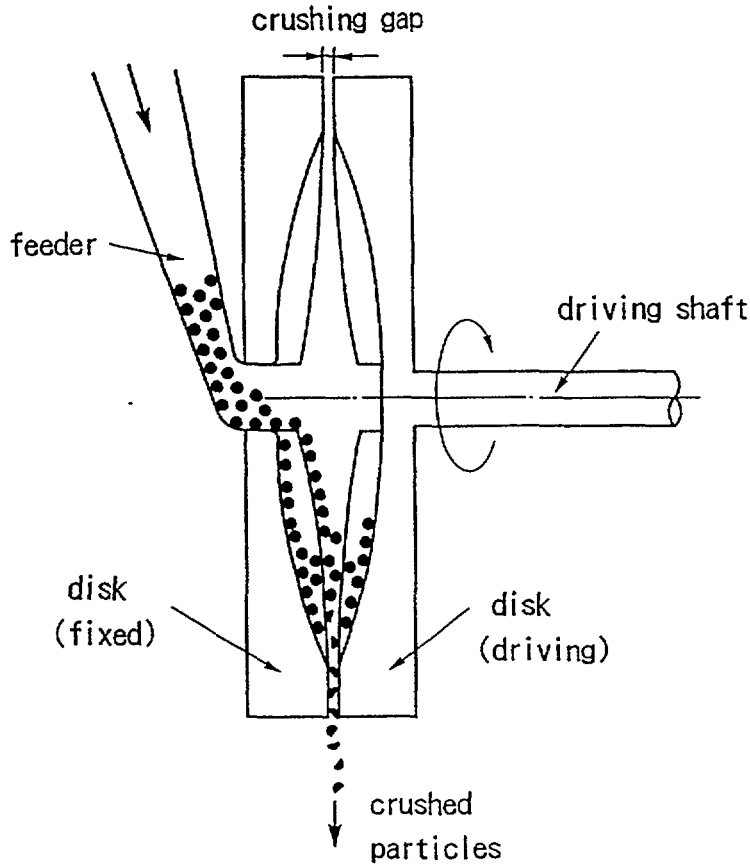

Fig. 9 Hard disk crusher in recovery process.

TABLE 4. CRUSHING EFFICIENCIES OF THE HARD DISK CRUSHER AS A FUNCTION OF CRUSHING GAP

Crushing gap	Crushed coated fuel particles	Crushed UO_2 kernels
0.50 mm	100 %	20~30 %
0.60 mm	100 %	~0 %
0.75 mm	~90 %	~0 %

(2) Uranium recovery experiences in fuel fabrication process

Fabrication of the first-loading fuel of the HTTR started July 1996. In the fuel fabrication process, uranium in the as-fabricated fuel compact is recovered by the process illustrated in ***Fig. 8***. Since this process is almost the same as burn-grind-leach method in the head-end reprocessing method of the spent fuel, useful data can be obtained to investigate the head-end reprocessing method. Then data have been accumulated. In this paper, as an example, the SiC layer removal efficiency from burned SiC-particles by grind-crush method is introduced.

A hard disk crusher used in the recovery process is shown in ***Fig. 9***. Crushing gap was selected to meet the following criteria: (1) crush efficiency of coated fuel particle should be as high as 100 % and (2) UO_2 kernel should not be crushed to prevent uranium contamination in the crusher. The crush efficiencies were investigated as a function of the crushing gaps. The results are summarized in ***Table 4***. In the recovery process, 0.60 mm of the crushing gap was selected. Through the fabrication of the HTTR first-loading fuel, about 100 kg of UO_2 was recovered. The experiences showed that 100 % of crush efficiency was obtained and no re-adjustment of the crushing gap was needed during the fabrication.

IV. Summary

A calculation study has been carried out to investigate the fission products release behavior from the HTTR fuel during long-period repository. The JAERI's head-end reprocessing technique was reviewed. The uranium recovery experience in fuel fabrication process of the HTTR first-loading fuel was also examined to investigate head-end reprocessing technique.

(1) The calculation result showed that the intact particle can retain krypton and iodine almost completely during long-period repository. Even from the through-coatings failed particle, remarkable release of krypton and iodine starts after 1000 years repository.

(2) JAERI studied graphite-CO_2 reaction and jet grind method as a head-end reprocessing technique to reduce CO gas release and maintenance for controlling roll-gap clearance in the roll grinder. The feasibility of this method was approved through laboratory scale experiments.

(3) In the fuel fabrication process, uranium in the fuel compact is recovered by the burn-grind-leach method and basic data have been accumulated. In the recovery process, almost 100 % of crushing efficiency was obtained by selected crushing gap and no additional adjustment of the gap was needed.

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GT-MHR SPENT FUEL STORAGE DISPOSAL WITHOUT PROCESSING

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Abstract

Possibility of GT-MHR spent fuel storage during long time without additional processing is discussed in this paper

Spent fuel elements discharged from this reactor type are ideal waste forms for permanent disposal in a geologic repository The graphite fuel elements and the ceramic coatings on the fuel particles are as-manufactured engineered barriers that provide excellent near field containment of radionuclides and minimize reliance on the waste package and surrounding geologic media for long-term containment Because of the high level of plutonium destruction and degradation achieved by GT-MHR, the isotopic composition of residual plutonium in spent fuel elements would not be practical for use in nuclear weapons and for energy production Dilution of plutonium within the relatively large volume of GT-MHR fuel elements provides excellent resistance to diversion throughout the fuel cycle This is accomplished without adversely impacting repository land requirements, since repository loading is determined by decay heat load and not by physical volume

These conditions of safe fuel storage criticality conditions, conditions of decay heat removing and radiation doses are discussed as well

BACKGROUND

An important issue for any plutonium disposition strategy, as well as for any high radiotoxicity waste, is the suitability of the final waste form for permanent disposal For assessing permanent disposal option, it is assumed that spent fuel will be placed in a deep stable geologic repository that is similar to Yucca Mountain in the USA, which is the candidate site for disposal of unreprocessed spent fuel from commercial light water reactors (LWRs) Disposal feasibility considerations may be categorized into

- Proliferation risks and safeguards requirement,
- Radiological risks to the general public for very long time period following permanent closure of the repository,
- Suitability and licensability of the final waste form for permanent disposal,
- Cost for disposal, including waste package costs, tunneling cost, land-area requirements, and disposal operation

Proliferation considerations can be categorized into short-term safeguard issues (e g diversion of material during temporary storage and transportation to a repository) and long-term issues (e g reclamation of material from a repository long after institutional controls and oversight have been abandoned) The high fuel burnup capability of the GT-MHR (approximately 65 % of the initial plutonium and ~ 90 % of the initial Pu-239) without requiring recycle is clearly more effective for destroying and degrading weapons grade

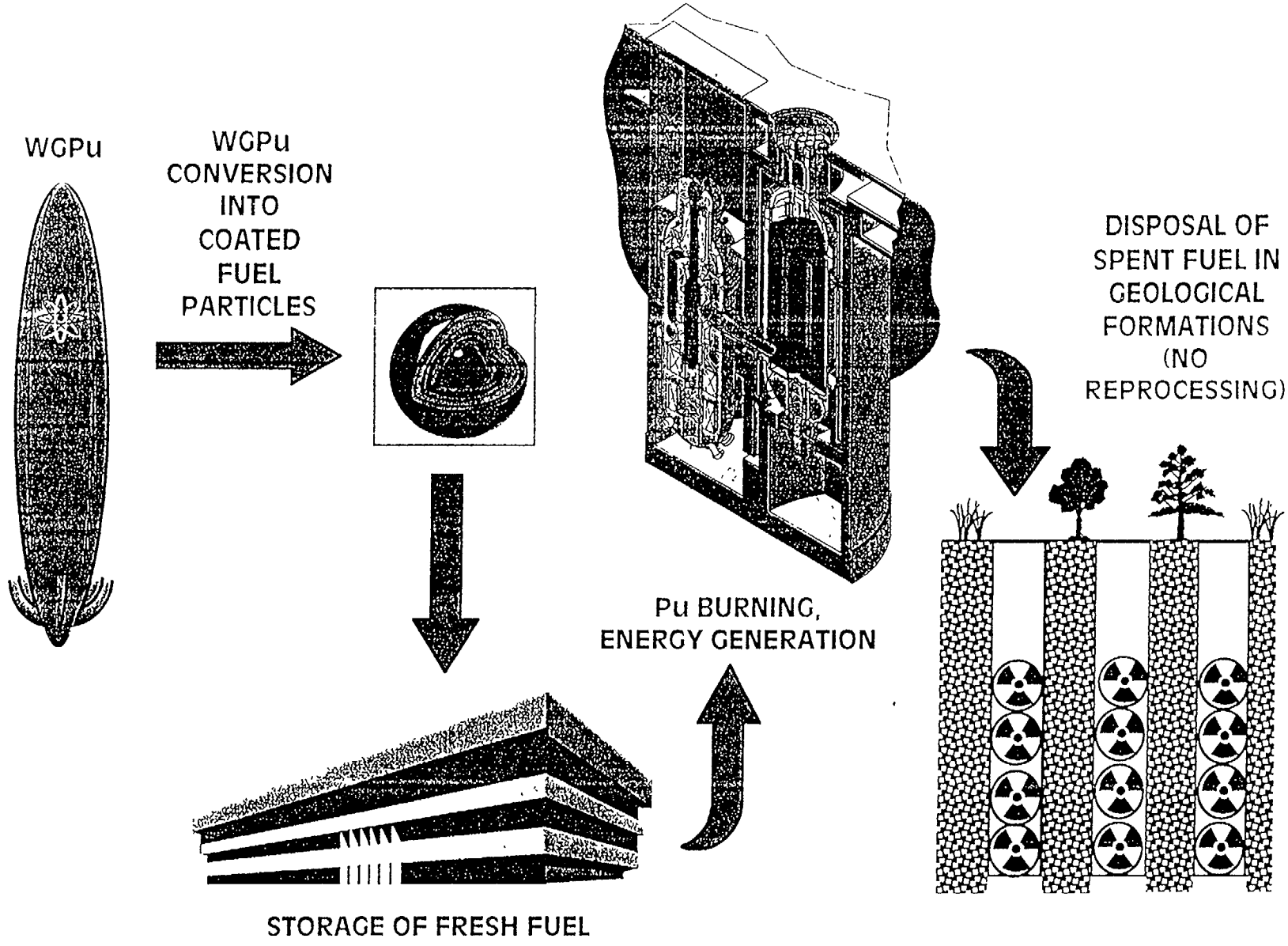

FIG. 1. Disposition of WGPu.

plutonium (WGPu) than direct vitrification, indefinite storage, or any other reactor-based strategy currently under consideration that incorporates an once-through cycle without reprocessing.

Whole - fuel elements disposal of GT-MHR spent fuel has been recommended as the preferred option because of advantages related to ease of implementation, proliferation risks, safeguards requirements, cost and schedule. This recommendation is corresponding with previous studies conducted by Oak Ridge National Laboratory [1] and in Germany [2] , which concluded that whole elements of HTGR spent fuel containing uranium and thorium coated fuel particles repository and should perform better than unreprocessed LWR spent fuel. Also, whole - elements disposal of spent GT-MHR fuel is directly analogous to options being developed for disposal of commercial LWR spent fuel in unprocessed, whole - assembly form.

Detailed evaluations of whole - elements were performed, including an assessment of the technical criteria for use of a multipurpose conister (MPC) as for LWR spent fuel storage to meet requirements for temporary dry on-site storage, transportation to the repository, and final disposal within the repository. It was concluded that spent fuel elements discharged from GT-MHR are ideal waste forms for permanent disposal in a geologic repository. The graphite fuel elements and the ceramic coatings on the fuel particles are as-manufactured engineered barriers that provide excellent near-field containment of radionuclides and minimize reliance on the waste package and surrounding geologic media for long-term containment. No technical issues should preclude whole - elements disposal of GT-MHR spent fuel within MPCs in a geologic repository.

The general scheme of WGPu utilization through GT-MHR is shown in Fig. 1.

WASTE DISPOSAL DESCRIPTION

The current conception of management with spent fuel from GT-MHR is the following:

- after interim storage in the local in-site storage facility of reactor plant area spent fuel is moved to on-site long term storage (see Fig. 2);
- some storage methods have been considered:
  - concrete storage casks (current preferred choice);
  - dual purpose casks;
  - expanded in-plant storage facility;
  - modular vault dry storage (currently used at Fort St. Vrain).

As a preferable spent fuel disposal method the following operations are considered:

- Place spent fuel in multi-purpose canisters (MPC) in reactor service building. A conceptual design has been developed for a multipurpose canister (see Fig. 3), which would be used for storage, transportation, and permanent disposal of spent fuel. The GT-MHR MPC would contain 42 fuel hexagonal graphite elements of 0,8 m in length and 0,36 m across the flats, arranged as seven columns with six fuel elements per column. Each fuel element contains ~ 20 million fuel coated particles.
- Load canister into concrete cask (Fig. 4);
- Move cask to storage facility;
- Store 10 years or until final disposal facility is available;
- Load multi-purpose canister into shipping cask (see Fig 5);
- Ship spent fuel in multi-purpose canister to final disposal facility.

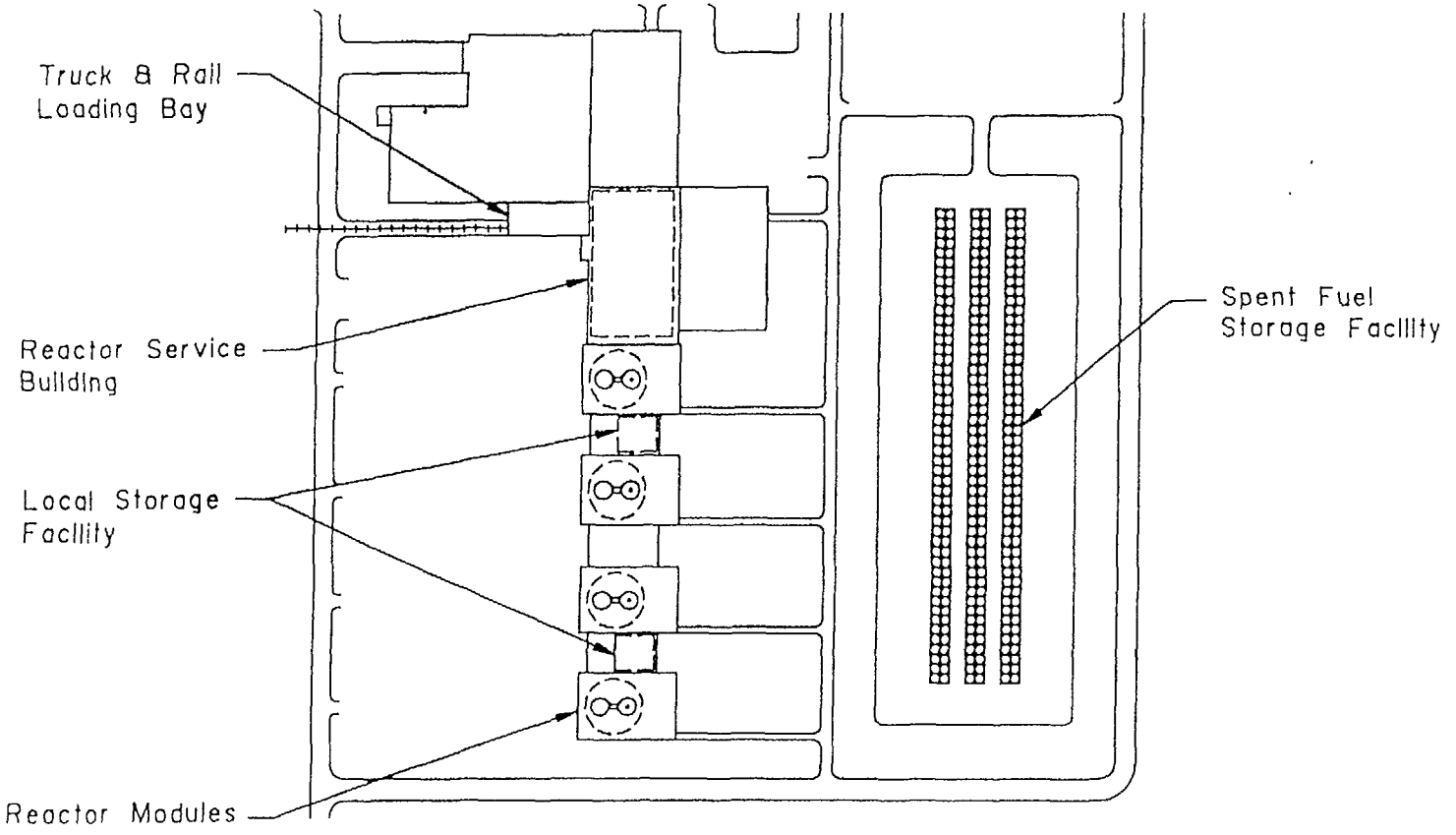

FIG. 2. Spent Fuel Storage Facility.

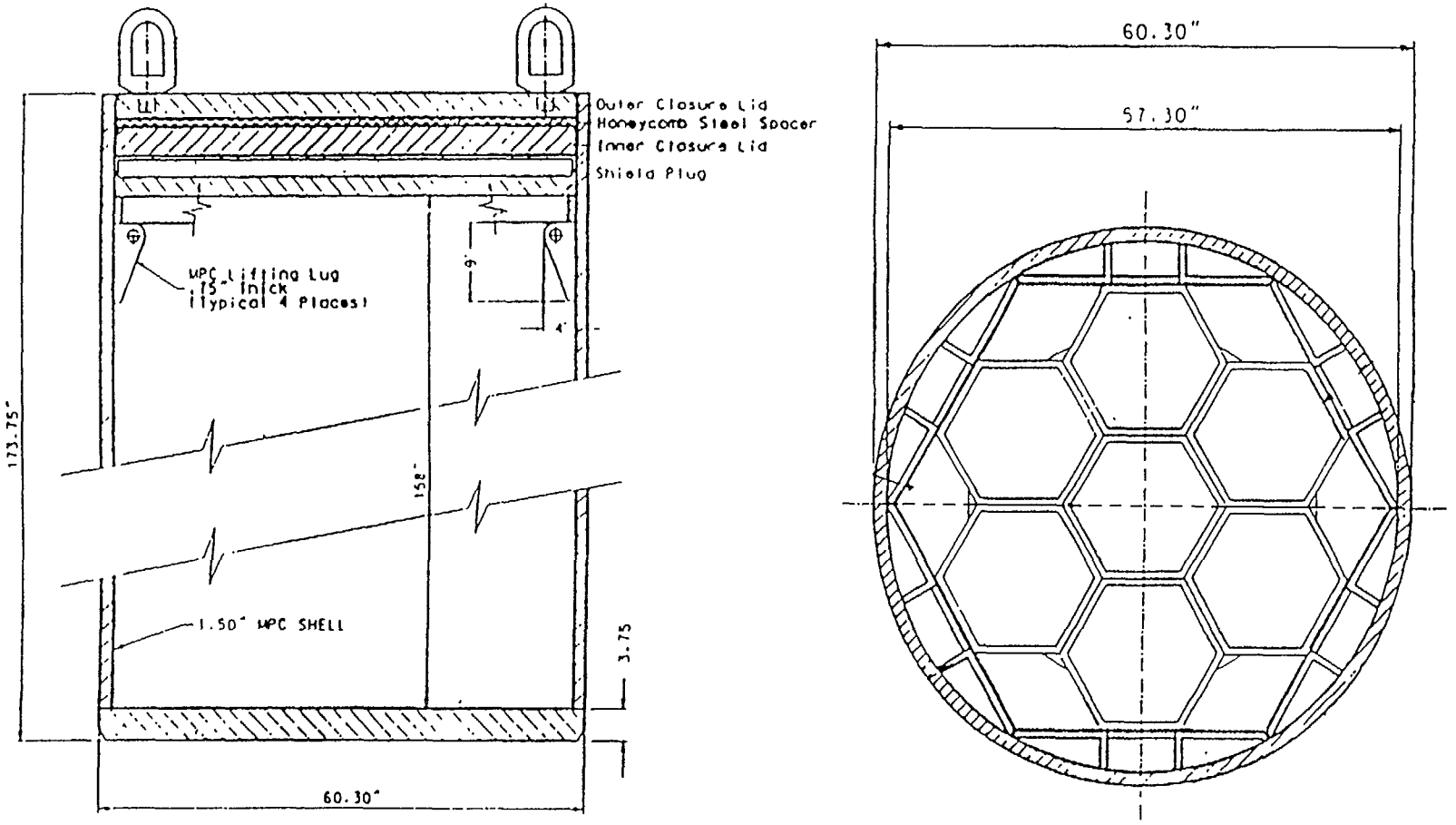

FIG. 3. Multi-Purpose Canister.

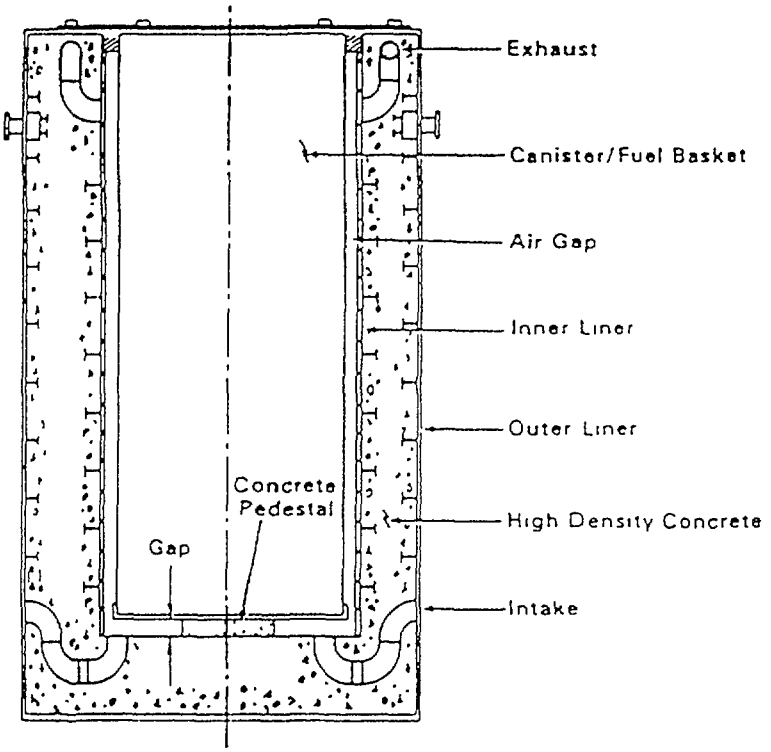

FIG. 4. Concrete Storage Cask.

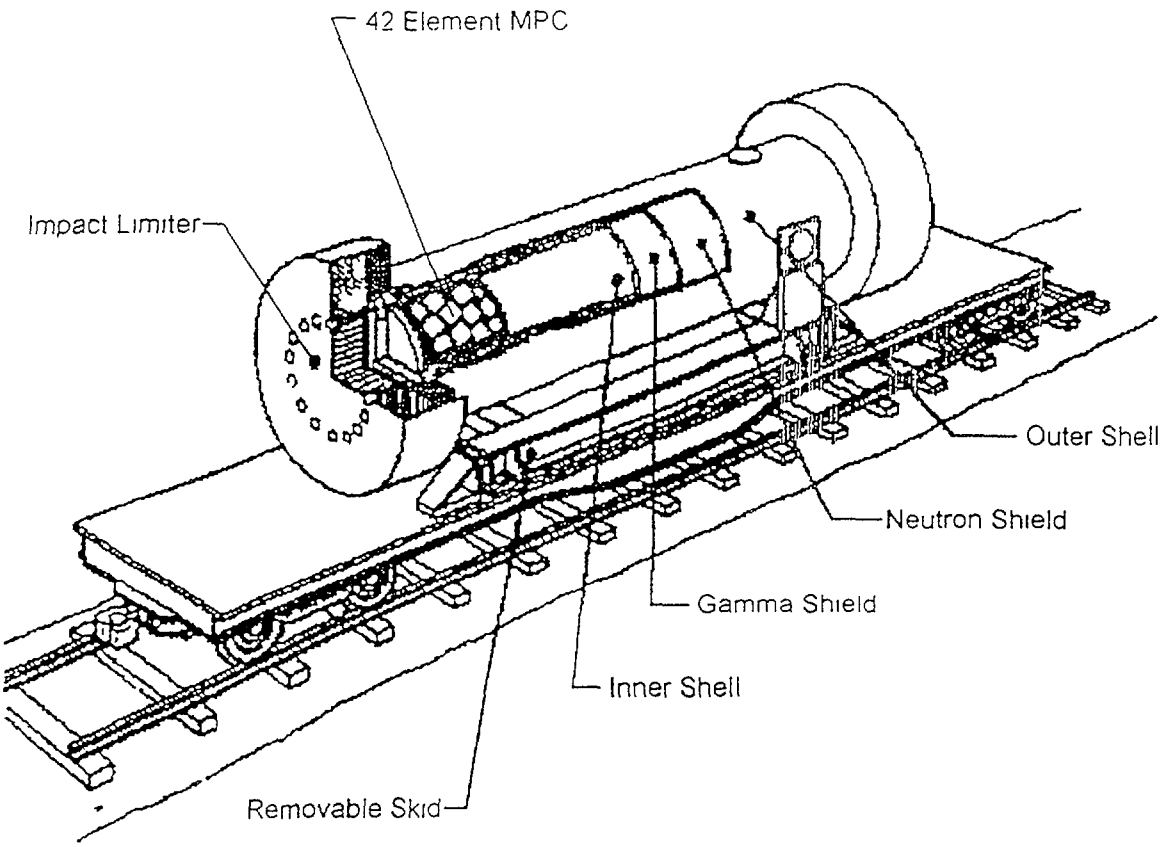

FIG 5 Shipping Cask

THE MAIN RESULTS FROM EVALUATIONS OF WHOLE-ELEMENTS DISPOSAL

The following results have been obtained from evaluations of whole-elements disposal:

- Graphite comprises most of the volume of GT-MHR spent fuel. Because of very low level of impurities in nuclear-grade graphite and excellent irradiation performance of coated particle fuel, the graphite does not become highly radioactive during irradiation. The high-purity, nuclear-grade graphite fuel elements are noncombustiable by conventional standards, and oxidation of graphite and other fuel element components at repository temperatures would be negligible over geologic time period.

- Because of the relatively low volumetric decay heat for GT-MHR waste packages, the peak fuel/graphite temperatures are significantly lower than the corresponding fuel/cladding temperatures within LWR waste packages. The GT-MHR peak fuel temperatures are $\sim 220\,^{\circ}C$ versus $\sim 350\,^{\circ}C$ LWR fuel.

- GT-MHR TRISO-coated particle fuel offers the benefit of long-term containment for radionuclides without having to rely on performance of the waste package or geologic media. Quantitative assessment show that the TRISO coating is capable to maintain its integrity for hundreds of thousands to millions of years in a repository environment. (For comparison, the expected lifetime of zircalloy cladding in a repository is less than 1000 years, even under dry conditions, [3]). Previous experimental studies [4,5] have shown that the corrosion rates of pyrocarbon, SiC, and nuclear-grade graphite are very low (even relative to waste glass) and are ideal components of an engineered barrier for a waste-management system. A key conclusion from the ORNL study [4] was that coated particle waste provides much better long-term containment of radionuclides than glassified waste forms. Key results of the ORNL investigations are described below:

  – Coated waste parcels (including those coated with pyrocarbon only) are leached at rates slower than could be detected by sensitive analytical techniques, including atomic absorption and inductively assumed to be at the detection limit, the rates are still 100 to 10.000 less than the rates measured for borosilicate glass.

  – Coated particle waste is especially effective for immobilizing cesium, which readily leaches from glassified waste because of its high solubility and tendency to partition into leachable phases.

- For release by groundwater transport, only nuclides with high mobility are expected to reach the accessible environment within the 10.000 year time period currently specified by the US Environmental Protection Agency (EPA). The nuclide of most concern is carbon-14, because nearly all of the inventory is external to the coated particles and can be releases by groundwater leaching of the graphite. Conservative estimates of release and transport show that the dose rates resulting from aqueous carbon-14 release are well below the applicable EPA criteria for any anticipated GT-MHR spent fuel disposition strategy. The preliminary estimations show that C-14 activity may be reach to 1,5 Ci/m^3 which is more than five times less of criterion given in EPA requirements (8 Ci/m^3) for low-level waste.

As for plutonium release, which limit requirement of allowable fuel fraction is $5{,}5 \cdot 10^{-5}$ it is necessary notice that in discharge fuel particles the fraction of failed particles not exceeds of 10^{-5} even after different abnormal conditions.

• Preliminary evaluations of a GT-MHR MPC conceptual design show that all regulatory requirements for storage, transportation and disposal are met with significant margin. The GT-MHR MPC containing 42 fuel elements has overall dimensions nearly identical to those for an LWR MPC designed for 21 pressurized water reactor (PWR) fuel assemblies, but the weight with spent fuel is be about one-half that of the PWR MPC. The GT-MHR MPC decay heat load is significantly lower: 760 W for GT-MHR vs. 13.200 W for the LWR MPC (with both values based on a decay time of 10 years following discharge from the reactor). These advantages allow to store MPC without active cooling. GT-MHR MPC contains only ~ 11,3 kg discharge plutonium and the following content of the most important isotopes:

Pu-239 - 27,81 %	Am-241 - 0,97 %	Cm-242 - 0,057 %
Pu-240 - 29,05 %	Am-242m - 0,16 %	Cm-243 - 0,001 %
Pu-241 - 29,10 %	Am-243 - 0,97 %	Cm-244 - 0,39 %
Pu-242 - 10,71 %		Cm-245 - 0,021 %
		Cm-246 - 0,001 %
		Cm-247 < 0,001 %
		Cm-248 < 0,001 %

The shielding and criticality-control requirements for the GT-MHR MPC are also less than those for the LWR MPC. As for criticality-control requirement (in any case the value of $K_{eff} \leq 0,95$), this requirement is performed for 42 fuel element MPC even under a flooded condition with the necessary margin ($K_{eff} \sim 0,89$).

The estimated maximum dose rate determined by of actinides activity after 1000 years disposal is amount to ~ 10 mrems/year that is less than dose-rate limit from EPA requirement for controlled area of fuel storage (~ 25 mrems/year).

These factors result in a simpler and less expensive MPC design for the GT-MHR. The unit cost for the 42 GT-MHR MPC is estimated to be about $ 75.000 vs. about $154.000 for the 21 LWR MPC.

• The large volume of GT-MHR spent fuel relatively to that of the LWR, which results primarily from dilution of the plutonium within the graphite fuel elements and results in significant safeguards and security advantages for the GT-MHR, does not adversely affect repository land requirements. Land requirements are determined primarily by decay heat load and not by physical volume of spent fuel. On a per unit electrical energy basis, the GT-MHR MPCs and thermal / mechanical design requirements for the repository itself, the GT-MHR will requires less repository area. For the current reference areal power density limit of ~ 14 kW/m^2 (57 kW/acre), GT-MHR spent fuel requires about one-half of the repository land area of LWR spent fuel per MWe-year. The corresponding number of waste package that can be loaded into the repository per a square meters are about 19 for the GT-MHR vs. only one for the LWR.

CONCLUSION

For any permanent high-level waste form, desirable qualities include the following:

1) the primary waste-containment barrier should provide defense-in-depth and should last as long as possible;

2) the quantity of waste contained by the primary barrier should be as small as practical, which minimizes the fraction of exposed waste if the barrier does fail;
3) the short-term and long-term safeguards requirements and the potential for reusing fissile materials should be minimized;
4) the risks of diversion and proliferation of fissile materials should be reduced as much as possible.

GT-MHR spent fuel elements, with their TRISO-coated particle fuel, achieve these qualities to a much greater degree than other waste forms, including spent zircalloy-clad fuel rods irradiated in the LWR. GT-MHR spent whole elements are an excellent waste form for permanent disposal.

