SRP REPRESENTATIVE TO THE FEDERAL REPUBLIC OF GERMANY (FRG)

Donald E. Clark

Monthly Report for April 1987

Summary

The principal activities during this reporting period included a continuing exchange of programmatic information and data on the Asse mine, the candidate nuclear waste repository at Gorleben, and the FRG direct disposal concept; consultative visit to the U.S.A. for discussions with the SRP; participation in CEC meeting on natural analogues; and preparations for upcoming U.S. interactions and meetings. Copies of very informative video tapes on the freeze shaft sinking technique being used at Gorleben and the Hope mine flooding experiment conducted in the FRG were provided to the SRP.

Introduction

Beginning in early 1987, the long-term assignment of a representative of the Salt Repository Project (SRP) to the nuclear waste disposal program in the Federal Republic of Germany (FRG) was established as part of the ongoing interactions between the two countries under the U.S./FRG Bilateral Agreement (Waste Management). Through day-to-day contacts and close association of a technically cognizant SRP representative with key aspects of the FRG program, the objective of having a systematic exchange of pertinent programmatic information and data on the nuclear waste disposal programs of both countries is being realized. During this reporting period, additional valuable contacts with key FRG personnel were established and direct communication with SRP management was maintained. A consultative visit to the U.S.A. to brief SRP staff on developments in the FRG provided important feedback for future concentration of efforts.

Gorleben Salt Repository Site

At the candidate Gorleben Salt Repository Site, activities are continuing on the current phase of characterizing the salt dome for its acceptability as a geologic repository for nuclear waste. Separated by about 400 meters (ca. 1,300 feet), two large-diameter shafts are being sunk to depths of 850 to 900 meters, using the freeze-shaft sinking technique. A status report on the shaft sinking activities which was recently submitted to the SRP is attached (see Attachment 1). Through the end of April 1987, no serious problems had been encountered during shaft-sinking operations. Shaft 1 had been sunk to a depth of about 230 meters and shaft 2 to about 30 meters.

An informative set of photographs was obtained on the Gorleben site (see Attachment 2). During future visits to the site, updated photographs of site activities will be obtained for use by the SRP.

Asse Salt Mine

The Asse salt mine is being used as an underground laboratory for research and development testing of domal salt as a medium for disposal of radioactive wastes. A major part of the field testing required for the license application for the Gorleben repository is being conducted in the Asse salt mine. This facility is being operated by the German Company for Radiation and Environmental Research, Institute for Underground Storage (GSF/IfT). A very nice booklet with excellent color photographs and figures has been produced by the GSF/IfT on the Asse salt mine; unfortunately, this booklet has only been available in German, but now it is being translated for eventual publication as a document in English. The English translation is included in this report (see Attachment 3), since it provides a good summary of this facility.

Karlsruhe Nuclear Research Center, KfK

The Karlsruhe Nuclear Research Center, KfK, was visited during this reporting period and meetings were held with staff of the Reprocessing and Waste Treatment Project (PWA) and the Program for Direct Disposal (PAE). An outline of the PWA activites is included (see Attachment 4); the PWA group has gained valuable experience in the reprocessing of irradiated fuel and vitrification of the reprocessing wastes in Karlsruhe and Mol, Belgium. The PWA experience directly supports the FRG reprocessing plant that is being constructed at Wackersdorf. The PAE is conducting research and development on the disposal option for directly disposing of packaged spent fuel in a salt repository. Important field tests are planned to be conducted in the Asse salt mine for this diposal concept, and these will be discussed in detail in later reports to the SRP. A pre-publication copy of the paper entitled "Concepts for Direct Disposal of Spent LWR and HTR Fuel in the Federal Republic of Germany" was obtained during this visit (see Attachment 5).

Visit to Mol, Belgium; CEC Meeting on Natural Analogues

A visit was made to the underground laboratory for disposal of nuclear wastes in a clay formation and to the PAMELLA vitrification facility in Mol, Belgium. Discussions were held with staff of the Commission of the European Communities (CEC) in Brussels, Belgium, and the CEC Symposium on Natural Analogues in Radioactive Waste was attended. Details of these activities will be presented in the next monthly report.

Consultative Visit to the U.S.A.

A consultative visit was made to Columbus, Ohio for the purpose of briefing SRP staff on developments in the FRG. Presentations were given to SRP management and technical staff; two very informative video tapes, one on the freeze shaft sinking technique being used at Gorleben and the other on a flooded mine experiment conducted at the Hope mine in the FRG, were shown to the SRP. Copies of these two video films were given to the SRP International Coordinator for future use by the SRP. During the discussions with SRP staff, valuable feedback was obtained concerning areas of particular interest in the FRG program. Another consultative visit to the U.S.A. is planned during the summer of 1987 (probably in late July).

Activities Planned for May 1987

The month of May will be very busy as there are important meetings and a number of U.S. visitors expected in the FRG. A U.S./FRG workshop on geotechnical instrumentation is planned for early May 1987. Attendance is planned at the IAEA International Symposium on the Back-End of the Nuclear Fuel Cycle - Strategies and Options in Vienna, and at the Bavarian government's symposium on reprocessing and waste disposal in the FRG. Important visitors will include Mr. Ben C. Rusche, Director of the DOE's Office of Civilian Radioactive Waste Management, and other key U.S. officials.

Attachments

1. Status Report on Shaft Sinking at Gorleben
2. Gorleben Salt Repository Site, 1984-87
3. English Translation of Booklet on the Asse Salt Mine
4. Outline of Activities of the Reprocessing and Waste Treatment Project (PWA)
5. Preprint of Paper Entitled "Concepts for Direct Disposal of Spent LWR and HTR Fuel in the Federal Republic of Germany

dec

STATUS REPORT ON SHAFT SINKING AT GORLEBEN

Donald E. Clark
SRP Representative to FRG

March 13, 1987

The potential suitability of the Gorleben salt dome was confirmed through extensive site characterization work which began in April 1979 and extended through 1983. Within a surface area of about 3000 kilometers (115 square miles), a total of 590 drillings (e.g., hydrological exploratory drillings, salt wash surface drillings, salt dome exploratory deep drillings, and shaft pilot drillings) and 540 geoelectrical surface measurements were made. In addition, about 156 kilometers (97 miles) of deep seismic profiles were recorded. As a result of these activities, a detailed knowledge of the overlying strata and the geohydrology of the Gorleben site was acquired, and it was concluded that sufficiently large rock salt regions exist in which the required disposal fields can be established. Accordingly, definitive plans were made for underground exploration and development of the Gorleben site.

Site-specific planning data required for construction of the repository will be obtained by the sinking of two shafts and subsequent exploration of the interior of this salt formation. Underground exploration will occur during the period of 1988-1992 upon the completion of two exploratory shafts, and will include linking the two shafts underground by means of a drift (they are 400 meters -- about 1,300 feet -- apart), and then driving more than 25 kilometers (15 miles) of exploratory galleries, along with drilling more than 50 kilometers of underground reconnaissance boreholes. Fan drilling will be done to test the salt prior to the driving of all main and cross drifts. The planned dimensions of the drifts are 3 meters by 6 meters, for a cross-sectional area of 18 square meters (about 190 square feet). This program will entail exploration of an area comprising 18 square kilometers (about 6.9 square miles) along the central zone of the salt dome at a depth of 840 meters (2,760 feet). Later on, when the disposal portion of the site is excavated, the exploratory drifts will be used for ventilation purposes (return air).

The disposal level at Gorleben will be 30 meters lower than the site exploration level, i.e., at a depth of 870 meters (2,850 feet), where it is intended that high-level waste (HLW) will be disposed of in vertical boreholes drilled to additional depths of 300-600 meters (985-1,870 feet). Spent fuel (SF) may be disposed of in galleries using the Pollux waste package concept that has been developed by the Deutsche Gesellschaft fuer Wiederfaufarbeitung von Kernbrennstoffen MBH (DWK -- German reprocessing company).

Using the well-established freezing shaft technique discussed below, two exploratory shafts are currently being sunk at Gorleben to depths of 850 to 900 meters. These shafts are located at shaft pilot borehole drillings designated as Gorleben 5001 (shaft 1) and Gorleben 5002 (shaft 2), respectively, which were previously sunk to depths of about 1,000 meters (about 3,200 feet). The 90-millimeter (3.5-inch) diameter drilling cores provide complete data sets on the underground strata and purity of salt at the two exploratory shaft locations. While the shaft excavation diameters are about 10 meters (nearly 33 feet), the inner diameters will be about 7.5 meters (about 25 feet), after final installation of the shaft liners. Extending to the salt region at a depth of about 260 meters (about 850 feet), the final liners will consist of concrete blocks on the outside, then steel sheet, bitumen, and an inner liner of steel reinforced concrete. No linings are necessary in the salt regions.

The strata covering the salt dome at Gorleben consist of Pleistocene and Tertiary sediments, so that for shaft sinking, the miners are dealing with incompetent ground water-bearing rocks. Furthermore, the top part of the salt is covered by a caprock derived from the dissolution of salt which can contain saturated brines. The nature of these overlying strata requires application of the freeze process for shaft sinking to the salt below. In this process, the section of loose or semi-consolidated sediments extending from the surface down to the rock salt is frozen prior to excavation.

At each Gorleben exploratory shaft location, directionally controlled holes were drilled at a distance of 1.3 meters (4.3 feet) in a circle with a diameter of 18 meters (59 feet) to the depth of the salt, 260 meters (about 850 feet). These holes were secured by steel casing and lined with freeze pipes through which a brine coolant (calcium chloride solution) of -40 degrees Celsius is continuously circulated. In extracting heat from the rocks, the returning liquid is warmed by about 1.5 degrees Celsius. Thus, a massive frost body is formed with temperatures of the order of -10 to -20 degrees Celsius in the regions of importance, and, under the protection of which, the shaft can be safely sunk.

Although it was earlier thought that a cutting and extraction method could be used for sinking of the shaft, excavation through the frozen rock layers has required application of a blast and dig-extraction method. Due to delicacy of the frost wall, the explosives must be used carefully (charge emplacement is very important). This excavation method is being successfully applied with no notable problems to date at Gorleben. The excavated shafts consist of open holes, the walls of which are secured by a preliminary lining of concrete blocks. After the shafts reach the salt formation, the final water tight linings will be put into place and the freeze process will be terminated. Excavation through the rock salt will then proceed through application of a standard cutting and extraction method.

To produce the required frost body takes several months or longer, depending on the strata. The only delays experienced at Gorleben have been the result of longer than anticipated times to reach the desired freeze conditions. Shaft 1, for which work is more advanced than for shaft 2, is ahead of schedule at this time. The excavation work at Gorleben in taking place around the clock (four shifts per day), seven days per week. The shaft sinking rate depends on the strata, but averages about one meter per day above the salt, and it will be much faster in the salt formation itself. Freezing operations began in October 1985 for shaft 1 and in 1986 for shaft 2. On March 9, 1987, shaft 1 had reached a depth of 218 meters (715 feet). Shaft 2 is at a depth of 27 meters (89 feet), and work is currently halted to await complete freezing of a clay layer. When the work again commences (this is expected to occur in about two weeks), it will be with the mining equipment, not the simple crane and associated equipment that are used down to depths of about 30 meters.

During the operational phase of the repository, shaft 2 will be used for transport of nuclear waste from the surface (and exhaust of ventilation air). Shaft 1 will be used for everything else, i.e., lowering of other materials and removal of salt to the surface, personnel ingress/egress, and intake of ventilation air.

Based on discussions with DBE staff, the exploratory shaft sinking operations at Gorleben appear to be going very well indeed. No major problems have arisen and it is anticipated that this phase will be completed in 1988, the underground exploration will be carried out from 1988 until 1992, and implementation of the licensing application and other regulatory requirements to obtain approval of the repository will taken from 1993 to 1996. Then, construction of surface facilities and underground preparations is scheduled for the period of 1997 to 2000. It appears that at the earliest the Gorleben repository will be in a position to accept nuclear waste by the year 2000.

GORLEBEN SALT REPOSITORY SITE
FEDERAL REPUBLIC OF GERMANY
1984-1986

1. View of the Gorleben facility, front entrance, head frame Shaft #1 (Fall, 1986).

2. First work at Gorleben, 1983-84. Note trailer facilities for work force.

3. Road construction.

4. Heliport at Gorleben site.

5. Warehouse and view looking to Shaft #2 (start of building the head frame; Fall, 1986).

6. Storage facility for explosives.

7. Social facility at Gorleben, includes offices, conference room, canteen, etc.

8. View of wall and fence, with tower frame for Shaft #1, from outside road at Gorleben.

9. Construction at Shaft #1, drilling of boreholes for the freezing operation (probably in 1984).

10. View of freezing cave for Shaft #1.

11. Beginning of shaft sinking for Gorleben Shaft #1.

12. Celebration for reaching first 30 meters of depth, Shaft #1, September 18, 1986. At this point, the construction facility is changed over to hoist machine, etc.

13. Flag on Shaft #1, commemorating reaching the 30 meter depth.

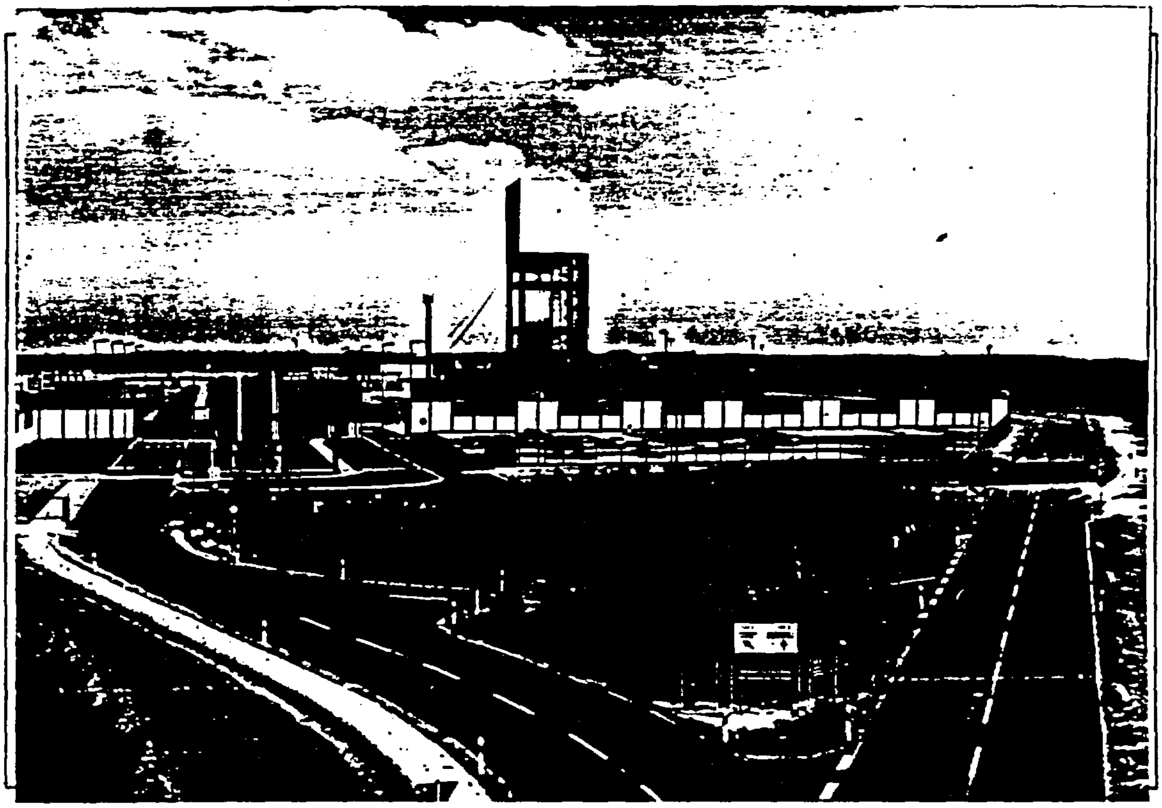
14. Tower for Shaft #2, head frame without the hoist (later on it will have a different head frame and become like Shaft #1 head frame with high weight capacity).

15. View of Shaft #1, early excavation.

16. View toward site for salt pile after construction of basement, looking from head frame of Shaft #1. Salt pile will be spread over water tight seal and designed for collection of rain water.

17. Road construction. View from salt pile site looking back toward Shaft #1.

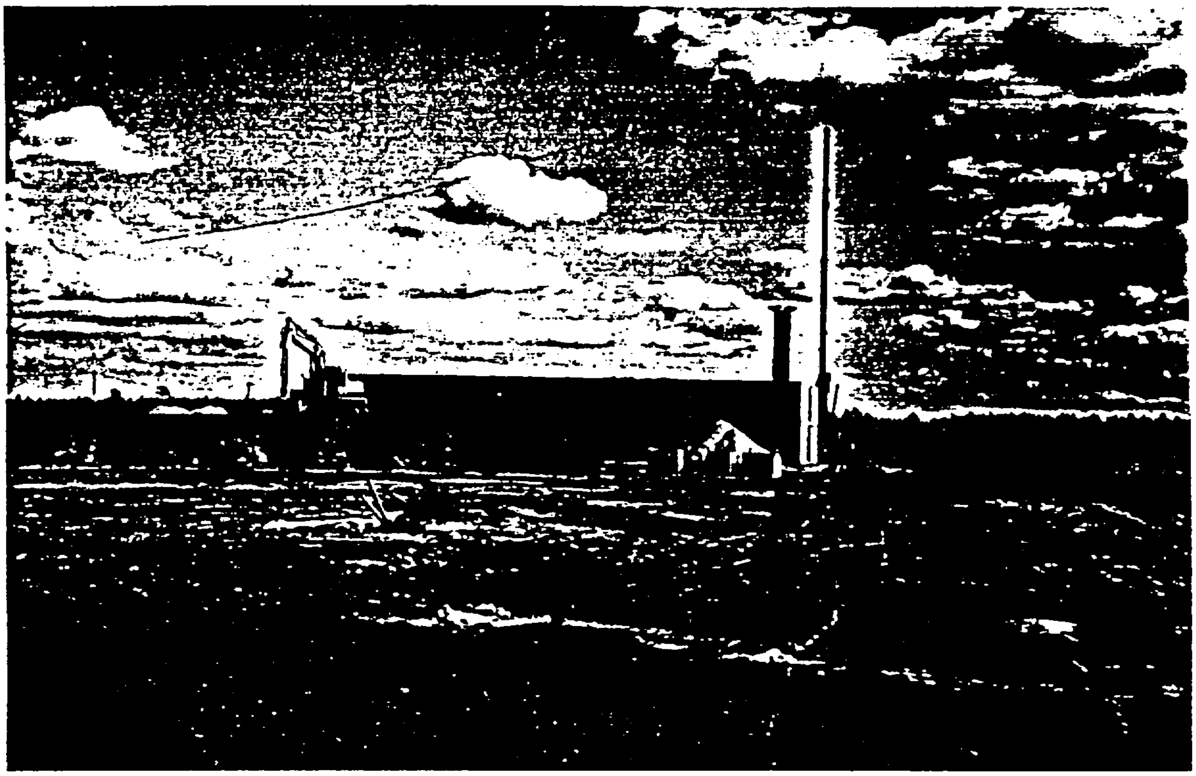
18. Aerial view of Gorleben site, April 1986. In background is Elbe River and border with East Germany.



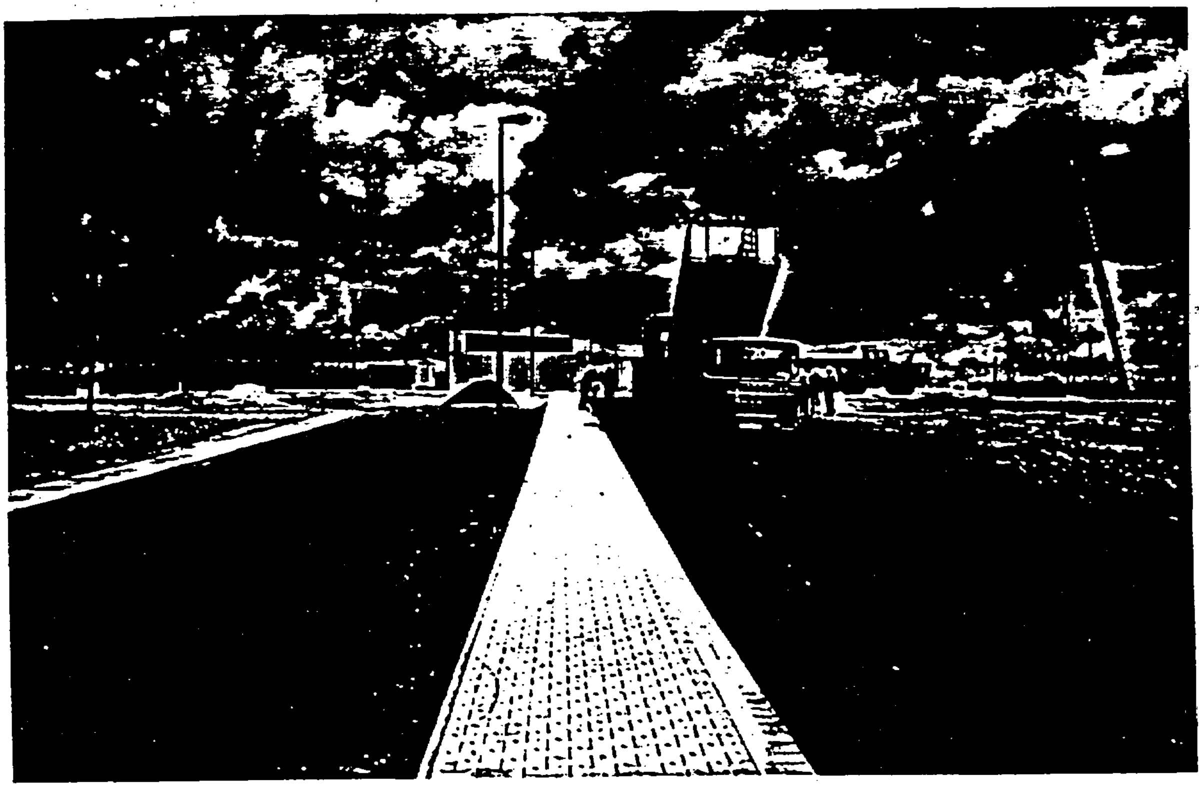
Bergwerk zur Erkundung

des Salzstockes Gorleben

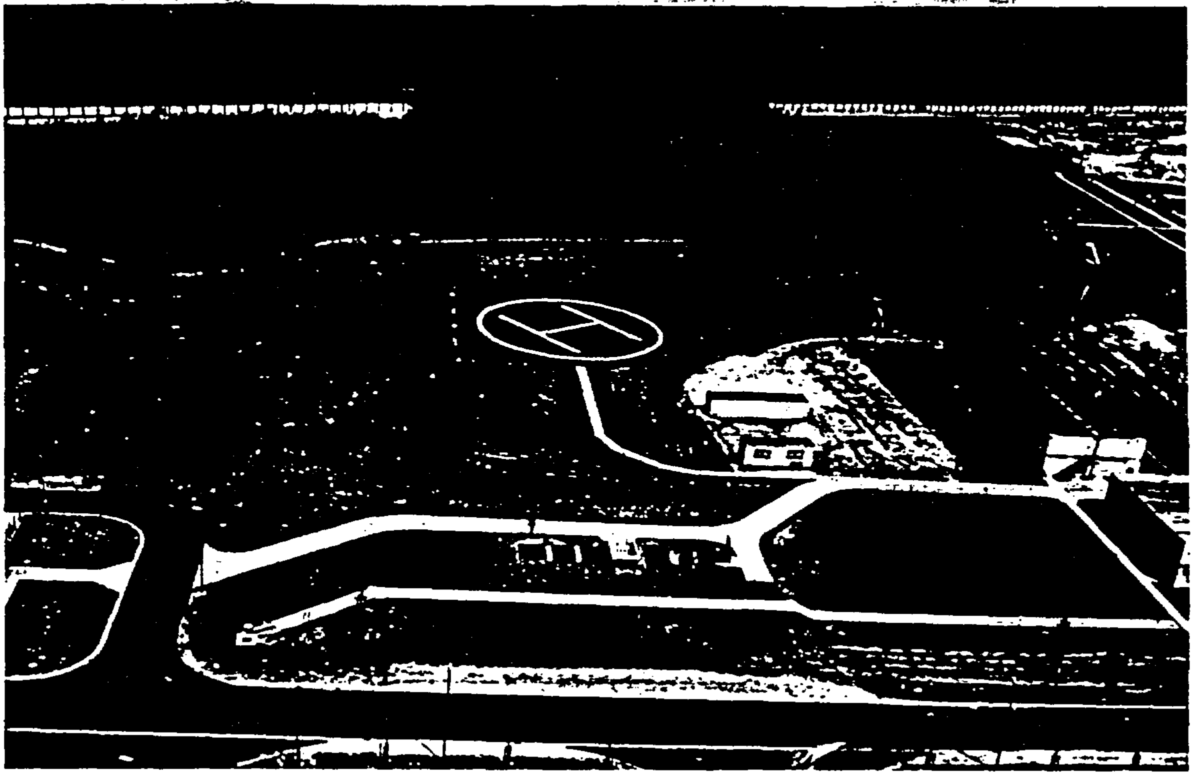
Durchgeführte Baumaßnahmen 1984 - 1986



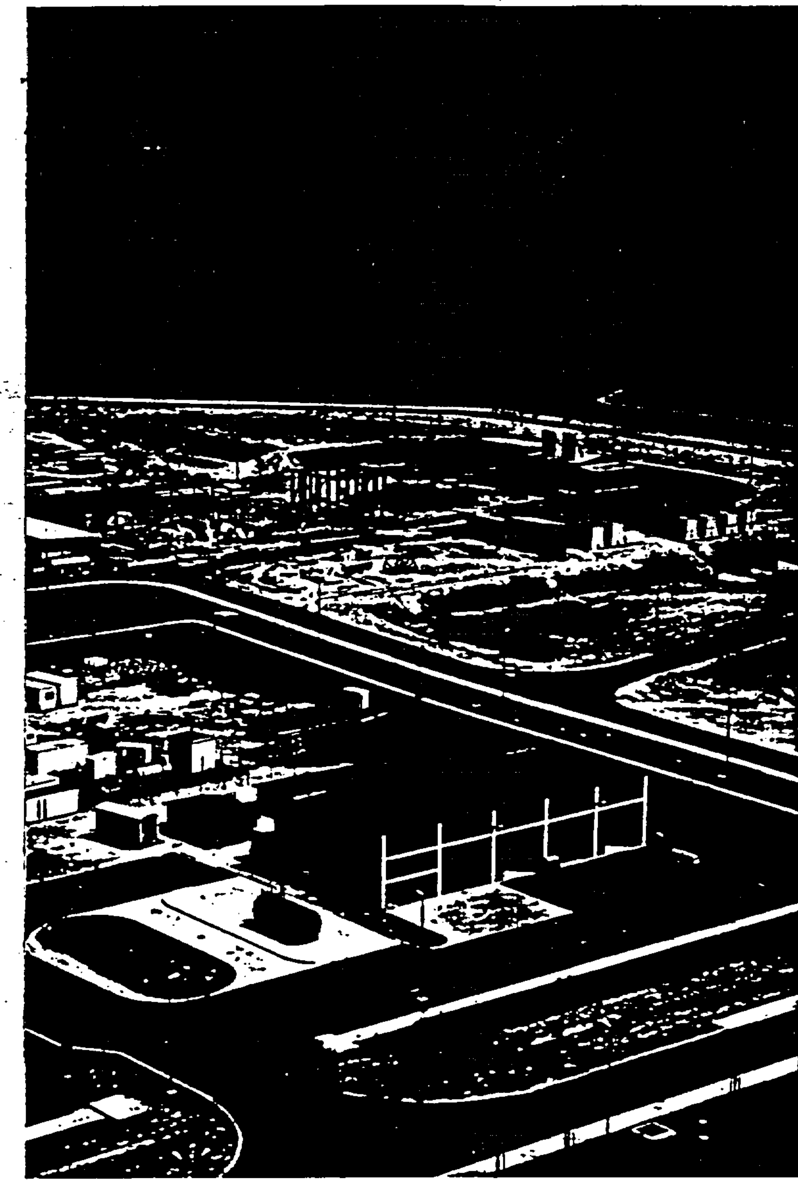
Erdarbeiten , Mutterbodenmieten zur Böschungsandeckung