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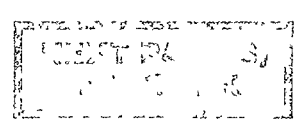
WASTE DISPOSAL AND DECONTAMINATION PRACTICES



GAS COOLED REACTOR DECOMMISSIONING — PACKAGING OF WASTE FOR DISPOSAL IN THE UNITED KINGDOM DEEP REPOSITORY

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Abstract

United Kingdom Nirex Limited has been established to develop and operate a deep underground repository for the disposal of the UK's intermediate and certain low level radioactive waste

The UK has a significant Gas Cooled Reactor (GCR) programme, including both Magnox and AGR (Advanced Gas-cooled Reactor) capacity, amounting to 26 Magnox reactors, 15 AGR reactors as well as research and prototype reactor units such as the Windscale AGR and the Windscale Piles Some of these units are already undergoing decommissioning and Nirex has estimated that some 15,000m^3 (conditioned volume) will come forward for disposal from GCR decommissioning before 2060 This volume does not include final stage (Stage 3) decommissioning arisings from commercial reactors since the generating utilities in the UK are proposing to adopt a deferred safestore strategy for these units

Intermediate level wastes arising from GCR decommissioning needs to be packaged in a form suitable for on-site interim storage and eventual deep disposal in the planned repository In the absence of Conditions for Acceptance for a repository in the UK, the dimensions, key features and minimum performance requirements for waste packages are defined in Waste Package Specifications These form the basis for all assessments of the suitability of wastes for disposal, including GCR wastes

This paper will describe the nature and characteristics of GCR decommissioning wastes which are intended for disposal in a UK repository The Nirex Waste Package Specifications and the key technical issues, which have been identified when considering GCR decommissioning waste against the performance requirements within the specifications, are discussed

1. INTRODUCTION

United Kingdom Nirex Limited (Nirex) is responsible for developing facilities for the safe disposal of intermediate and certain low level radioactive waste (ILW and LLW) within the UK

In due course, Nirex will issue waste acceptance criteria and all waste packages will have to comply with these before being accepted for disposal The acceptance criteria will be determined principally by the safety standards to be achieved, including requirements specified in the authorisation for disposal, but will also take account of design constraints, legal, operational and economic factors

Since the authorisation will not be granted until very much closer to the commencement of repository operations and waste producers wish to package wastes prior to the availability of the Deep Waste Repository (DWR), Nirex is producing a suite of specifications and guidance documentation in order to permit wastes to be packaged in a form which is compatible with plans for transport and disposal as currently envisaged

This paper describes the key technical issues within the Nirex Waste Package Specifications which have been found to be of significance when considering GCR decommissioning waste against the performance requirements for safe transport and disposal.

2. GAS COOLED REACTORS IN THE UK

The UK has a significant GCR programme, including both Magnox and AGR (Advanced Gas-cooled Reactor) capacity, amounting to 26 Magnox reactors and 15 AGR reactors (including shutdown reactors). These reactor types have graphite cores and are cooled using carbon dioxide. The Magnox reactors are fuelled by natural uranium in metallic form and take their name from the magnesium alloy (Magnox) fuel cans. The later generation AGRs are fuelled by enriched uranium dioxide, clad in stainless steel.

In addition there are a number of research and prototype reactor units, the most notable being the Windscale Piles and Windscale AGR.

The principal GCR Reactors in the UK are listed in Table I. The Table notes the date of operation, whether the reactor is decommissioned or operating, and if operating the envisaged date for shutdown is given.

TABLE I. PRINCIPAL GAS COOLED REACTORS IN THE UK

Name	Units	Type	Commissioning date	Status	Shut Down date
Windscale Pile 1	1	air cooled	1950	decom	1957
Windscale Pile 2	1	air cooled	1951	decom	1958
Calder Hall	4	Magnox	1956	op	2006
Chapelcross	4	Magnox	1959	op	2009
Hunterston A	2	Magnox	1964	decom	1990
Berkeley	2	Magnox	1962	decom	1989
Bradwell	2	Magnox	1962	op	2000
Dungeness A	2	Magnox	1965	op	2003
Trawsfynydd	2	Magnox	1965	decom	1993
Hinkley Point A	2	Magnox	1965	op	2002
Sizewell A	2	Magnox	1966	op	2001
Oldbury	2	Magnox	1967	op	2004
Wylfa	2	Magnox	1971	op	2005
Windscale AGR	1	AGR	1963	decom	1981
Dungeness B	2	AGR	1983	op	2013
Hinkley Point B	2	AGR	1976	op	2006
Hunterston B	2	AGR	1976	op	2007
Heysham I	2	AGR	1983	op	2014
Hartlepool	2	AGR	1983	op	2014
Heysham II	2	AGR	1988	op	2018
Torness	2	AGR	1988	op	2024

3. GCR WASTES FOR DISPOSAL

A key factor in disposal of GCR and other wastes is the volume that will arise for disposal during the operating lifetime and decommissioning of GCR facilities. The basis of information on waste arisings is the UK Radioactive Waste Inventory ('the Inventory') which is maintained jointly by Nirex and the UK Department of the Environment, Transport and the Regions.

The UK Radioactive Waste Inventory contains information on volumes and radioactivities of wastes, either in stock, or predicted to arise in the future, and companion documents provide detailed information on radionuclide content of wastes and on their physical and chemical characteristics. The Inventory presents information separately for operational wastes that arise in power stations or other nuclear facilities during their operational lifetimes, and decommissioning wastes that are generated after the facility has shut down.

The Inventory records all radioactive wastes arising in the UK and is not specific to those destined for deep disposal at the deep repository planned by Nirex. Hence further information is required before the Inventory can be used by Nirex as the basis for designs and safety cases.

As most commercial GCR cores will be subject to a 'deferred safestore' strategy and will remain in a safestore structure until 135 years after shut-down [1], Nirex does not include core decommissioning wastes from commercial reactors in its planned repository. GCR decommissioning wastes predicted to come forward for disposal to the deep repository include: graphite (fuel 'struts' and sleeves, fuel boats and dowels, core blocks from prototype reactors); Magnox fuel can components such as splitters and end pieces; activated steel components (control rods, flattening bars, AGR stringer components); sludges and ion exchange resins from clean-up operations. A total volume of some 15,000 m^3 of these wastes (when commissioned) are predicted to arise up to 2060.

Most of these wastes are activated items which contain a variety of activation products depending upon the material concerned and its chemical composition. One radionuclide of particular significance to post-closure safety and for which Nirex has put in place a major research programme to improve confidence in the disposal inventory, is chlorine-36. In addition, will also be contaminated with fission products and potentially with uranium residues from failed fuel elements. The wastes are already arising from on-going decommissioning programmes within the UK and therefore require packaging and conditioning for safe storage and disposal now.

4. WASTE PACKAGE SPECIFICATIONS

A key component of any decommissioning strategy is the definition of waste packages and specification of their performance requirements. In the absence of a deep disposal route for intermediate level waste, Nirex has defined a range of standard packages and has specified dimensions, key features and minimum performance requirements in a suite of Waste Package Specifications. This documentation has been fundamental in permitting waste packaging to commence in advance of the issue of repository Conditions for Acceptance.

Range of Standard Packages

The standard containers defined by Nirex for packaging ILW and LLW are shown in Table II. The number of containers in the standard range has been limited to six, as this is the minimum which best meets the needs of the UK waste producers. Standardisation has been shown to produce economic and safety benefits throughout the waste management lifecycle.

The 500 litre drum, $3m^3$ box and $3m^3$ drum are manufactured from relatively thin-walled stainless steel and are not designed to provide any radiation shielding. Handling and storage of these packages requires remote handling facilities and for transport, re-usable shielded transport containers. The use of a re-usable transport container has the major advantage that shielding and containment to meet IAEA Type B Transport Regulations can be invested in the re-usable item rather than the disposable one. The 500 litre drum, $3m^3$ box and $3m^3$ drum are described as 'unshielded containers'.

The 500 litre drum is already in widespread use in the UK, particularly for the packaging of operational type wastes. It may find limited uses in decommissioning applications, but it is expected that the other unshielded containers, the $3m^3$ box and drum, may be more suited to the packaging of decommissioning wastes in view of their larger payload. The $3m^3$ drum is designed for in-drum mixing applications such as sludges and resins, whilst its box counterpart, having a large square aperture is particularly suited to the packaging of solid items.

The 4m ILW box in contrast to the unshielded containers, is a transport package in its own right and will be disposed of at the repository without the need for any unpacking or unloading operations. The box will be restricted in radioactivity content to that which can be classed as Low Specific Activity (LSA) or Surface Contaminated Object (SCO) and packaged into an IAEA 'Industrial Package'. As a consequence, shielding can be economically provided within the package itself. The 4m box is designed around freight

TABLE II. NIREX STANDARD CONTAINERS

Intermediate Level Waste		
500 litre Drum	the normal container for most operational ILW	0.8m diameter x 1.2m high
$3m^3$ Box	a larger container for solid wastes	1.72m x 1.72m plan x 1.2m high
$3m^3$ Drum	a larger container for in-drum mixing and solidification of liquid and sludge type wastes	1.72m diameter x 1.2m high
4m ILW Box	for large items of waste especially from dismantling operations	4m x 2.4m plan x 2.2m high
Low Level Waste		
4m LLW Box	for general LLW	4m x 2.4m plan x 2.2m high
2m LLW Box	for general LLW	2m x 2.4m plan x 2.2m high

container principles and has a maximum gross weight of 65t. It is envisaged that the 4m box will find widespread use for the packaging of decommissioning wastes from GCR in view of the relatively low activity content of many of these wastes. However, higher activity items such as flux flattening bars may require packaging in the $3m^3$ box which can then be transported in shielded flasks as a Type B package.

The two LLW boxes are also available which are classified as Industrial Packages and may be suitable for decommissioning rubble. Further information on standard containers is available in Reference 2.

Waste Package Specifications and Guidance Documentation Performance Requirements

In the UK, Nirex has developed a strategy to facilitate early waste packaging, whilst minimising the risk of future reworking of packages, by providing guidance to its customers through the issue of a suite of Waste Package Specifications. Additional support for waste packagers by the formal assessment of specific packaging proposals is also provided. The Waste Package Specifications [3] are comprehensive and cover all aspects of the waste package including dimensions, handling and other key features, performance requirements, wasteform characteristics, QA and data recording.

Waste Package Specifications are not in general mandatory and are primarily issued for the guidance of waste packagers to assist in the development of packaging proposals which will be considered in detail by Nirex. The specification of waste packaging QA requirements is an exception to this and waste packagers' QA arrangements are subject to routine surveillance by Nirex.

Waste Package Specifications are independent of any particular repository site or design, and are based on bounding conditions which form a benchmark against which Nirex can provide advice and assurances. The bounding conditions are derived from design and safety considerations of the generic deep disposal system proposed for the UK.

Specifications are provided for: Waste Package, Wasteform, Quality Assurance, Data Recording Requirements and, the Package Identification System. Further information on the role of waste package specifications can be found in references 3, 4 and 5.

The vast majority of GCR decommissioning wastes should meet the requirements for disposal as defined by Nirex waste package specifications, if appropriately packaged. Of all the specifications, those relating to the wasteform are of particular interest from a GCR decommissioning view point and a number of wasteform issues have now been addressed by Nirex and are described in the following section.

5. TECHNICAL ISSUES

Assessments have been carried out by Nirex in support of GCR decommissioning packaging proposals. Issues requiring consideration with respect to performance under storage and disposal conditions have arisen due to the chemical composition of GCR components, their radionuclide content or their potential to release energy under normal or accident conditions. This section addresses some of the issues considered for various materials and components of GCR decommissioning wastes.

Magnox Fuel Cladding

The use of natural uranium as a nuclear fuel requires that neutron economy is carefully preserved Magnesium metal, in the form of a range of low additive alloys, provided a material that is essentially transparent to neutrons However, it is chemically reactive and this has to be considered in determining the appropriate wasteform for storage and eventual disposal.

Magnox reacts readily with water, producing hydrogen and a comparatively voluminous corrosion product, $Mg(OH)_2$ Both the hydrogen gas and the corrosion product will tend to provide internal pressure on the wasteform. Although the wasteform is not required to retain its integrity for a defined time after disposal, early failure can prejudice post-closure performance and may not meet regulatory requirements.

One simple approach to maintenance of package integrity is to limit the contents of the reactive metals to a level that can be shown not to prejudice the long-term performance of the package Current studies are showing that water limitation may also be significant in preventing corrosion rates rising to theoretical maxima.

For Magnox wastes arising from GCR decommissioning, the consequences of early package failure following emplacement in the repository are low because of the low inventory of short-lived soluble activity in these wastes. Therefore efforts are directed primarily to ensuring integrity of the packages for the interim surface storage period prior to disposal. This is achieved by use of high quality containers and carefully formulated cementitious wasteforms.

It must also be noted that these wastes also contain long-lived soluble activity in the form of chlorine-36. The inventory of chlorine-36 must be known since it contributes to the repository total inventory. Nirex has carried out a major research programme to support an overall assessment of the chlorine-36 activity of wastes destined for the repository.

The approach chosen was to base the estimate on activation calculations rather than direct measurements of chlorine-36, due to the difficulty that this latter approach would pose. The programme consisted of:-

- precursor measurements - on the surface and/or bulk of 1421 representative samples of relevant materials, using specially developed methods (mostly based on neutron activation analysis, NAA)
- transfer studies - to quantify the potential for transfer between waste streams during irradiation of graphite and reprocessing of fuel
- theoretical assessments - to support the calculational methodology

The results of the precursor measurement programme showed that the dominant precursor in Magnox was chlorine-35, present in natural chlorine at a level of 76%. A total of 145 measurements on samples from 124 batches has resulted in the characterisation of the four Magnox alloys by two probability density functions (PDF) for the mean chlorine concentration.

By deriving the inventory of chlorine-36 in Magnox (and other GCR decommissioning wastes) the effect on disposal risks has been established. It should be noted that it is unlikely packages will be able to retain their integrity for long periods comparable to the half-life of chlorine-36, hence risk is primarily ameliorated by a combination of dilution and long return times from a potential repository.

Graphite

Graphite wastes arise from their use in the construction of reactor cores and in fuel elements and fuel stringers, where the graphite has acted as the neutron moderator. Graphite was also used as physical support for fuel elements in prototype reactor systems (boats) and as spacers and coolant flow modifiers between fuel in the core (dowels).

Chemically, graphite is expected to be a stable material in the disposal environment, being resistant to oxidative corrosion across a wide range of conditions, and providing an unattractive medium for microbiological growth. Thus releases of carbon-14 from such wastes are expected to be low.

However, graphite has several properties that require consideration prior to packaging:

- it can act like a noble metal, promoting galvanic corrosion in more reactive metal systems;
- it acquires stored (Wigner) energy on neutron irradiation;
- it contains long-lived activation products (Cl-36);
- it acts as an effective moderator, potentially promoting neutron chain reactions.

Graphite may react electrochemically with other materials. Acting like a *'noble'* metal, graphite can promote accelerated corrosion of other metals by electrical (galvanic) coupling, in which local electrolytic cells driven by potential differences lead to increased dissolution and oxidation of less noble metals. Graphite is more electronegative even than stainless steel, so that direct contact between graphite wastes and stainless steel containers can lead to premature penetration and loss of integrity. Experimental studies have shown that corrosion rates can be increased by factors of up to 10. A number of preventive measures have been identified, including use of cement grouts and baskets to isolate graphite from stainless steel waste containers.

The irradiation of graphite leads to the accumulation of stored energy which can be released by heating to temperatures above the original irradiation temperature. In graphite from most reactor systems, this energy is only accessible at temperatures of several hundred degrees centigrade. However, in some prototype and experimental reactors, where the irradiation temperature of the graphite was not much above ambient, the stored energy can be released at relatively low temperatures. Where the received dose was significant, say greater than 10^{20} neutrons/cm^3, a considerable amount of stored energy can be acquired and can amount to more than 1kJ/g.

In order to prevent the rapid release of this stored energy, it is necessary to ensure that the graphite does not experience temperatures above the 'initiation' temperature. Following emplacement in the repository control of package temperature can be achieved by careful analysis of the radiolytic and chemical energy input to the repository system, repository design and waste emplacement strategy. The geothermal gradient provides a background temperature of about 30-35°C at depths of about 500 metres, and other heat sources can yield a peak temperature in the region of 50-100°C. These temperatures could be sufficient to release a significant fraction of the stored energy in low temperature irradiated graphite.

Nirex is sponsoring studies to provide additional understanding of the parameters controlling the release of stored energy under disposal conditions, in order to be able to advise whether such components could be suitable for direct disposal with currently envisaged repository

design concepts. Experimental studies are under way to determine the amount of stored energy remaining in low temperature irradiated graphite after 40 years quiescence, and its release characteristics. In these studies the total stored energy and its response to slow heating rates is being examined in detail in order to ensure that theoretical analyses are accurately estimated.

In addition to leading to stored energy, irradiation of graphite results in activation of impurities. Residual, low levels of the chlorine used to purify graphite are converted to the long-lived chlorine-36. The research programme referred to previously has also addresses the chlorine-36 content of graphite.

For graphite, a total of 458 measurements on samples from 57 batches have been performed to provide a detailed understanding of the composition of graphite. The work has resulted in the characterisation of three classes of graphite by probability density functions (PDF) for the mean chlorine concentration. Transfer studies have shown that a significant fraction of the chlorine is released from the graphite during irradiation both in precursor and activated form. The release rate of chlorine and chlorine-36 has been modelled to allow the calculation of residual chlorine-36 inventories.

The moderating properties of graphite for which it was originally used can also apply within the disposal environment. The co-disposal of graphite with fissile material (e.g. Pu-239, U-235) must be considered in the case of the UK repository. Assessments have been carried out to study the potential for accumulation of a critical mass within a well moderated and reflected system. Nirex has assessed the potential for development of a neutron chain reaction after disposal within repository and waste package safety assessments.

Steels

Steels are widely used for construction of reactor components, including both pressure vessel structures and in-core components such as control rods, flattening bars and fuel stringer components. Steel items may arise both as decommissioning and SPD wastes.

The irradiation of steels in a neutron flux results in activation of alloying elements and impurities and the packaging of steels must take account of the inventory of activation products. The key issue for GCR steel is heat production and radiation dose due to the activation products cobalt-60, iron-55 and nickel-63. Generally for normal steels the dominant activation product for heat and dose is Co-60 but it is important to understand the chemical composition of steels used inside a reactor as special grades or alloys may introduce unusual radionuclides which could be important contributors to heat, dose and other aspects of package performance such as post-closure risk.

The first choice for packaging steel components is likely to be the 4m ILW box in view of its large volumetric capacity (11m^3 assuming 200mm concrete wall thickness) and high payload (65t gross weight). However, for materials to be acceptable for packaging in such a container, the requirements for LSA (or SCO) must be met and in addition the waste must be suitable for packaging in an industrial package. The LSA criteria, for instance, impose a requirement for uniformity, impose limits on specific activity (2,000 A_2/t for LSA III) and on external dose rate. For packaging in an industrial package, limits are placed on the unshielded dose rate at 3m.

The decision whether to package in the 4m box or the alternative $3m^3$ box must take account of the constraints imposed by LSA and IP criteria and have to be balanced against the thickness of shielding that can be provided and the potentially beneficial effect of radioactive decay. If LSA and IP criteria cannot sensibly be met then the $3m^3$ box should be considered. This can be transported with up to about 280 mm steel shielding as a Type B package.

Unlike magnesium, steels are comparatively inert in an alkaline cementitious grouts environment, although their corrosion can be increased by galvanic coupling to more noble metals and graphite.

The three examples above are illustrative of a number of GCR waste components that have been addressed when assessing GCR waste packages for disposal.

6. CONCLUSIONS

In the absence of a finalised repository site, design or associated safety cases, Nirex is not in a position to issue Conditions For Acceptance. Nirex has therefore developed a strategy to facilitate early waste packaging by providing guidance through the issue of Waste Package Specifications supported by the formal assessment of specific packaging proposals on a case by case basis. The Waste Package Specifications are comprehensive and cover all aspects of the waste package including dimensions and other key features, performance requirements, wasteform characteristics, QA and data recording requirements.

The operation and subsequent decommissioning in the UK of experimental, prototype and commercial gas-cooled reactor systems, has led to the production of a variety of wastes. These wastes have been assessed against Waste Package Specifications. Most are suitable for direct deep disposal if appropriately packaged. Some issues remain for certain items, particularly low-temperature irradiated graphite containing Wigner energy, however research programmes are on-going to address these specific issues.

A clear conclusion is the need to develop a good understanding of the physical and chemical characteristics of wastes so that waste packages of appropriate performance can be specified.

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EVALUATION OF GRAPHITE SAFETY ISSUES FOR THE BRITISH PRODUCTION PILES AT WINDSCALE: GRAPHITE SAMPLING IN PREPARATION FOR THE DISMANTLING OF PILE 1 AND THE FURTHER SAFE STORAGE OF PILE 2

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Abstract

The two air-cooled graphite-moderated production piles at the UK Sellafield site (formerly Windscale) have not operated since the late 1950s following the fire in Pile No. 1 in 1957. Although limited graphite trepanning took place immediately following the fire (mainly in the undamaged reactor, Pile No. 2), it has been deemed necessary to obtain new samples and extensive data to support the intended programme, which is for further "Safe Storage" of Pile 2 and for the dismantling of Pile No. 1.

The reactors (especially Pile No. 2) contain significant Wigner energy, and elaborate precautions have been applied to ensure that intrusive operations cannot lead to energy releases. The paper discusses the visual surveys, trepanning procedure and the analytical data which have been obtained from the samples. There appear to be no obstacles to the continued storage of Pile No. 2. For Pile No. 1, the data will be used to support the safety case now being developed for dismantling and will define acceptable handling, storage and disposal procedures for the graphite blocks.

1. INTRODUCTION

The Windscale Piles are early graphite-moderated air-cooled production reactors operating at low temperature. Each consists of around 2000 tonnes of extruded blocks of AGXP graphite stacked vertically to form a horizontal cylinder approximately 15.3 metres diameter by 7.4 metres long. There are 36 graphite blocks along the length of each channel. There are 2660 fuel channels in each Pile, arranged in groups of four at a specified channel reference around a central experimental channel (Figure 1). The four fuel channels in each position are denoted as TL, TR, BL, BR for Top Left, Top Right, Bottom Left and Bottom Right.

After the 1957 fire in Windscale Pile No. 1, both Piles ceased further operation. A consequence of the low graphite temperature was the build up of Wigner energy, and periodic anneals were undertaken. It was during one of these events that the fire occurred, although it should be noted that it is probable that it was failed fuel and not the graphite which was the principal source of combustion. However, the gradual "hardening" of the Wigner-energy spectrum provided sufficient concern and uncertainty about the exact course of events in Pile 1 led to a decision to close Pile 2 also.

As a result of the fire, the residual Wigner energy in Pile 1 is considered to be less that in Pile 2; the several anneals which had previously taken place in each reactor also mean that the distribution of the residual energy within the cores is not easy to predict.

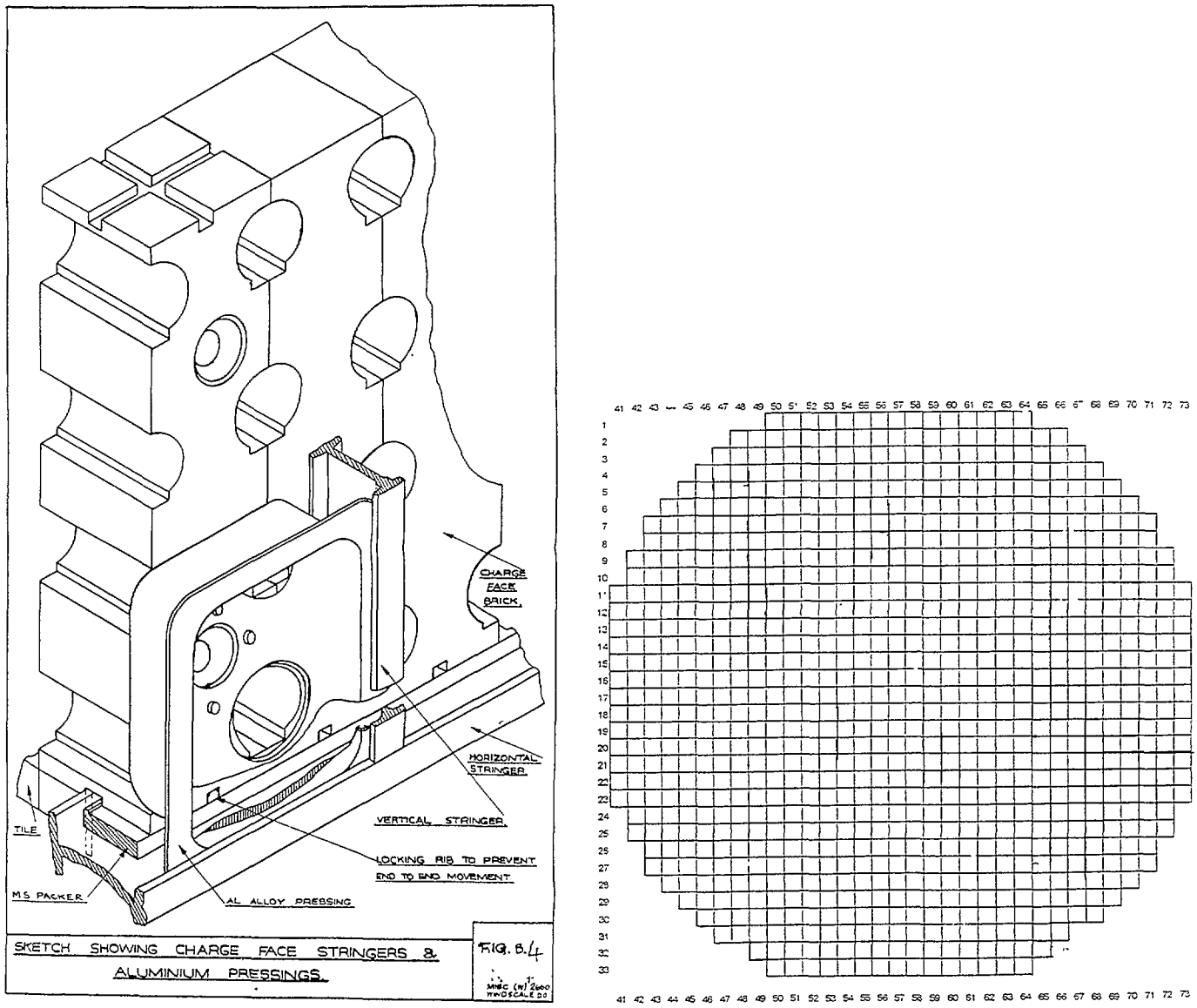

Figure 1. (a) General arrangement of fuel and experimental channels within Windscale Pile graphite blocks. (b) Charge-face grid plan. Each square represents four fuel channels (Top Left, Top Right, Bottom Left, and Bottom Right), and may also include one central experimental channel.

Shortly after the close-down of the reactors, a limited graphite trepanning programme took place, principally (but not exclusively) in Pile 2. Initially this was to justify a re-start of Pile 2. Later, it was decided on the basis of the Wigner-energy results that it was acceptable to leave the reactors shut down and that the hazard of so doing was minimal, although the provision of facilities to inject large quantities of water was considered prudent.

Now, some 40 years later, it has been decided that the present position is unacceptable and that the damaged reactor must be fully decommissioned. Separately, the safety case for the retention of Pile 2 in a "Safe Storage" mode requires extensive modernisation, including the provision of a seismic analysis. This present paper briefly describes the activities which have taken place in support of these programmes with respect to the graphite.

2. REACTOR SURVEYS

The objectives of the reactor surveys of the two Piles are somewhat different, since Pile 1 is to be dismantled whilst Pile 2 is to be retained intact for up to 40 years. However, there are common features and, in each case, a thorough survey of the state of the cores has been made.

Initially, a visual inspection of the charge and discharge faces of the reactors was undertaken. After the shutdown, the intention had been to remove the fuel and isotope cartridges. In fact, numerous fuel elements and isotope cartridges were found to have lodged at channel exits, and it was necessary to retrieve these and dispose of them. The next stage was to use through-channel (non-intrusive) viewing equipment to establish that channels were indeed empty.

In the case of Pile 1, there exists an extensive "fire-damaged zone" within the lower central region of the core. In many channels in this area, light transmission from the charge to discharge faces is not possible and it must be concluded that debris remains within them. It has been speculated that there may be a void within this region, although the current view of the authors is that, whilst graphite at high weight loss may be present, it is most likely to have retained its basic geometrical structure despite the greatly reduced density in some regions. It is known from small-sample oxidation experiments that extremely high weight losses can be achieved before loss of geometrical form occurs [1].

Intrusive surveys have been conducted in selected visually-clear channels in both Piles, to view the channels and to obtain additional radioactivity and chemical information prior to renewed trepanning of samples for more comprehensive analysis. These surveys have been conducted under contract by Magnox Electric plc, using specially designed equipment. The principal machine carries forward, sideways and vertical TV cameras responding for near-infra-red LED illumination, together with pneumatically deployed pre-weighed paper sample swabs, and instrumentation for monitoring local radiation levels. Verification of block gap dimensions has been made. Videotape records of the surveys have been retained.

Figures 2 and 3 illustrate the typical channel structure, showing the vertical block junctions which embrace the fuel channels, the horizontal block junctions, and the slot feature in the base of the channel in which the graphite fuel "boats" were located. Figure 3 shows an example of extruded material, possibly lead, which has reached the fuel channels at a number of locations from adjacent isotope channels either during normal operation or during Wigner-energy anneals.

No channels in the fire-damaged zone of Pile 1 have been entered. However, the present programme does include channels above, below and to the side of this region. In those above, in order to guard against the risk of encountering a weakened region, a lighter-weight tool containing only a forward viewer was first deployed. The wisdom of this decision was revealed when in one particular channel - 16/55 TR - above the centre of the fire-damaged region, an apparent void was encountered at the block 12/13 junction (Figure 4). Reference to drawings established that this feature, and a corresponding feature at the block 25/26 junction, represented breakthrough into the fuel channel from a vertical shut-down-rod channel. The rod may be clearly seen within these channels. Clearly, the vertical channel has acted as a conduit for hot gases from the fire zone during the incident, and sufficient graphite oxidation has occurred to allow the vertical and horizontal channels to merge. However, the transit of both the light-weight and full-weight machines caused no additional damage. In a nearby channel - 16/60 BR - trepanning had previously taken place shortly after the fire. The present TV survey showed that all of the old trepanning holes were clearly defined (see Figure 5), a further indication that extensive friability was not present.

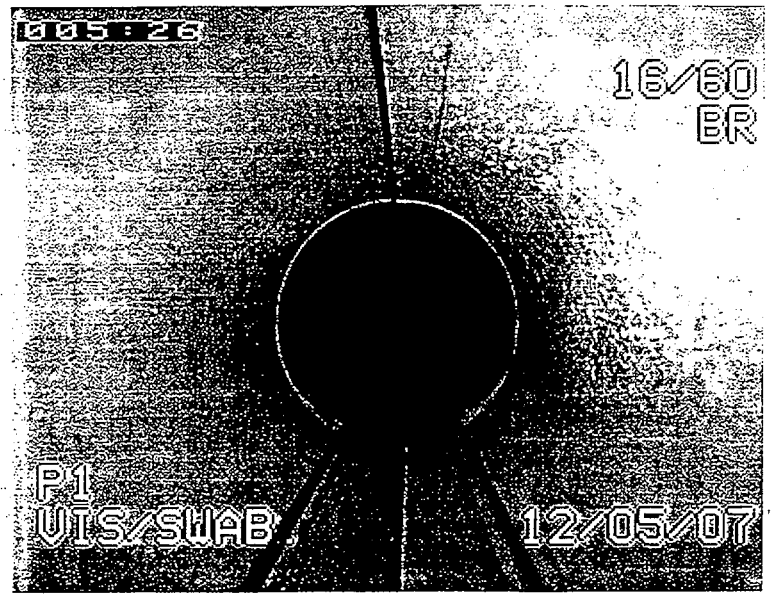

Figure 2. Video image of Pile 1 channel 16/60 BR. Note lateral and vertical brick joints and channel for fuel-element boats. Grainy patch on right-hand side of nearest block may suggest surface oxidation of graphite, resulting from local fuel-element overheating.