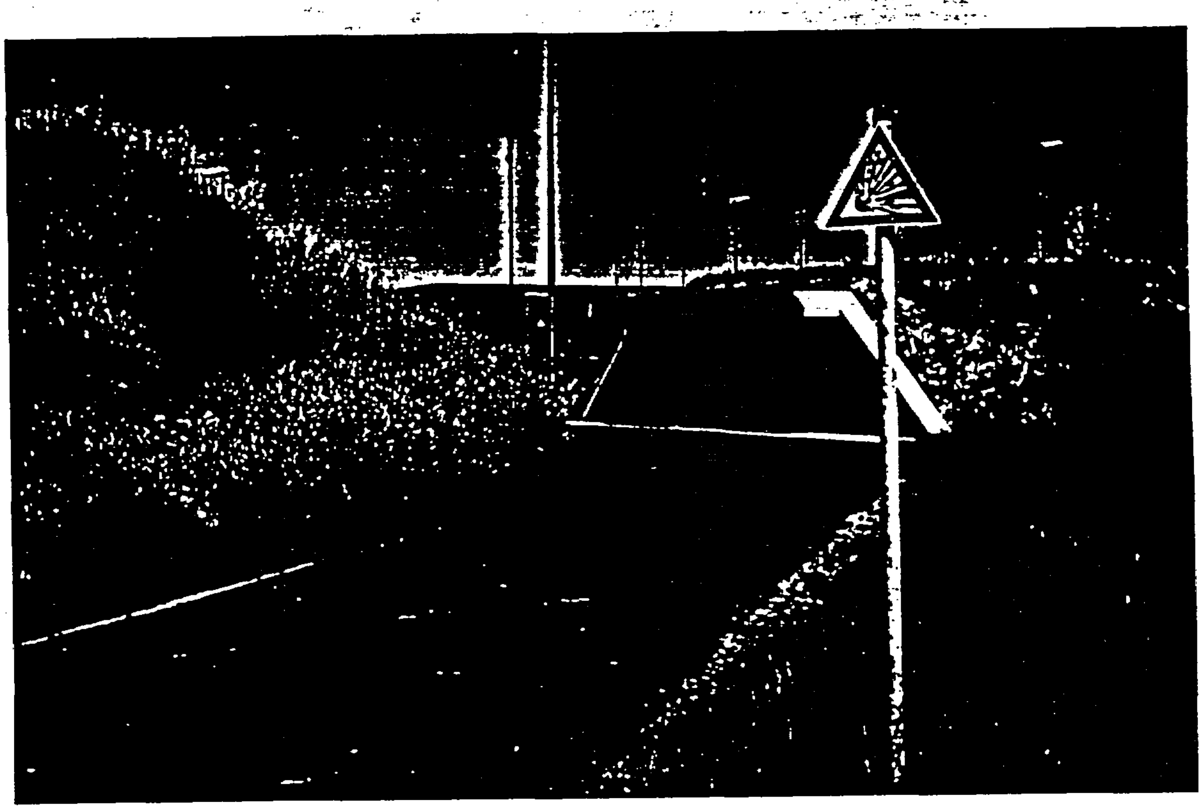
Straßenbauarbeiten



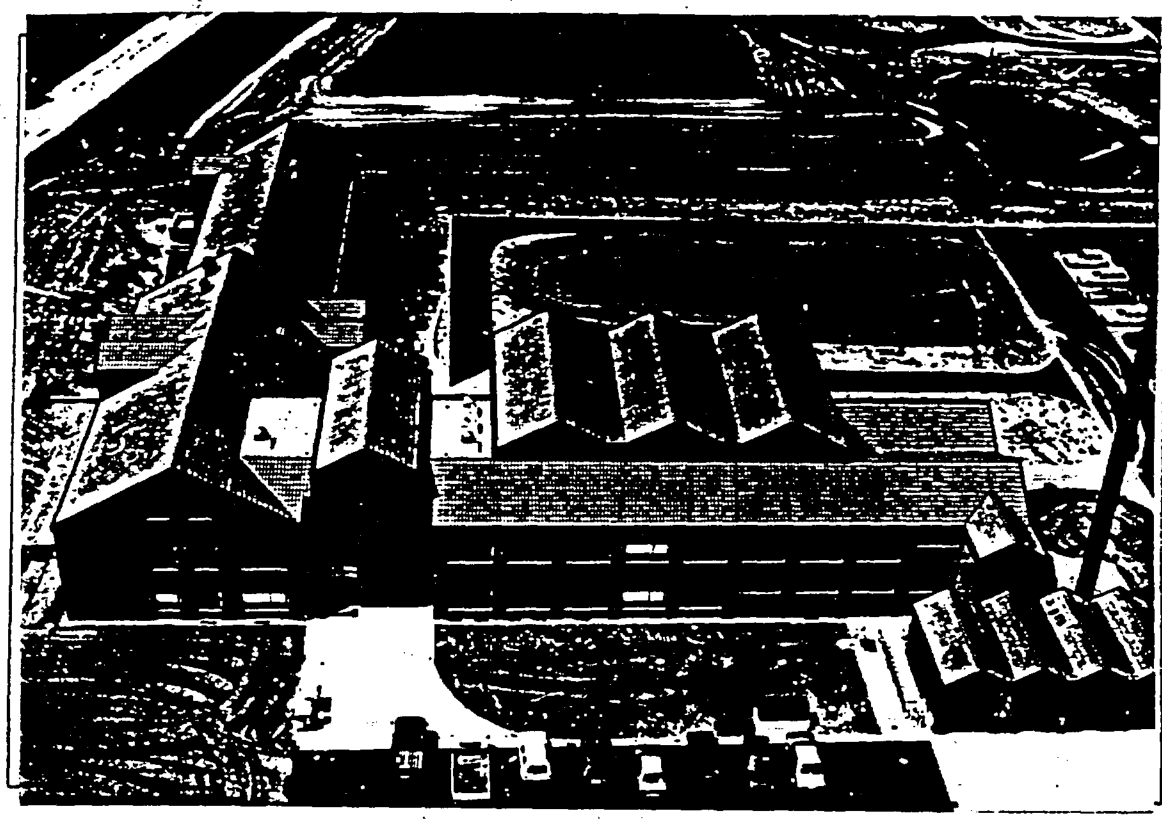
Geländebearbeitung , Hubschrauberlandeplatz



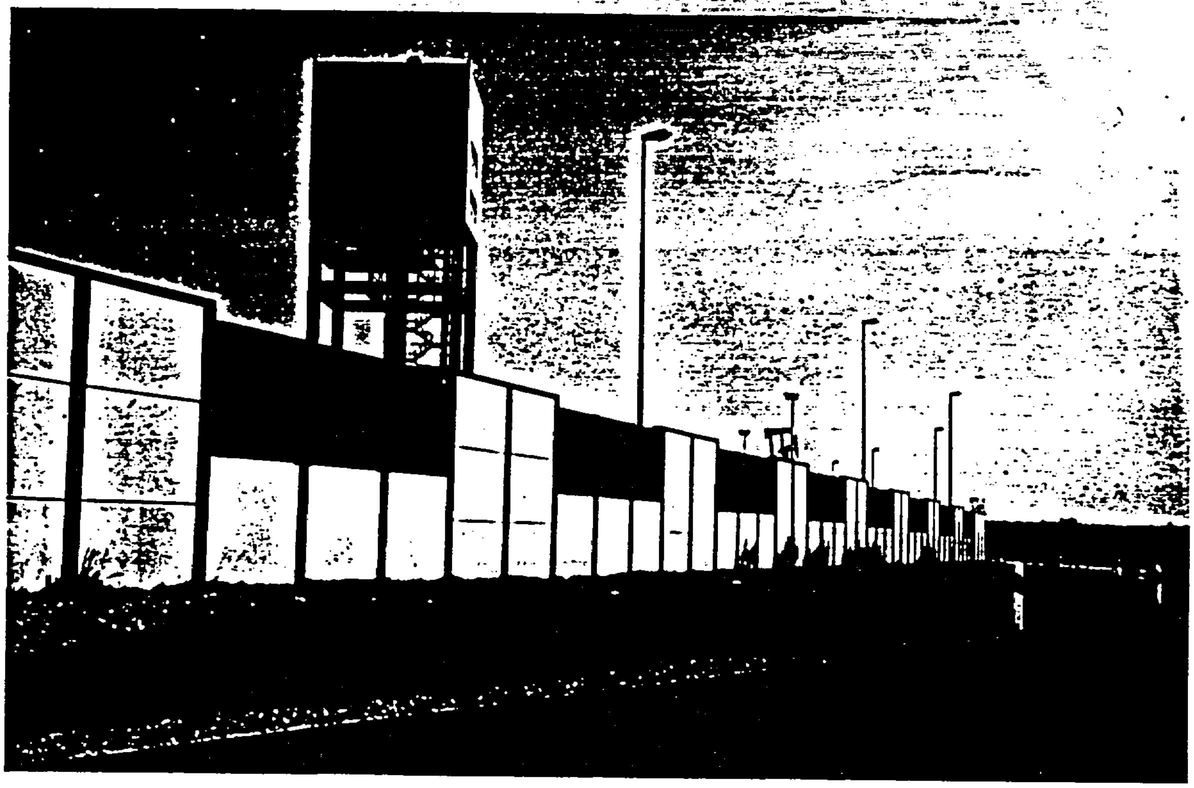
Materiallager , Blick in Richtung Schacht 2



Sprengmittellager

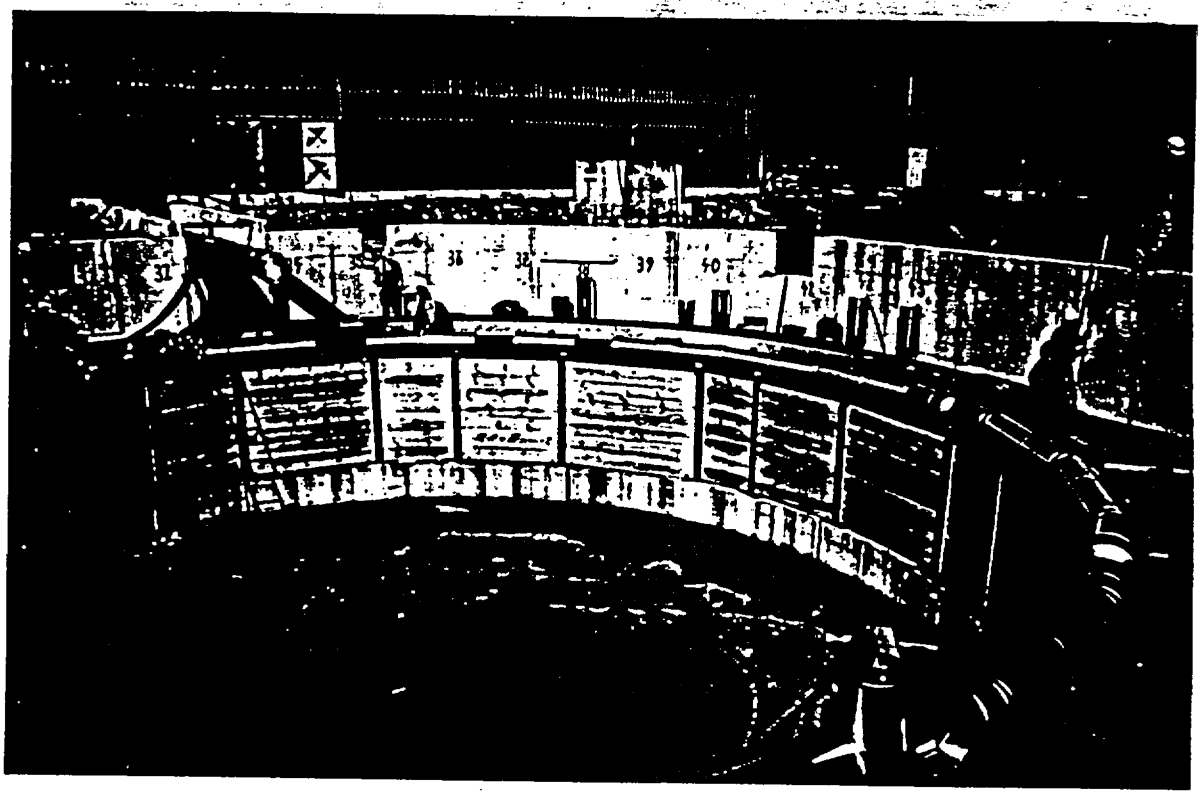


Kauen , Büro - und Sozialgebäude

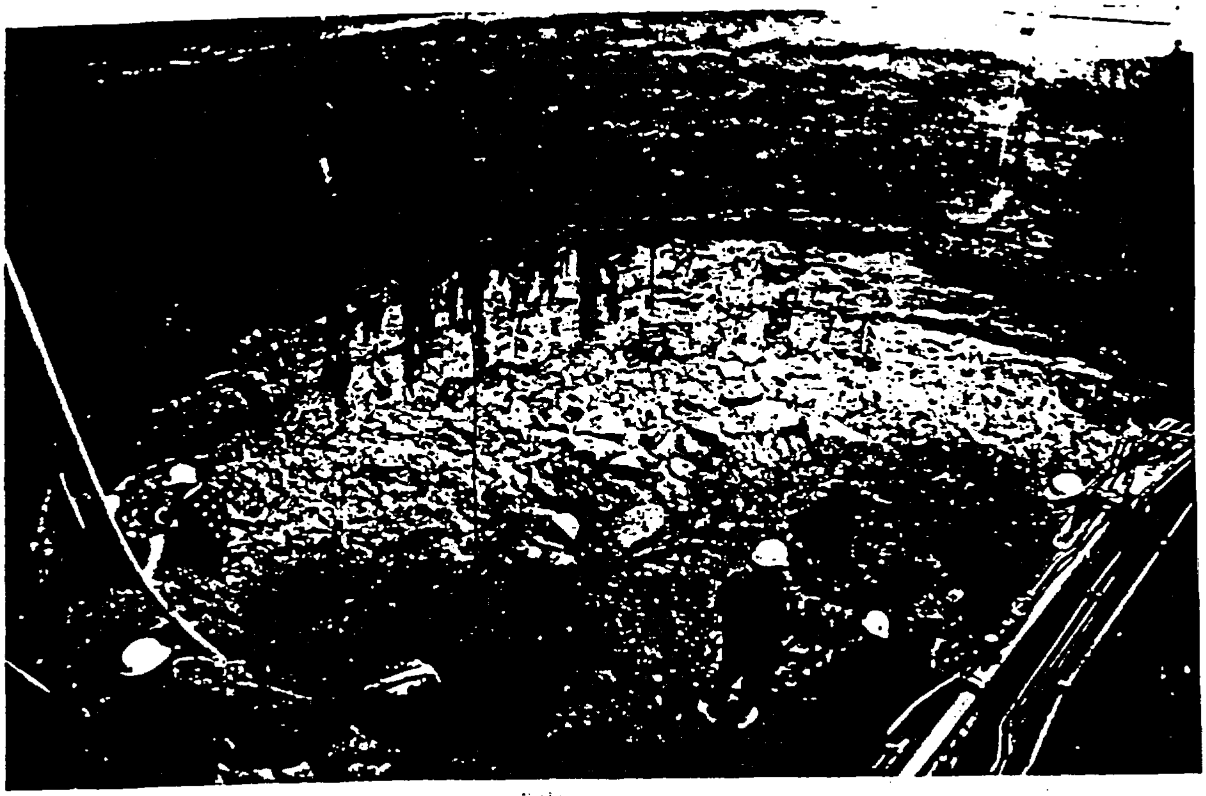


Sicherungseinrichtungen

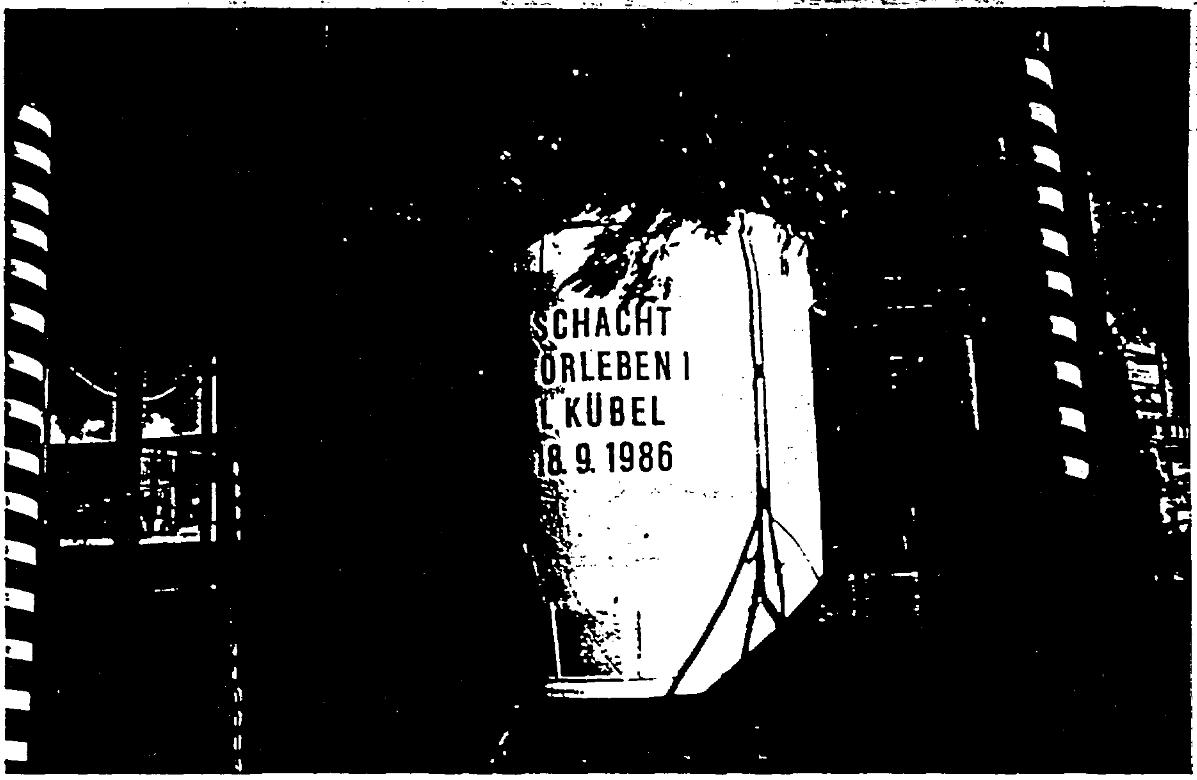
Gefrierlochbohren , Schacht 1



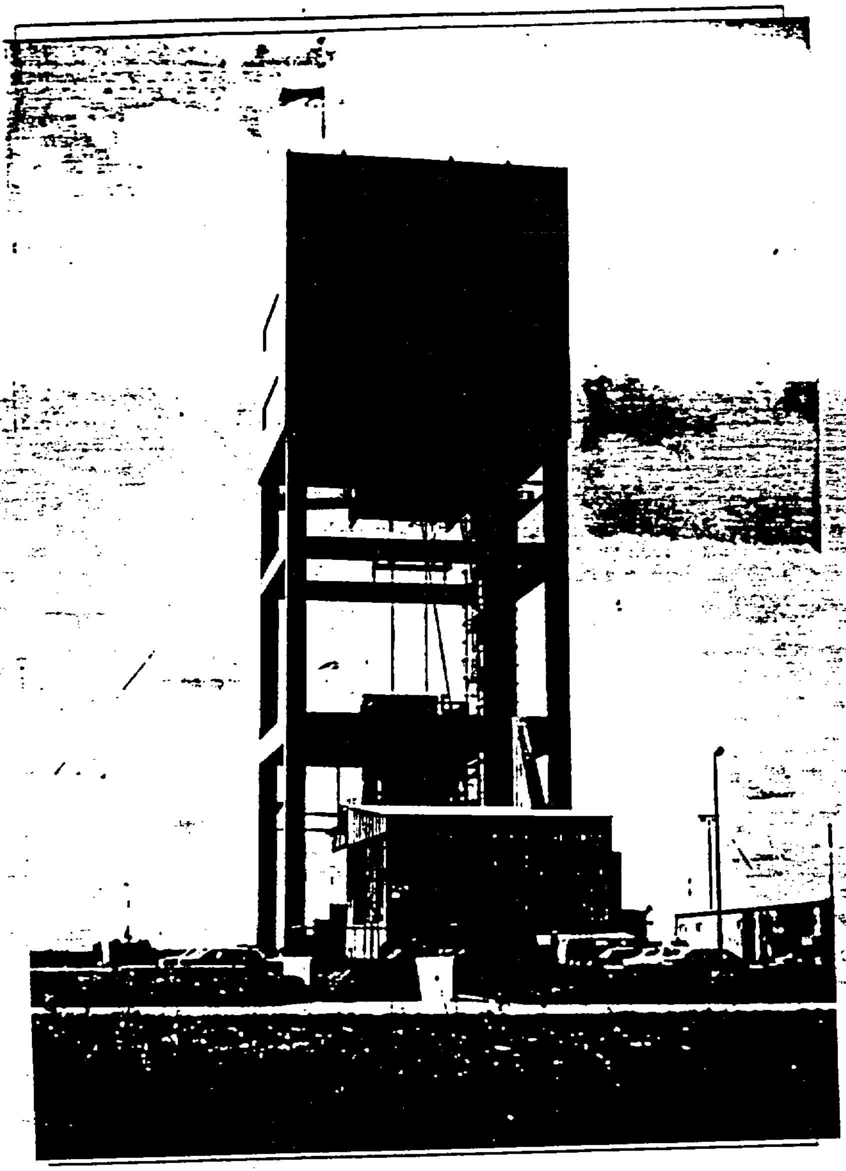
Aufbau Gefrierkeller , Schacht 1



Vorschachtteufen , Schacht 1

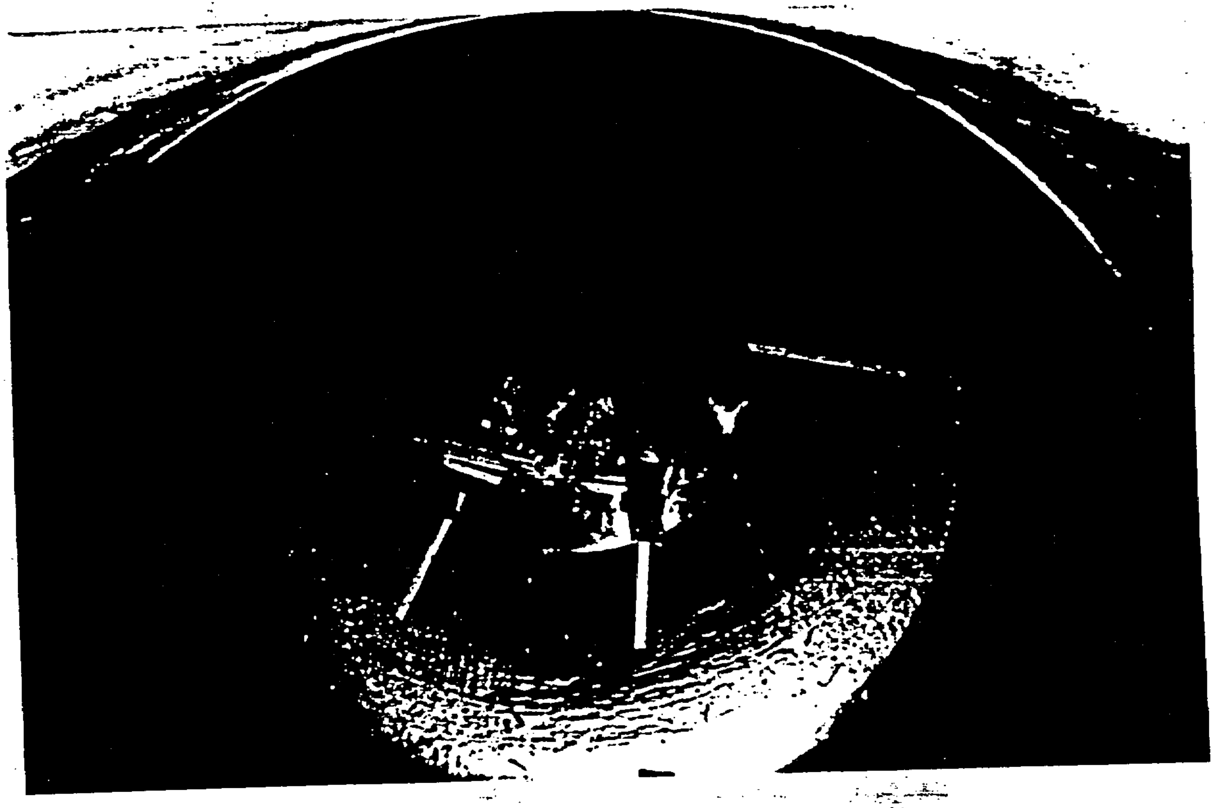


1. Kübel , Schacht 1

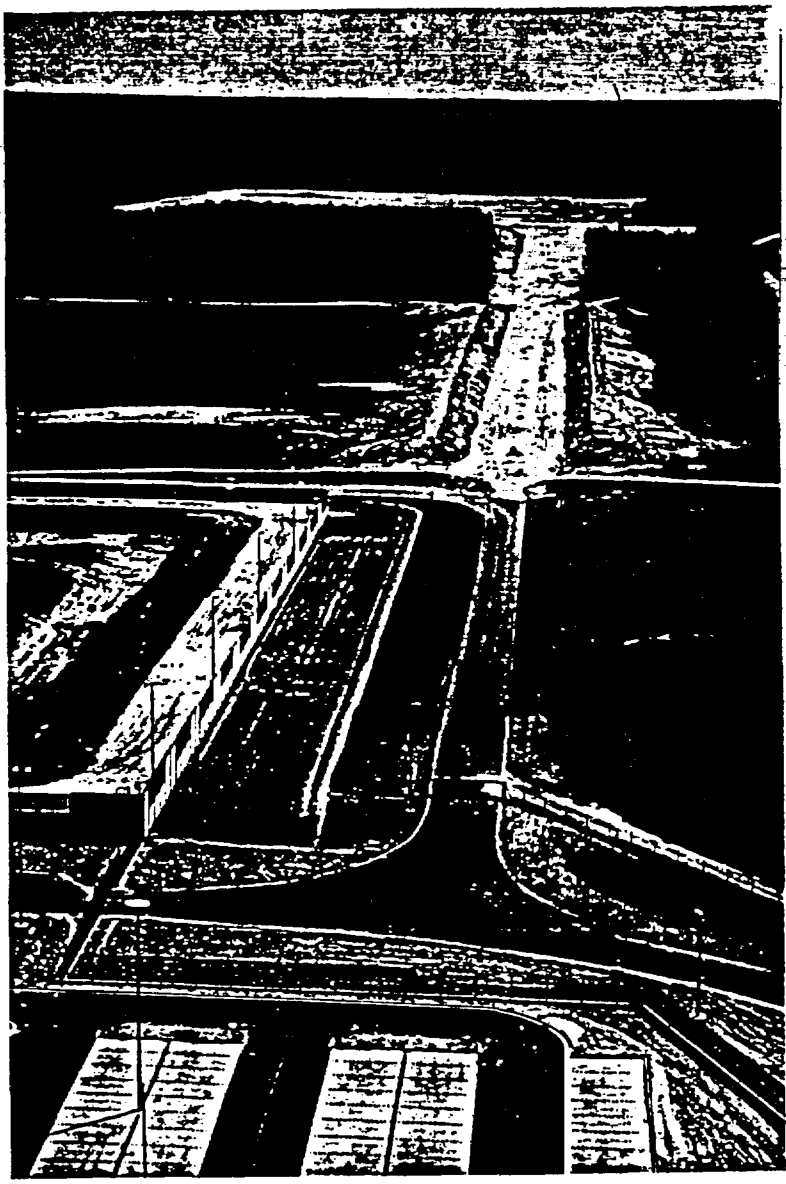


Schachtförderanlage Schacht 1

Abteufanlage Schacht 2

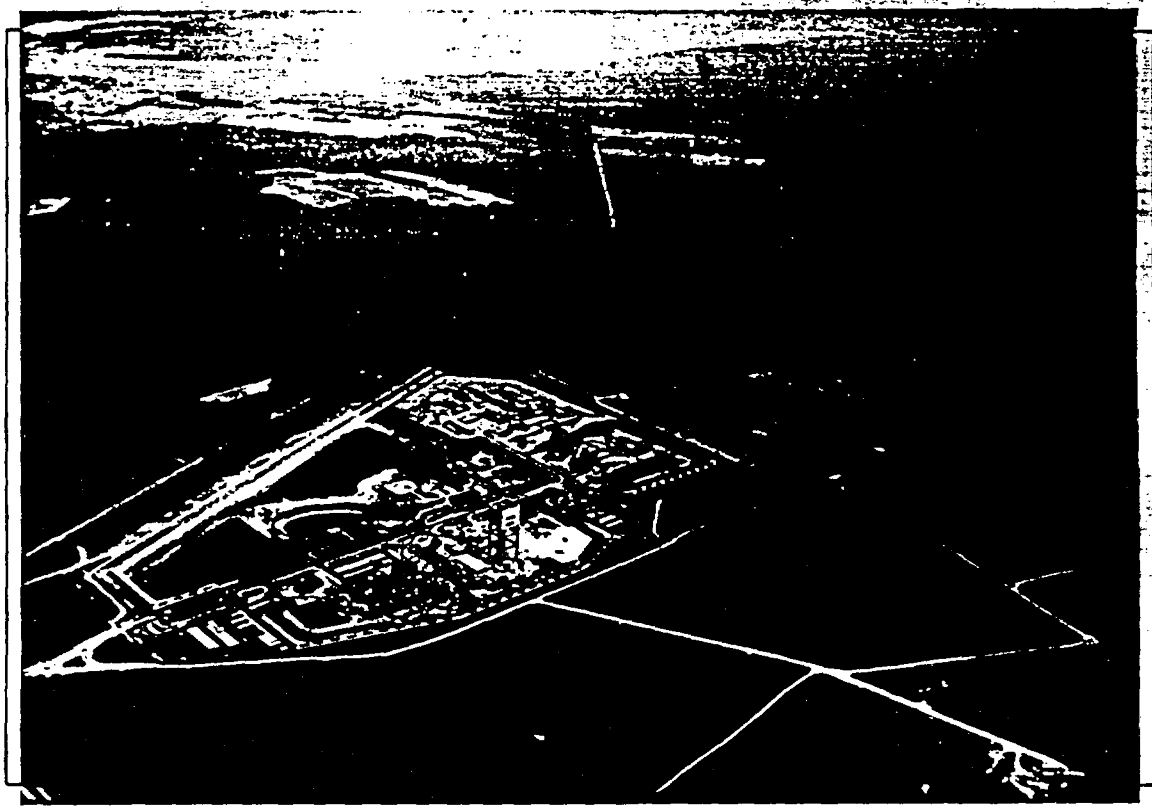


Einhängen der unteren Schwebebühnenetage



Blick von der Schachtförderanlage Schacht 1
auf die im Bau befindliche Salzhalde

Straßenbauarbeiten an der Salzhalde



Luftbildaufnahme vom April 1986 , Bergwerk zur Erkundung
des Salzstockes Gorleben ,
Bezirksregierung Braunschweig Freigabe Nr. 6039/235

GORLEBEN SITE VISIT
March 19, 1987

Photographs Numbered 3-27

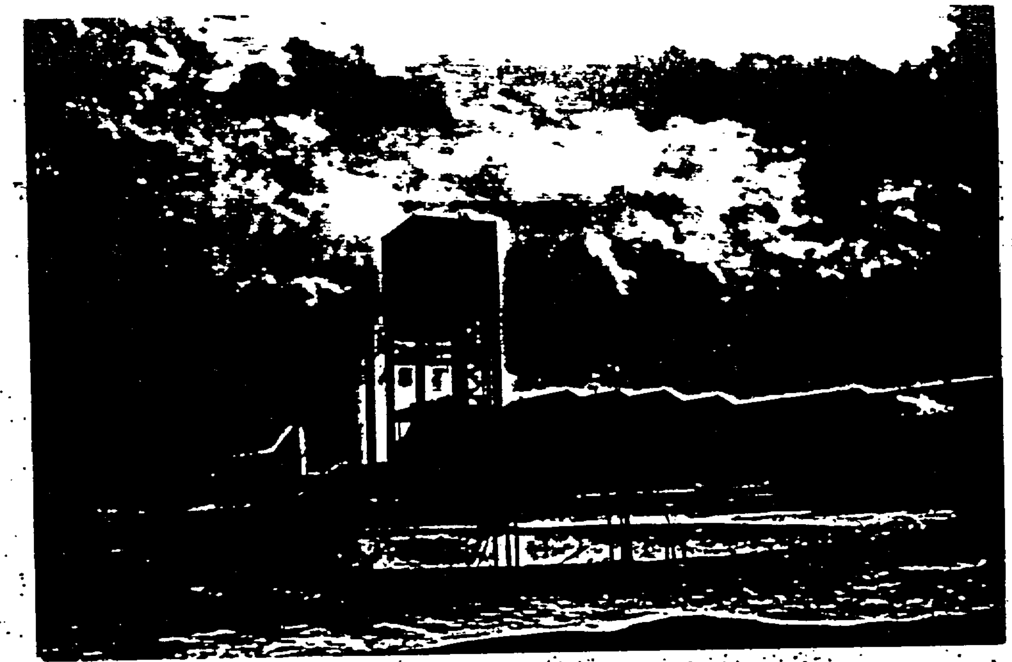
3. Head frame, Shaft #1, and office buildings.
4. Looking toward Shaft #2.
5. Head frame, Shaft #1, and office buildings.
6. Entrance to the Gorleben site.
7. Pump house; head frame, Shaft #2.
8. Shaft #1 with head frame, hoisting machine.
9. Shaft #1, with head frame, hoisting machine.
10. View of old trailers (now used by guard force) and Shaft #2.
11. Shaft #2, with head frame.