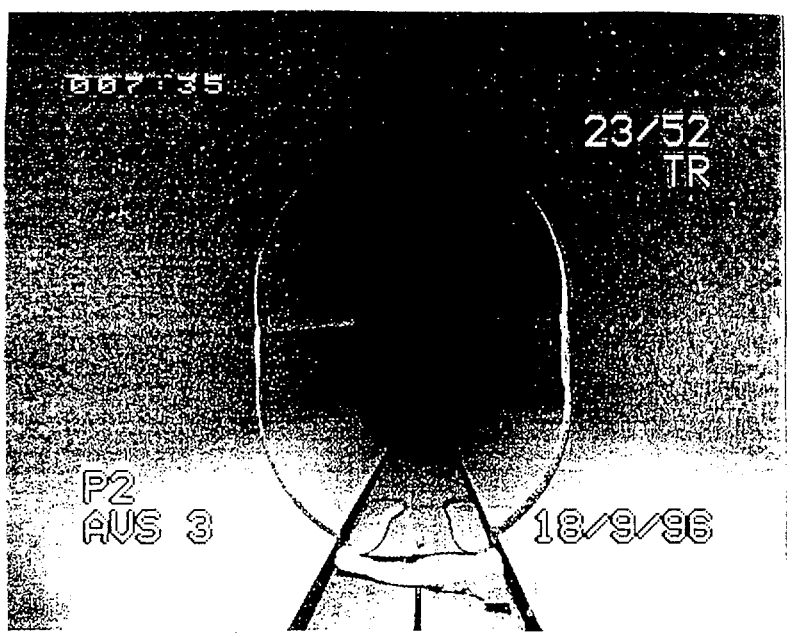

Figure 3. Video image of rear of channel 23/52 TR (Pile 1) showing possibly extruded material at rear of block 35.

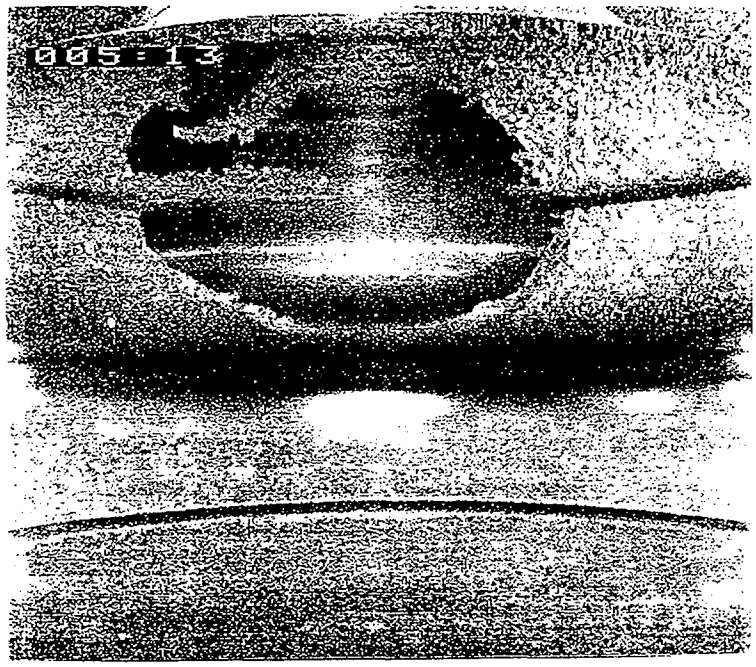

Figure 4. Cavity at block 26 channel 16/25 TR (Pile 1) observed with side viewing camera. Convection during the fire in the vertical shut-down rod channel has conveyed hot gas sufficient to allow oxidation of graphite through to the cool channel. The shut down rod may be seen through the aperture.

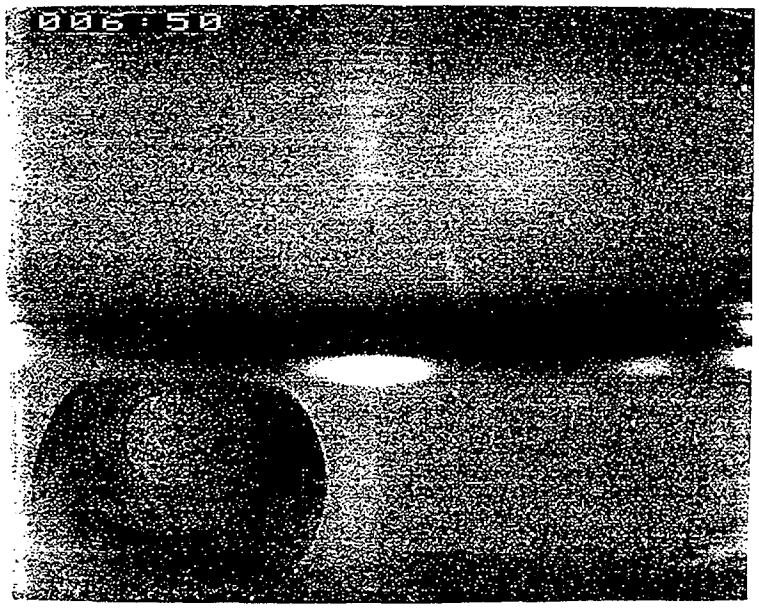

Figure 5. 1958 trepanning hole in block 30 channel 16/60 BR (Pile 1) observed in the recent survey.

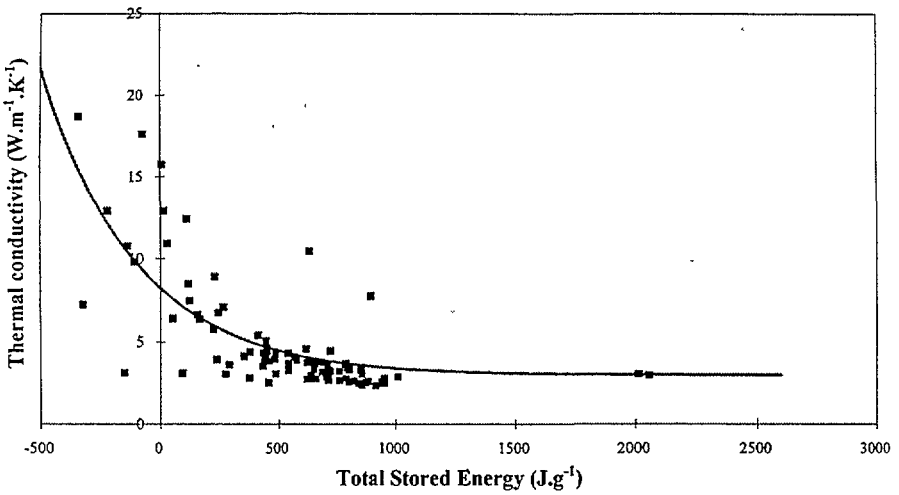

Figure 6. Relationship between total stored energy and thermal conductivity in Pile 2 samples. Apparent negative total stored energy results arise from an uncertainty in the total heat of combustion of the AGXP graphite and will be corrected following the annealing of suitable samples.

Radioactivity readings, and activity measurements on the swab samples, reflect the long period of decay which has taken place since the last operation of the plant. Typical Pile-2 radiation readings, sampled at many places within the core, lie in the range 2 - 4 $mSv.h^{-1}$ ($\beta\gamma$), whilst Pile-1 results were more variable and reached values of 18 $mSv.h^{-1}$. In typical Pile-2 swab results, gamma spectroscopy identified ^{60}Co, ^{137}Cs, ^{152}Eu and ^{154}Eu on all samples. Total α and ^{137}Cs (from fission-product contamination) is maximised towards the bottom and outer faces of the pile. ^{60}Co, the dominant γ-emitter, increases along channels (*i.e.* from front to back) and from the bottom and sides of the pile towards the centre, consistent with the neutron-activation profile. The maximum α activity is approximately 0.3 Bq/swab, maximum ^{137}Cs and ^{60}Co activities (separately) approximately 30 Bq/swab and maximum ^{152}Eu activity approx. 15 Bq/swab. Although the swab pick-up is rather variable, we estimate a typical value to be between 0.7 and 2.0 mg.

Pile 1 radioactivity readings appear much higher than Pile 2 in some regions of the Pile. This is not entirely surprising since a fuel fire has taken place in the Pile-1 case. Local α and β activities two orders of magnitude higher than in Pile 2 have been found.

X-ray Fluorescence and Inductively-Coupled Plasma Mass Spectroscopy measurements have been made on the Pile 2 swabs and the latter along with Atomic Absorption Spectroscopy on the Pile 1 samples. The principle interest in the chemical data is the possibility of the presence of catalysts for graphite oxidation in air, which might conceivably increase the risks associated with dismantling operations. The principle contaminants in Pile 2 in descending order are calcium (up to 6.5% by weight in one example), aluminium (1.5%), sodium (0.8%) and magnesium (0.46%), apparently derived largely from concrete dust with some contribution from fuel cladding. None of these substances is regarded as a powerful catalyst (in the manner of some transition metals, for example) but, in the quantities present, may certainly enhance the observed reaction rates. Typical data on this are given below. A formerly-identified "reactive zone" within the core is thought to have arisen as a result of water-borne chemical contamination from the air ducts, and larger concentrations of these concrete-derived contaminants and sodium were found in and around this region.

Direct comparison of Pile-2 with Pile-1 data is hampered by the change in analytical method used and the different degree of systematic and random uncertainties considering the extremely small amount of graphite present on the swab samples. Pile 1 graphite appears to contain some higher local concentrations of calcium than Pile 2 and also more aluminium, the latter arising presumably from the fuel fire. There is less slightly less sodium in Pile 1, but significantly more iron. The presence of small amounts of lead on isolated samples is consistent with the transport of lead within the core as already mentioned.

3. THE SAFETY OF GRAPHITE TREPANNING

None of the results from the visual and swab survey caused significant anxieties about the ability to obtained trepanned samples of the graphite safely. Although it was accepted that trepanning had previously taken place in both of the reactor cores, it was still considered appropriate to review the proposed operation in terms of modern safety standards and it was necessary to obtain the approval of the UK nuclear regulatory body.

The following issues were addressed prior to new trepanning:

(i) Acceptable graphite temperature in the light of the perceived propensity of the graphite to release Wigner energy. If the uncertain history of anneals is ignored, it could be argued that a stored-energy release could be initiated if the graphite temperature is raised more than around 50K above the original irradiation temperature. Existing data suggested the presence of graphite capable of undergoing a self-sustaining release under these circumstances *i.e.* where the rate of release of Wigner energy exceeds the specific heat capacity of the graphite at the initiating temperature. Laboratory tests using the trepanning equipment demonstrated that the trepanning itself using the modern equipment would raise the temperature by a few degrees at most. An overall graphite temperature limit of 35 °C was therefore imposed, with an argon atmosphere in the vicinity of the region being trepanned;

(ii) The propensity of the Piles graphite to undergo a dust explosion was reviewed in detail. Pure nuclear-grade graphites are formally classified as "non-explosible" [2], in contrast to less pure carbon materials. Indeed, graphite is commonly safely used in extreme applications such as arc lamps and electric motor brushes. Previous studies of the potential explosibility of UK nuclear graphites have confirmed this "official" view except in one extreme case when a powerful oxygen-rich chemical ignition system was employed. Ref. 2 defines the criteria which must simultaneously be satisfied before a dust explosion can occur -

a) The dust must be combustible
b) The dust must be airborne, implying a need for a turbulent gas flow
c) The particle size must be optimised for flame propagation
d) The dust concentration must fall within an explosible range (neither too high nor too low)
e) An ignition source of sufficient energy to initiate flame propagation must be present
f) An oxygen-rich atmosphere must be present

and, if a *disruptive* explosion is to occur -

g) The suspended dust must be in a confined space which inhibits the relief of the pressure rise resulting from ignition

It may also be noted that a French assessment of the explosibility risk in a rather more potentially hazardous operation of incinerating spent nuclear graphite in a fluidized bed plant has also given a favourable result [3] . It is considered that the procedures adopted for the trepanning of Piles graphite, particularly the controls on temperature and the presence of an inert atmosphere in the vicinity of the cutting operation, effectively eliminate the explosibility risk;

(iii) The best available information about graphite oxidation rates, and graphite thermal conductivity, measured in the early 1960s on the original samples, was re-assessed. It was considered extremely unlikely that these parameters would give rise to any hazard although it was accepted that there was sufficient uncertainty about the current values (*i.e.* following 40 years of standing) to require the acquisition of new samples prior to any dismantling of Pile 1.

In Pile 1, it was further decided that no trepanning would take place within the zone previously defined as the "fire-damaged region". Ultimately, when Phase 2 of the decommissioning programme is reached and the graphite core is dismantled, a case for a closer inspection of this region may become necessary.

4. RESULTS FROM PILE 2 TREPANNED SAMPLES

A total of 134 samples approximately 15 mm diameter by 20 mm long were recovered from a total of 13 channels previously identified as completely free of obstructions. The associated dust was also recovered. 11 of these channels were located along lines joining the centre of the core to the base (vertical section) and to the right-hand side (horizontal section). One criterion by which the channels were selected was to attempt to include regions in which Wigner energy was likely to be maximised on the basis of irradiation dose and temperature.

The following determinations were made:

4.1 Density

The mean measured density of the samples was 1.58 $g.cm^{-3}$ compared with a virgin average value of 1.63 $g.cm^{-3}$, representing an average weight loss of 3% through radiolytic oxidation in air during operation of the Pile.

4.2 Sample Activity

The activity of all samples was recorded before they were sectioned for other tests. Specific beta activities were in the range 200-1100 $Bq.g^{-1}$ with one isolated result of 2938 $Bq.g^{-1}$.

4.3 Total Stored Energy

Bomb calorimetry tests, primarily on the dust samples, provided a heat of combustion for comparison with a virgin value and hence the contribution of stored energy to the combustion of the sample. No reliable virgin value for this material could be identified and, as impurity concentrations may be quite high (as previously noted), this represents a significant potential error in the derived total stored energy values.

As expected, the greater values of stored energy were found towards the front and base of the pile, consistent with the most favourable combination of temperature and dose for Wigner-

energy accumulation. Values up to 1000 $J.g^{-1}$ have been returned; it is intended to improve the accuracy of these data by subjecting some of the trepanned material to a total anneal at 2300K and hence to obtain a more precise value for the base heat of combustion of the AGXP graphite.

The correlation of these results with the perceived dose and temperature distributions within the core during operation is not very successful: this is attributed to the several Wigner-energy anneals which took place towards the end of the reactor's operating period.

4.4 Thermal Conductivity

Thermal conductivity, which is an important parameter for thermal assessment of the core behaviour in the safe-storage safety case, is derived from thermal diffusivity measurements which employ a laser heat-pulse technique defined by a British Standard method (BS 7134: Part 4.2, 1990).

The minimum result obtained was 2.27 $W.m^{-1}.K^{-1}$. Figure 6 shows the relationship between total stored energy and the thermal conductivity. For other irradiated graphites a linear relationship is found between the fractional change of thermal conductivity and total stored energy, and this is tested in Figure 7. An unirradiated thermal conductivity value for the radial direction in the graphite (perpendicular to extrusions) has been taken to be 109 $W.m^{-1}.K^{-1}$ for this purpose. The data display a great deal of scatter; both parameters would be expected to be affected by the Wigner-energy anneals which took place during operation of the reactor, but not necessarily to the same degree. In the present context of data collection to support a safestorage safety case, it is the absolute values of the parameters which are important rather than relationships between them.

4.5 Rate of Release of Wigner Energy

Initial data were obtained on a differential scanning calorimeter using a heating rate of 2.5 $K.min^{-1}$, but this was later increased to 10 $K.min^{-1}$ with no significant effect upon the results. Typical rate-of-release curves are shown in Figure 8. Nearly all specimens show a large peak in stored energy release at around 200°C, especially in those samples taken towards the edges of the pile. No significant release was observed in any sample below 100°C. Comparison with the limited data available from the 1950s suggests that little self-annealing has taken place in the intervening 40 years. Again, the effects of partial anneals during pile operation have confused the interpretation of the results in terms of the temperature and dose profiles assumed for the pile.

4.6 Thermal Oxidation Rate

Measurements to date have been made at 673K using a gas-chromatograph to determine the off gases during oxidation in air, with the results then confirmed gravimetrically. Additional tests have been performed at a range of temperatures in order to obtain an activation energy. Generally, all results are in the range expected for this intrinsically reactive material. There was a high variability, which correlates with different concentrations of potential catalysts. The majority of oxidation-rate results fell within the range 30-700 $\mu g.g^{-1}.h^{-1}$, with a few isolated results reaching a maximum of 9450 $\mu g.g^{-1}.h^{-1}$. These high values, interestingly, lie some way outside the previously-defined "reactive zone", but nowhere match the high results previously obtained within it.

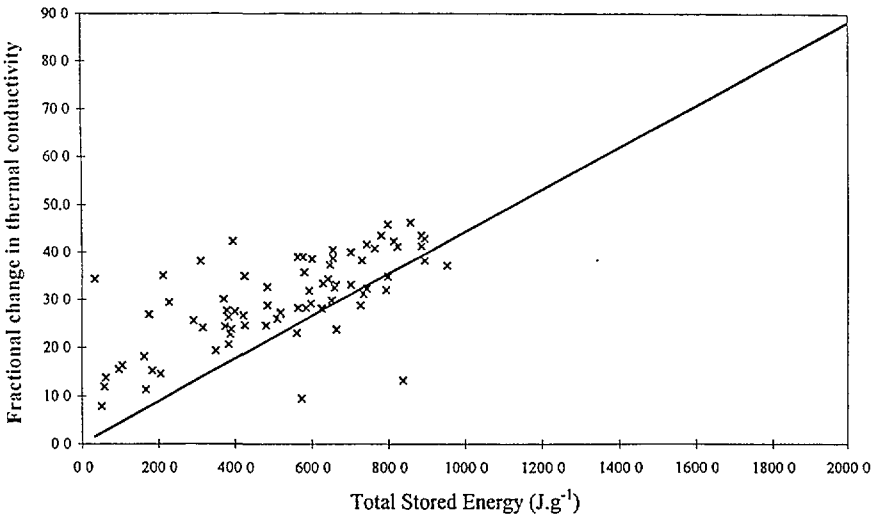

Figure 7. Test of linear relationship between fractional change in thermal conductivity and the total stored energy in Pile 2 graphite samples.

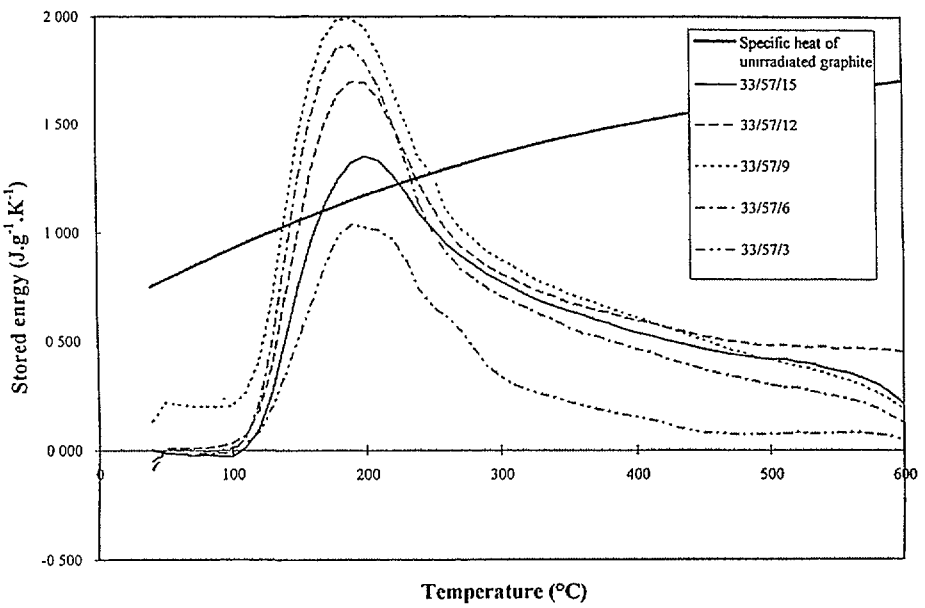

Figure 8. Typical curves for rate of release of stored energy for a number of blocks in Pile 2 channel 33/57 TR. Note that five out of six curves exceed the specific heat capacity of the graphite and thus that a spontaneous energy release is theoretically possible.

4.7 Summary of Pile 2 Graphite Data

In no case has any result been obtained which alters the previous perception of the status of Pile 2. This is extremely encouraging since it is considered that it will be relatively simple to prepare a safety case to modern requirements which will justify the continued safe storage of this reactor core without the imposition of special conditions. The experience gained from the dismantling of Pile 1 (where the potential problems associated with Wigner energy are less than in Pile 2) will ultimately be employed in the disposal of this reactor

5. PILE 1 TREPANNED SAMPLES

Trepanning of samples from Pile 1 is now in progress, although it has been severely delayed by mechanical problems associated with the reactor charge hoist which gives access to the channels. Measurements of thermal conductivity, stored energy and stored-energy release rate, and thermal-oxidation rate are expected to commence in mid-September 1997. An analysis of the implication of these data for the dismantling of Pile 1 will therefore be published later.

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GRAPHITE CORE STABILITY DURING "CARE AND MAINTENANCE" AND "SAFE STORAGE"

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Abstract

The current decommissioning strategy for the graphite-moderated reactors operated by Magnox Electric plc, Nuclear Electric Ltd and Scottish Nuclear Ltd is to delay dismantling and to initiate a monitored period of care and maintenance followed by a period of safestorage totalling up to 135 years. This philosophy has the considerable advantage of permitting the majority of radionuclides to decay, thereby minimising personnel dose during dismantling which itself will require far less complex remote-handling equipment. It also defers the disposal of the graphite and other components so that the provision of a deep land-based repository can be achieved.

A comprehensive review of all relevant data on the chemical, physical and mechanical properties of the graphite and its potential reactions, including radioactivity transport, has been undertaken in order to demonstrate that there are no potential mechanisms which might lead to degradation of the core during the storage period.

It is concluded that no significant experimental work is necessary to support the safestorage philosophy although, since the ingress of rainwater over long periods of time cannot be assumed incredible, a number of anomalies in chemical leaching rates may be worthy of re-examination. No other potential chemical reactions, such as the radiolytic formation of nitric acid leading to corrosion problems, are considered significant.

1. INTRODUCTION

The current policy of the United Kingdom electricity utilities Magnox Electric plc, Scottish Nuclear Ltd and Nuclear Electric Ltd, for the decommissioning of their Magnox reactors and Advanced-Gas Cooled Graphite-Moderated Reactors (AGRs), is to delay dismantling and allow a period of care and maintenance followed by a period of safestorage. Initially it is planned that these periods will total around 135 years, during which the majority of radionuclides will have decayed sufficiently to permit dismantling of the active cores with a greatly reduced requirement for shielding compared with that required should the work be undertaken immediately. In consequence, it has been important to consider the potential for degradation of the graphite cores during a period of at least 135 years (and therefore any consequences for the dismantling operation), with particular attention being paid to structural integrity and the release of radioisotopes.

The work described in this paper was undertaken as part of the UK programme on nuclear safety issues sponsored by the Health and Safety Executive, where specific concerns relating to the long-term storage of Pile Grade 'A' (Magnox) and gilsocarbon (AGR) graphite have been expressed. In particular, these concern the presence of Wigner energy (a form of potential energy

arising from the displacement of carbon atoms by colliding neutrons) and the potential for fire. The first of these is of no actual significance for the advanced gas-cooled-reactor graphite, which is operated at too high a temperature for there to be any significant accumulation of stored energy. Although the lower part of Magnox reactor cores does see some build up of energy over the lifetime of the plant, the total quantity is everywhere insufficient to lead to self-heating in any circumstances and is, in point of fact, well below the specified fraction of the specific heat capacity of the graphite set after the Windscale Pile fire of 1957 [1]. Furthermore, there is ample evidence from trepanning activities that there can be no possible energy-release hazard resulting from the eventual rough handling of this graphite, from frictional heating or any other means [2].

The present work has identified the more significant issues, taking due note of contemporary evidence from decommissioning activities in progress on other graphite-moderated plant. Within the United Kingdom these include the Windscale Production Piles, where Pile No. 1 is being prepared for dismantling [3], and the Windscale prototype AGR. Six Magnox reactors have also been shutdown and are in various stages of preparation for Care and Maintenance. The lengthy history of changes in the graphite properties which has been compiled from routine monitoring studies has also been utilised to assist the prediction of future behaviour.

The principal issues identified are discussed in the following sections.

2. CHEMICAL CHANGES

Graphite is generally extremely chemically inert, as its use in high-temperature applications such as crucibles and electrodes demonstrates. It is unaffected by strong alkalis and many strong acids and reactions are possible only in the presence of strong oxidising agents. In the present context it is necessary only to consider the potential for oxidation in the ambient environment of the reactor vessel during storage.

2.1 Consequences of Nitric Acid Formation

Nitric acid may be formed by the radiolysis of moist air and is principally of concern for the long-term integrity of the steel structures in the reactors during long-term storage. The residual radiation within the core during most of this period is insufficient to cause the formation of a significant quantity of nitric acid by radiolysis of the moist coolant (which during safe-storage is ambient air). It is in fact during the de-fuelling period when the greatest risk from this reaction occurs. Measurements obtained during this phase of decommissioning the Magnox reactors at Berkeley and Trawsfynydd. Coolant samples at Trawsfynydd contained a minimal 0.002 wppm, whilst metallic samples placed in the reactors collected small quantities - typically around 0.45 $\mu g.cm^{-2}$ nitrate on copper and much smaller quantities on both unoxidised and pre-oxidised steels. These low levels were also confirmed by swabbing of internal reactor ductwork, and do not represent any potential risk to storage.

Samples of graphite present in the Berkeley cores for most of the operating life yielded less than 0.47 µg nitrate per gram of graphite. This observation is important for two reasons:

(i) It validates earlier decisions that periodic purging of reactor vessels in care and maintenance or safestorage is unnecessary. The graphite core offers a substantial adsorption "sink" for removal of any nitric acid formed in the early stages during and immediately after refuelling (when the potential gamma-irradiation dose is at its highest value after final reactor

shutdown). This conclusion may be supported even when substantial pessimisms are incorporated into the estimates of nitric-acid production rate;

(ii) The very low nitric-acid concentration predicted, combined with the ambient storage temperature, make degradation of the graphite through the formation of intercalation compounds insignificant.

2.2 Oxidation of Graphite and Carbonaceous Deposits in Air and Water

Nuclear graphite is a material of low chemical reactivity because of the generally low concentrations of chemical impurities which might otherwise assist oxidation catalytically. The routine monitoring programmes during reactor operation have regularly confirmed the very modest rate of reaction - typically less than 100 $\mu g.g^{-1}.h^{-1}$ at a standard temperature of 723K - and also verified the activation energy for the reaction which is typically around 35-40 $kcal.mol^{-1}$ (145-170 $kJ.mol^{-1}$). These data have been required for verification of safety studies which model the hypothetical ingress of air into an operating reactor [2].

There exists, in addition, a vast literature on the oxidation of graphite and other carbons in air which has been the subject of numerous reviews - for example [4], complemented by extensive studies within the nuclear industry of the behaviour of relevant graphites. The catalytic activity of specific inorganic impurities is also well researched, for example [5]. This has enabled us to review the propensity of likely catalytic materials either initially present in small quantities, or potentially entrained into the reactor vessel through inleakage of rainwater during storage. These are likely either to be leached from concrete (alkaline earth materials) or washed-in steel-oxidation products (transition-metal compounds). These could give rise to potential problems both during the storage and during eventual dismantling. Even with pessimistic assumptions of the likely concentrations of these materials, the risk of encountering graphite of enhanced chemical reactivity during dismantling will remain remote even if cutting equipment inadvertently impinges upon the graphite. There also exists the possibility of introducing oxidation *inhibitors* at this final stage, based upon the experience with other uses of carbons such as in rocket nose cones [6].

Deposited carbon which is significantly more chemically reactive to air than the graphite is present within the graphite in all reactors which have been operated in coolants containing carbon dioxide, in which significant carbon monoxide is produced which then undergoes radiolytic polymerisation. The largest quantities are found in the lower part of the cores of the Magnox reactors, where the temperature is most favourable for the retention of this material. Existing data on the quantities and reactivities of these deposits again suggest that their presence is of no significance for safestorage or eventual core dismantling.

As already indicated, the presence of water (or water vapour) in the stored reactor vessels cannot be discounted. Direct reaction between carbons and water requires extremely high temperatures - typically >1200K - even when catalysts are present [4], and even over the long timescale associated with safestorage is of no consequence. Radiation-induced reaction may also be discounted at the low residual dose rates. Water itself has only a small effect on the rate of graphite and carbon oxidation in air, although the available data are rather ambiguous, with some indications of an increase in rate and others of a decrease.

2.3 Ignition Potential and Dust Explosibility During Core Dismantling

Graphite dust is classified as non-explosible by the UK's standard Fire Research Station (FRS) criteria [7] and it is considered that irradiation under Magnox or AGR conditions is not likely to affect this. Even for a formally "combustible" substance (such as an impure form of carbon), criteria relating to concentration, particle size, turbulence, the presence of an ignition source and a suitably oxidising atmosphere must be simultaneously satisfied before a flame front can propagate and give rise to an explosion, and this must in turn be within a confined space if it is to have significant consequences for structural stability. Furthermore, an explosibility assessment for the proposed French graphite-incineration process has been made, where the potential risk is far greater than in a stored core, and found to be acceptable [8].

Within the UK nuclear industry, tests have also been made to demonstrate that heating graphite to red heat, spattering it with weld metal and subjecting it to the standard FRS tests represented no hazard. Only in the presence of a powerful chemical ignition system has an explosion been achieved, and it is difficult to envisage how this circumstance could arise within a reactor vessel.

It is possible to build upon the experience of existing operating safety cases for the scenario of air ingress to demonstrate that temperatures of many hundreds of degrees Celsius would have to be achieved in a stored core before the aggregated heat-loss pathways could be overcome by the rate of chemical heat generation. This opinion is reinforced by new tests on the combustibility of nuclear grade H-451 graphite conducted at Los Alamos [9] which confirm the present authors' view that there is little evidence for combustion of graphite either in the Windscale Pile accident of 1957 (primarily a fuel fire, which recent observations of the channels adjacent to the fire zone tend to confirm [3]) or in the Chernobyl accident; oxidation of graphite occurred purely as a high-rate chemical reaction and without the production of flames or fire propagation. These views are also shared by researchers from Brookhaven National Laboratory [10] and it considered that five conditions must be satisfied before burning is possible; a high surface area-to-volume ratio; a temperature of at least 923K; adequate oxygen supply but not sufficient gas flow to cause cooling; a high intrinsic reaction rate and low heat losses. Satisfying these conditions is considered to be incredible either in the storage regime or during subsequent dismantling.

3. GRAPHITE PROPERTY CHANGES

The UK nuclear industry has assembled a comprehensive database of virgin and irradiated graphite properties for the PGA and gilsocarbon graphites employed in the Magnox reactors and AGRs respectively, but there is no systematic study of the potential for change over long periods of time or under changing atmospheric conditions. This possibility was highlighted as early as 1959 [11] when it was noted that graphite was stronger under vacuum than in air, with various other atmospheres giving different results. However, it has subsequently been suggested that it was, in fact, the water concentration which was critical in these results, and this view is supported by more recent work from the UK and Japanese nuclear industries.

The changes are of no significance for the present purpose because most data on graphite properties have in any case been obtained in apparatus where the graphite was exposed to moist laboratory air. There are no other observations of changes in mechanical or physical properties which are likely to arise as a result of long-term graphite storage in the reactor vessels.

4. RELEASE OF RADIOACTIVITY

4.1 Leaching from Graphite

The rate of leaching of radionuclides from graphite is likely to be governed by:

(i) The rate of corrosion of the graphite itself;
(ii) Selective dissolution from other phases within the graphite cores;
(iii) Desorption from the graphite (or other phases); and
(iv) The presence in the leachate solution of substances such as complexants and acids or alkalis.

Process (i) is expected to be the major process affecting the dissolution of radionuclides present in the graphite lattice (principally ^{14}C) or strongly adsorbed thereon. Processes (ii) or (ii) could result in more rapid leaching of the radionuclide than the graphite.

There are numerous French and US studies [12-15] and also analyses for a UK Magnox reactor [16-17]. There are also studies based on graphite particles collected from the vicinity of Chernobyl [18]. Overall, leaching data suggest that the results are strongly influenced by the nature of the graphite under examination. In most cases the leach rates both for carbon atoms and entrained isotopes appear to have reached an equilibrium rate within 140 days exposure to water in approximately neutral pH, but the data are variable and sometimes inconsistent. Generally, the relative leach rates of the isotopes present in irradiated graphite after shorter-lived isotopes have decayed are:

$$^{137}Cs > ^{133}Ba > ^{60}Co \approx ^{63}Ni > ^{134}Cs > ^{36}Cl > ^{154}Eu > ^{3}H > ^{14}C$$

For radioelements for which no release data are available, it is possible to make pessimistic estimates based upon thermodynamic solubility calculations. A suitable database has already been created by UK NIREX in connection with their studies on the proposed deep waste repository. In combination with kinetic data, the results may be used to estimate the release of ^{3}H and ^{14}C into the air and, in combination with water flowrate data (where relevant) to assess the overall release of soluble radioactivity from stored graphite. The integrated release of ^{3}H is unlikely to be a significant problem in safestorage because of the relatively short half life. It has been suggested that bench-top tests on graphite samples removed from the reactors in question would enable the accuracy of leach-rate estimates to be considerably improved although there appears to be little prospect of a significant problem on the basis of existing results.