12. Near Shaft #2, cables and ventilation pipe.
13. Views looking down into Shaft #2; divisions indicate different lining structures. At this level, the separate structures are < 10 meters in height; at lower levels the heights range to 6 meters.
14. Views looking down into Shaft #2; divisions indicate different lining structures. At this level, the separate structures are < 10 meters in height; at lower levels the heights range to 6 meters.
15. View of top of Shaft #2, showing refrigeration lines.
16. Group of teachers touring Gorleben site, at Shaft #2.
17. Hoist at Shaft #2, not yet in operation. Dr. Schrimpf of DBE.
18. Refrigeration unit for Shaft #2.
19. Refrigeration unit for Shaft #2.
20. View of fence surrounding the Gorleben site.
21. Shaft #1, kennel for later use for air intake to the repository.
22. One of the grapples used to load up dump container in Shaft #1, Don Clark, and view looking back at Shaft #2.
23. One of two buckets used at Shaft #1 for personnel ingress and egress, as well as transfer of material.

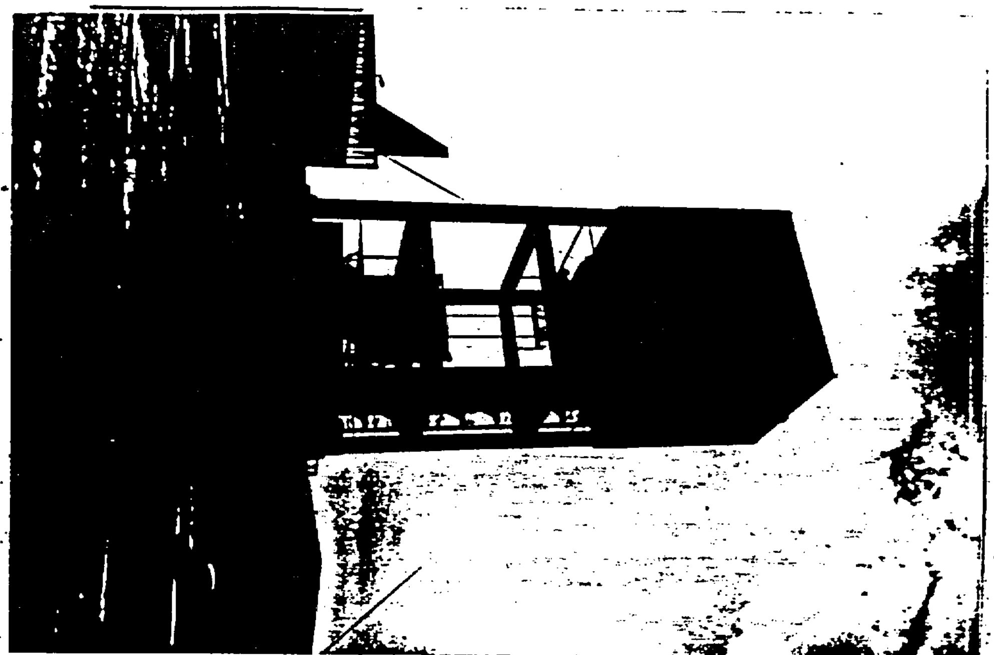
24. Bricks used for initial shaft lining at Gorleben.

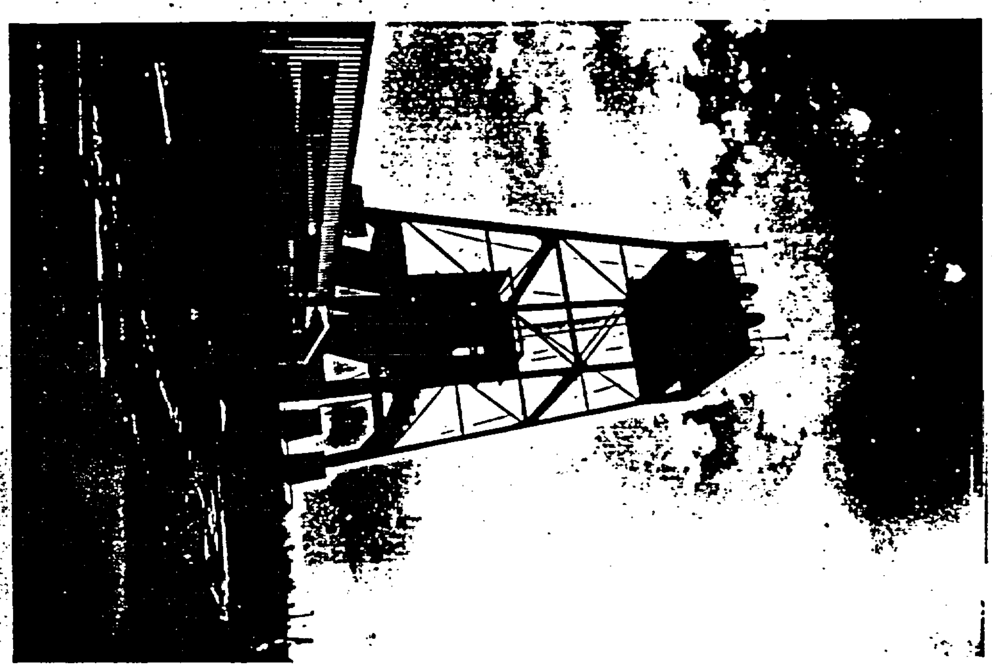
25. Hoisting machine and brake, Shaft #1.

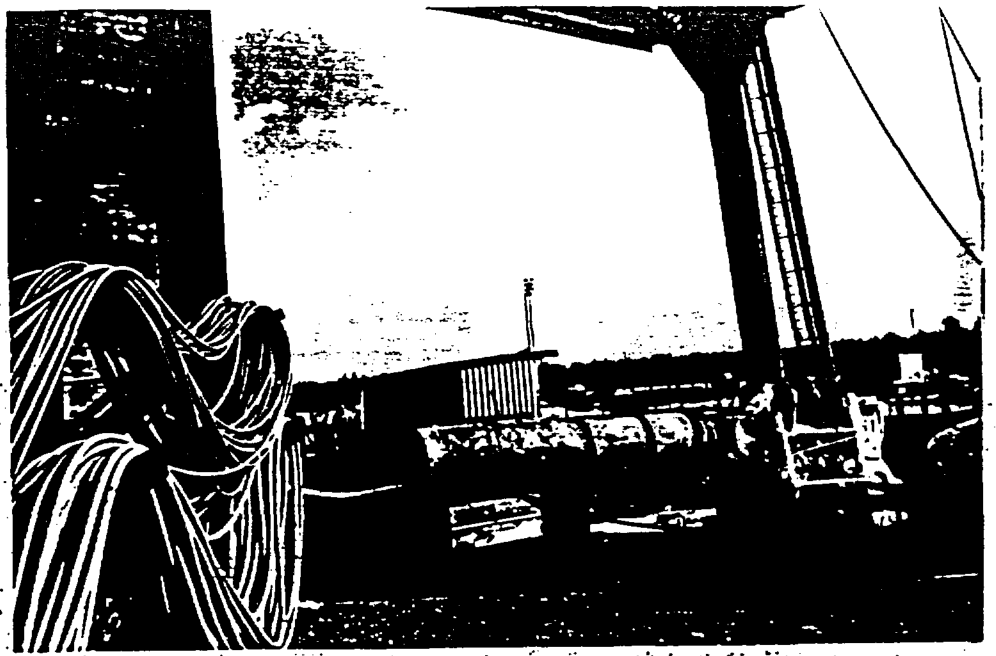
26. Exhibit at information center, Gartow (not far from Gorleben site).