4.2 Gas-Phase Activity Release

^{14}C is retained in the graphite and associated carbon deposits primarily as elemental carbon; on surfaces, some oxygen will be associated with it. ^{14}C produced in reactions other than from ^{13}C (*i.e.* from ^{15}N or ^{17}O) is likely only to be present in surface oxide and in the carbonaceous deposits (C/H/O compounds of high molecular weight), which are potentially more mobile in a turbulent flow or leaching situation, but the total activity in this form will be relatively small when safe storage commences.

In order for ^{14}C to be released in significant quantities by exchange with gas-phase species, replenishment by internal solid-state diffusion would be required. There is no significant diffusion of carbon within carbon until a temperature of over 2000K is reached [19], an observation

confirmed by the extreme temperatures needed to "label" graphite with ^{14}C from ^{14}CO, a technique used in our own laboratories to prepare samples for radiolytic oxidation tests.

The sources of tritium are lithium impurity in the graphite, and the fission process in the fuel followed by diffusion in the fuel cladding and absorption by graphite from the coolant. Tritium is transferred from the graphite surface to the coolant with extreme ease [20, 21]. This implies that tritiated species on the graphite surface will have exchanged tritium with atmospheric moisture and that this tritium will have been lost very quickly after reactor operation ceased - probably within a matter of weeks. Further releases will depend upon the rate of diffusion of tritium within graphite, which is very much slower than the surface-exchange rate but remains significant at reactor operating temperature [22]. However, the high activation energy for the process implies a rapid reduction in diffusion rate with diminishing temperature, and it is considered to be of low significance. The possibility of biological processes leading to releases of active gases has been considered in parallel studies and is considered remote.

4.3 Particulate Release

The greatest potential for particulate release during reactor vessel safestorage undoubtedly arises from the corrosion of metallic components within the core. The graphite may also release particulates directly, through degradation (friction and wear) mechanisms, or from deposited carbon or other entrained materials such as oxide dusts.

Carbonaceous deposits are generally adherent to the graphite and are only likely to be released under conditions where the graphite is itself abraded, or with severe atmospheric disturbance within the vessel, or through graphite oxidation. It is extremely difficult to envisage how any of these processes can occur. Such friction data as are available (J. Skinner, Magnox Electric plc, personal communication) appear to support this view: coefficients of friction on graphite tend to fall as surfaces "run in", and the low temperature of storage favours the retention of adsorbed gases, which also reduces the friction coefficient.

In the low probability of particulate release, evidence from the routine graphite-monitoring programmes indicates that the principal emitters will be ^{60}Co, ^{14}C, ^{3}H and ^{35}S, with the proportions of those with shorter half-lives declining with time. The potential releases will be quite different in different reactors, as a result of their operating histories, but it should also be remembered that safestorage requires the effective elimination of release pathways in any case. Activity release during graphite incineration, if such a route were eventually taken, has been extensively considered but is outside the scope of this work.

Attrition of graphite by biological activity is possible but, again, complementary work suggests that the probability is low even over >100 years.

5. CONCLUSIONS

This paper has presented a brief summary of the detailed evidence evaluated in connection with the behaviour of graphite cores during the care and maintenance and safestorage periods of reactor decommissioning and has in particular considered aspects relating to both structural integrity and activity release. The retention of the graphite core within the vessel for periods of up to 135 years does not present any significant hazard either during the storage period or during subsequent dismantling. This conclusion is based upon consideration of the potential for changes

in physical, chemical and mechanical properties, with parallel work having covered the possibilities of biological action, and upon consideration of events such as water inleakage and the potential for radioactivity releases.

Additional reassurance would be available from a limited number of leaching tests on samples of the irradiated graphite, and by comparing the final graphite properties (particularly the oxidation rate in air) with the predictions after storage but ahead of the dismantling operation.

The present UK strategy of delaying full dismantling of reactor cores has numerous advantages and does not appear to be compromised in any way by the degradation of the graphite core.

ACKNOWLEDGEMENTS

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THE CARBON 14 CYCLE

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Abstract

The purpose of this report is to assess the carbon 14 releases from a contaminated graphite incinerator and to compare these releases and their radiological impact with other ways in which carbon 14 is produced, both naturally and artificially in waste produced by the nuclear industry.

1. Recapitulation - Ways in which carbon 14 is formed

Carbon 14 has a half-life of 5730 years. It would have disappeared from the earth's atmospheric carbon dioxide a long time ago if it were not re-formed by the action of cosmic neutrons on the nitrogen in the atmosphere.

$$^{14}_{7}N + ^{1}_{0}n \rightarrow ^{1}_{1}H + ^{14}_{6}C$$

The carbon 14 formed is oxidized in the air. It is then converted into CO_2 and penetrates by photosynthesis into the metabolism of the plant and animal world. Assuming constant intensity of the cosmic radiation, the proportion in the carbon dioxide in the air, $^{14}C/C^{12}$, remains constant and equal to:

$$1.2 \times 10^{-12}$$

In plants assimilating CO_2, the ratio of the concentrations in all their organic substances equals 1.2×10^{-12} while the plants are alive. The same is true for animals and humans feeding on plants.

After death, injection of carbon dioxide into the plants ceases and the ratio $^{14}C/C^{12}$ decreases following the formula:

$$^{14}C/C^{12} \rightarrow 1.2 \times 10^{-12} \times e^{\frac{-0.69\,t}{5730}}$$

where t = time in years.

Measurement of radioactive carbon 14, which emits a β particle with an energy of 0.156 MeV, and comparison with the C^{12} concentration allows the age of the organism (plant, animal, human) to be estimated, provided it lies between 1000 and 50 000 years.

Carbon 14 is produced continuously in the atmosphere at a rate of 27 000 curies or 10^{15} Bq per year by the action of cosmic neutrons.

The following three main modes of production of carbon 14 can be identified in nuclear reactions.

Reaction	Capture cross section in barns (10^{-24} cm²)	Abundance of isotope in %
^{14}N (n, p) ^{14}C	1.8	99.63 N^{14}/Nitrogen
^{13}C (n, γ) ^{14}C	0.9 x 10^{-3}	1.1 C^{13}/Carbon
^{17}O (n, α) ^{14}C	0.235	0.04 O^{17}/Oxygen

1 g of carbon 14 represents an activity of 4.6 curies.

2. Carbon 14 releases into the atmosphere

The nuclear tests carried out in the atmosphere since 1945 and particularly in the period from 1954 to 1962 were an important source of releases.

The United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) has made an inventory of the 423 atmospheric explosions that produced a large quantity of radionuclides in the atmosphere (see Table 1).

Country	Period	Number	Power (Mt)
USA	1945-1962	193	138,6
USSR	1949-1962	142	357,5
UK	1952-1962	21	16,7
France	1960-1974	45	11,9
China	1964-1980	22	20,7

Table 1 - Main radionuclides due to nuclear tests

Radionuclide	Half-life	Activity
^{3}H	12.3 years	$2,4.10^{20}$ Bq
^{14}C	5730 years	$2,3.10^{17}$ Bq
^{54}Mn	313 days	$5,2.10^{18}$ Bq
^{55}Fe	2.7 years	$2,0.10^{18}$ Bq
^{85}Kr	10.7 years	$1,6.10^{17}$ Bq
^{90}Sr	28.6 years	$6,0.10^{17}$ Bq
^{89}Sr	50.5 days	$9,0.10^{19}$ Bq
^{106}Ru	368 days	$1,2.10^{19}$ Bq
^{131}I	8 days	$7,0.10^{20}$ Bq
^{137}Cs	30.2 years	$9,6.10^{17}$ Bq
^{140}Ba	12.8 days	$7,2.10^{20}$ Bq
^{144}Ce	284 years	$3,0.10^{19}$ Bq
^{238}Pu	87.7 years	$3,3.10^{14}$ Bq
^{239}Pu	24 100 years	$7,8.10^{15}$ Bq
^{240}Pu	6570 years	$5,2.10^{15}$ Bq
^{241}Pu	14.4 years	$1,7.10^{17}$ Bq
^{241}Am	433 years	$5,5.10^{15}$ Bq

Carbon 14 is produced by the intense neutron fluxes on the nitrogen in the atmosphere and the cumulative production of carbon 14 by atmospheric explosions is estimated at 6×10^6 curies, i.e. 220 years of natural production.

Assuming that 90 % of the carbon 14 is projected to high altitude following the explosion, UNSCEAR has calculated that the integrated dose over the average human lifetime due to exposure to carbon 14 fallout would be in the order of **0.3 mSv, i.e. 30 mRem.**

In power reactors, the materials subjected to neutron flux contain nitrogen as an impurity and also carbon and oxygen as major constituents of structures, fuel or the moderator.

For each type of reactor, it is possible to draw up an approximate inventory of the production of carbon 14, in either solid or gaseous form.

Table 2 gives the annual production of carbon 14 in both forms for the main types of reactor assuming operation of each reactor for 7000 hours a year.

Table 2 Carbon 14 production by different reactors

Type of reactor	Power in MWe	Total releases in curies/year	Gaseous effluents		Solid waste	
			Reactor	Reprocessing	Reactor	Reprocessing
PWR	900	50	4	14	18	14
	1300	70	5	20	25	20
Magnox or	200 (1)	77	10	15	50 (1)	2
			10	30	72 (2)	3
Gas-cooled	450 (2)	115				
AGR	660	110	8	5	72	25
PHWR						
CANDU	600	150	110	22		19
BWR	1000	75	8	16	35	16

(1) Example: Chinon A2: 1700 tonnes of graphite - 20 years operation
Pile activity = 50 x 20 = 1000 curies

(2) Example: SL A: 2400 tonnes of graphite - 18 years of operation
Pile activity = 72 x 18 = 1296 curies

Specific activity of graphite = 2×10^4 Bq/g

As an example, in PWRs carbon 14 is produced mainly in solid form by interaction between neutrons and the nitrogen in the stainless steel used to make the reactor internals, the nitrogen content ranging from 0.04 to 0.08 %.

Gaseous carbon 14 is produced mainly by activation of nitrogen entering the primary coolant and is released in the gaseous effluents. The average PWR release rate is about 5 to 10 curies per year of gaseous carbon 14.

For graphite reactors, the main carbon 14 production route is through irradiation of the moderator and, out of the 50 curies per year produced by a 200 to 250 MWe gas-cooled graphite-moderated reactor, 60 % is produced by interaction with the nitrogen impurities in the graphite and 40 % by interaction with the carbon 13 contained in the graphite pile.

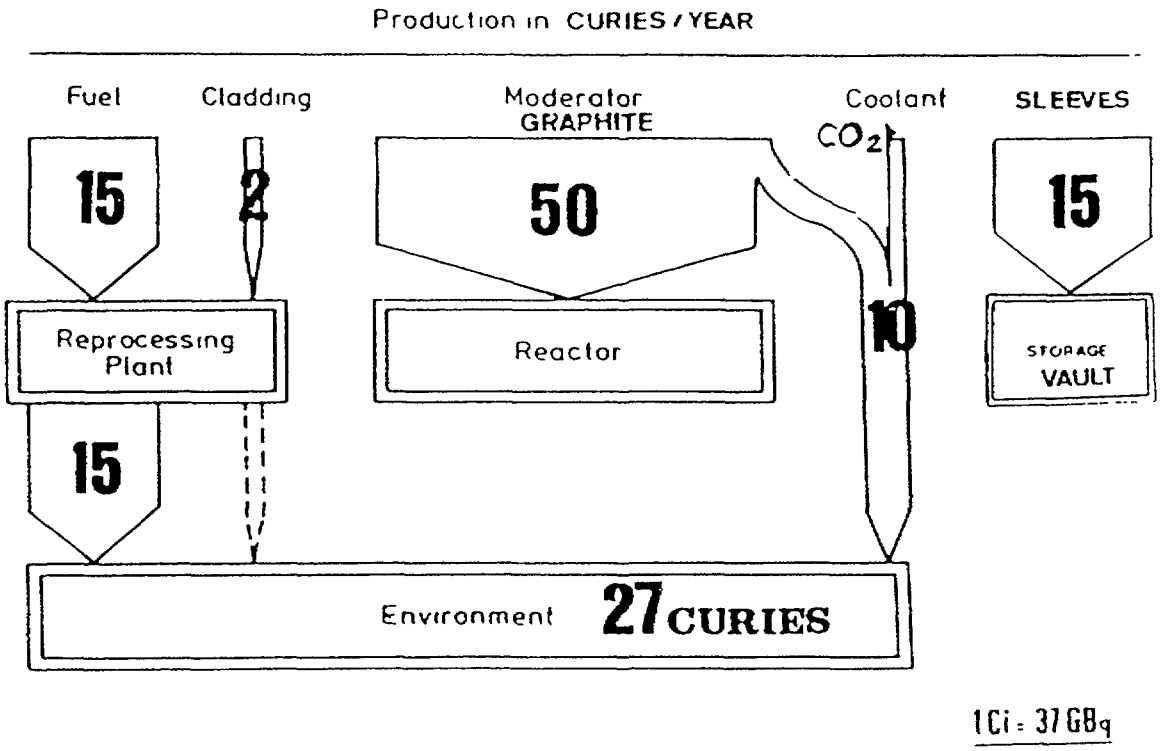

Fig 1 Schematic diagram of flows of ^{14}C produced in a 200 MWe Magnox reactor

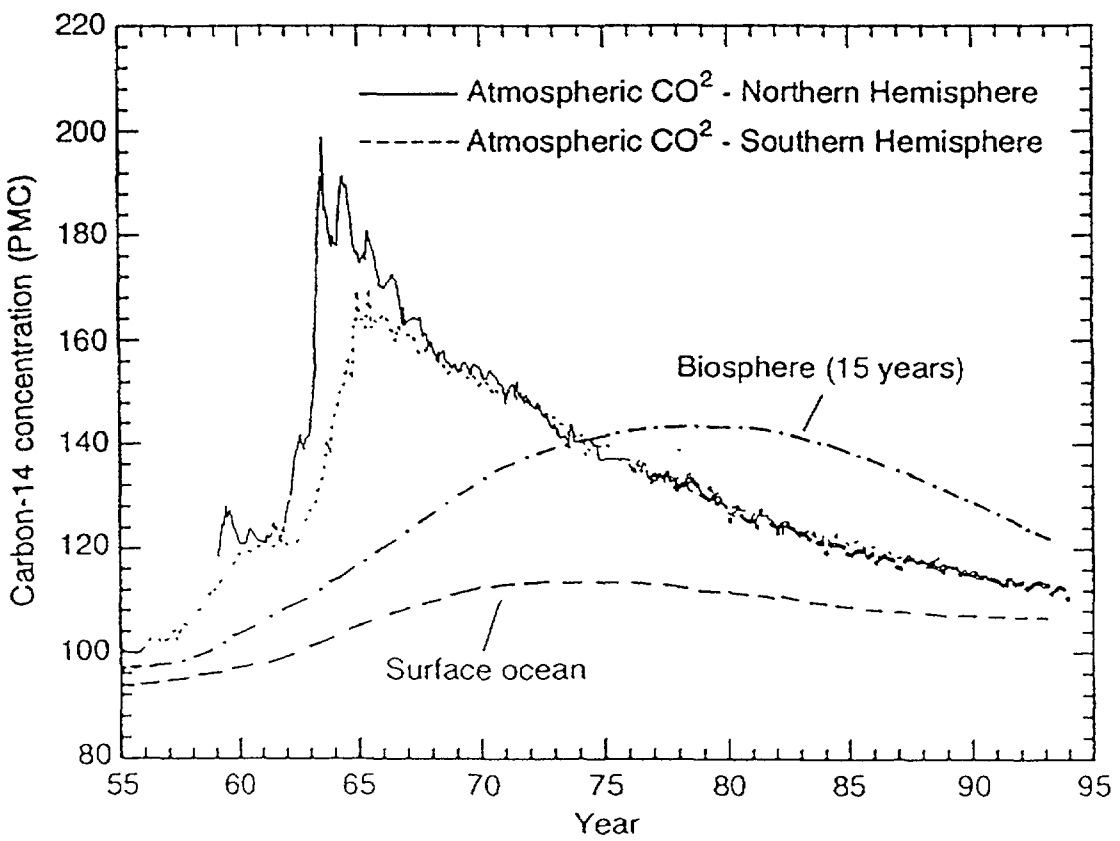

Fig. 2 Carbon-14 concentration in atmosphere; 1955 to 1994

A large portion of the carbon 14 gaseous releases from gas-cooled reactors comes from the purification of the CO_2 circuits used to cool the reactor and from the isotopic exchange between the moderator and the CO_2 circuit.

A gas-cooled reactor discharges into the environment about 27 curies of carbon 14 a year on average, i.e. 10^{12} becquerels (see Figure 1).

3 Estimate of releases from an incinerator for contaminated and irradiated graphite

Fluidized bed incineration of contaminated graphite permits capturing the radionuclides contained in the graphite, except for the releases of tritium and carbon 14, which is converted into CO_2.

Since the average concentration of carbon 14 in the moderator graphite is 2×10^4 Bq/g, the gaseous releases from an incinerator capable of processing 600 tonnes of graphite a year will be 12×10^{12} Bq, i.e. 320 curies per year.

This corresponds to an increase of 1 % of the natural annual production; however, the impact must be measured over time, because of the long half-life of carbon 14 (5730 years).

Saturation can be considered to be reached after a time period equivalent to four half-lives, i.e. 20 000 years in the case of carbon 14.

The atmosphere's carbon 14 inventory is around 1.2×10^{17} becquerels, i.e 3.1×10^6 curies.

An incinerator operating for 50 years at a rate of 600 tonnes per year, i.e. incinerating a total of 30 000 tonnes of graphite, will produce an increase in the inventory of 5×10^{-3} times that from natural sources, which is small compared to the natural fluctuation of carbon 14 in the atmosphere (± 2%) in the last one hundred years.

The radiological impact calculated by several writers, including the IPSN, shows that, provided that precautions are taken during discharge (stack height, daytime or nighttime conditions), gaseous releases remain acceptable for the most exposed group according to ICRP (International Commission on Radiological Protection) criteria; in any case, it is much lower than the limit of 1 mSv/year established for the general public (ICRP 60) being, in fact, about 4 % of the allowable limit.

CALCULATION OF NEUTRON FLUXES AND RADIOACTIVITIES IN AND AROUND THE TOKAI-1 REACTOR PRESSURE VESSEL

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Abstract

This paper describes work that has been performed by NNC Ltd and Fuji Electric for the study of decommissioning of Tokai power station (Tokai-1). The objective was for NNC to provide an independent validation of a representative selection of the existing Fuji Electric calculations the results of which were obtained by the methods generally used in Japan based on the discrete ordinate code DOT 3.5 [1], in estimating full power neutron fluxes and reaction rates in components located within the reactor biological shield of the Tokai 1 reactor. The calculational methods and modelling assumptions are described for the four regions in which fluxes and reaction rates were required, namely in regions above the core, regions to the side of the core, regions below the core and regions in the concrete walls of the bioshield gas duct penetrations. NNC has considerable experience in performing similar analyses for UK reactors and the methodology and computer codes employed here are based on experience gained in carrying out such work for AGR, PWR and Magnox reactor types. Thus, much of the component modelling has been achieved using the Monte Carlo code MCBEND [2] supplemented, in the case of the gas duct penetrations, by the iterative kernel albedo code MULTISORD [3].

Above, below and to the side of the core, results were obtained in some detail in nearly all of the structural components. In the case of the bioshield concrete, results were obtained in many regions at various depths and axial heights. Along the gas ducts, results were calculated at the concrete wall surfaces of the penetrations to the point where the total flux had reduced to a level of 10^3 n/cm^2/s, this being the level at which the induced concrete activity can be regarded as negligible. Preliminary calculations were carried out using the duct streaming code MULTISORD in order to establish the approximate location where flux levels dropped to this level. MCBEND was then used to model the geometry in detail up to this point. The MCBEND calculations were very computationally intensive. In total several hundred hours of CPU time was required.

The timescale of the project has only allowed a limited amount of comparison between the DOT 3.5 and MCBEND predictions at this point in time. An initial comparison has been made within the biological shield in the bottom corner region of the reactor vault where good agreement was obtained. It is envisaged that the neutron fluxes derived from this work may be used to determine the activity of neutron activated structures in order to evaluate the dose rates that are likely to be experienced during dismantling. This data should also be useful in calculating the costs of waste disposal and in estimating worker and public dose exposure during decommissioning.

1. Introduction

Fuji Electric have used the Discrete Ordinates code DOT 3.5[1] to calculate neutron flux levels and reaction rates in components within and including the reactor biological shield of the Tokai-1 reactor. A brief description of the modelling of Tokai-1 used for the DOT 3.5 calculations is given in section 2. In order to validate these calculations, Fuji Electric asked NNC to perform prescribed analyses based on current UK practices using Monte Carlo methods based on the Monte-Carlo code MCBEND [2] supplemented where applicable by the duct streaming code MULTISORD [3]. Neutron transport calculations were performed around the bottom, top and sides of the core and along the bio-shield gas duct penetrations. Detailed descriptions of the MCBEND and MULTISORD modelling assumptions are given in section 3.

At this point in time only a limited comparison between the DOT 3.5 and MCBEND predictions has been possible. A limited comparison of these results for the bottom corner concrete bioshield region of the reactor vault is given in section 4 which shows good agreement.

2. The Discrete Ordinates DOT Calculational Model

The Tokai-1 reactor is shown in section below in Figure 1 together with the representative two dimensional model used for the DOT 3.5 calculations in Figure 2.

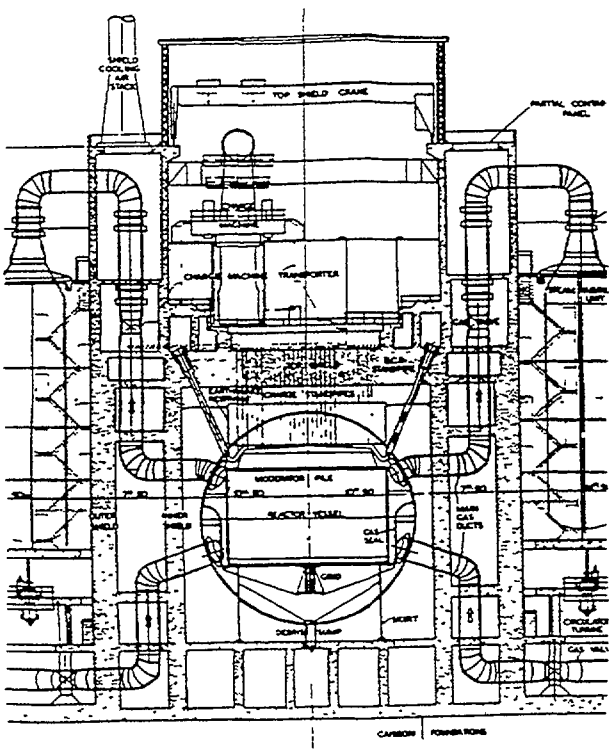


FIG. 1. Section of the Tokai reactor

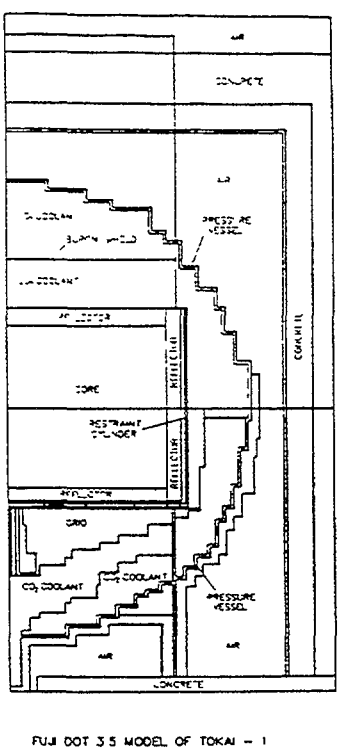


FIG. 2. Fuji DOT 3.5 model of TOKAI-1

The neutron production rate source terms were derived from a common origin, this being polynomial fits to measured axial and radial profiles supplied by JAPC. The data was taken from reactor information and normalised to a core average of 6.327×10^{10} n cc^{-1} s^{-1}. A 15 group neutron energy scheme was used for the output fluxes is given below in Table 1 and fifteen selected reaction rates were also calculated for those reactions listed below in Table 2.

TABLE 1. ENERGY GROUP SCHEME

Group	Upper	Lower
1	17.3	5.22
2	5.22	3.165
3	3.165	1.82
4	1.82	1.185
5	1.185	0.7065
6	0.7065	0.4076
7	0.4076	0.2472
8	0.2472	0.1576
9	0.1576	5.656E-2
10	5.656E-2	2.418E-2
11	2.418E-2	7.103E-3
12	7.103E-3	3.035E-3
13	3.035E-3	9.611E-4
14	9.611E-4	3.536E-4
15	3.536E-4	1.301E-4
16	1.301E-4	3.727E-5
17	3.727E-5	1.068E-5
18	1.068E-5	3.059E-6
19	3.059E-6	8.764E-7
20	8.764E-7	4.14E-7
21	4.17E-7	0.0

TABLE II. NEUTRON REACTIONS CALCULATED

Reaction	Reaction	Units
1	Total Flux	n/cm2/s
2	Neutron Dose Rate	μSv/h
3	Co59(n,γ)Co60	reactions/atom/s
4	Fe54(n,γ)Fe55	reactions/atom/s
5	Ni58(n,γ)Ni59	reactions/atom/s
6	Ni62(n,γ)Ni63	reactions/atom/s
7	Nb93(n,γ)Nb94	reactions/atom/s
8	Eu151(n,γ)Eu152	reactions/atom/s
9	Eu153(n,γ)Eu154	reactions/atom/s
10	Eu154(n,γ)Eu155	reactions/atom/s
11	Ho165(n,γ)Ho166	reactions/atom/s
12	Ag107(n,γ)Ag108m	reactions/atom/s
13	Li6(n,α)H3	reactions/atom/s
14	Cl35(n,γ)Cl36	reactions/atom/s
15	Ca40(n,γ)Ca41	reactions/atom/s

3. The Monte Carlo Calculational Models

3.1 General

Experience has shown that attempting to represent the whole of the reactor in one, vast model would have been impossibly demanding of computer resources and so, as mentioned above in the introduction, the distribution of neutron fluxes was determined in four separate calculations corresponding to the modelling of

(a) the regions above the core

(b) the regions to the side of the core

(c) the regions below the core and

(d) the gas duct penetrations of the bio-shield

The neutron source terms used for the core regions were similar to those used in the DOT 3.5 calculations as described above in section 2. Also the output 21 group neutron energy scheme was as detailed above in Table 1 and the same reaction rates as listed above in Table 2 were prescribed.

For the gas duct streaming calculations the neutron source terms at the mouth of the ducts were derived from fluxes already calculated at this location by the core side and sub-core models. For each of the regions listed above, this paper describes the modelling that was performed and diagrams are given to illustrate how some of the components have been represented. Computer screen dumps are also included to confirm the modelling actually being used in the calculations. The engineering drawings on which the models are based supply 'cold' dimensions and these have been used in this work assuming that:

(a) There is perfect concentric alignment in the fuel channels, standpipes and guide tubes and that the expansion at the temperatures of the operating reactor will not significantly affect streaming gap dimensions (typically ~ 1% increase in cross-sectional area)

(b) The design of the Tokai fuel channels is such that they are essentially open without restriction until well into the above core concrete, where annular streaming paths are stopped with plug/penetration overlaps sufficient to accommodate expansion effects.

(c) The below core geometry relevant to radiation streaming is essentially un-attenuating apart from the coolant gas. For the larger structures there is no need to account for expansion effects, provided the overall mass of the materials is represented accurately.

3.2. Monte Carlo Calculations in the Regions Above the Core

3.2.1. General Description

The Tokai-1 reactor top shield comprises the following civil structures:

(a) graphite stack (core and reflector)

(b) channel guide pots

(c) boron steel plate

(d) pressure vessel

(e) primary concrete shield.

All the structures from the boron steel plate and above are penetrated by a vertical fuelling standpipe which allows fuel and control rods to be inserted into, and removed from, the reactor. Each fuelling standpipe serves 12 fuel channels clustered around a central control rod channel as shown in Figure 3. Multiples of the array shown in Figure 3 can be interlocked to cover the whole core completely.

The standpipe penetration through the primary concrete shield is stepped (at the half thickness level) and contained within a roof above which is similarly stepped. In addition, at the level of the steps, the roof sleeve is enlarged to accommodate a shield muff. Within the standpipe, above and below the step, are large shield plugs, the diameter of the upper being significantly larger than the internal diameter of the lower part of the standpipe. Thus the neutrons which stream through the penetration are considerably attenuated.

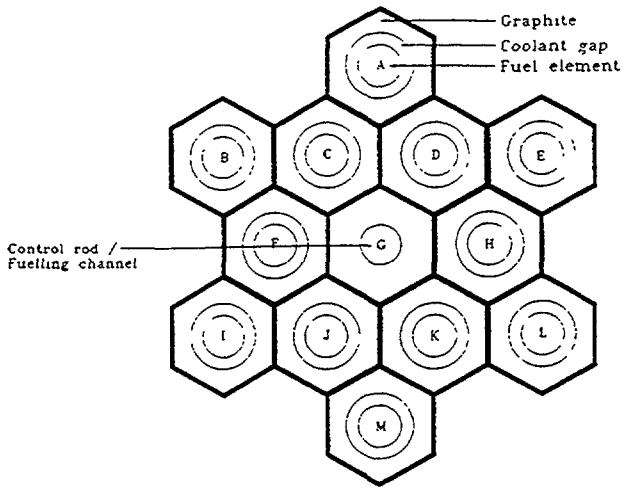


FIG. 3. Above-core Model, Standard Array Of 12 Fuel Channels and 1 Control Channel

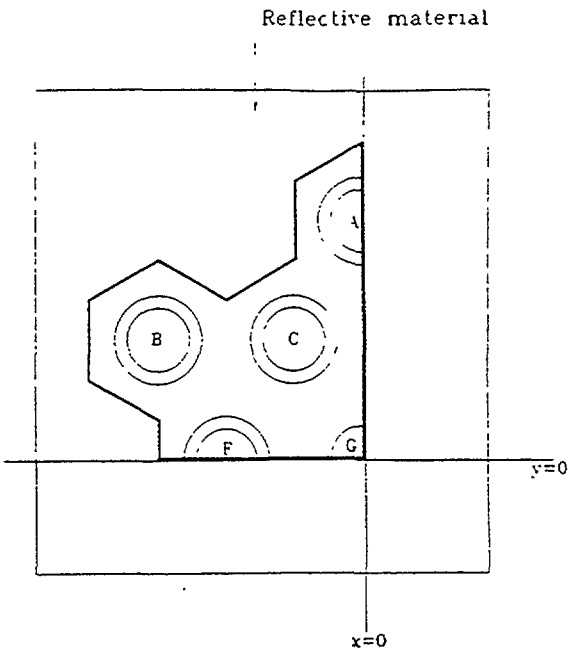


FIG. 4. Above-Core model, MCBEND Representation

Due to the spherical design of the pressure vessel, the standpipes which penetrate it and their associated nozzles are closer to the graphite stack and up to three times longer, at the core edge channel positions, relative to those at the core centre.

In general the fuel channels contain fuel elements in a uniform array, but the occupancy of a control rod channel may vary in accordance with its purpose for core control. It may contain either a fine or coarse control rod at an appropriate insertion level, an emergency shutdown device (ESD) or other core viewing or monitoring devices. The possible occupancy of the 185 control rod channels is given below:

(a) 34 fine control rods at around 35% insertion

(b) 8 sector control rods at around 10% insertion

(c) 67 coarse control rods at around 0% insertion

(d) 44 ESD units (open channels)

(e) 32 other viewing and monitoring units.

3.2.2 Geometry Modelling Details

The MCBEND model is based on a unit array of 12 fuel channels and 1 control rod channel, as shown in Figure 3. A further refinement was made whereby advantage was taken of the fact that the array is symmetrical about the X and Y axes and by the use of reflecting material, the reduced, equivalent arrangement shown in Figure 4 could be used.

Two calculations were performed:

(a) For the core axis location. This representation is shown as the shaded area close to the reactor axis in Figure 5. It was assumed that the control rod channels were empty, corresponding to the situation of the control rods being fully withdrawn.

(b) For the core edge location. This representation is shown as the shaded area at the edge of the core in Figure 5. The control rod channels were again assumed to be empty. As well as the source term data being different from the core centre case, the geometric details of the model were also different since the axial heights of some of the components are not the same as those on the core axis due to the curvature of the reactor pressure vessel.

For both cases the above core model is composed of various regions which are described in the following sub-sections and shown in Figure 6. The representation of these regions is shown in more detail in Figure 7.

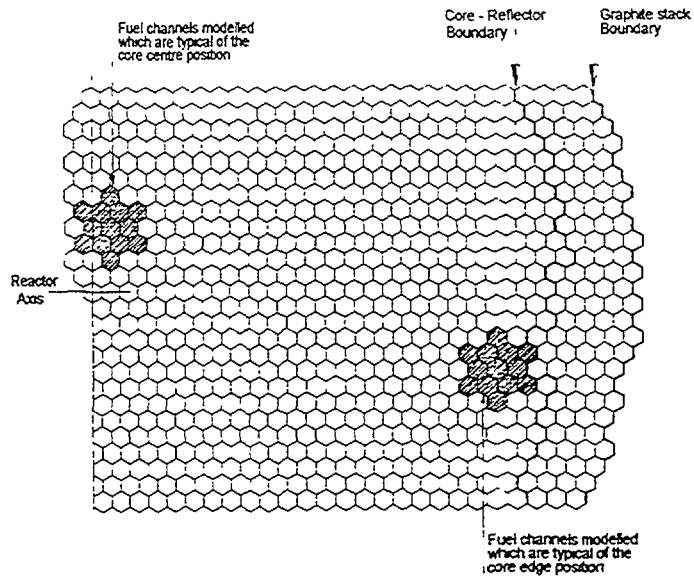


FIG. 5. Part Plan of Reactor Core

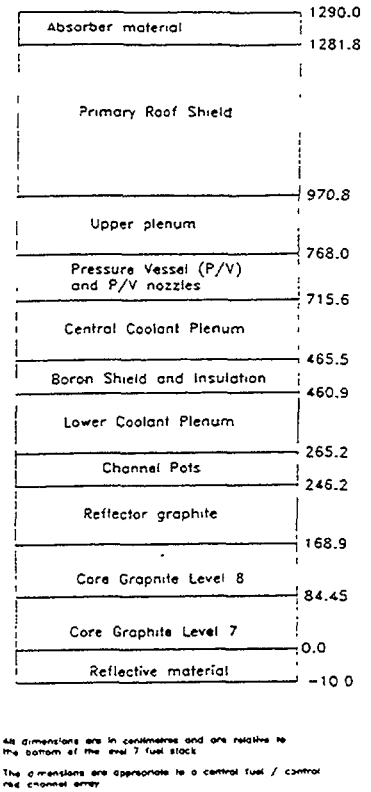


FIG. 6. Above Core Model, General Section

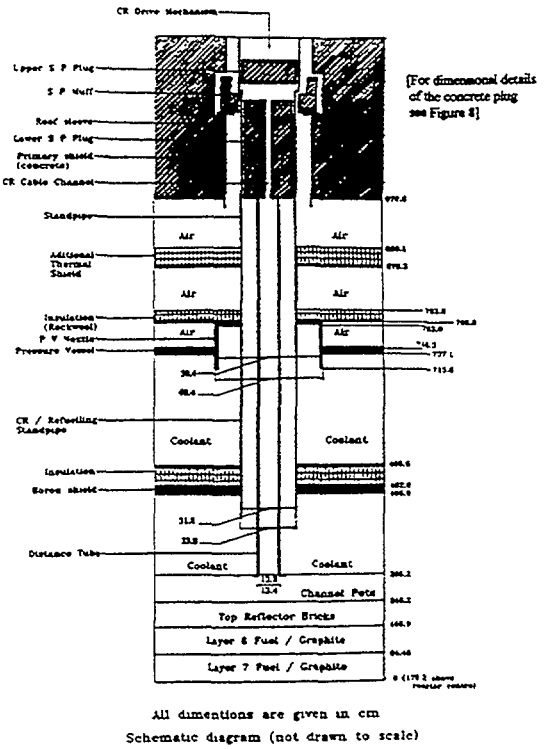


FIG. 7. Above Core Model, Detailed Section

3.2.3 Core Graphite Regions

For both the on-axis and core-edge cases, the top two fuel elements only were represented as it was anticipated that no significant increase in streaming neutrons will be contributed by fuel graphite layers 1-6. Thus the models start at the base of layer 7. A sensitivity study to test this assumption is described in section 3.4.1 since it also applies to the calculations described there.

No allowance was made in the model for inter-brick gaps. The contribution to the neutron fluxes from streaming up these gaps was judged to be insignificant (<2%) compared to the contributions coming from the open channels through the reflector. Inter-brick gaps are narrow (nominally 0.6 mm) and short due to the brick design of interlocking lugs and rebates. Their significance is a judgement based on the relative cross-sectional areas of the gaps around the top fuel channel reflector brick and the area of the central cooling hole in comparison. The comparison of the relative contributions based on areas alone leads to an overestimate of the effect since no account is taken of the fact that neutrons can stream directly from the fuel up the open fuel channel, whereas in order for neutrons to stream up the inter-brick gaps, graphite penetration and a scatter are first necessary.

3.2.4. Channel Guide Pots and Boron Shield Regions

The conical shape of the fuel channel pots which allow for ease of refuelling was taken account of in the computer models. The boron shield is a sandwich of 2% boron steel between wide steel plates which, for MCBEND modelling purposes, was homogenised. Also included in this region was the metal foil insulation.

3.2.5. Pressure Vessel and Nozzles Region

The two models described in section 3.2.2 above were assessed. These corresponded to:

(a) Where the pressure vessel and the nozzle is at its further possible distance from the top of the graphite stack (i.e. on the core axis) and

(b) Where the pressure vessel and the nozzle are at the least possible distance from the top of the graphite stack (i.e. at the core radial edge). This model allowed for

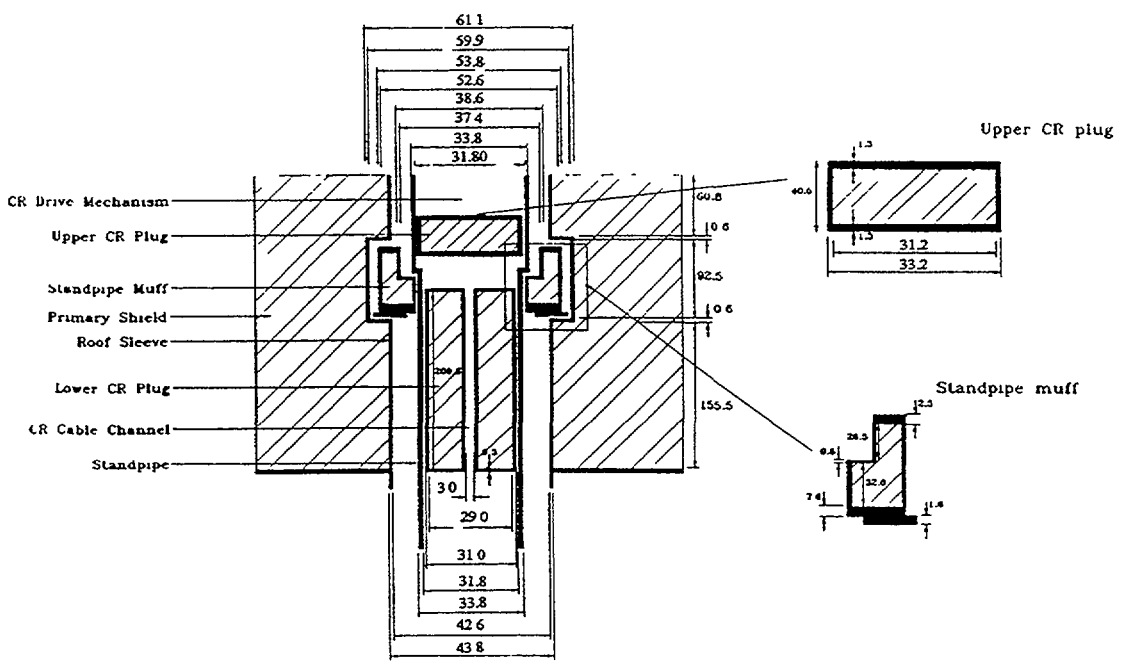
(1) the effects of lower neutron sources due to the radial form factor

(2) the reactor pressure vessel being closer to the graphite stack and

(3) the reactor pressure vessel nozzle at the core edge being approximately three times the length of the nozzles on the core centre line.

3.2.6. Concrete Roof Shield and Closure Plugs

The details of the standpipe penetration through the concrete roof shield is shown in detail in Figure 8. The presence of the shield muff, the steps in the standpipe and roof sleeve and the shield plugs are such as to prevent any direct streaming lines-of-sight, even when allowances are made for expansion and/or distortion from concentric alignment due to hot conditions.



Schematic diagram, not drawn to scale

All dimentions are given in cm

FIG. 8. Standpipe Penetration Roof Shield

Several computer generated sections of the above core models are shown below.

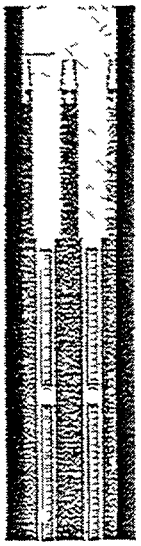


FIG. 9. Above Core Model- Top of Fuel Channels

FIG. 10. Above Core Model- Section through Fuel and Control Rod Channels

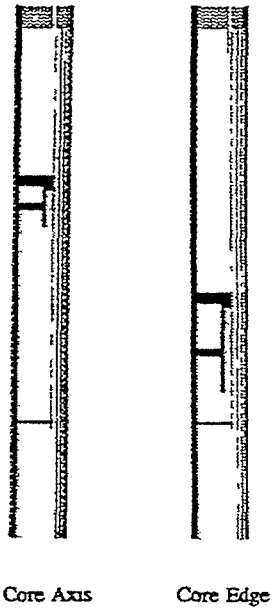
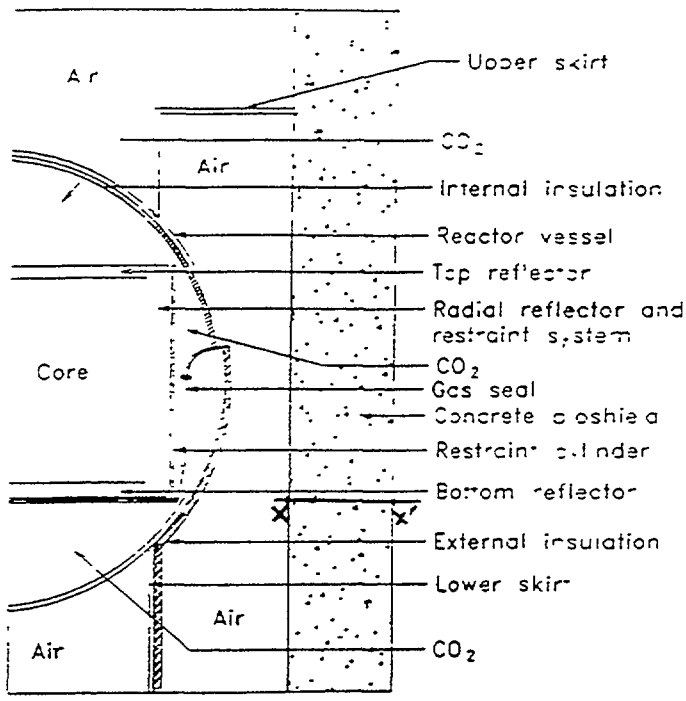


FIG. 11. Above Core Model-Showing Differences in RPV And Nozzle Heights

Fluxes and reaction rates were calculated in the channel pots, the boron shield, the pressure vessel, the pressure vessel nozzles, the concrete roof liner, the roof sleeve, the standpipe divided into several axial regions, the refuelling penetration muff shield and the concrete roof divided into several axial regions. In all, fluxes and reaction rates were calculated in 75 components.

3.3. Monte Carlo Calculations to the Side of the Core

The modelling to the side of the core consisted of the core itself, the reflector, restraint system, gas seal, pressure vessel, internal and external insulation and concrete bio-shield. Axially the model extends for the full height of the reactor vault but azimuthally credit is taken for symmetry and a sector of width 6.203 degrees with reflecting material was specified to achieve the required modelling. The core was represented as an equivalent cylinder of radius derived from the total area of all the fuel channels The outer edge of the reflector bricks was modelled to represent the reflector at its thinnest, since this will maximise the neutron fluxes in the locations of interest Figure 12 shows a section through the model and illustrates the components that have been represented. Figure 13 shows the computer representation of this



Vertical section along 90° azimuth

FIG. 12. Side Core Model - Section

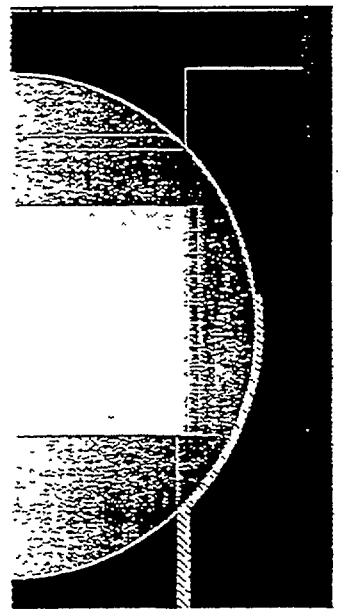


FIG. 13. Side Core Model - Computer Generated Section

The concrete in regions such as the bottom corner of the reactor vault will experience neutron flux from both the side core calculation and the sub core calculation. In order to be able to distinguish between these, it was important in the side core case not to 'double count' neutrons that will be taken account of in the sub core case. In the side core model therefore, the bottom core reflector region was designated as a black (perfect absorber) material. The purpose of this was to eliminate from the side core case, the flux scores due to neutrons which would have penetrated the bottom reflector. This penetration path was evaluated in the sub-core model and the fluxes in the bottom corner regions from the two cases can therefore be summed.

Results were obtained in the restraint cylinder, in the gas seal, pressure vessel and bio-shield concrete all of which were divided axially into 1m high regions. Additionally, for the concrete, a radial subdivision of scoring regions was specified over a 105 cm depth, these being divided into a region of 5 cm width, two of 10 cm, and four of 20 cm. Results were calculated in 164 separate regions.

3.4 Monte Carlo Calculations in the Region Below the Core

Simultaneous modelling of all the fuel channels in complete detail was not practical and computationally not feasible. The neutron leakage from each fuel channel will however be the same and therefore the overall calculation was achieved in two, linked stages:

(a) the angular distribution and spectrum of the neutron leakage from the bottom of the core was determined using a single channel model

(b) the single channel angular distribution and spectrum were used to define a disc source in a separate model which represented the sub-core region of the reactor.

3.4.1. The Sub-Core Single Channel Leakage Calculation

The angular distribution and spectrum of the neutron leakage from the bottom of the core was calculated using a MCBEND model of a single fuel channel. The MCBEND model represented a hexagonal core brick containing the Magnox fuel surrounded by a reflective material at the boundaries of the hexagon in order to simulate an infinite array of such channels. Axially, two fuel elements were modelled together with the dummy fuel in the bottom reflector brick. The core support plates were included as was the gag. Above the level of the bottom of the support plate the coolant gas was represented at the appropriate density, but below the support plate the void was represented as a vacuum so that there was only geometric attenuation from that point downwards. A hemispherical flux scoring region was included, divided into 5° polar bins and placed at a large distance so that the fuel channel was essentially equivalent to a point source. Neutron source term data derived as described in section 1 above, was used to specify the neutron production rate in the two fuel elements. The flux scores in the 5° bins gave the required leakage angular distribution which was fed into the sub-core model described in the next section.

The calculation described above and those in section 3.2.3 used a fission source represented axially by two fuel elements since it was considered that this would be adequate for the determination of the neutron leakage from the fuel channels. In order to assess the effects of representing two fuel elements rather than three, the single channel case described above was repeated with an extra fuel element making three fuel elements in all. In order to make the study computationally feasible within a realistic CPU time, the MCBEND option of a diffusion solution was adopted rather than the full Monte Carlo treatment which was used for the main calculations. The basis of the assessment was to compare the reaction rate scores for those reactions given in Table 2 in the region corresponding to the hole in the core support plate which is directly under a fuel channel. This location was chosen because any neutrons entering the sub-core region would have to pass through it preferentially rather than through the bulk of the core support plate and it would therefore be useful in assessing the fuel channel leakage from the two cases. It was observed that the reaction rates with three fuel elements were greater than that with two fuel elements by an average of 11% for the reactions in Table 1. It was concluded therefore that the basis of the calculations reported here were justified and that the results were fit for purpose.

3.4.2. The Sub-Core Region Reactor Model

The sub-core model represented the geometry of the components which are below the core support plate ie the core support members, the skirt, debris cone, pressure vessel, insulation and concrete. Figure 1 gives a section of the reactor showing the sub-core region. The graphite stack is supported by 24 support members as shown in Figure 14. The dimensions of one of these is shown in Figure 15 and its computer representation in Figure 16. In plan, symmetry was used to maximise the efficiency of the Monte Carlo calculations. A 30 degree sector with reflecting boundaries as shown in Figure 14 was used to reproduce the whole of the sub-core region.

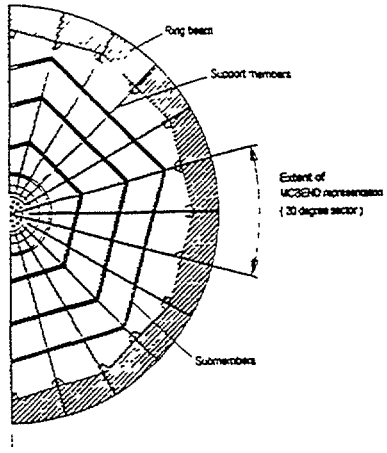


FIG. 14. Sub-Core Model - Plan of Supporting Members and Ring Beam

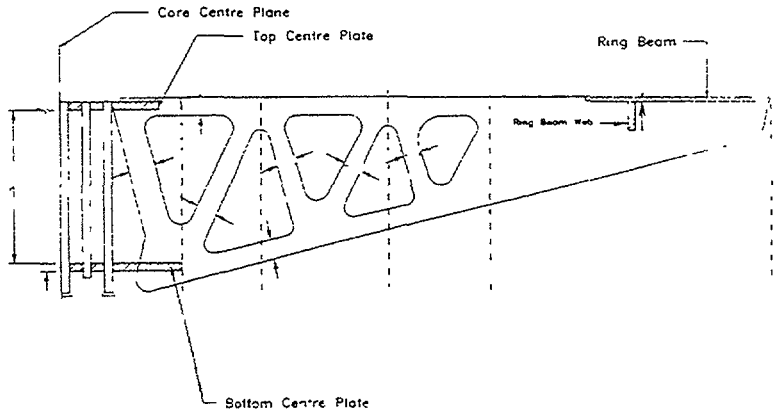


FIG. 15. Sub-Core Model - Dimensions of A Core Support Member

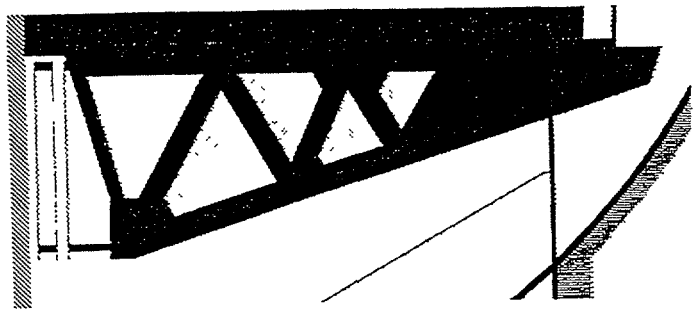


FIG. 16. Sub-Core Model - Computer Representation of A Core Support Member

The results from the single channel model described in section 3.4.1 were used to define the angular distribution and spectrum of a lamina disk source placed at the bottom of the core support plate. The radius of this disk source was based on the equivalent area of all the fuel channels and the source strength was normalised to give the correct neutron leakage from the whole core.

The sub-core flux and reaction rate results were determined for the support members, the debris cone, the pressure vessel divided into several polar regions, the support skirt and the vault floor concrete divided into several radial and axial subdivisions. Results were also produced for the bio-shield concrete below the core support plate level in the bottom corner of the reactor vault in the same regions as those defined for the side-core model described in section 3.3. Results were calculated in 95 separate regions in the sub-core model.

3.5. Monte Carlo Calculations In the Gas Duct Penetrations of the Bio-Shield

This section describes the calculational methods and modelling assumptions used for the neutron transport calculations which established the neutron flux and reaction rate distributions along the boiler duct penetrations through the bio-shield.

In order to optimise the extent of the MCBEND model of the boiler duct penetrations, preliminary neutron streaming and scattering estimates of the flux variation were made for each of the hot and cold ducts using the code MULTISORD [3]. This calculates the streaming of radiation within a rectangular duct system by a kernel approximation to the albedo transport equation. Part of the scope of the work was to identify the location along each duct where the neutron flux had reduced to a level of 10^3 n/cm^2/s. At this level of flux, the structural activation of the materials is regarded as negligible and it was important that what is essentially non-active concrete could be identified since this has an impact on decommissioning costs.

3.5.1. MULTISORD Calculations Used for Scoping Purposes

The modelling assumptions for this calculation are outlined below. They are based on the assumption that detailed modelling was not really necessary since the purpose was to roughly establish the likely extent that would be required for the more rigorous Monte Carlo calculations to be performed later. In general both the hot and cold ducts are made of mild steel of internal diameter 1.820 m. The thickness of the cold ducts is 16 mm for straight sections and 23 mm for bends, and that for the hot ducts is 22 mm (ducts 3 and 4) and 27 mm (ducts 1 and 2) for straight sections and 32 mm for bends. The value of 22 mm is assumed for all straight sections of the hot duct. The dimensions of the hot and cold ducts are shown in Figures 17 and 18.

MULTISORD represents ducts of a rectangular cross-section and so, for the purposes of the scoping calculation, the hot and cold ducts were represented as having square equivalent cross-sectional area i.e. 1.613 m^2. The MULTISORD mouth currents are based on the neutron fluxes calculated in the sections above at the positions of the pressure vessel where the hot and cold duct penetrations' nozzles are situated.

The neutron fluxes entering the mouth of the cold duct were taken to be the sum of those calculated by MCBEND from the side and sub-core calculations described above at the cold gas ducts entrance location in the pressure vessel.

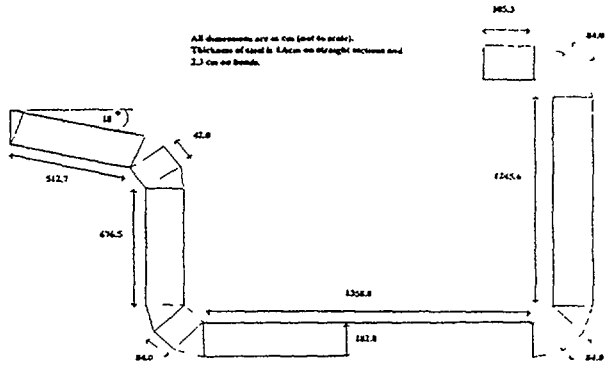


FIG. 17. Dimensions of the Cold Gas Duct

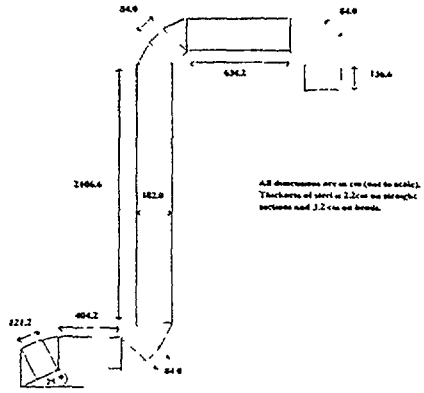


FIG. 18. Dimensions of the Hot Gas Duct

Since the above-core calculations represented an infinite array of fuel channels, they did not include those parts of the pressure vessel in the vicinity of the hot gas duct penetration and therefore no account was taken in the above-core calculation of the neutrons which scatter around the top corner of the graphite stack and enter the hot gas duct penetration area. The neutron fluxes in the vicinity of the hot duct entrances were therefore taken to be equal to the fluxes calculated for the side shield model with a contribution added to account for the top-core leakage component. This component was scaled from the sub-core results and is considered to probably result in pessimistically high flux values for the hot gas duct penetration since the neutron flux levels below the core are generally higher than those above.

The MULTISORD code requires mouth currents rather than fluxes to be specified and the values also needed to be transposed from the 21 energy groups of Table 1 into the standard 16 group energy used by the code. The mouth currents were assumed to be equal to half of the corresponding fluxes. Input currents from the sides of the ducts were not included in the MULTISORD models since it was possible to demonstrate for both cold and hot ducts, that the mouth values dominate.

The neutron differential (energy and angular dependent) scattering albedos for both the hot and cold ducts were taken to be the values for an effectively semi-infinitely thick steel surface which had previously been calculated for other work using the neutron diffusion code REDIFFUSION [4] In practice the cold and hot ducts are steel cylinders of thickness varying between 16 mm and 32 mm within concrete walled rooms. A comparison was made with albedos derived for 6 mm of steel backed by concrete. The appropriate values for the cold and hot ductwork surfaces would lie between the two but the semi-infinite values were, in the main, higher by around 8% and so these were used for the MULTISORD calculations, being bounding values which would give conservatively high flux levels along the ducts.

The results of the MULTISORD calculations were that for the cold duct the 10^3 n/cm^2/s flux threshold was reached approximately half way along the third straight section situated at the lowest level of the building, and for the hot duct it was reached at the start of the third straight section situated at the top of the building under the shield cooling air stack. This was taken into account in the following section which describes the Monte Carlo calculations.

3.5.2. MCBEND Calculations for Neutron Flux and Reaction Rate Predictions

The MCBEND geometrical model represents a quadrant of the reactor containing a coolant loop external to the steel reactor pressure vessel. Adjacent ducts are accounted for by the use of reflecting boundaries. Figure 19 shows the computer representation of the MCBEND model

where the extent of the gas ducts has been determined from the results of the MULTISORD calculations described above plus some additional lengths to accommodate any potential differences in the MULTISORD and MCBEND flux predictions.

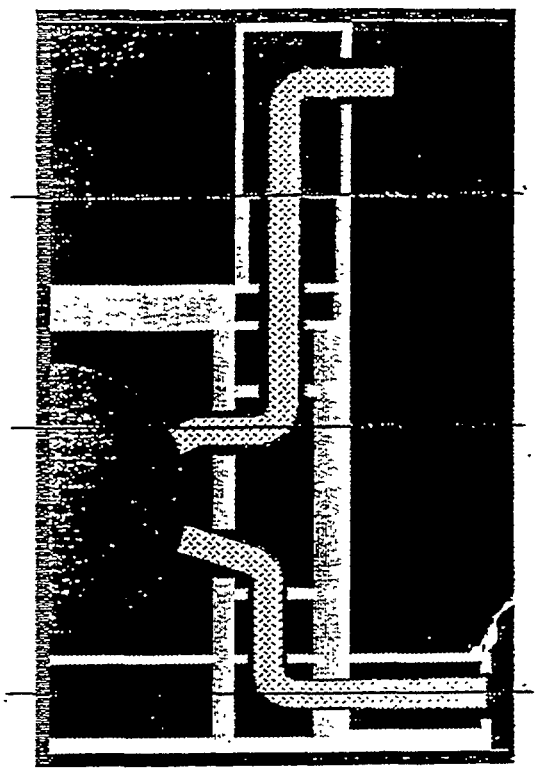


FIG. 19. Section of MCBEND Computer model used to predict the cold and hot gas duct Neutron flux levels

The following civil features were included in the modelling:

(a) the reactor steel pressure vessel with the inlet and outlet gas duct penetrations for one coolant loop

(b) a cold inlet steel gas duct

(c) a hot outlet steel gas duct for the same coolant loop

(e) the concrete primary and secondary shields and gas duct penetrations in the vicinity of the coolant loop

(f) intermediate concrete floors and gas duct penetrations between the concrete primary and secondary shields in the vicinity of the coolant loop.

Other duct support structures and bellows expansion units in the steel ductwork have not been included in the geometric model. It was considered that these omissions would result in conservative predictions of the neutron fluxes and reactions rates but the degree of conservation will not be overly high.

The neutron source terms for the MCBEND calculation assumed the same input fluxes at the mouths of the cold and hot ducts as for the MULTISORD calculation described above, but for MCBEND, the 21 energy group scheme given in Table 1 was used. Since the MCBEND geometrical model was large and in order to obtain good results of low statistical uncertainty, two separate calculations were performed. The first calculation incorporated a neutron splitting map with importances that favoured the flow of neutrons upwards and along the path of the hot duct. Similarly the second calculation was performed with neutron importances that favoured the flow of neutrons downwards, along the path of the cold duct. The MCBEND model was divided into scoring regions to allow neutron fluxes and the reaction rates specified in Table 2 to be calculated at many positions in the concrete walls. These scoring regions are shown schematically in Figure 20.

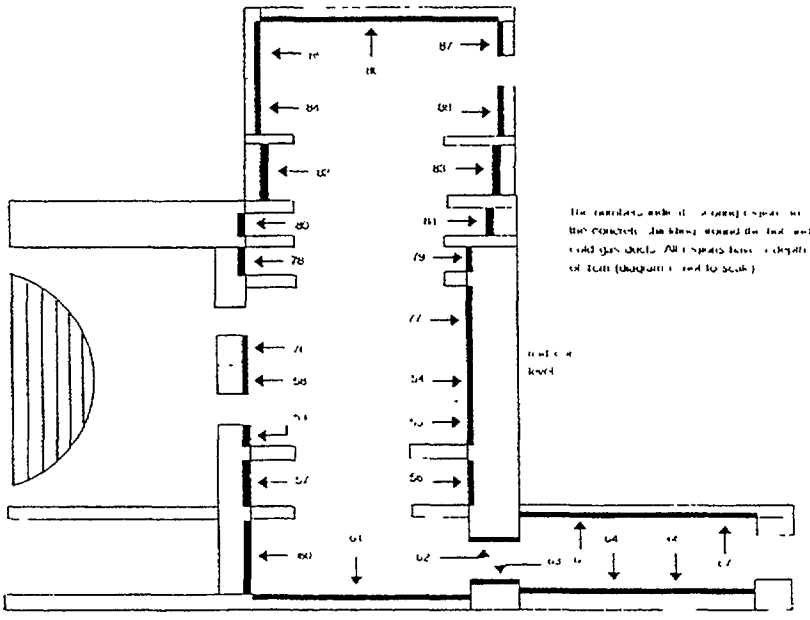


FIG. 20. Gas Duct MCBEND Calculation - Concrete Wall Scoring Regions

The neutron fluxes and reaction rates at scoring locations above the mid-core plane were taken from the first MCBEND calculation and those at locations below the mid-core plane taken from the second. In addition to the wall regions shown in Figure 20, additional scoring in the void regions between the walls was specified. In all, results were obtained in 39 regions. The positions of the 10^3 n/cm^2/s threshold predicted by MCBEND were in good agreement with those predicted by MULTISORD.

4. Comparison of Monte Carlo and DOT Calculations

The Monte-Carlo calculations have only recently been completed and there has been a limited amount of time for collaboration between Fuji and NNC to compare results. Consequently at this point in time the results available for publication are limited to a comparison of the thermal neutron fluxes extending radially through the bulk concrete bioshield at the level of the bottom of the core (along the line x - x in Figure 12).

The sum of the NNC MCBEND contributions from the side shield and bottom shield calculations (the contribution from the top shield calculations being judged to be relatively insignificant) are compared with the Fuji DOT 3.5 predictions in Figure 21.

It can be seen that there is good agreement between the MCBEND and DOT predictions with the MCBEND values being very close to the DOT values at the inner surface and rising to around a factor of two higher at depths of up to 40 cms into the concrete and above.