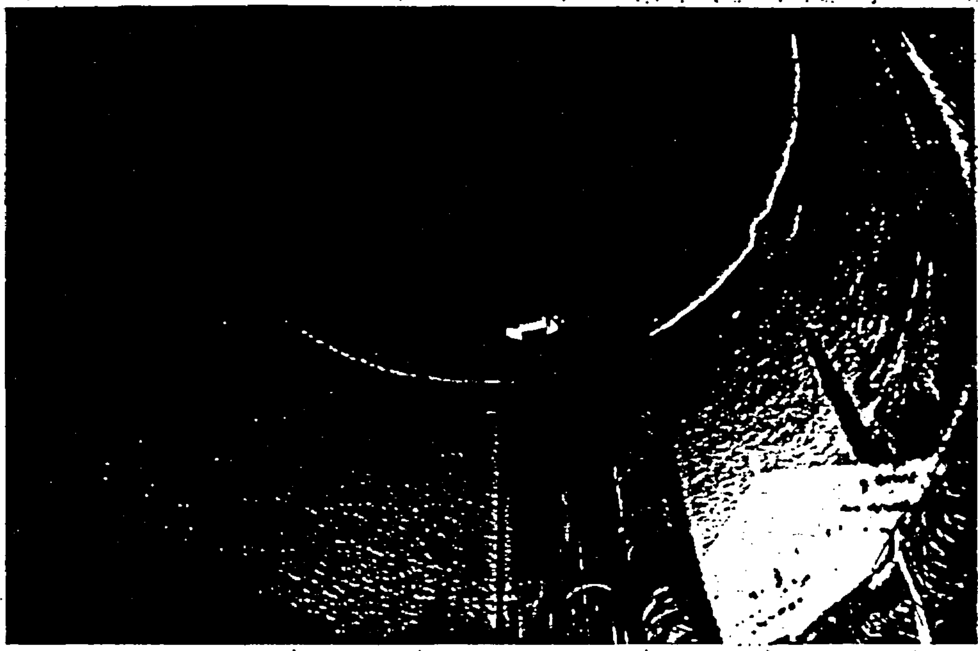
27. Large piece of slat from the Asse mine, at information center, Gartow, Don Clark.

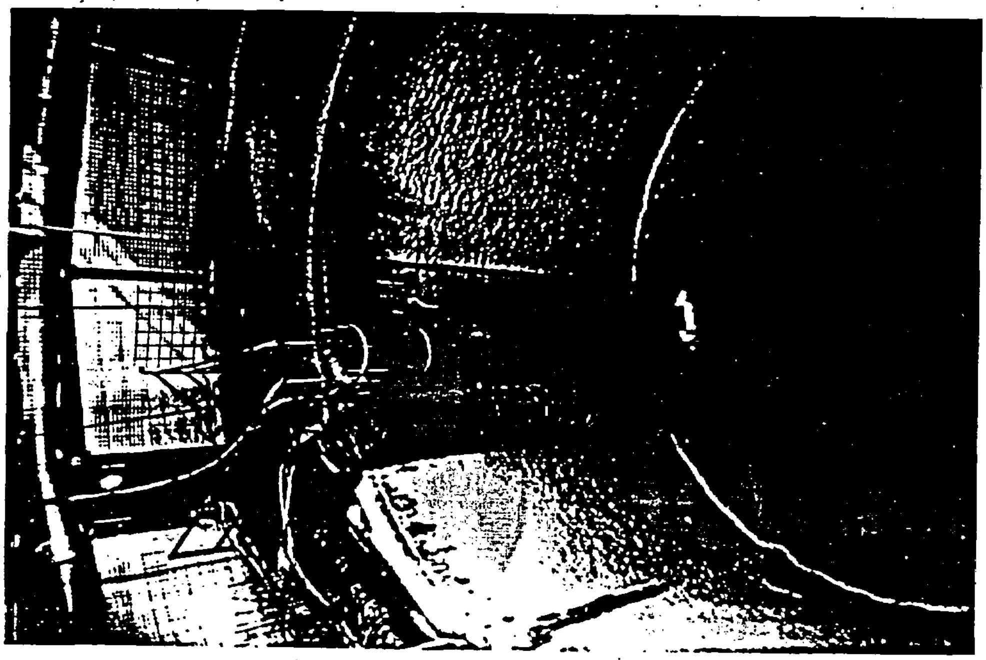


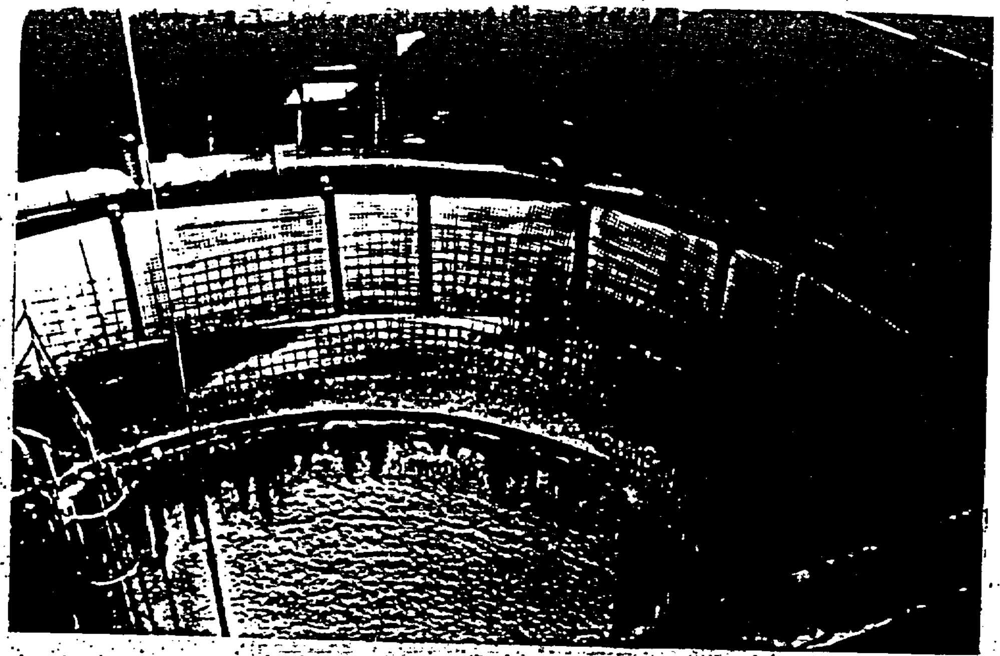


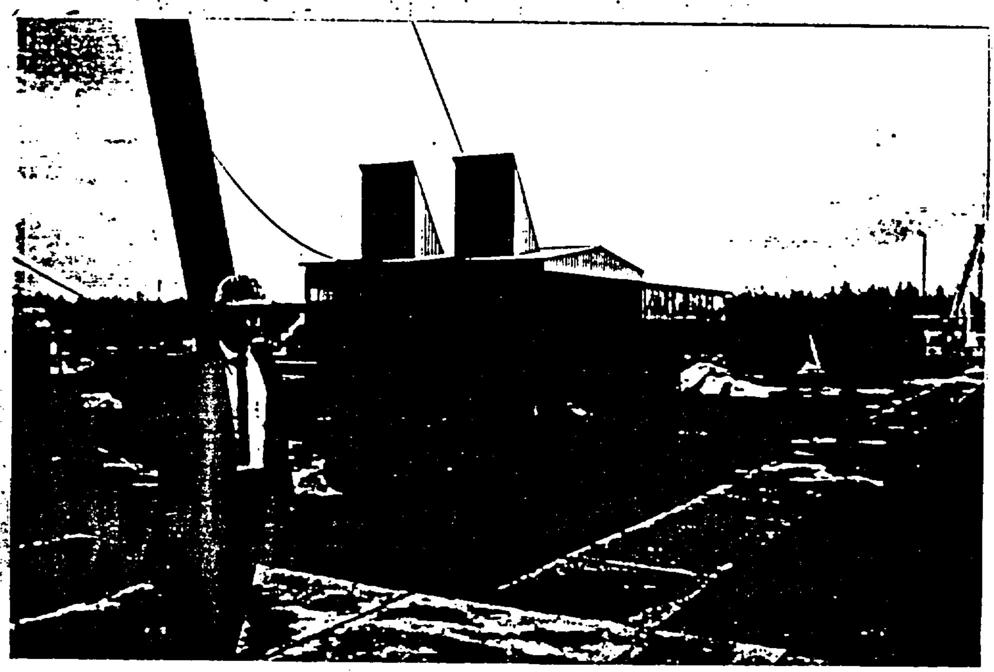


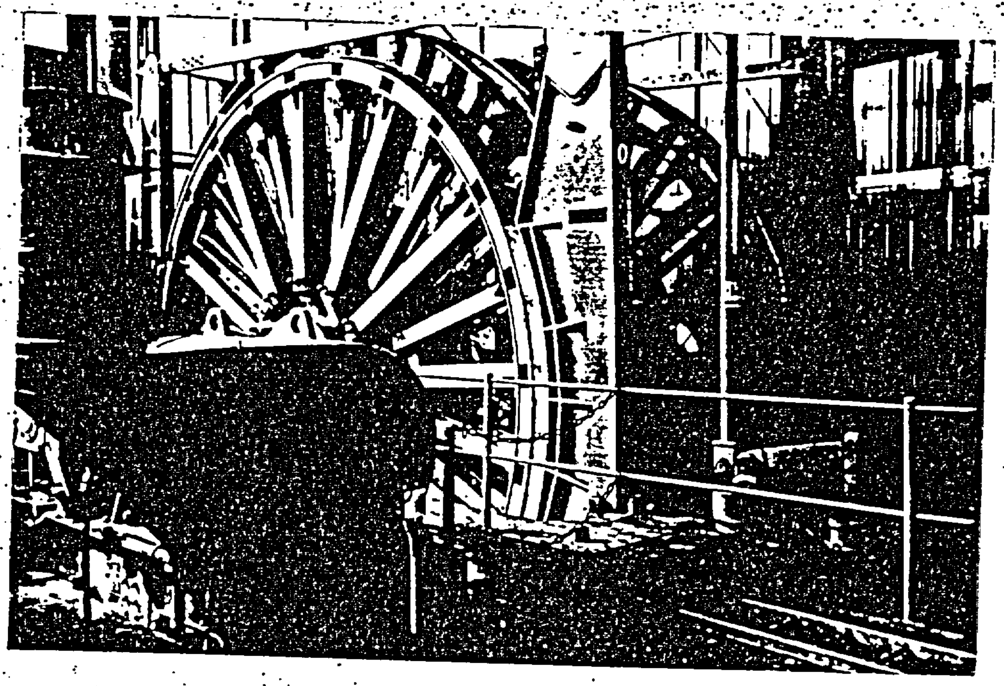


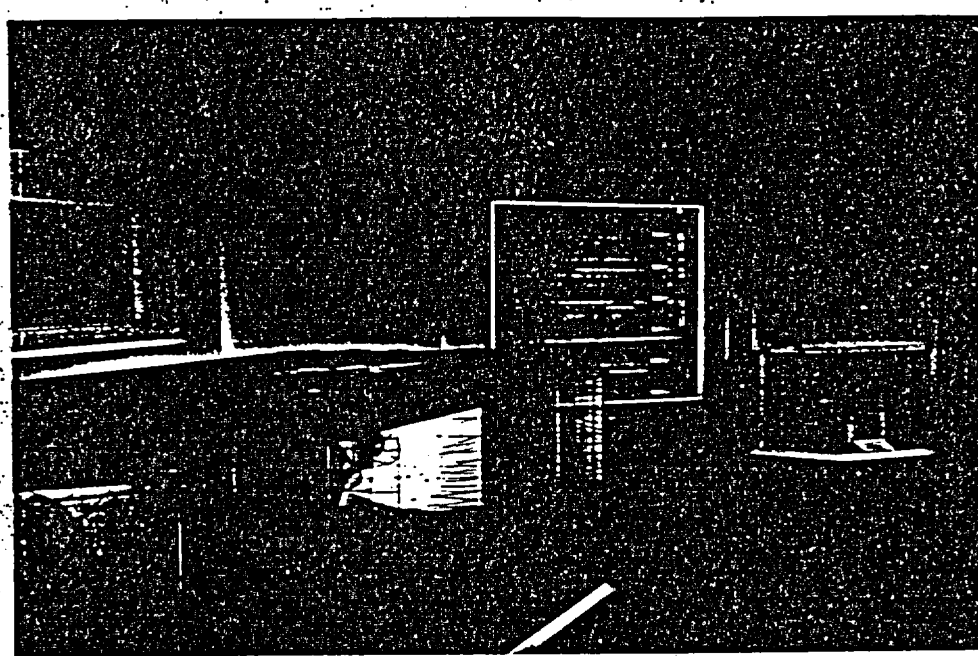












Attachment 3

The Asse Salt Mine

Research for the Final Disposal

Research for the Protection of Man and the Environment

Gesellschaft für Strahlen- und Umweltforschung München

Publisher:
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FORWORD

Of the 13 main research institutes in the Federal Republic of Germany (FRG), the Gesellschaft für Strahlen- und Umweltforschung (GSF) is the centre for environmental and health research. In accordance with objective of the GSF to carry out research into the protection of man and the environment, the topic of management of radioactive wastes became part of its programme as early as the beginning of the 1960's.
In the former salt mine Asse near Wolfenbüttel, scientists of the Institut für Tieflagerung of the GSF have been concerned with investigations on possibilities and risks of the final disposal of radioactive wastes in salt formations since 1965. During the 1970's the technical feasibility of the disposal concepts developed by GSF was demonstrated for low- and intermediate level wastes. These research activities are internationally considered to be a guideline and form part of the waste management concept of the Federal Republic of Germany.
Aim of the research into radioactive waste disposal is the long term safe isolation of radioactive materials from the biosphere. Main concern in this regard has always been the safety both of persons living in the vicinity of a repository as well as of the operating personell.
To accomodate public interest in its work, besides keeping both press and politicians constantly informed, GSF is also engaged in looking after a large number of visitors to the mine; world-wide the Asse salt men is the only research facility of its type which is open for visits by the general public. Every year more than 15 000 people take the opportunity of seeing the mine. With the aid of this publication GSF hopes, moreover, to contribure towards a better understanding of those interested in the topic of the final disposal of radioactive wastes.