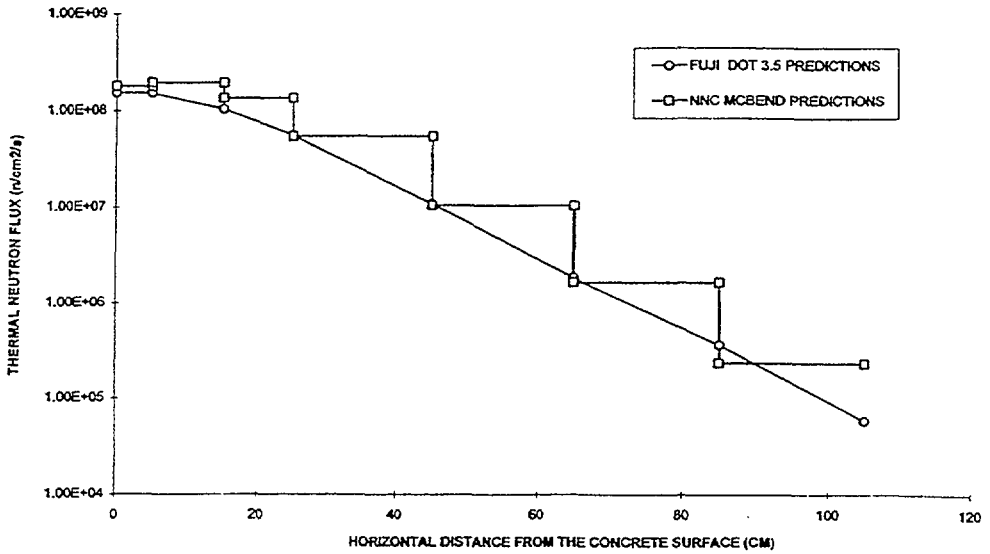


Fig. 21. Comparison of Fuji (DOT) and NNC (MCBEND) Predictions of the Thermal Neutron Fluz in the Bottom Corner of the Bulk Concrete Side Shield

The results also indicate that the MCBEND neutron spectrum at the concrete inner surface is harder than the corresponding DOT spectrum. In the MCBEND case the thermal flux rises with bulk penetration of the concrete indicating the thermalisation of higher energy neutrons whereas in the DOT case it is level for up to 15 cms penetration before falling steadily. A possible reason for this difference may be that the bottom shield MCBEND model allows for the streaming of neutrons along the void between the members of the support structure whereas in the DOT model the support structure is homogenised. The MCBEND results indicate that at the flux measurement position the contribution to the thermal neutron flux from the bottom shield model is around 90% of the total.

5. Conclusions

Comparison between MCBEND and DOT has only been made at this stage for the thermal neutron flux extending radially through the bulk concrete shield at the level of the bottom of the core. The results of this comparison are as follows:

(a) good agreement to within a factor of 2 is achieved

(b) MCBEND 3-dimensional model predictions are generally higher than the DOT 3.5 2-dimensional model predictions

(c) MCBEND results predict that approximately 90% of the total thermal flux arises from the bottom shield contribution

(d) the limitations of the 2-dimensional DOT code to represent streaming paths in detail may underestimate thermal fluxes in shielding below core level due to homogenisation.

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DEVELOPMENT AND LICENSING OF A MELTING PLANT FOR CHERNOBYL SCRAP

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Abstract

One decade after the accident at unit 4 of the Chernobyl nuclear power station, a melting plant for radioactively contaminated metallic materials, the so-called SURF facility, is being planned and licensed for erection in the direct neighbourhood of the NPP area. Main goal is the recycling of the material, largely decontaminated by the melting process, by means of manufacturing of casks and containers for waste disposal and of shielding equipment.

The melting plant will be part of the Ukrainian waste handling centre (CPPRO). The technology is based on the long-term experience gained at Siempelkamp's CARLA plant in Krefeld.

Within 1995 and 1996 the licensing conditions were defined, the licensing documents prepared and the formal procedure initiated. The complex is scheduled to start operation in 2001, in case the necessary financing is allocated. To this end, the proposed site of the facility has undergone the state assessment. The technical documentation for construction is at the stage of development.

MOTIVATION AND DESIGN BASIS

Within the remediation and restoration programme for the Chernobyl exclusion area, the erection of a waste handling centre (CPPRO) in the close vicinity of the Chernobyl nuclear power plant is being realised step by step. A melting facility, the so-called SURF pant (Smelter for the Ukrainian Radwaste Facility) is defined as a core element for handling of radioactively contaminated metallic material, by means of which the waste volume shall be drastically reduced and the largely decontaminated materials could be recycled.

The Central Waste Processing Enterprise was established by the GOSKOMATOM directive No. 380 of November 16, 1995. The supplement to this directive (No. 28 of January 21, 1997) specifies the goals and functions of the Enterprise as follows:

- Collection, reprocessing, interim storage and transport of radioactive wastes originated from the 1986 accident, plant operation as well as wastes arising from decommissioning operations at the CNPP units and the Ukrytie.

- The Central Enterprise has been called upon to implement the waste management provisions set forth in the Ukrainian Laws „Utilisation of Nuclear Power and Nuclear Safety", „Radioactive Waste Management", as well as the State Program for radioactive waste management.

On May 6, 1997, the President L. Kuchma signed the order No. 388/97 which transforms the Ukrainian State Committee on the Utilisation of Nuclear Power (GOSKOMATOM) into a State Department of Nuclear Power under the Ministry of Energy.

Therefore, the Ministry of Energy takes over the management of radioactive wastes and the respective facilities located in the Chernobyl Exclusion Zone.

This presidential order lifted the inter-institutional confusion caused by the „Ministry of Accidental Situations and Protection from the Consequences of the Chernobyl Accident" controlling the waste management aspects.

As part of the solution of the Chernobyl Zone problem, the Central Enterprise will include a complex for smelting of radioactive metal (the SURF Project). In the course of analysis of the required capacity and technologies, the plant offered by Siempelkamp has been selected. The smelting complex comprises scrap cutting and blasting shops, and a melting hall which contains induction and electric arc furnaces.

The melting technology is based on the separation effect of the radiologically dominant nuclides during the melting process, which leads to a transfer of these radionuclides from the basic material into the process waste such as slag and filter dust. Slag and filter dust representing only a few percent of the initial mass can be safely disposed off after additional volume reduction in a permanent final repository.

The technology is based on the long-term experience gained at Siempelkamp's CARLA plant in Krefeld. This plant has been licensed per § 3 of the German Radiation Protection Ordinance and used commercially since 1988. Up to now, more than 10,000 Mg of contaminated scrap from European nuclear installations have been melted and recycled in the nuclear field [1].

In order to provide adequate input data for the SURF plant design and the suitable equipment, an on-site evaluation was performed within the EU-sponsored TACIS programme [2]. It showed that radioactively contaminated metallic material is being stored openly at 48 locations within the 30 km exclusion zone. The main part with approx. 60,000 Mg is located in the immediate vicinity of the Chernobyl plant consisting of spare parts such as pipes, vessels and electronic equipment. In total, an overall mass of metal scrap of minimum 100,000 Mg was estimated, not including the unregistered emergency dumps and the metallic structural materials expected to arise from plant decommissioning.

The on-site measurements showed furthermore the dominance of Cs 137 and Sr 90 leading to a maximum specific β-/γ-activity of approx. 400 Bq/g as well as the very low portion of α-activity.

THE PLANT DESIGN

According to these results a plant throughput of approximately 10,000 Mg/a was defined with the option to enhance the capacity by modular units.

Main data and the general arrangement of the overall plant are presented in Fig. 1, the process flow with the main procedural steps in Fig. 2. The main elements of the SURF plant can be roughly described as follows:

- The melting shop:
In order to ensure a high degree of availability and with regard to the expected broad range of incoming material and their dimensions, two different furnace types have been planned for, i. e. an induction and an electric arc furnace. Crucible induction furnaces which are operated at medium frequency are the ideal melting aggregate for bulky parts composed of various materials. The advantage of the discontinuously charged electric arc furnace lies with its specific larger crucible diameter which permits charging with larger pieces of scrap. For redundancy reasons, both furnaces are housed in separate low-pressure enclosures each with their individual filter technology. They are controlled from a joint control station (Fig. 3).

- The blasting shop:
All incoming materials ought to be blasted prior to further treatment, especially in order to cover highly contaminated parts. Blasting takes place in a tight enclosure where a negative pressure is produced by strong suction, so that no dust particles can escape to the environment. The blasting material envisaged is medium grain-sized steel shot which has good stripping qualities.

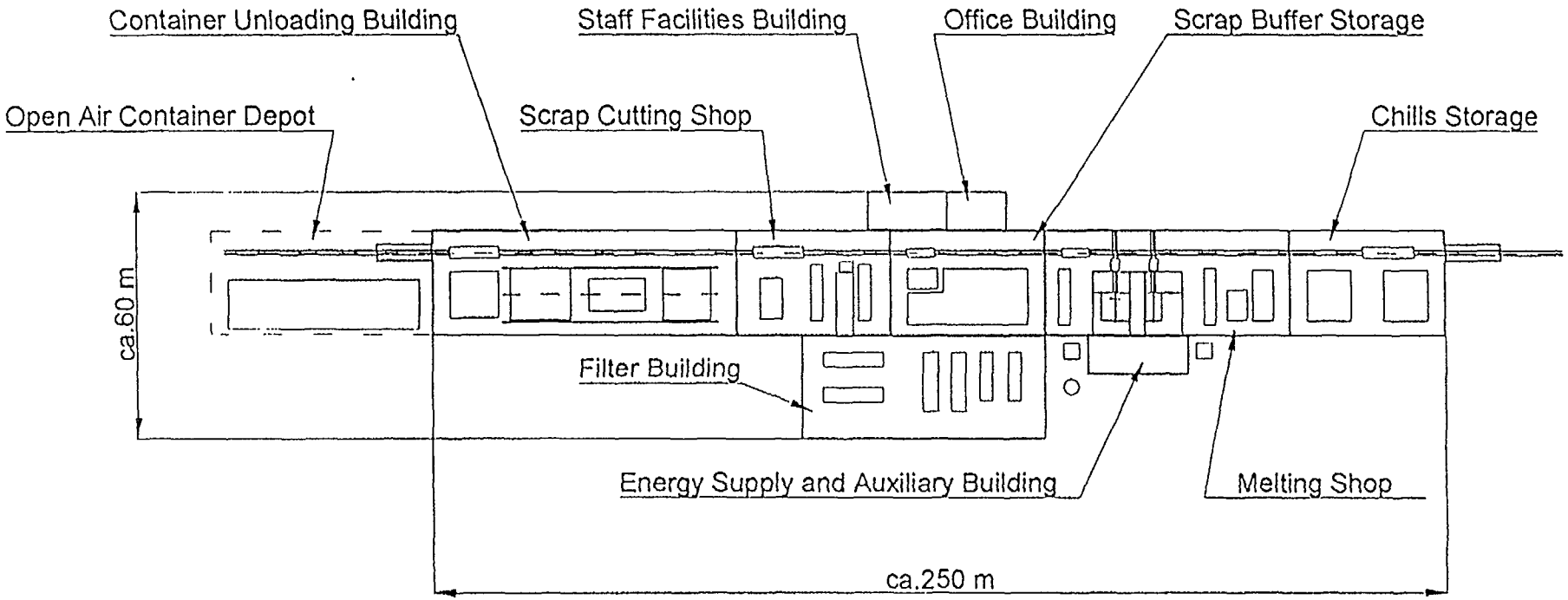
- The granulator:
As an alternative utilization concept for the cast iron, granulating is implemented. The main process of the granulating technique is given by the injection of the melt via a special nozzle into a water bath.

- Main auxiliary systems:
The air is filtered by 6 simultaneously operating filter systems with an overall discharge rate of 390,000 m^3/h. The filter systems for the blasting shop, the cutting area and the furnace casings consist of a high efficiency cyclone, a bag filter and a HEPA-filter and meet nuclear standards.

Besides torch-cutting devices two saws are foreseen for cutting thick-walled parts such as shafts. The main equipment is a heavy-duty hydraulic scrap shear with a cutting force of approx. 600 Mp.

The transportation system consists of a single track which covers the whole length of the plant. Flat battery-operated trolleys run on this track.

- Radiation protection concept:
The melting plant has been planned under consideration of the ALARA principle. This means that all radiation exposure is kept as low as reasonably achievable considering economic and social factors.



melting aggregates:	1 induction furnace
	1 electric arc furnace
melting capacity:	10.000 t/a
filter capacity:	390.000 m^3/h
power requirement:	20 MW

FIG.1. SURF - melting plant Chernobyl. Main data and general arrangement.

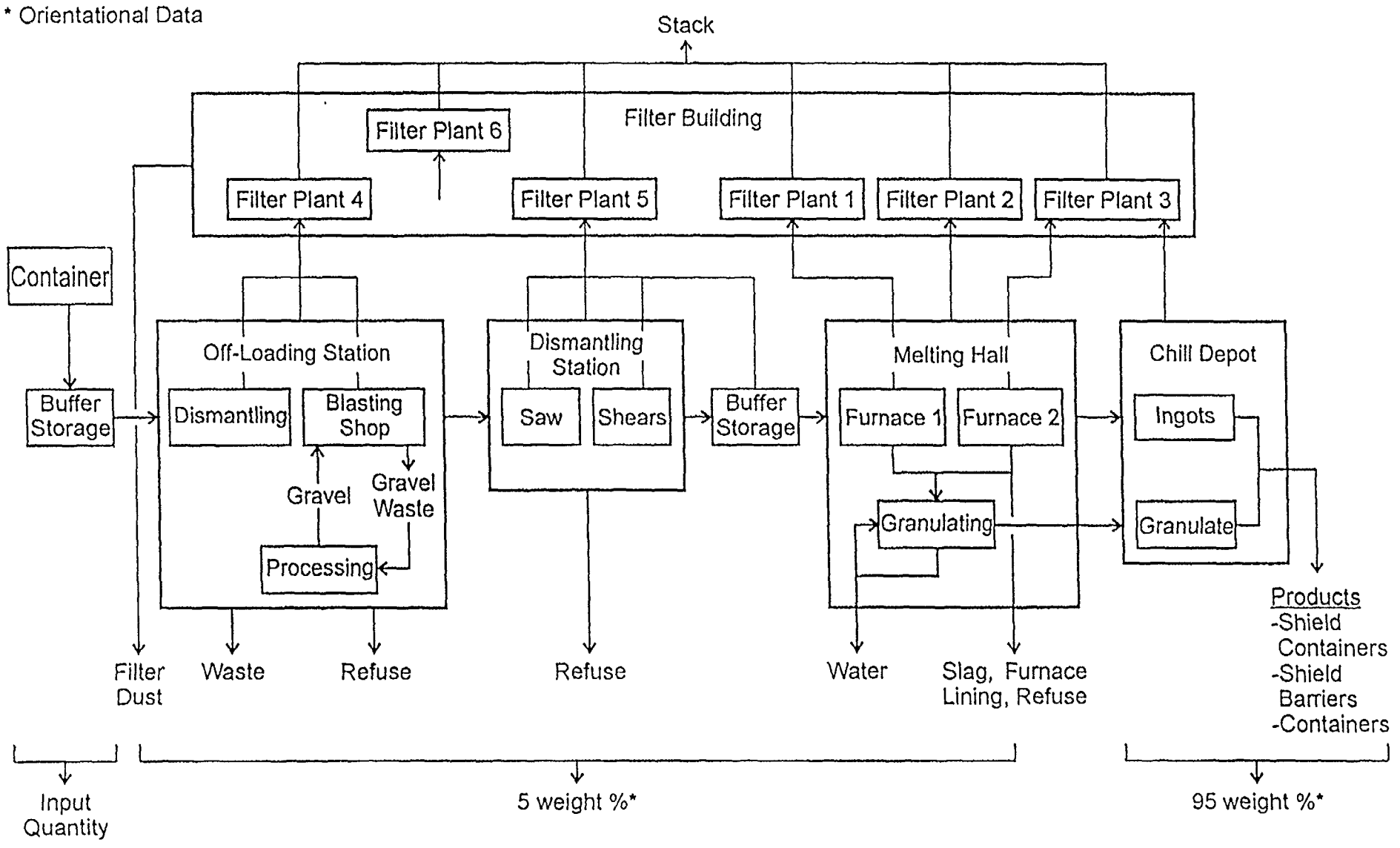
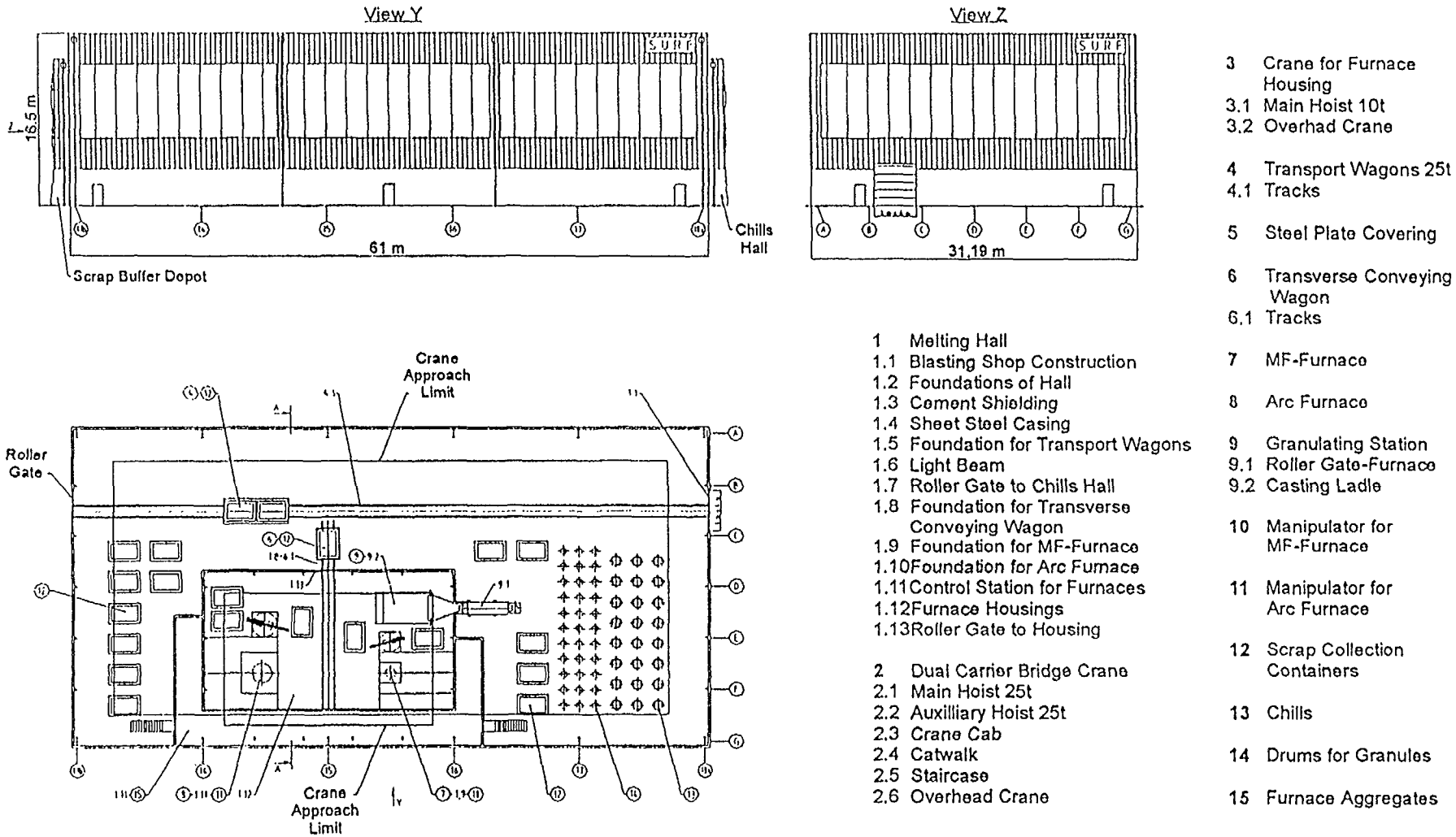


FIG. 2. SURF - melting plant Chernobyl. Flow diagram of material stream.



1 Melting Hall
1.1 Blasting Shop Construction
1.2 Foundations of Hall
1.3 Cement Shielding
1.4 Sheet Steel Casing
1.5 Foundation for Transport Wagons
1.6 Light Beam
1.7 Roller Gate to Chills Hall
1.8 Foundation for Transverse Conveying Wagon
1.9 Foundation for MF-Furnace
1.10 Foundation for Arc Furnace
1.11 Control Station for Furnaces
1.12 Furnace Housings
1.13 Roller Gate to Housing

2 Dual Carrier Bridge Crane
2.1 Main Hoist 25t
2.2 Auxilliary Hoist 25t
2.3 Crane Cab
2.4 Catwalk
2.5 Staircase
2.6 Overhead Crane

3 Crane for Furnace Housing
3.1 Main Hoist 10t
3.2 Overhad Crane

4 Transport Wagons 25t
4.1 Tracks

5 Steel Plate Covering

6 Transverse Conveying Wagon
6.1 Tracks

7 MF-Furnace

8 Arc Furnace

9 Granulating Station
9.1 Roller Gate-Furnace
9.2 Casting Ladle

10 Manipulator for MF-Furnace

11 Manipulator for Arc Furnace

12 Scrap Collection Containers

13 Chills

14 Drums for Granules

15 Furnace Aggregates

FIG. 3. SURF - melting plant Chernobyl. Melting hall.

Possible danger to the personnel of the melting plant exists in form of external exposure from the contaminated scrap as well as the incorporation of radionuclides during handling. To minimize the exposure risk, special clothing and breathing masks (type P 3) are required for work areas with higher radiological impact. Model calculations have shown that even under conservative assumptions regarding α-activity the dose rate can be evaluated as very low in comparison with the annual dose limits of 50 mSv/a defined for the category A personnel working in the Chernobyl area.

THE RECYCLING CONCEPT

Due to the special situation in the Chernobyl area and the general requirements for the handling of nuclear waste within the 30 km-zone, the recycling concept for the melted material has been defined as follows:

- manufacturing of casks and containers for storage and final disposal of radioactive waste,

- production of shielding equipment such as plates, beams and cubes of different geometries and sizes to be used for the sarcophagus work at unit 4 and other shielding tasks within the 30 km-zone (e. g. storage facilities).

In both recycling regimes casting of ingots and granules as intermediate products can be realized for the following final product line. With view to the present conditions granulation seems to be the favourable procedure.

THE FIRST LICENSING STEPS

Licensing requirements
In continuation of the basic design work, in January 1995 the European Commission ordered a follow-up study on preparation of the licensing procedure for the SURF plant with the following main objectives [3]:

- Elaboration of the frame conditions for the licensing procedure
- Evaluation of the relevant Ukrainian regulations including present regulation drafts
- Adequate transfer of the German regulations with special emphasis on adaptation to specific local conditions in the Chernobyl area
- Compilation of the licensing file.

According to the previous conceptual work, and, under consideration of the legal requirements, the CARLA plant with its supplementary facilities at Siempelkamp's Krefeld location has been confirmed as a pilot technique. This procedure assured a direct know-how transfer and enabled the implementation of the German licensing know-how.

Consequently, the Ukrainian authorities approached the German local licensing authority, the Bezirksregierung Düsseldorf, for assistance and consultant tasks, especially to obtain a formal statement on licensibility of the technical equipment of the SURF plant according to German regulations.

The adaptation to the specific Ukrainian conditions and the formal licensing procedure would then be covered by the Ukrainian authorities.

The joint adjustment of the main design criteria was successfully elaborated as can be shown with Fig. 4 giving the basic survey of design requirements for the SURF plant in comparison to the CARLA values.

Subject	Design values	
	CARLA	SURF
1.) Handling	200 Bq/g α , β , γ 100 Bq/g U 233, 235, Pu 239, Pu 241 (fission material) open and sealed 1000 Bq/g "naturally" contaminated materials	1000 Bq/g 1000 Bq/cm²
2.) Utilization after melting		
a) within the nuclear field	200 Bq/g α , β , γ 100 Bq/g fissable	1000 Bq/g 100 Bq/g fissable
b) released	old: 1 Bq/g α , β , γ new: $10^{-4} A_{Fr}$ Bq / g (Residue regulation)	α : 7,4 Bq/g β , γ : 74 Bq/g
3.) Waste (slag, dust, furnace lining)	old: 5 Bq/g released new: $10^{-4} A_{Fr}$ Bq / g i.e. Co 60: 5 Bq/g	7,4 Bq/g α 74 Bq/g β , γ
4.) Exhaust air	$10^{-6} A_{Fr}$ (Bq / m³) i.e. Cs 137: 0,5 Bq / m³ Co 60, Sr 90: 0,05 Bq / m³	0,15 Bq / m³ α , β , γ
5.) Liquid waste	not applicable	optional: Granule waste water into NPP liquid waste system

FIG. 4 a. SURF - melting plant Chernobyl. Design requirements.

Subject	Design values	
	CARLA	SURF
6) Dose rate	1,5 rem/a 15 mSv/a (2000 h/a)	5 rem/a 50 mSv/a (only Cat. A required) (1700 h/a)
7.) Areas		
Surveillance area	x	-
Control area	x No prohibited area *Full protection during open handling*	x No prohibited area *Full protection during open handling*
8.) Throughput	4000 t/a	20000 t/a
9.) BImSch		
Dust	$\leq 1\ mg\ /\ m^3$ (Law: $\leq 20\ mg\ /\ m^3$)	$20\ mg\ /\ m^3$
Noise	55 dB	80 dB
UVP	No	Yes
CO , NO_x , SO_2 negligible	x	x

*) A_{fr} = free limit for activity

FIG. 4 b. SURF - melting plant Chernobyl. Design requirements.

Site approval

According to the Ukrainian regulations the formal detailed plant licensing application has to be preceded by the licensing of the plant site on the basis of an assessment of three potential sites. Main criteria are safety aspects and infrastructural issues, but also technical-economic topics.

Fig. 5 shows the three assessed sites around the power plant area. As result of the detailed evaluation, option 3 has been selected, offering especially advantages for the erection of the overall CPPRO plant.

Formal licensing procedure

On the basis of all the boundary conditions described above a licensing file has been elaborated consisting of the technical plant concept and a detailed description of the safety characteristics with main emphasis on radiation protection activities. The German local authority attested the licensability according to German regulations. This led to the initiation of the formal licensing procedure by the official application placed on December 28, 1995.

Within 1996, detailed discussions with the Ukrainian authorities took place giving their positive expertise. This expertise was presented to the Ukrainian Board of Ministers, which gave a positive vote for this site. A decision of the site of the SURF-facilities is made by this act.

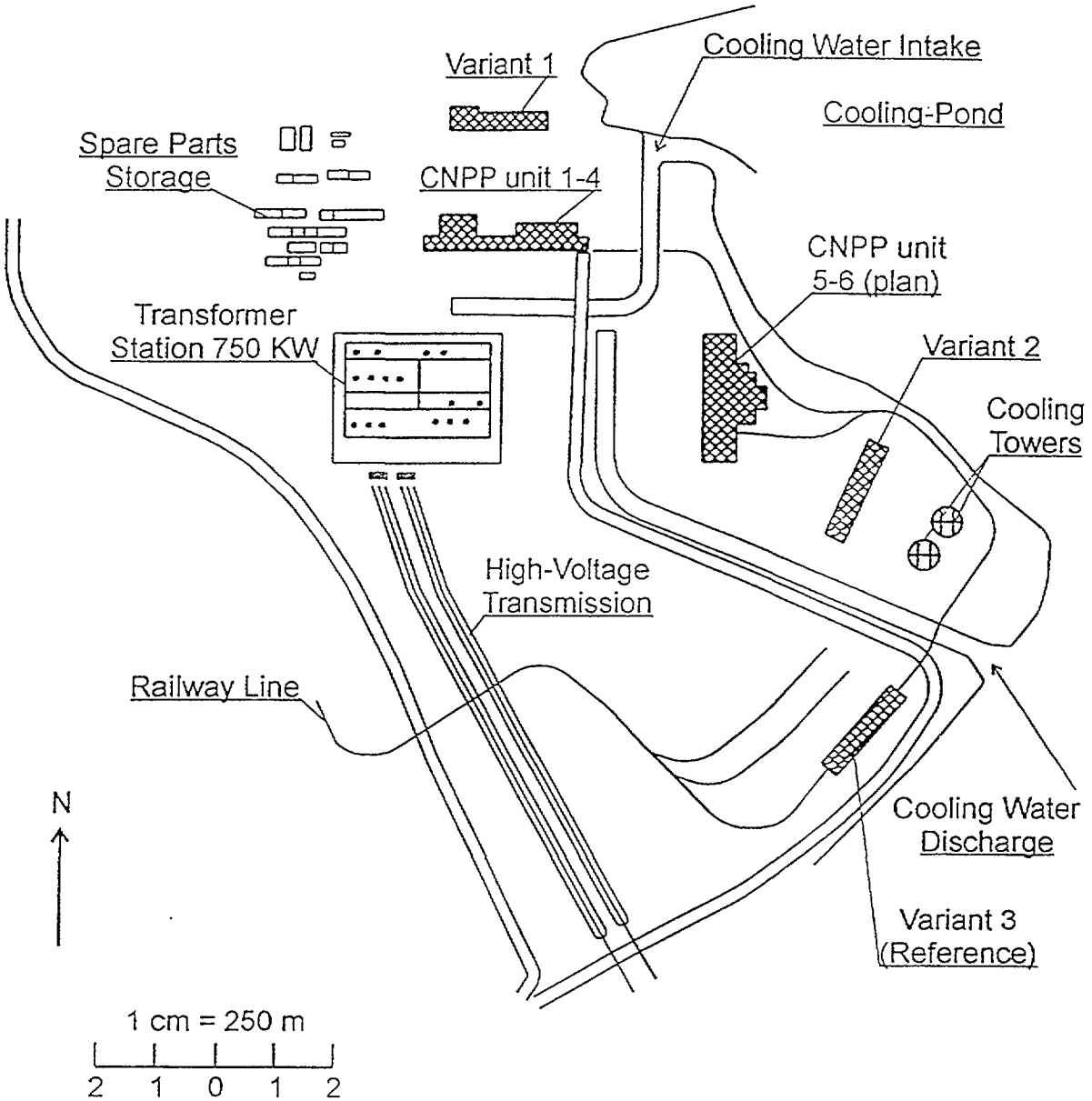


FIG. 5. SURF - melting plant Chernobyl. Site variants.

An other very difficult step is to come to a financial concept which guarantees the establishment of the SURF-facility at Chernobyl. One proposal is to obtain money from the G7-states where the money is distributed by the European Community, or to receive money directly from the European Community, using TACIS fundings.

More promising is the alternative of financing the plant by a credit being secured via HERMES. The Ukrainian government has already decided to ask for such a credit; today, the amount of 15 % of the total project volume as a first rate has to be paid.

Provided with proper funds, the engineering work is scheduled to be finished in 1998, the construction to start the same year. Commissioning is feasible in 2001.

ACKNOWLEDGEMENTS

The work reported on has been sponsored by the Commission of the European Communities within the TACIS and the NUCLEAR STUDIES programmes.

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RUSSIAN GRAPHITE NUCLEAR APPLICATION. HISTORY, STATE OF THE ART, FUTURE

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Abstract

Near 40 nuclear reactors based on graphite moderator were built in Russia during 50 years.

They are: RBMK, AMB, production reactors. Reactor graphites grades GR-220, GR-280, EGP-6, GR-94, GRP-2-125 were used for this reactors construction. Some grades of new graphites were developed on the Russian HTGR program such as GR-1 showing best radiation stability.

About 20 reactors were stopped due to exhaust of cladding resourses and/or metalwork structures and other reactor systems.

Production reactors (AI, AD, AV types) graphite claddings must to be dismounting but at the present time this work is in preliminary study stage. These reactors will not be demolished during 30 years.

The waste recovery program is developing now. There are two possible ways to reclaim graphite products:

- to born in special furnaces;
- to keep them in containers under special preserve compound.

At the same time very important to define the impurities in graphite claddings half-life in order to detrmine minimal period before its processing and conservation.

Near 40 uranium-fueled graphite moderated reactors of different types were built in Russia during 50 years. 20 reactors were shutdown.
As a rule, the graphite GR-220 and GR-280 are used as moderator in RBMK, AMB and production reactors.

The decommission problem of graphite reactors divides to two main ones: when reactor must be shutdown and how it will be decommissioned.
Our approaches to first problem are follow.

There are three criteria for evaluation the reliability of the channel reactor graphite stack:

- degradation of physical and mechanical properties of graphite as material;
- preservation of the graphite brick integrity;
- degradation of the graphite stack as a structure.

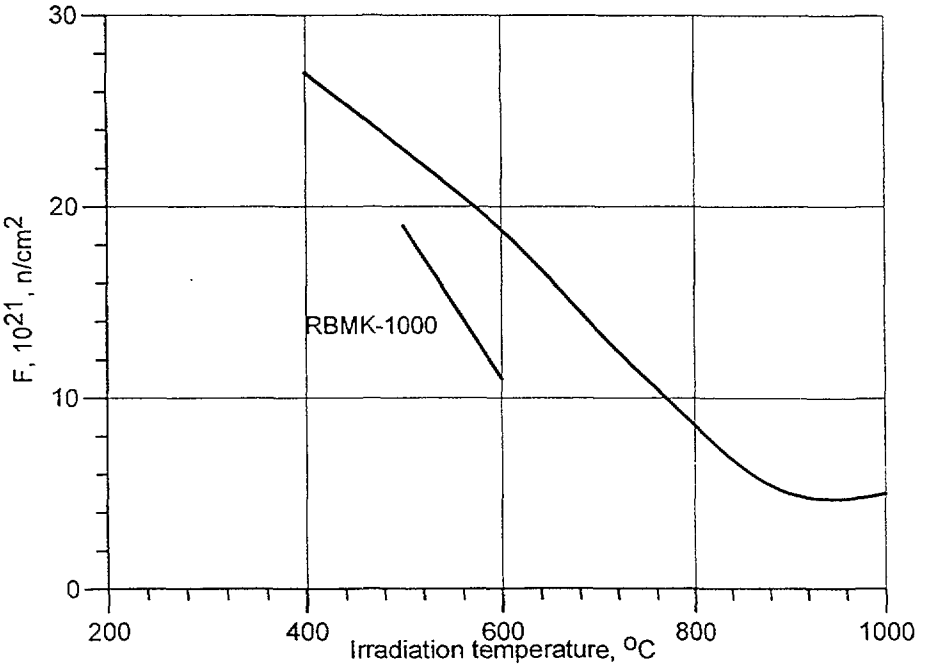

Fig.1. GR-220/GR-280 critical neutron fluence vs. irradiation temperature and operation condition of RBMK-1000.

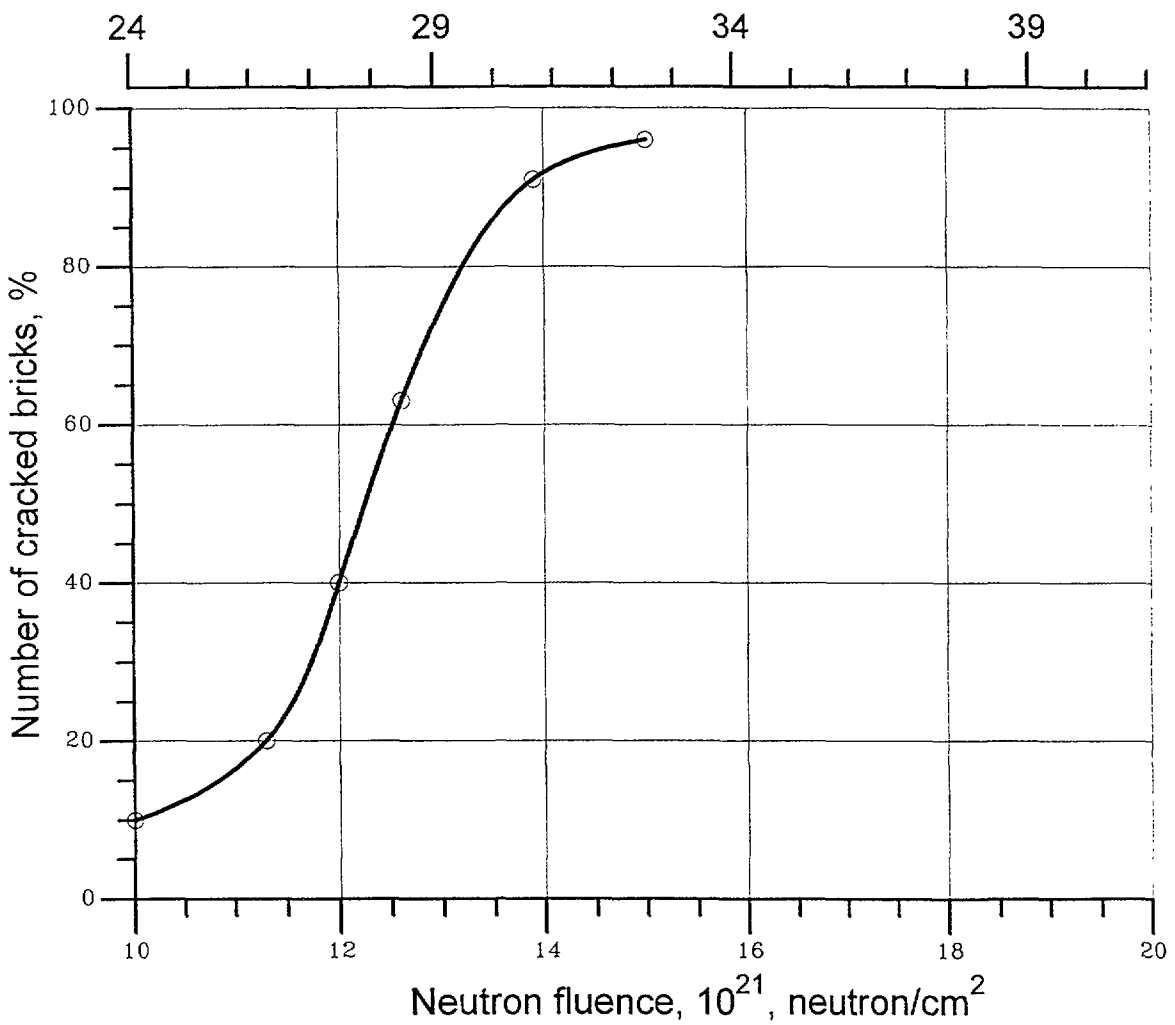

Fig.2. The cracking of graphite bricks during reactor AV-3 operation.

As a criterion of graphite degradation, the critical fluence has been taken ([1]). For the temperatures characterized for existing Russian graphite reactor the critical fluence is that fluence, when the shrinkage of material is changing to swelling. In the most part of operating reactors the critical fluence will be not reached even after prolongation of design lifetime, see Fig.1.

The reactor stack consists of free standing columns, consisting of the bricks of square section. The fracture of graphite bricks does not bring to threat to the operational reliability of the stack as a whole.

In the long time operated reactors the swelling of inner brick layers cause the tensile stresses on external brick surface and subsequent cracking during temporary shutdown (for preventive maintenance) when the cladding cooling, see Fig.2.

The fracture of even most part of bricks in such core structure does not cause a disruption in the graphite stack functions as a moderator but the stack temperature arising that cause the increasing of graphite swelling rate. 5-7 years pass from the beginning of cracking to the beginning of change in configuration of the whole stack. This changes of stack configuration up to critical value of graphite column binding passed approx. 8 years. So, 15 years passed from the beginning of cracking to the reactor shutdown in consequence of the stack degradation, see Fig.3.

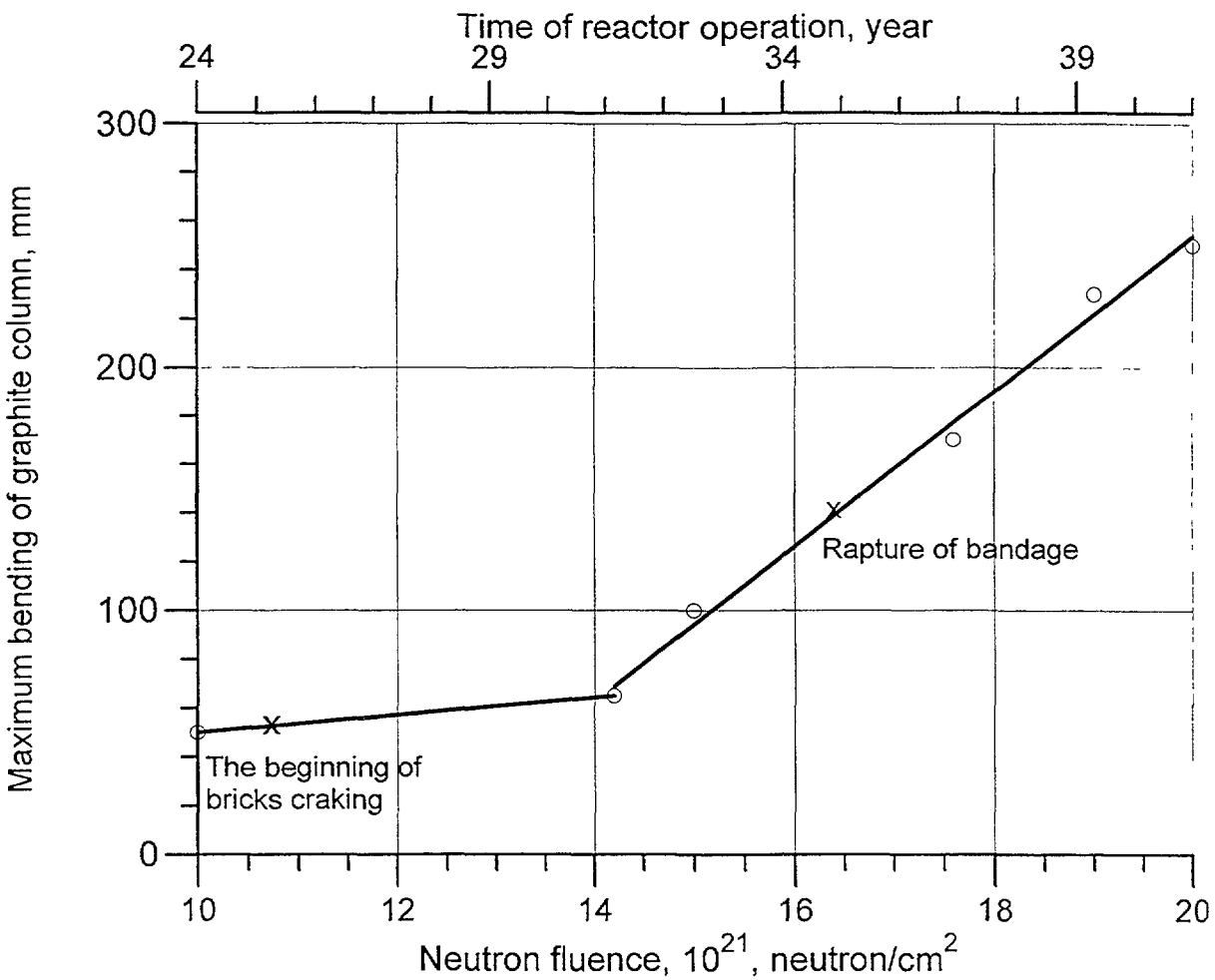

Fig.3. Maximum bending of graphite columns during AV-3 reactor operation.

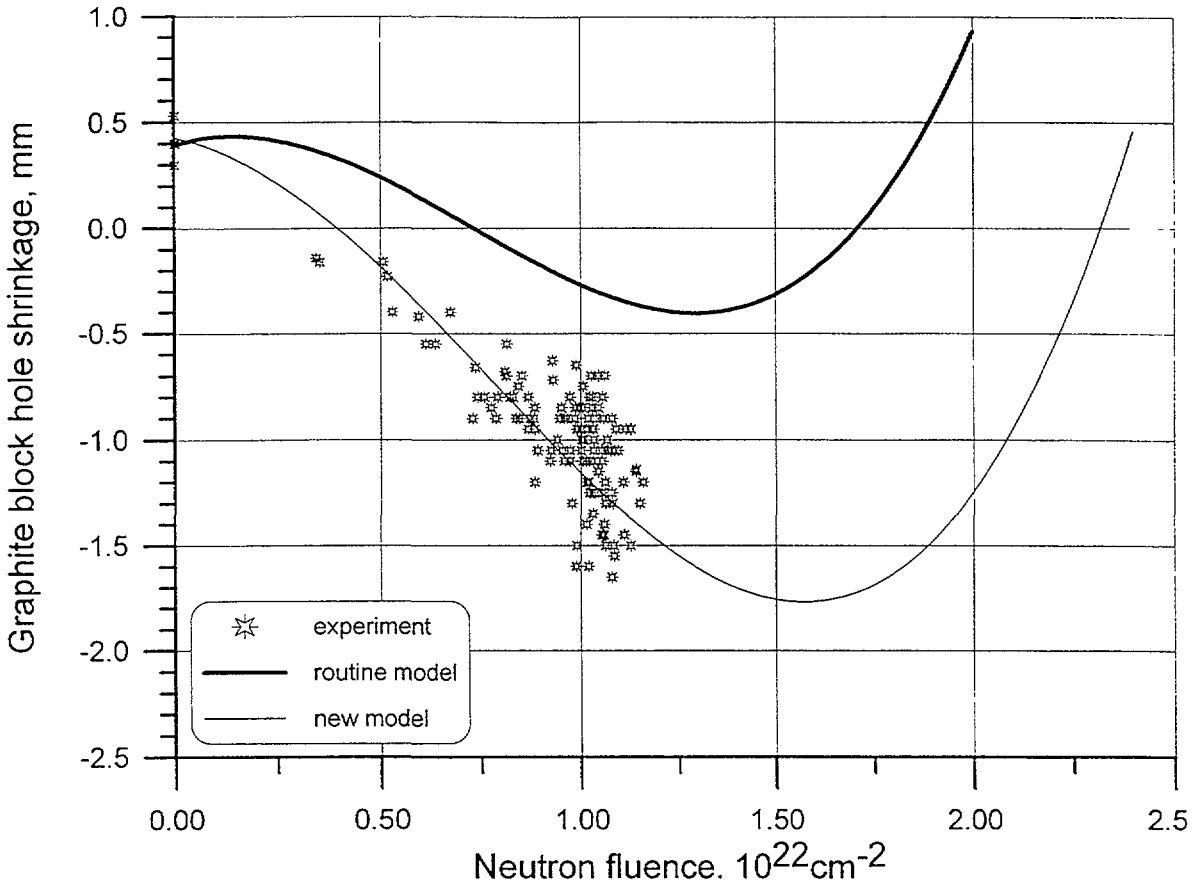

Fig.4. Changes of graphite block hole under irradiation. Leningrad NPP, Unit 2.

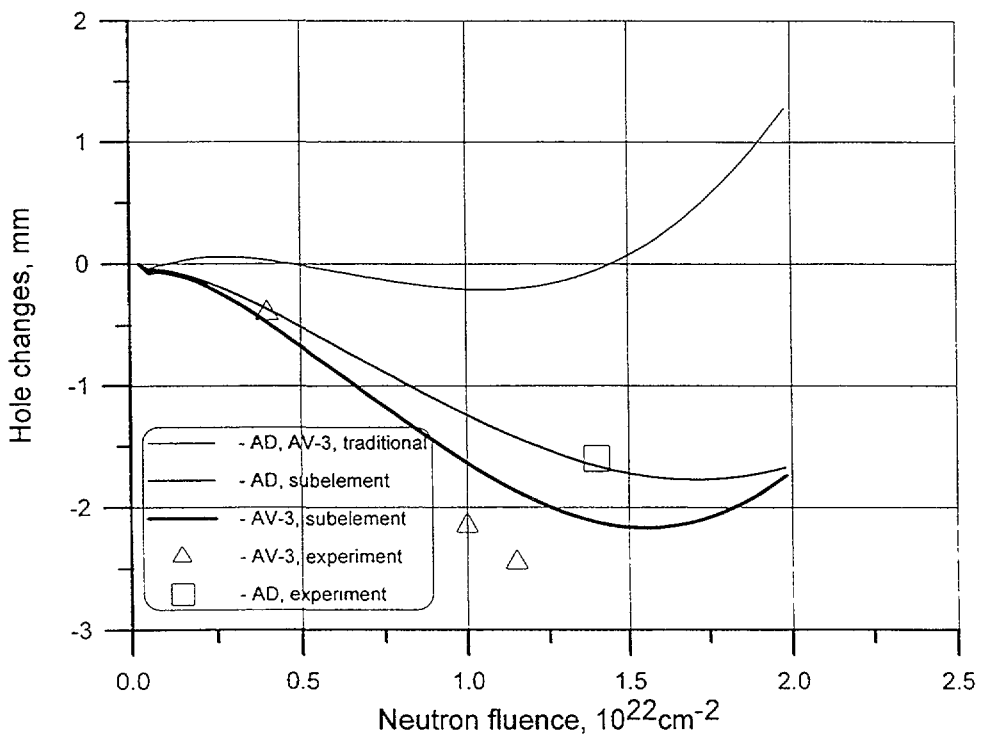

Fig.5. Changes of brick hole calculated on the base of traditional and subelement models and experimental data vs. neutron fluence.

What we will doing with long operating reactors graphite stack? It is necessary to define the condition of each graphite stacks for the operating its to the utmost. By the program we will take the probes (trepans) from the bricks, the whole graphite bricks from central part of core (RBMK), investigate its, including additional advanced irradiation in the test reactors, in order to determine the existing stack condition and its residual lifetime.

Other problem of this work is the calculation code revising. It is well-known that the traditional models and calculation codes cannot gives correct results of graphite bricks stress-strain conditions. That models are consider the graphite as solid continuos matter. And calculations gives the results differ from direct experiments.

On the base of electron microscopy investigations, making in our Institute from early 80-th, showing the mechanism of radiation damage of graphite [2], were made new calculation model for describing the existing processes in the structure of graphite under irradiation. The crystallites of graphite filler and binder, differed by sizes and radiation growth rates, are viewing as different subelements deformed and cracked by mutual interaction.

The verification of this model was made on the calculations of RBMKs and numerous production reactors stress-strain condition, displacement changes of brick holes and its rate. The results of RBMK-1000 graphite bricks shrinkage are presented on Fig.4.

In the Fig.5. the calculated brick hole changes made by traditional and subelement models and experimental data are shown.

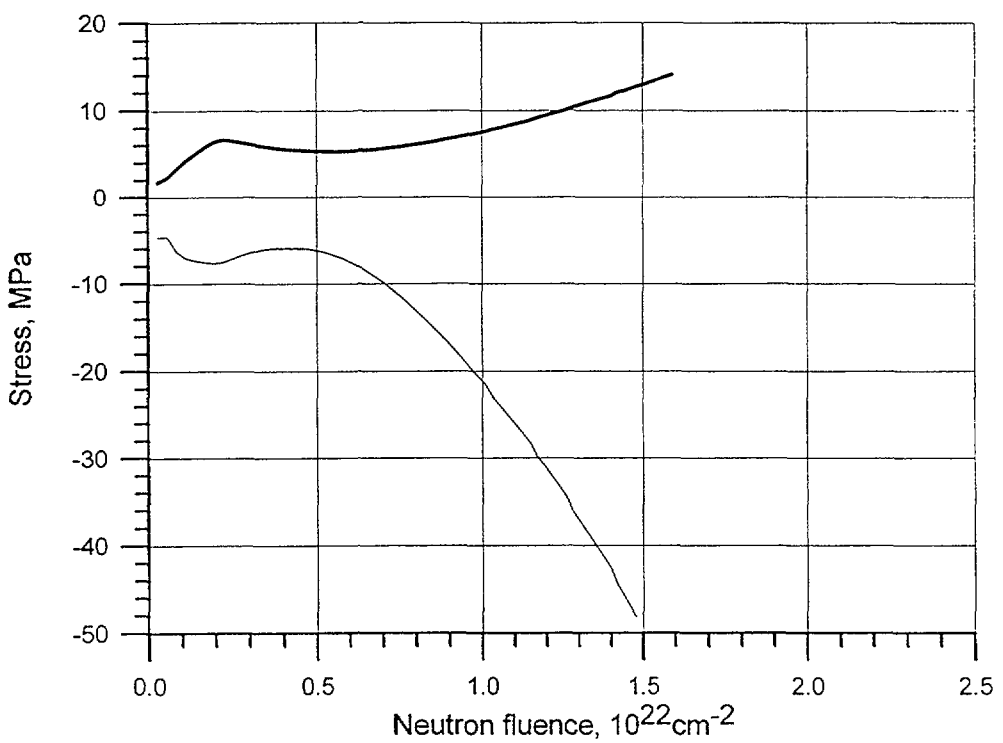

Fig.6. The summarized stress-strain condition of graphite brick calculated by F-model. Heavy line- external surface.

The results of stress-strain condition of AV-3 reactor graphite brick, shown in Fig.6 are successfully corresponding with experimental results of bricks cracking (see Fig.2.): after reaching neutron fluence of $1.2 \cdot 10^{22}$ cm^{-2} the stress reached the limit level (≈10 Mpa) and bricks begin cracking.

The problem of shutdown reactor decommissioning in the part of nuclear graphite is studied in package with all tasks on investigations of graphite as material, as stack and taking account with interactions it with fuel channels and helding bandages of cladding.
Generation of principles, criteria and technologies of RBMK graphite utilization are the main aims of such Russian program. Common quantity of graphite in builded RBMK is 30,000 tons from 50,000 tons of all Russian graphite reactors. By the program, in order to developing principles on safe handling with irradiated graphite will be working out the techniques of:

- definition of radioactivity, its distribution (C-14, fuel, fission fragments);
- investigation of graphite properties;
- technology of graphite bricks dismounting;
- technologies of chemical and physical influence on radioactive graphite (breaking, milling, cutting of layers with fuel, impregnation by conservants, burning of graphite etc.);
- technologies of storage.

Some of these technologies are developed.

The technologies of AMB-100 reactor will be developed on the first stage, then this technologies will be adapted to RBMK-1000.

The cladding of production reactors will not be dismounted during at least 30 years.

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PACKAGING REQUIREMENTS FOR GRAPHITE AND CARBON FROM THE DECOMMISSIONING OF THE AVR IN CONSIDERATION OF THE GERMAN FINAL DISPOSAL REGULATIONS

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Abstract

Green field decommissioning of HTGR results in a large amount of activated graphite and carbon installations which have to be treated for final disposal in a safe and economic manner. In case of AVR decommissioning, 67 Mg of reflector graphite and 158 Mg of carbon isolation are highly contaminated and activated. A study for waste treatment recommended to package this graphite and carbon into cast iron moulded containers qualified for the planned German final depository KONRAD. The reference concept results in 364 cubic containers using up a final waste disposal volume of about 1.700 m^3.

1. *Introduction*

After a successful operation for 21 years, the experimental HTGR AVR plant, having been developed in Germany, had been shut-down in 1988. The AVR operator applied for safestore decommissioning after shutdown. The way to safestore is splitted into two phases :

Phase 1 covers defuelling of about 100.000 spherical fuel elements and dismantling of the turbine hall and of other buildings outside the reactor. This phase 1 will be finished in the end of 1997.

Phase 2 includes dismantling inside the reactor building to yield a safe sealing of the remaining radioactive areas.

The safestore decommissioning as applied for and licensed is a kind of minimum cost programme. Having in mind that no burden should be left for future generations, a decision to apply for a reorganisation and extension of the safestore path has already been made [1].

Studies concerning green field decommissioning have been carried out in 1995. AVR commissioned Siempelkamp / WTI to develop a concept for treatment of all kinds of waste from decommissioning. About 60 % of the remaining activity after defuelling is concentrated in the graphite / carbon installations. These 225 Mg of solid waste have to be treated in a safe and economic way.

2. *Graphite and Carbon, a HTGR problem*

To meet the high safety features and high temperature application of HTGR, graphite has been chosen as basic material for fuel elements and core structures. The good behaviour of this material during long reactor operation time could be demonstrated. The video inspection of the AVR top reflector showed pictures like a fresh reflector. The carbon layers surrounding the graphite reflectors isolate the core region to protect the metallic structures from high temperatures, especially in case of loss of coolant accident. From the view of decommissioning HTGR, some disadvantages result using graphite / carbon. The typical low core power density leads to large core dimensions following a large waste amount of graphite and carbon structures. Main activation of carbon is yielded by the neutron capture reactions :

C 13 (n, γ) C 14
N 14 (n, p) C 14
Li 6 (n, α) H 3
Co 59 (n, γ) Co 60

More long-lived radionuclides have to be considered for final disposal of the waste. Another problem is the dust production in the HTGR-pebble bed core -type from the circulation of the spherical fuel elements. The expected dust mass in the AVR primary circuit is about 70 kg which is expected to be evenly distributed on the primary system surfaces. Reactor designers in the early '60ies did not properly take into account the aspects of decommissioning and waste disposal. Specifications of the impurities in graphite and carbon can reduce these problems.

3. *Waste balance from AVR deocmmissioning*

For the green field decommissioning of AVR, the solid waste is amounting to 33,314 Mg. Fig. 1 shows the solid primary waste paths. Most of it is rubbish from buildings of which a free release of 90 % is expected. Free release is also expected for 546 Mg of 1,696 Mg of metallic waste; 643 Mg fulfil the limits for melting at Siempelkamp CARLA melting shop and can be recycled to containers or components. A rest of high active metallic waste of 480 Mg has to be packaged for final disposal.
The graphite and carbon structures are shown in Fig. 2. 67 Mg of reflector graphite and 158 Mg of carbon isolation, in total 225 Mg of ceramic waste, has to be treated in a licensable and economical way.

4. *Activity distribution*

After defuelling the reactor core and finish the safestore decommission, the remaining activity is mainly concentrated inside the reactor vessel. The activity inventory in graphite and carbon results from the neutron capture reaction on carbon itself and on impurities like cobalt, iron, lithium, etc.

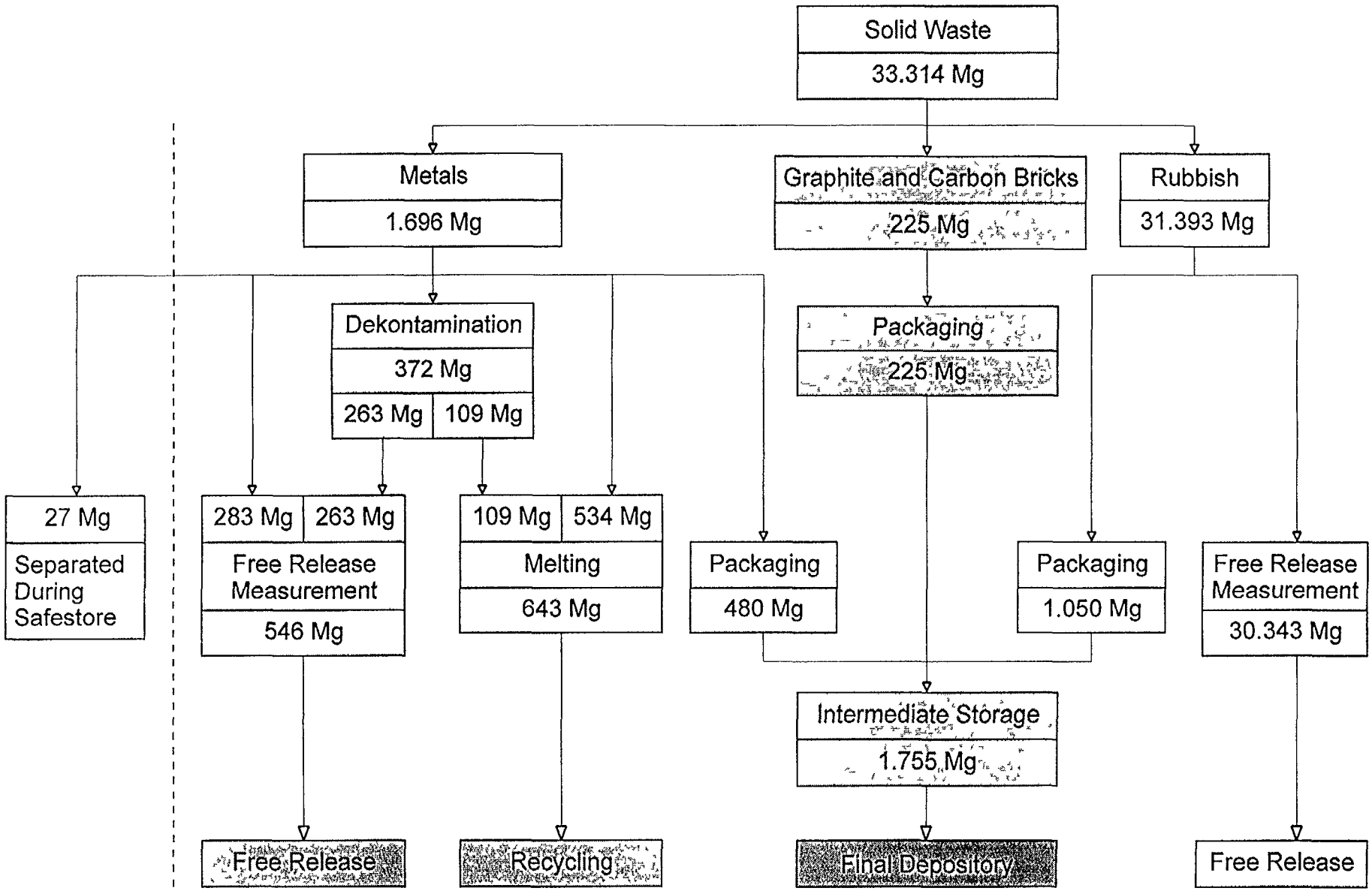


Fig. 1: Waste Balance AVR-Decommissioning (Primary Waste)

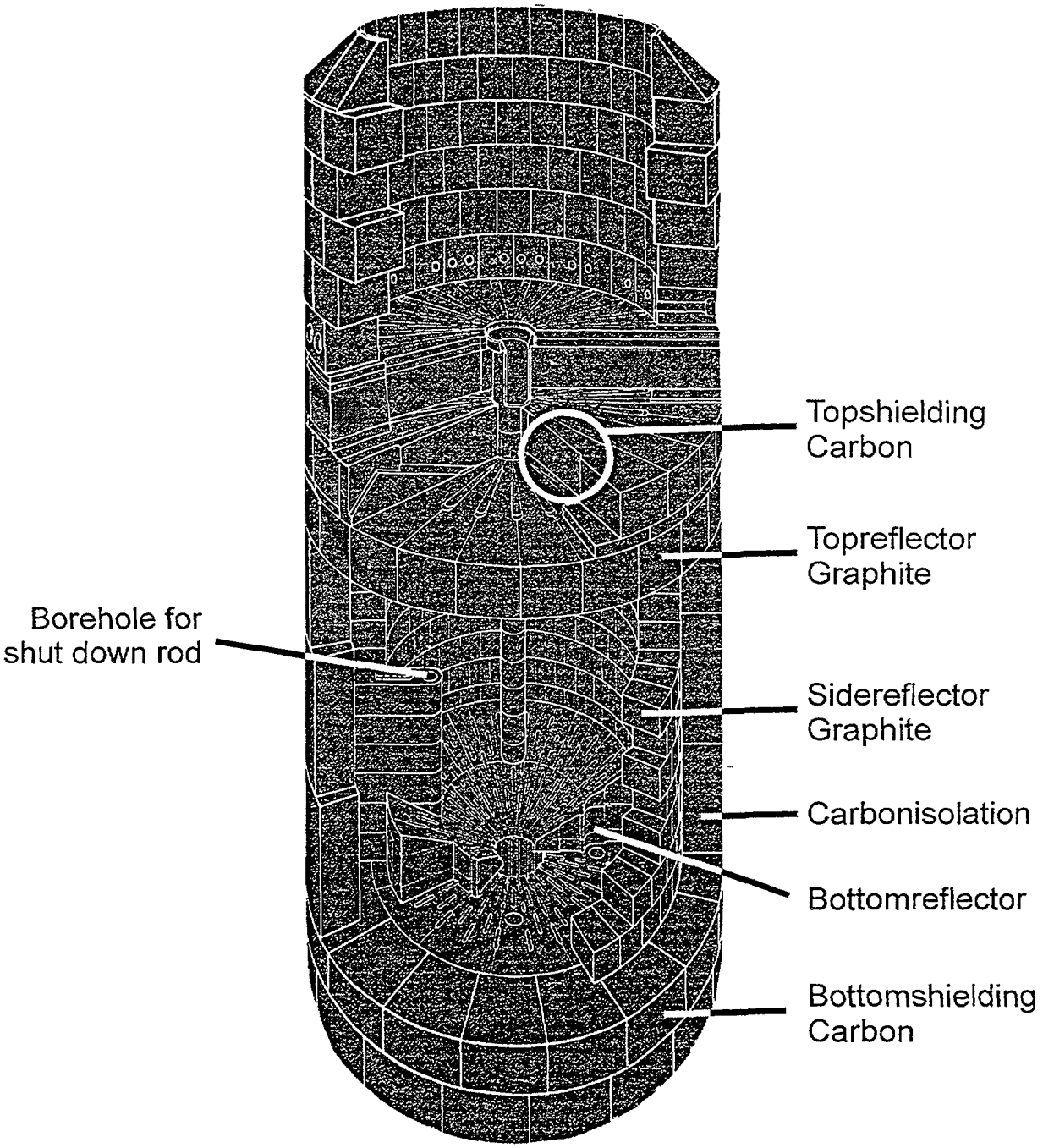


Fig. 2: AVR Graphite- and Carbon Installations

Production and decay routes for the activation have been examined by ORIGEN II calculations. A calculation of two-dimensional r, z - activity distribution inside the reflector and carbon layers has been performed for normalized impurities of 1 ppm each element [2].