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Origin of radioactive waste

Radioactive wastes and wastes contaminated with radioactive material arise from the operation of nuclear power plants and from the use of radioactive substances in medicine, research and industry. Because of the wide range of waste-producing processes, untreated waste arises in many different forms (e.g. contaminated clothing and equipment, cleaning materials, construction rubble, laboratory effluents, spent radioactive soureces, filters, exchange resins, fuel element hulls, highly active fission product solutions). The waste must be converted into a form suitable for final disposal by appropriate conditioning (processing, fixation, packaging). Preliminary processing of untreated waste basically consists of volume reduction by way of purification, compaction, incineration or evaporation. In order to obtain a solid endproduct which is easy to handle and package and is not or at most only very slightly soluble in water or brine, cement, bitumen, polystyrol or glass are used as solidification matrices. Finally, packaging in steel drums, redundant reinforced concrete shielding (VBA), cast steel canisters or high-grade steel containers provides safe containment of the waste during interim storage and transport to and within the repository. Selection of the conditioning process depends on the waste type and the disposal method envisaged.

Properties and classification of radioactive wastes

The main difference between radioactive wastes and other toxic materials (e.g. poisonous chemical wastes) is that the toxicity of the former is determined predominantly by the ionizing radiation emmitted by the waste nuclides and that this radioactivity decays spontaneously according to a natural physical law. The characteristic decay rate for individual nuclides is given by the half-life, i.e. the time required for half the original activity to decay away. The waste nuclides of significance for final disposal have half-lives ranging from a few to several tens of thousands of years.

Fig. 2:
Radioactive waste from a research laboratory

Fig. 3:
Handling of intermediate-level radioactive wastes with manipulators

Fig. 1: Medicinal application of radioaktive substances

Fig. 4.
Shielded canister for intermediate-level radioactive wastes

Radioactive wastes generally contain a mixture of short- and long-lived radionuclides of differing radiotoxicity. However, for handling, transport and interim storage of the conditioned wastes, the radiation emitted by the waste containers is of most significance. In addition, the content of long-lived, i. e. mainly alpha-containing, radionuclides and the nuclide concentration-dependent heat production are of decisive importance to disposal. In accordance with the recommendations of the International Atomic Energy Agency (IAEA), the following waste classes can be defined:

- Low-level radioactive wastes (LLW) can be handled without additional shielding. 0.002 Sv/h (200 mrem/h) is accepted as the upper limit for surface dose. The wastes are disposed of in chambers which can be entered by operating personnel.
- Intermediate-level radioactive wastes (ILW) require shielding or remote control handling because of their higher radiation levels. They are disposed of in chambers of boreholes which are not accessible to operating personnel.
- High-level radioactive wastes (HAW) produce such a high-level of radiogenic heat that heat conduction during transport and at the disposal site must be considered. Borehole disposal can produce temperatures of appr. 200° C in the salt.

The conceptes of low-, intermediate- and high-level radioactive wastes are often applied to untreated wastes, in which case delimitation between the categories is mainly on the basis of activity concentration.

To allow full description of waste products and assessment of disposal suitability, this simple waste classification must be supplemented by information on origin, type of waste, radionuclide inventory, solidification matrix and container properties. As the institution responsible for final disposal, the Physikalisch-Technische Bundesanstalt (PTB) keeps a record of all wastes arising in the FRG.

Annual waste arisings

Within the framework of long-term capacity planning for disposal of radioactive waste in the FRG, the PTB estimated the quantities and activities of waste to arise annually from various areas of utilization. An electrical energy production of 2500 GWa in the power plants during a 50 years operational period was assumed (see Table 1).

The greatest volume of waste comes from the power plants and consists mainly of low-level radioactive waste. Besides low- and intermediate-level radioactive wastes reprocessing of spent fuel also produces a relatively small proportion of high-level radioactive waste which contains by far the largest activity proportion of all radioactive wastes.

Fig. 5:
Interim of low-level radioactive wastes at collection points

Waste origin		Waste volumes		Activity	
		m^3	%	Bq	%
Reprocessing of spent fuel[1]:					
High-level radioactive waste (HLW)		150	0,5	1,7.10^{19}	80,40
Low- and intermeadiate-level radioactive waste (LLW + ILW)		12 350	38,5	0,4.10^{19}	19,59
Nuclear power plants	LLW + ILW only	16 200	50,4	2,1.10^{15}	0,01
Major research		2 210	6,8	8,1.10^{13}	
Industry		563	1,7	1,0.10^{15}	
National collection points		731	2,3	3,4.10^{13}	

Table 1:
Annual waste arisings in the Federal Republic of Germany (PTB aktuell, number 11).

[1] A reprocessing plant of this size will come into operation in the FRG only as from the mid-1990's.

Legal background to final disposal

In the federal Republic of Germany the Atomic Act forms the legal basis for handling of radioactive materials and disposal of radioactive wastes. According to the Act, radioactive residues must either be exploited harmlessly or, if this proves impossible or uneconomic on a scientific/technological basis, disposed of appropriately as radioactive aste. There is a basic duty to deliver radioactive waste and the Federal States therefore have to provide collection points for interim storage (see Fig. 5) while the Federal Government has to provide installations for the disposal of the wastes. The Federal Government's responsibilities are taken care of by the PTB which is, in turn, assisted by the Deutsche Gesellschaft zum Bau und Betrieb von Endlagern für Abfallstoffe mbH (DBE).

The fourth amendment (1976) to Atomic Act provides für a so-called "Planfeststellungsverfahren" as a licensing procedure for final disposal of radioactive wastes. This means that all officials whose area of responsiblility is affected by plans for disposal, all communities and citizens affected are involved in the procedure.

Disposal of low- and intermediate-level radioactive wastes in the Asse salt mine up to the end of 1978 was authorized on the basis of the legal situation prior to 1976 (first Radiation Protection Regulation in conjunction with the Atomic Act).

Salt has the best general prerequisites

For various reasons, many countries involed in the problem of radioactive waste management look upon disposal in deep geological formations as the safest solution. Particularly when considering the long-term isolation of long-lived and high-level wastes from the biosphere, a multiple barrier concept with independently functioning natural and engineered barriers comes into consideration. The wast matrix, the canister, backfilling of repository voids and the surrounding rock formations are to prevent leaching, release and transport of radionuclides. The most important component of the safety

The salt structures of nortwest Germany

■ 0–400 m

■ deeper than 400 m

Fig. 6: Location of salt formations in northwest Germany (due to Jaritz, BGR)

system is the geological barrier consisting of the actual host rock and its overburden, both of which must be tectonically stable and have a minimum of waterbearing strata. The demands on the barriers depend on the quantity and radiotoxocity of the waste to be emplaced. In each case, the effectiveness of the overall barrier system must be confirmed by a safety analysis based on formation- and site-specific data.

Basically, a whole series of rock types is suitable for waste disposal, with local conditions being of additional significance in each case. According to their availability in the countries working in the field of waste disposal, rock salt, granite, clay etc. are being investigated. In the first place, this involves acquisition of data for safety analyses, development of suitable disposal techniques and demonstration of the technical feasibility of disposal. Research mines and rock laboratories have therefore been (and are currently being) set up at various locations worldwide for the purpose of carrying out in-situ experiments with a view to solving important geological, technical and nuclear-related problems. Internationally, the Asse mine occupies a leading position as a pilot facility for specific investigations in rock salt.

The considerable advantages of salt as a host rock for radioactive waste are

- good plasticity leading to sealing of fissures and cracks and, on the long term, to sealing of mined cavities,
- high heat conductivity which, in comparison with other rocks, allows more extensive use of the repository for heat-producing waste,

Formation Country	Salt	Anhydrite	Granite	Clay	Sedimentary rock	Basalt	Tuff
Belgium				X			
Denmak	X						
FRG	X		X		X		
GDR	X						
England		X	X	X			
Finland			X				
France	X		X		X		
India			X				
Italy				X			
Japan			X				
Canada			X				
Netherlands	X						
Sweden			X				
Switzerland		X	X		X		
USSR	X						
USA	X		X			X	X

Table 2:
Summary of host rocks under investigation in different countries

- extremely high impermeability to water inflow, provided that the salt body is homogeneous,
- satisfactory performance during mining operations, which allows construction of large disposal chambers.

Not only these properties, but also those which can have a detrimental effect on repository safety, are being investigated at Asse. The latter include, inter alia, the presence of inhomogeneities, high water solubility, low radionuclide retardation capacity and the corrosive effect of salt solutions on waste canisters. On the other hand, with more than 200 salt domes in the FRG, there is no shortage of raw materials.

For several years, an international project in the disused iron ore mine at Stripa in central Sweden has been testing the suitability of granite as a non-saline host rock. Similar work is going on in the Climax mine (USA) and a further underground laboratory in granite is presently being constructed in Canada. At the end of 1983, the Grimsel test site in Switzerland came into operation. Here, the National Cooperative for the Storage of Radioactive Waste (Nagra) is working together with the Bundesanstalt für Geowissenschaften und Rohstoffe (BGR) and the GSF on investigation of the water-bearing capacity of fissures and the mechanical effects of a temperature increase in granite.

Work on disposal in clay is well advanced in Belgium. Near Mol, a 220 m deep experimental shaft with a test section has been sunk to allow the hydrogeological and rock

Fig. 7:
Tunnelling of a new section in salt

mechanical properties of the clay (Tertiary Boom clay) to be investigated.

Mention should be made in this connection of the research carried out by the GSF in the disused Konrad iron ore mine at Salzgitter, where extensive clay formations isolate the disposal formation from water inflow.

This survey shows that research into geological disposal in being carried

Fig. 8
Upfolding of a salt formation in the course of geological time

out for a range of different host rocks in several underground laboratories. The GSF Institut für Tieflagerung (IfT) has the longest tradition and widest experience in this area with its work in the Asse salt mine. The results obtained here are also of significance for development of disposal concepts in other countries, particulary the USA and the Netherlands.

The Asse salt mine

Geological and hydrogeological conditions of the Asse

The Asse is a range of hills of approximately 8 km lenght in the northern Harzforeland. In the course of geological time, the Zechstein salt deposited on the ocean floor some 240 million years ago migrated, due to its plasticity and low density, into the core of the Asse structure. The Zechstein salt pierced upwards through the falt-lying Lower and Middle Buntsandstein, merged with the salt of the Upper Buntsandstein of the southwest flank and pushed the overburden into a steeply dipping position.

The actual salt anticline is made up primarily of salt formations of the Stassfurt and Leine series. In the vicinity of the Asse salt mine, the inner structure of the salt body consists of a main culmination the acis of which dippis evenly towards the southeast, and a southern satellite culmination which is separated from the main culmination by a syncline.

The flat-lying base of Zechstein formation lies some 2200 m below the earth's surface and deep drilling has revealed that sandy sediments of the Rotliegend underlie the formation.

The sedimentary cover of the Asse salt culmination comprises Triassic strata of the Buntsandstein, the Muschelkalk and the Keuper, with sporadic occurrences of Quaternary unconsolidated deposits. It is made up of an interstratification of almost impermeable clays and argillaceous silts, low permeability silts and marls and, in the outermost flank areas, permeable sand- and limestones. The more recent salt strata in the flanks have been leached near the surface only, creating isolated local water flow paths. At depth, they have remained intact, thus proving the barrier effect of the aquicludes.

Deepdrilling Remlingen   Shaft   Shaft 2

Cretaceous
Jurassic
Keuper
Muschelkalk
Halite in the Middle Muschelkalk
Upper Buntsandstein (Röt)
Halite in the Buntsandstein (Röt)
Middle / Upper Buntsandstein
Carnallitite
Zechstein salt series
Zechstein basis series
Rotliegendes

Fig. 9
Geological section through the Asse range

Groundwater movement occurs mainly in a structure-parallel north-west direction in the various water channels. A crosswise movement through the strata of the overburden is possible only at geologically faulted zones on the southwest flank.
Pumping tests in hydrogeological test boreholes rendered only negligible amounts of water in the adjoining rocks, which confirms the low water conductivity of the overburden. Lowering of the groundwater level was not noticeable in the neighbouring boreholes of the collapsed overburden and no coherent groundwater table could be detected.
At several positions a fairly close range, saturated salt solutions (brines) were discovered or drilled in the Asse mine. Their constant temperature, chemical composition, limited availability and pressure conditions in boreholes show that the are only isolated occurrences within the salt body. There are therefore no links to ground water-bearing strata in the overburden.
The strata series encountered in the salt give evidence that over geological time periods no water was able to reach the interior of the salt formation. Together with the many extensive argillaceous aquicludes in the overburden, the salt sequence forms an effective barrier system between the Asse mine and water-bearing strata.