The main impurities have been assumed as follows :

	impurity [ppm]		
	cobalt	**nitrogen**	**lithium**
graphite	0,1	100 [1)]	15
carbon	40	100 [1)]	1

[1)] surface contamination

A calculated total activity as of 31.12.1996 amounts to :

	mass	**activity [Bq]**		
	[Mg]	**C 14**	**H 3**	**Co 60**
graphite	67	8.6×10^{12}	1.0×10^{15}	5.2×10^{12}
carbon	158	3.9×10^{12}	4.5×10^{14}	9.4×10^{14}

An averaged activity for each brick has been calculated and used as basic data to choose packagings. The calculated activity inventories have to be confirmed by measurements on samples taken from the reflectors. This is scheduled for 1998 after finishing the defuelling.

5. *Treatment options for graphite*

In 1984, a study of „Assessment of management modes for graphite from reactor decommissioning" has been performed in the framework of the EU-R&D programme on decommissioning of nuclear power plants [3]. The large amount of graphite expected from the future decommissioning of Magnox (F, UK) and AGR (UK) reactor systems was the topic of this study.

Waste routes like

- incineration
- ocean disposal
- deep geological disposal
- shallow land burial

have been considered.

The authors came to the conclusion that the lowest collective dose results from the deep geologic inland disposal option. However, an optimum management mode has to be made in a specific context depending on specific circumstances. National aspects of waste disposal concepts as well as the availability of waste treatment facilities and depositories and last but not least the public acceptance determine the waste route.

In Germany, the geological disposal facility Morsleben will be in operation until 30.06.2000 - extension up to 2005 is under discussion. To meet the time schedule for AVR decommissioning, a disposal in the planned KONRAD geological disposal facility - a former iron ore mine - is assumed. Packaging requirements for KONRAD have to be considered.

6. *KONRAD final disposal regulations*

The Bundesamt für Strahlenschutz (BfS - federal office for radiation protection) has established waste acceptance requirements for the KONRAD repository. These requirements were developed on the basis of the results of a site specific safety-asessment. They include general requirements on waste packages as well as specific requirements on waste forms and packagings and limitations for activities of individual radionuclides [4]. To choose a package, different limits have to be fulfilled.

These are :
- package geometry
- total activity inventory of repository
- activity limits per package
- dose rate limits
- weight limits and
- mass limits

In view of the relevant activation products C 14, H 3 and Co 60 in the graphtie / carbon structures, the limits are :

Package geometry
Tab. 1 gives the accepted types of containers for KONRAD - repository. There are cylindrical containers manufactured of concrete or cast ductile iron and cubic containers manufactured of iron sheet, concrete or cast ductile iron.

Total activity
The maximum disposal activity at the end of the KONRAD operation is defined to less than

6×10^{17} Bq Tritium and

4×10^{14} Bq Carbon 14.

Compared to the calculated inventories in the AVR graphite / carbon (ref. Chptr. 4), only 0,25 % in case of H 3 and 3 % in case of C 14 are demanded.

TABLE 1. DIMENSIONS AND GROSS VOLUME OF THE CONTAINERS ACCEPTED BY KONRAD-REPOSITORY

No.	Container Type	Outer Dimensions			
		Length / Diameter mm	Width mm	Height mm	Gross Volume m^3
1	Concrete Container Type I	Ø 1060		1370	1,2
2	Concrete Container Type II	Ø 1060		1510	1,3
3	Cast Iron Container Type I	Ø 900		1150	0,7
4	Cast Iron Container Type II	Ø 1060		1500	1,3
5	Cast Iron Container Type III	Ø 1000		1240	1,0
6	Container Type I	1060	1700	1450	3,9
7	Container Type II	1000	1700	1700	4,6
8	Container Type III	3000	1700	1700	8,7
9	Container Type IV	3000	1700	1450	7,4
10	Container Type V	3200	2000	1700	10,9
11	Container Type VI	1600	2000	1700	5,4

TABLE 2. KONRAD LIMITS — RESULTS FROM SAFETY ANALYSIS FOR NORMAL OPERATION AND INCIDENTS FOR THE RELEVANT NUCLIDES H-3, C-14 AND Co-60

	AVR Inventory Total (31.12.1992)	Normal Operation (guarantee values)	Incidents		Influence on basic rock Cast-mould -Container Type VI
			APG 02		
			ABK I	ABK II	
	Bq	Bq / Package	Bq / Package		Bq / Package
H-3	1,5 E 15	3,0 E 09 - 7,4 E 10 [1)] 3,3 E 09 - 9,3 E 12 [2)]	2,1 E 14	5,4 E 16	3,8 E 15
C-14	1,2 E 13	1,8 E 08 - 1,8 E 10 [1)] 2,0 E 08 - 2,0 E 10 [2)]	1,0 E 12	1,7 E 14	4,7 E 13
Co-60	3,2 E 15	3,7 E 15 - 3,7 E 17 [3)]	1,7 E 11	1,2 E 14	1,1 E 13

1) for non specified density and depending on the total activity of non-metallic solid matters.
2) only for specified density and depending on the the total activity (rel. leakage rates : < 1 %/a; < 0,01 %/a) for non-metallic solid matters.
3) for β / γ - emitter in waste products with residual moist < 1 %.

Abbreviations :

APG 02	-	waste form group 02 (solid waste)
ABK I	-	waste containers class I
ABK II	-	waste containers class II

Activity limit per package
Guarantee values for specified radionuclides (H 3, C 14, Kr 95, I 126 and Ra 226) and radionuclide groups (α-emitter and Pu 241, β/γ-emitter) are defined. Table 2 gives the relevant range of limits for C 14, H 3 and for β/γ-emitters representing Co 60. These limits result from safety assessment for normal operation. Investigations for incidents, thermal rock influence, criticality and long-time radiological affects leads to the limits in Tab. 2, column 4 - 6 for the radionuclides relevant within this work. The yearly accepted activity of a radionuclide or group of radionuclides is 10^4 higher than the guarantee values. That means for H 3 : 4,2 x 10^{14} Bq (HTO by 10^{10} $\leq$ Ages $<$ 10^{12} Bq), C 14 : 1,8 x 10^{14} Bq (C 14 < 1 % volatile) and Co 60 : 3,7 x 10^{19} Bq.

Dose rate limits
At the time the waste is delivered to the repository, the average surface dose rate has to be less than 2 mSv / h. A local max. dose rate is limited to 10 mSv / h. In 1 m distance of a cylindrical waste container and in 2 m distance of a cubic waste container the dose rate should be less than 0,1 mSv / h. These dose rate limits are compatible to GGVS / ADR limits (Regulations for transporting radioactive goods).

Weight limits
The delivery of cylindrical waste containers is only accepted on a shipping unit. The weight of waste containers including the shipping unit should not exceed 20 Mg. Cubic containers can be handled without a shipping unit directly up to a max weight of 20 Mg.

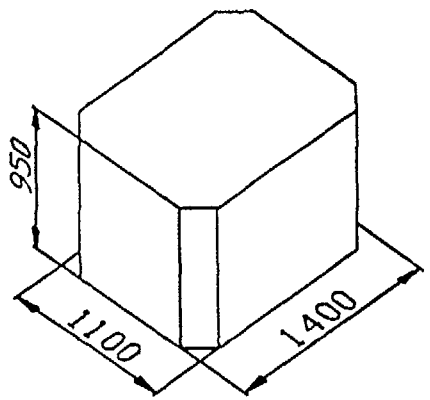
Mass limits
For moderator and reflector materials, a mass limitation has been defined to prevent criticality in the waste container or in the repository. In case of the absence of fissionable nuclides, the max. mass of graphite per package is 420 kg. A higher mass / package has to be approved of by BfS.

7. *Packaging the AVR graphite / carbon*

To meet the KONRAD requirements and the GGVS / ADR transport dose rate limits, cast ductile iron (CDI) with the highest shielding efficiency has been chosen as container material [5]. Three types of CDI containers are accepted for KONRAD disposal (ref.Tab.1; no. 3 - 5).

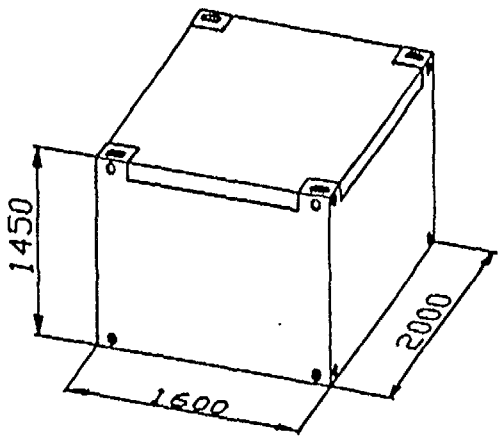
The MOSAIK II-15 container with a gross volume of 1,3 m^3 has been studied at first to store the graphite / carbon from AVR (packaging variant A). This container type has no optimal geometry for packaging the cubic bricks from AVR reflector and isolation layers. The investigations result in a total number of 2,511 MOSAIK II - containers, 1,378 of them have to be equipped with additional lead shields inside to fulfil the abovementioned dose rate limits. A further disadvantage is that most of the bricks have to be cut into smaller sizes to be stored.

Internal Box



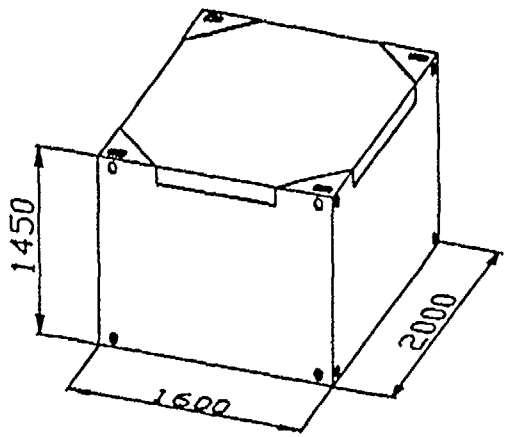
V = 1672 ℓ

Shielding Container for Internal Transportation



Cast Iron Container

Transport and Storage Container



Container Type VII

Fig. 3 : Package for Ceramic Structures (Graphite, Carbon Bricks)

For an optimized packaging, a CDI-container type VII has been designed (see fig. 3). This container is under licensing procedure for transport and storage of high level waste from the reprocessing of fuel elements. An extension of the licence to graphite / carbon seems possible. This variant B packaging concept results in a total number of 364 type VII containers with a cast iron wall thickness of 16 cm. 299 containers need an additional lead shield. Only 150 of 1,031 bricks have to be cut into smaller sizes for packaging. At the dismantling site, the bricks are put into internal boxes to avoid contamination during internal transport in shielding containers to the loading facility.

The packaging concepts taken into consideration need intermediate as well as final storage volume of

-	variant A (MOSAIK)	3,264 m^3
-	variant B (container, typ VII)	1,689 m^3

AVR decided that variant B is the reference packaging for graphite and carbon waste from decommissioning. The licensing procedure for type VII containers has to be extended to graphite / carbon.

8. *Conclusion*

Green field decommissioning of the HTGR results in a large amount of graphite / carbon installations which have to be treated as radioactive waste in a licensable, safe and economic manner. For the 225 Mg of graphite / carbon from AVR decommissioning, investigations to package the waste into containers qualified for transport, intermediate and final storage, have been performed. An optimized packaging can be realized by using cubic containers manufactured of cast ductile iron. The type VII container is under licensing procedure to fulfil the GGVS / ADR regulations as well as the acceptance for the planned German final repository KONRAD.

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REVIEW OF SOME ASPECTS OF RADIOLOGICAL INTEREST DURING THE ESTABLISHMENT OF THE SAFE ENCLOSURE OF THE THTR 300 PLANT

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Abstract

One of the first activities with the establishment of the safe enclosure was the disassembly of the reactor of the burn-up measurement facility. This was a graphite-moderated, air-cooled reactor with strip-shaped fuel elements made of an aluminium uranium alloy. The reactor contained 3.9 kg of high-enriched uranium (93% U-235), the thermal power output was 500 W. Because of the highly cramped conditions, the acceptable dose level and the limited number of fuel stripes, the decommissioning was executed almost exclusively manually. To reduce the collective dose of the personnel, an extensive training with a 1:1 scale mock-up was carried out prior to decommissioning. The removed fuel elements were put into special baskets and were shipped to the interim storage facility BZA in two CASTOR THTR/AVR casks.

In order to clear place for the installation of components of the new ventilation system and other systems, the components for high-purity helium compression and storage had to be dismantled. More than 90% of the metal were unconditionally released as iron scrap.

Extensive measurements had to be carried out on the dismounting and inspection equipment which had been mostly already in use during the 3 year time of operation. As a result 3 Mg had to be stored in the remaining controlled area, app. 183 Mg were stored within the supervised area and app. 49 Mg were released as free of contamination.

Due to the high tritium inventory, two containers with barrels filled with waste could not be shipped to external storage sites and therefore had to be stored in the remaining controlled area within the envelope of the safe enclosure.

Another interesting aspect of the low contamination level of the THTR 300 plant was the release of buildings from the restrictions of the Atomic Energy Act and reduction of the controlled area to a supervised area. Based on statistical methods we were able to prove the low-level contamination status with an acceptable amount of measurements.

Finally a new system for monitoring of released radioactivity with the new exhaust air system was designed and built. Government authorities requested a system with advanced sensibility for low emissions of tritium and carbon-14. The design especially had to consider the highest mean time between failures and the lowest mean time to repair possible.

1. Introduction

This paper is to give a brief review of some aspects of radiological interest during the establishment of the safe enclosure of the THTR 300 plant operated by the Hochtemperatur Kernkraftwerk GmbH (HKG). During the establishment of the safe enclosure, the consortium KSE (Noell-KRC and STEAG Kernenergie) as a general contractor was also responsible for radiation protection organization and waste management and provided one of the radiological health and safety officers and all of the health physics personnel.

2. Dismantling of the burnup measuring reactor

Loading and unloading of the THTR reactor core was carried out while the burnup of the pebble-shaped fuel elements was monitored by means of a burnup measuring system. The main component of this system was the burnup measuring reactor (Solid Moderated Reactor) in which a reactivity effect is caused by operating elements as they pass through the reactor. Evaluation of this effect permits determination of the type of element and, in case of fuel elements, the burnup of the element.

Unloading of the THTR reactor core was completed by 28 October 1994 with the establishment of the state "reactor core free of nuclear fuel". By then, the task of the burnup measuring reactor was completed so that dismantling of the burnup measuring reactor could be initiated in order to remove the nuclear fuel contained therein.

2.1. Initial situation

The burnup measuring reactor was a graphite moderated thermal reactor with a rated output of 500 W, arranged in the reactor hall below the prestressed concrete reactor vessel. The reactor core (1.0 m · 1.2 m · 1.1 m) consisted of various graphite plates provided with grooves for accommodation of the 767 strip-shaped fuel elements. The fuel elements have a rectangular cross-section (15 mm · 1.1 mm) and a length of between 89 and 711 mm. They contain 93% enriched uranium in a U-Al alloy (20% uranium, 80% aluminum). Total uranium content of the core was 3.9 kg.

The core was enclosed by a graphite reflector consisting of plates similar to those of the core. Outside dimensions of the SMR were thus 1.8 m · 2.0 m · 2.0 m. The operating element guide tube, used to guide the operating elements rolling through the core by gravitation, passed through the center of the reactor core. The entire reactor composed of graphite plates was mounted on a steel slab anchored to the floor and was supported by a steel structure installed around the reactor. Reactor instrumentation, absorber rods and the neutron source were arranged in twelve vertical drill holes through the reactor core. Figure 1 gives a general idea of the burnup measuring reactor.

The initial radiological situation was determined by a burnup of approx. 3.1 MW·h/kg U after unloading of the THTR reactor core. Overall activity, originating mainly from the fission products, totaled $2 \cdot 10^{12}$ Bq/kg U. The measured dose rate in the room of installation of the SMR ranged from 500 to 800 µSv/h as regards gamma radiation and was below 1 µSv/h as regards neutron radiation. Values ranging from 1.5 to 20 Bq/cm² were determined for non-fixed contamination. When dismantling work was initiated, indoor air activity concentration was below 40 Bq/m³.

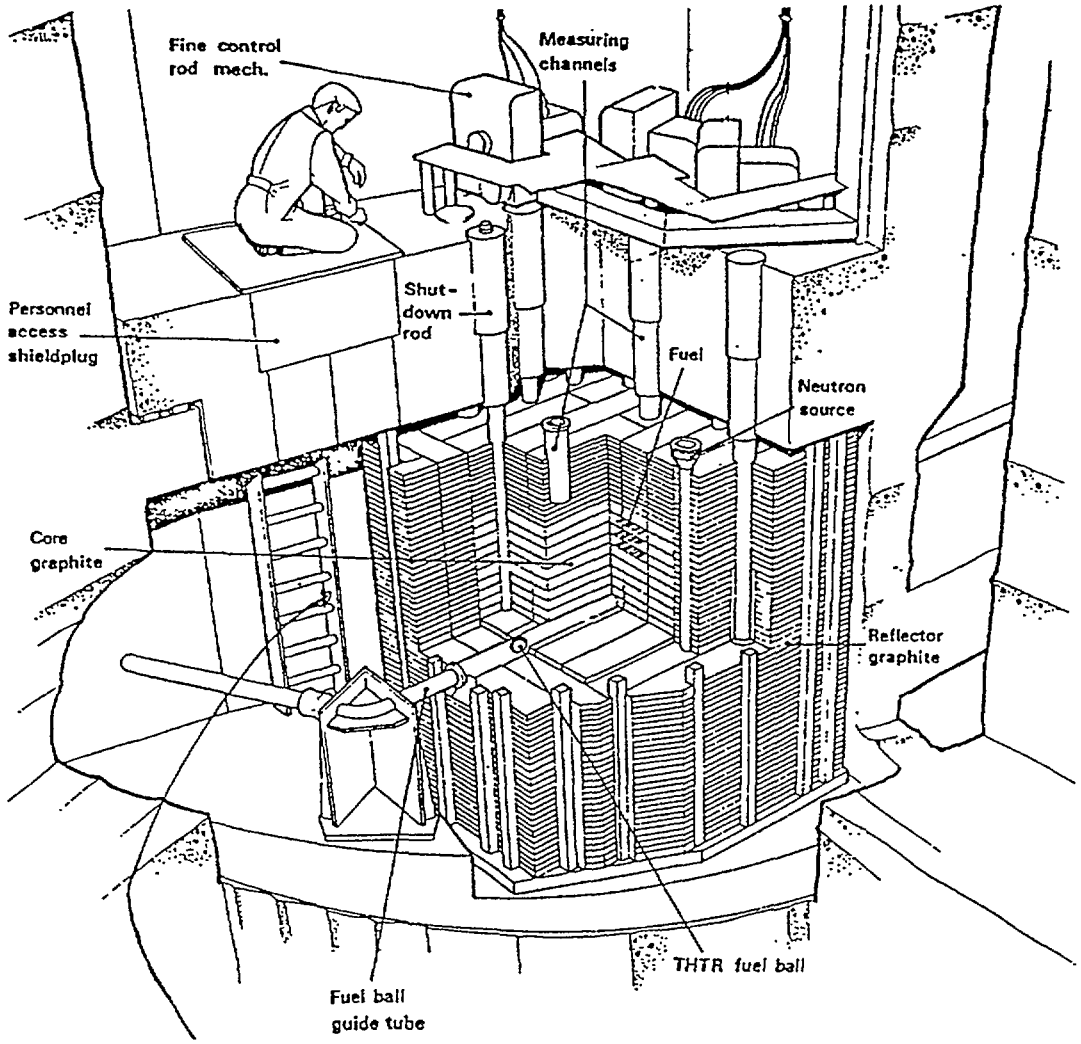

FIG. 1. Installation of the burnup measuring reactor

In terms of technology to be used for dismantling, the initial situation was characterized by extremely cramped spatial conditions in the SMR room of installation and difficult access to this room.

2.2. Preparatory activities

Under the given radiological and spatial conditions, a dismantling concept was chosen which was based on manual dismantling using suitable auxiliary equipment and installations. Preparatory activities included mainly the installation of a 1:1 mock-up, staff training and testing of the auxiliary installations.

The mock-up consists of an SMR room of installation and the access area arranged above. Height of the mock-up totals approx. 7 m.

The mock-up was used to test all equipment and installations designed specifically for dismantling of the reactor (particularly the mobile shield) both individually and in interaction and the devices were optimized.

Testing of the devices was followed by training of the dismantling personnel. Training concentrated on

- assembly of the auxiliary installations under the cramped conditions;
- video-monitored handling of the fuel elements and graphite plates by means of auxiliary equipment;
- handling of the fuel element shielding container (up to storage of the fuel elements in the THTR 300 fuel element storage facility);
- conduct in case of incidents and management of abnormal situations.

The work papers and step sequence plans used for testing and staff training were revised on the basis of the experience made with the mock-up and provided thus the basis for a comprehensive set of work papers and step sequence plans for dismantling of the SMR.

2.3. Dismantling of the burnup measuring reactor

The burnup measuring reactor had to be dismantled only to the extent necessary for removal of the nuclear fuel.

In addition to the equipment and installations that had already been installed at the mock-up during testing, the following preparatory activities had to be developed in the THTR 300 nuclear power plant:

- preparation of the transport path leading up to the SMR room of installation, including assembly of the transport means (inclined haulage and hoist);
- provision of a charging aid for the shielding device, reloading into transport and storage cask transport cages;
- installation of an auxiliary ventilation system for the SMR room of installation.

The activities in the THTR 300 plant - from preparation via removal of fuel elements, loading into shielding device and until transfer to the THTR fuel element storage facility - were carried out by a staff of approx. ten persons in one shift over 30 work days. The collective dose for the personnel was only approx. 10% of the maximum value of 200 mSv stated in the application that had been filed for the licensing procedure under nuclear law, and approx. 20% of the dose expected according to the initial step sequence plans.

The SMR fuel elements were loaded into two transport and storage casks CASTOR THTR/AVR. Shipping of the two CASTOR casks to the Ahaus fuel element interim storage facility on 10 March 1995 completed the activities for management of the SMR fuel elements.

3. Some main sources of waste from decommissioning

The wastes arising from the relevant work for the implementation of the safe enclosure and their destination - except waste containing nuclear fuel – are compiled in figure 2. The following are some of the main sources which are discussed in detail.

License	Year	Works	Shipped				Stored in THTR	
			Unrestr. Release		Radioact. waste			
			[Mg]		[Mg]		[Mg]	
7/12a								
	1994							
		Defueling of the core	14	1)	16		56	2)
		Final inspection						
Am.No.1	1995	Disassembly of the measurement reactor	16		5	3)	17	
Am.No.2		Release of the steam-feedwater-circuit	88	4)	-		-	
		Removal of mufflers	168		-		-	
Am.No.3, 4		Sealing components, Helium compressor	62		1		60	
	1996	Dismount. equipment	64		2		186	
7/12b		New vent systems	88	5)	-		1	
Am.No.1		Evaporator plant, Change status rooms	-		33	6)	-	
		Reconnect vent syst.	21		3		10	
		Remove vent stack	79		-		-	
	1997							
		Total: 7)	**730**	8)	**83**		**350**	

1) Pulverized resins from the condensate cleaning system
2) Graphite and absorber elements within the spent fuel element storage
3) Solid organic waste
4) Removed parts are included only.
5) Rubble
6) Evaporator concentrate and mud
7) Including works not listed above.
8) + 7,400 Mg structural steel and components
+ 44,400 Mg reinforced concrete

FIG. 2. Sources of waste during decommissioning

3.1. No-contamination-measurements for components of the secondary system

With a second amendment to the core unloading license (7/12a), no-contamination-measurements of components in the turbine hall and in the adjacent feedwater tank building and the disassembly of the steam-feedwater-cycle mufflers on the roof of the reactor hall were permitted. Only the waste water discharge station in the supervised area of the turbine hall continued to be subject to measurement after having achieved the state of safe enclosure.

The no-contamination-measurements of components of the steam-feedwater-circuit were performed at this early stage to enable reuse of these components at any other site.

Theoretical investigations based on measurements with pulverized resins from the condensate-cleaning system and certain experience from AVR led to the conclusion that the complete steam water cycle would stay clearly below the threshold value for unconditional release of iron scrap. The number of measurements to be made was comparatively small. Prior to granting of the second amendment to license 7/12a, HKG took 10 measurements at components that were easy to exchange (e.g. valves) and components easy to access (e.g. low

pressure section of turbine above the condenser) to verify the theoretical model. After granting the second amendment an additional 20 measurements were carried out together with the experts. These gamma-spectrometric measurements were partly made in the laboratory and partly in situ. For the in-situ-measurements, the background level was subtracted and the calibration factor was calculated for the actual geometry. The detection threshold referring to Co 60 was 0.006 Bq/cm² (equivalent to 0.001 Bq/g at a minimum thickness of 8mm). The nuclear supervising authority approved the release of these components on October 20, 1995.

Six non-contamination-measurements of the mufflers were performed under the supervision of the Technical Inspection Service at two representative mufflers. Approval for disassembly of all 6 units and their unconditional release for scrapping was given on July 19, 1995. The total masses of iron scrap arising from these activities are shown in figure 2.

3.2. Disassembly of the high-purity helium compressors

The 4th amendment to license 7/12a was issued on October 27, 1995. It permitted the disassembly of components of the high-purity helium compression and storage system, aiming to clear space for the installation of components of the new ventilation and activity monitoring system.

The components to be removed were installed in the supervised area of the plant. Thus they actually had to be non-contaminated. It was known, however, that certain inner surfaces of pipes had been slightly contaminated due to back streaming gas during plant operation.

Contaminated and non-contaminated piping segments had to be determined. During these measurements, it was found that the two heavy four-stage helium compressors were slightly contaminated in their first stages. They were disassembled prior to being subjected to a thorough investigation. As far as necessary they were decontaminated.

Parts of the system for which no-contamination-measurements were easy, were brought to closed containers installed outside and stored there until approval by the authorities had been obtained. Parts for which the state of no-contamination was to difficult to prove, were packed into 200 l-barrels and stored in the supervised area for the time of safe enclosure.

The final no-contamination measurements for the helium compressors were made in March 1996. More than 90% of the material (approx. 62 Mg) was unconditionally released as iron scrap.

3.3. Measuring of the dismounting and inspection equipment

The THTR nuclear power plant was equipped with a partly shielded dismounting and inspection equipment. This equipment was used for work on the fuel circulating system, the helium purification system, the absorber rods and for the inner inspection of the prestressed concrete reactor vessel. In parts, this equipment had already been in use and was therefore contaminated.

Due to the fact that some of the equipment had already been disassembled, the number of contamination-measurements amounted to several hundred. As a result of the measurements, the single parts were classified into three groups: parts with a contamination higher than

5 Bq/cm² were stored within the remaining controlled area; parts with contamination between 5 and 0.5 Bq/cm² were stored in the supervised area of the remaining plant; and parts with a contamination level below 0.5 Bq/cm² were unconditionally released for scrapping.

As a result, 3 Mg had to be stored in the controlled area, approx. 183 Mg were stored within the supervised area, and approx. 49 Mg were released as free of contamination.

3.4. Waste from external conditioning

In September 1996, two containers with barrels filled with tritium-contaminated waste were stored within the cover of the safe enclosure. The 16 D350-barrels of high-grade steel were filled and sealed at the Karlsruhe research center. Due to the high tritium inventory of the barrels, storage at the final repository Morsleben is not possible today. The tritium-activity amounts to 2.9E+12 Bq per Barrel.

4. Downgrading from controlled area to supervised area

With the first amendment to license 7/12b (establishment of the safe enclosure) issued on July 15, 1996 HKG was allowed to change the status of rooms outside the cover of the safe enclosure from controlled to supervised area. Therefore it had to be proved that the surface contamination of the buildings and components did not exceed 5 Bq/cm² and that the dose rate did not exceed 7.5 µSv/h in this area. In addition, HKG demanded that in rooms which should be accessible without any restrictions to persons who are not occupationally exposed to radiation, the dose rate should not exceed 2 µSv/h.

For a total area of about 12000 m² (170 rooms) the fulfillment of the above conditions had to be proved. The proof was provided in two steps: through the analysis of the history of operation of the plant and through measuring at representative locations. For regions on the floor with high probability of contamination, the number of measurements was 1 measurement per 2 m²; for regions with low probability of contamination and for walls up to a height of 2 m the number of measurements was 1 per 10 m². For components, the number of measurements was 1 per m². Dose rate measurements were done in every room. Spots with higher dose rates were either decontaminated or shielded. Components that continued to fail meeting the specifications of the supervised area even after decontamination were dismantled and stored in the remaining controlled area.

Due to the low-level contamination of the former controlled area the change to a supervised area was achieved with a relatively small amount of measurements. The total number of contamination measurements was 2316. Only 87 measurements showed values higher than 0.5 Bq/cm². All measurements were taken in the presence of members of the Technical Inspection Service.

5. Release of buildings from the scope of nuclear legislation (AtG)

The first amendment to license 7/12b allowed initiation of measurements for the release of buildings from the area of application of nuclear legislation. All buildings of the site outside of the safe enclosed plant were to be released from the restrictions of the Nuclear Energy Act, that is they were no longer subject to nuclear legislation.

From the analysis of the history of plant operation the buildings were divided into three classes:

- AE: slightly contaminated;
- BE: probably not contaminated;
- CE: clearly not contaminated.

All buildings outside the supervised area belonged to class CE. The turbine hall and the feedwater tank building belonged to class BE just as the health physics laboratory and some rooms of the access and safety building. Only the waste water disposal duct and parts of the waste water discharge station in the turbine hall were known to be contaminated and therefore allocated to class AE.

After thoroughly cleaning the waste water disposal duct of contaminated mud and dismantling of the contaminated parts of the waste water piping, it was possible to provide proof of no-contamination. For that proof, it had to be shown that the surface contamination was below the limits of the German radiation protection ordinance: 0.50 Bq/cm² for most of the beta/gamma-nuclides and 0.05 Bq/cm² for alpha-nuclides. From experience it was known that only Co 60, Cs 134, Cs 137 and Sr 90 (and in special cases H 3) were relevant in most cases.

Material samples to prove falling below the mass-specific clearence level of 0.1 Bq/g were taken from the waste water duct and from several sumps in the turbine hall and the feedwater tank building. In other cases, proof was provided by means of gamma-spectrometric in-situ measurements.

The number of contamination measurements was 1 per 25 m², however, at least 3 per room, in order to permit evaluation of representativeness. Locations with an increased probability of contamination were chosen as measuring points, such as floor drains, sumps and transport paths. At the outset, the number of measurements to be taken at certain components was increased to 1 measurement per 10 m².

All gamma-spectrometric measurements showed values of contamination by Cs 137 exceeding significantly the clearence level. By means of the relations of activities of Cs 137/Cs 134, it was possible, however, to prove that this contamination was due to the Chernobyl incident and not due to the THTR 300 operation.

A total of 729 contamination measurements had been carried out. About 30 samples were tested with the gamma-ray spectrometer and 10 in-situ measurements were taken with a portable gamma-ray spectrometer.

6. Summary

In general, exposure to radiation during all activities for the establishment of the safe enclosure was significantly lower than expected. Due to the low level of contamination in the controlled and supervised areas, it was possible to furnish the radiological proof required for downgrading of the controlled area to the supervised area and for release of buildings from the supervision under nuclear law by means of a relatively small number of measurements.

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