Fig. 10:
The Asse research mine alt Remlingen

The history of the Asse salt mine

Salt mining began at Asse with the sinking of the Asse 1 shaft at Wittmar betWeen 1899 and 1901. In order to mine the potash salt lying below the apex of the salt culmination, several different levels were constructed. Due to inappropriate stoping in the uppermost section of the salt anticline and in ignorance of the hydrogeological conditions, water inflow occured in 1906, as a result of which the shaft was flooded.

After abandonment of the first shaft, the Asse 2 shaft was sunk to a depth of 765 m, some 1.5 km to the east at Remlingen between 1906 and 1908. Initially, potash salt was mined exclusively, but under proper conditions and very deep in the salt core. In 1916 mining of rock salt began and this was also carried out at all times with apporpriate safety precautions against hydrological hazards. Because of a crisis in the German potash industry after the first world war, mining of potash salt was discontinued at the end of 1925 and all future mining was restricted to rock salt. This was discontinued for economic reasons on 31st March 1964 and, since then, no exploitation activities whatsoever have been carried on in the Asse salt formation.

To equip the mine with two independent shafts leading to the surface, work began on the sinking of the Asse 3 shaft at Klein-Vahlberg in 1911, i.e. shortly after commencement of potash salt mining in the Asse 2 shaft. Due to unforesseen difficulties in sinking the shaft and delays caused by the first wold war, the work was not completed until 1921, with a shaft depoth of 725 m. Because of the already mentioned crisis in the potash industry, mining never actually commenced in this shaft and, after its closure in 1924, so much water collected in the shaft due to lack of maintenance that today the water level is some 9 m below the shaft cover.

In 1965 the GSF aquired the former Asse salt mine on behalf of the FRG in order to carry out research and development work with a view to safe disposal of radioactive waste.

The mine and its technical facilities

The Asse salt mine consists of a vertical area between 490 and 850 m depth, divided into 16 levels. At present, the lower area of the salt structure is being mined to the planned level of 930 m.

Besides mineral dump rises, personnel shafts for the salt mining activities which served as vertical links for supply, transport and mine ventilation. Today these are used mainly for ventilation.

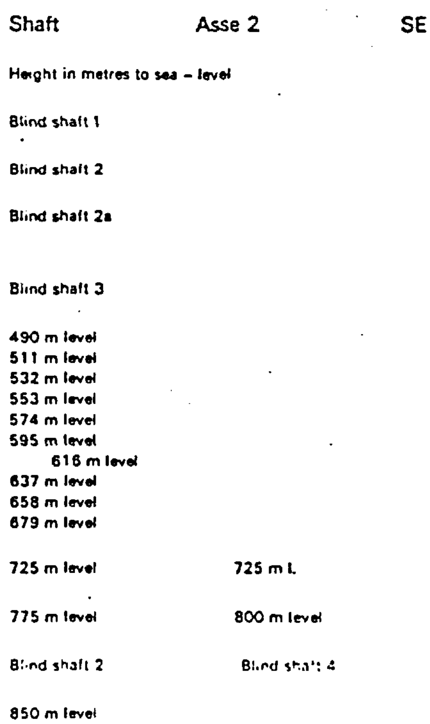


Fig. 11:
Longitudinal section through the mine

The cavities left from the mining of potash salt have been completely backfilled, but the 142 chambers from rock salt mining (with a total cavity volume of around 3.5 million m^3). A standard chamber is 60 m long, 40 m wide and 15 m high. Between adjacent chambers there are 12.5 m thick pillars and, between chambers arranged one above the other, there are 6 m thick stopes.
After conversion of the mine for experimental emplacement of radioactive waste between 1965 and 1967, a spiral section (average gallery incline 10 %) was constructed for transportation between the 490 and 800 m levels; in 1986 this was extended to the 930 m level. While the mining cavities lie mainly above 750 m, the test areas are mainly below this level.
The salt from construction of new test areas and tunnelling of new drifts is removed to the mining cavities by front end loaders or with a pneumatic conveyor.
A drilled shaft (shaft Asse 4) with a utilizable diameter of 1.5 m was sunk to allow further development of the intermediate level waste emplacement procedure and development of new emplacement methods. The shaft ends in a cavern with 10,000 m^3 volume at a depth of around 1000 m. Above the 750 m level the shaft also serves as a second independent exit from the mine.
Shaft Asse 2 is the most important installation as regards the utilization aspect of the mine. From the overburden into the salt over a lenght of some 410 m, the shaft tube is equipped with tubbing. In 1969 this cast iron shaft lining was additionally strengthened to a depth of 320 m with a reinforced concrete liner with bitumen backfill. The electrically operated hoist in the shaft can handle loads up to 10 t.

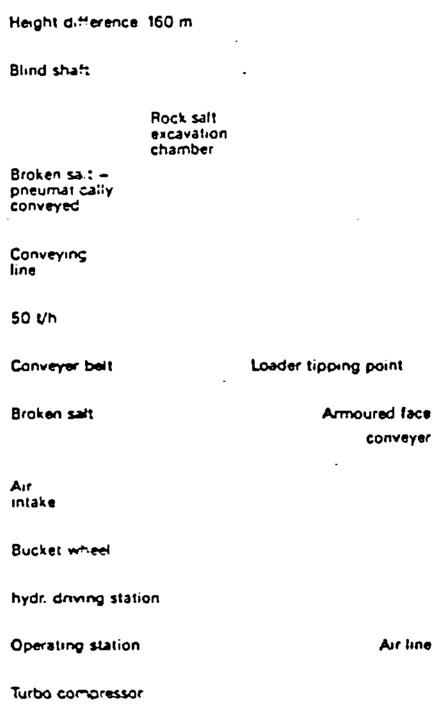


Fig. 12:
Principle of the pneumatic conveying plant for salt from deep excavation

A large number of diesel or electrically operated vehicles and machines is in use underground in connection with the many aspects of research and development, mine maintenance, operational safety and conveyance of visitors.

Radiatior protection and environmental surveillance

The same guidelines and protection regulations apply to both emplacement of radioactive waste in a mine and actual use of radioactive materials. Thus the dose to personnel and the activity concentration in the mine air are measured regularly. In addition, many surveillance measurements are carried out on the local dose rate and possible contamination. To date neither the dose limits for persons exposed in the course of employment nor permissible activity concentrations in the mine atmosphere have been exceeded.
As all radioactive waste is in a solid dry form or fixed in bonding agents and packed in stable canisters, release of significant amounts of radioactive material is not to be expected. Regular checking of the exit air from the mine showed only the smallest amounts of tritium, carbon-14 and successive products of the noble gas radon. From the beginning of emplacement, soil and vegetation samples have been investigated for their radioactive material content. In this case also no values above the level of natural background radioactivity were determined.

Low level waste (LLW) – no technical problem

In 1967 during the period of reconstruction of the Asse salt mine, low level waste was already being emplaced for experimental purposes. According to the licence issued by the relevant officials, the maximum dose at the canister surface was not to exceed 2 mSv/h (200 mrem/h).
Each canister being emplaced had to be labelled and indexed. Immediately upon arrival above ground, the integrity of each container had to be checked using a wipe test, this being repeated upon arrival in the underground disposal chamber.
Over a period of two years at least 50% of the accessible canisters had to be examined at intervals using the same wipe test.
The latter requirement in particular meant that the containers had to be stacked standing vertically in several layers on top of one another with access paths being left open between the rows of containers to allow the test to be carried out. No leakage as a result of emplacement in the mine could be detected.
In later emplacement phases, the wipe test was no longer made compulsory by the licensing authorities but stricter regulations were placed on the canister content which had to fulfil the following requirements:

- It was not to be capable of biodegradation,
- no violent chemical reactions were to be expected,
- it was not to cause corrosion of the canister from the inside,
- it was to be free of volatile nuclides and spontaneously combustible materials.

In subsequent emplacement phases, the containers were stacked on theur sides rather than being placed on top of one another.
The experience acquired in the early emplacement phases was compiled in the GSF's "Emplacement conditions for low active waste in the Asse salt mine"; these had to be observed by waste producers.
The severally revised emplacement conditions were valid up to the end of the test disposal and expiry of the last licence in 1978. They were mainly concerned with the type and properties of low level waste to be disposed of, permissible activity levels and dose rates and waste processing and packaging.
The introduction of these emplacement conditions brought regulations on the use of particular canister types and fixation of the wastes in the canister, for example with bitumen or cement. It also enabled introduction of a new emplacement technique in the form of dumping. Waste containers loaded on a frontend loader were tipped over an incline into the disposal chamber and the covered with loose salt. Canister damage leading to release of radioactivity could be excluded by adhering to the emplacement conditions.

This new technique was used up to the end of experimental emplacement in 1978 and led to considerable reduction in individual doses due to the short time spent by operating personnel in the vicinity of the waste containers and by the large distance between the loaded shovel and the driving position. It also led to an increase in disposal capacity and was therefore more economical. Between 1967 and 1978 a total of around 125,000 canisters containing low level waste were emplaced without any notable technical problems, and the technique was development to a fully operational standard. This improved disposal method was adopted by the PTB in its plans for a federal repository. Since expiry of the licence, test disposal using radioactive waste is no loger possible because, in terms of the revised Atomic Act, a so-called "Planfeststellungsverfahren" would be necessary.

Fig. 13: The stacking technique used initially for low active waste

Fig. 14: ... the dumping technique was a first optimization

Fig. 15: Backfilling of emplaced waste containers with salt

Fig. 16: De: very of intermediate level waste to the mine in a shielded collective container

A technique for intermediate level wastes (ILW) is developed

The properties, pre-treatment and packaging of intermediate level wastes are similar to those of low active waste but handling of the waste canisters is possible only with additional shielding and remote control because of the higher activity inventory (up to 10^{13} Bq per 200-l-canister) and the higher dose rate (up to 10^2 Gy/h) at the canister surface. For this reason it is not permissible for personnel to enter the emplacement chambers and a completely new underground manipulation and emplacement technique had to be developed and tested. The technique is shown in Fig. 17. The former excavation chamber 8 a at the 511 m level was converted into a disposal chamber, and the only connection between this chamber and the rest of the mine was sealed off with an 80 cm thick concrete shielding wall with a lead glass window. A supply chamber housing all technical facilities for conveying the waste to the disposal chamber below was constructed at the 490 m level. The two chambers are separated by a 6 m thick salt stope which is penetrated by several shielding boreholes.

Intermediate level waste was originally transported to Asse in individual shielded canisters and later in shielded collective containers. Because of the 10 t weight restriction on the conveyor in the Asse 2 shaft, the waste canister was transferred frome the vehicle onto the radiation protection slider of the transport borehole. When the slider was opened, the waste drum was lowered from the shielded transport canister into the chamber below. The canister grab was released from the waste drum electromagnetically and then withdrawn into the shielding container, allowing the radiation protection slider slider to be closed.

The heavy shielded containers, the 6 m thick rock salt cover and the general thickness of the chamber walls provide sufficient radiation protection. The dose to emplacement personnel always remained below the dosimeter limit of 4×10^{-4} Sv/month (40 mrem/month).

Fig. 17:
The emplacement technique for intermediate level waste in chambers

6 m 490 m level 511 m level

The technical facilities were such that during the intermediate level waste transport and emplacement procedure in the mine the required level of radiation protection was maintained at all times.
To ensure that no radioactive dusts or aerosols could escape from the disposal chamber, a ventilation plant was installed to extract air from the chamber and filter it prior to mixing it again with the ventilation air of the mine.
"Conditions for experimental emplacement of intermediate level wastes" were also formulated for this waste category and had to be observed by waste producers.
Between 1972 and 1978 around 1300 intermediate level waste containing canisters were emplaced with no appreciable technical problems in the course of the test disposal. The concept has proved itself completely in practice and has also been adopted by the PTB in its plans for a federal repository.

Borehole disposal of higher activit categories of intermediate level wastes

Some intermediate level waste at the upper activity limit of this category has a low heat production. It consists mainly of hulls and structural parts of fuel elements and feed clarification sludges, all of which arise from the reprocessing of spent fuel. These waste materials have to be solidified in concrete, packed in stable steel drums and emplaced in suitably dimensioned boreholes.

As part of its research and development program, the GSF is working in cooperation with the Jülich nuclear research centre on plans for a test disposal in the Asse which involes, retrievable emplacement of wastes of this category in boreholes.
The general aim of the intermediate level waste test disposal is to develop suitable techniques for disposal of these waste types in boreholes in salt, to test prototypes and to demonstrate safety.

The focal point of the work consists of:

- developing equipment for construction of the deep, large diameter boreholes,
- developing and testing borehole seals which prevent infiltration of brines and escape of radioactive gases, as well as providing radiation shielding,
- investigation of the interactions between waste containers and rock salt,
- determination of radiation dose to personnel as a basis for respective optimization of disposal technique.

Fig. 18:
Control desk with monitor above emplacement chamber 8 a

Fig. 19:
View through the lead glass window into the disposal chamber

Disposal and in-situ solidification of ILW-LLW in underground caverns

It is also a task of the Institut für Tieflagerung of the GSF to develop and test supplementary or alternative techniques not yet included in the plans of the PTB for a federal repository. A corresponding example is in-situ solidification of selected waste types.

Analysis of work performed to date has shown that direct emplacement of low and intermediate level wastes without canisters has advantages from the point of view of safety, technical feasibility and costs.

In 1976 work was begun with the Karlsruhe nuclear research centre, NUKEM GmbH, Hanau and the company Gatys, Neu-Isenburg on a first project phase aimed at demonstrating basic feasibility and defining a technical reference concept. After successful completion of this phase, the second phase involving experimental testing of important technical components was begun in 1978. The third phase concerned with preparing and carrying out the test a fully operational scale begun in 1982.

The main features of in-situ solidification are:

- direct emplacement of pre-treated granulated LLW-ILW from surface proportioning, maxing, and conveying plants into an underground cavern, using a vertical conveying line,
- in-situ solidification of the product in the cavern to form a quasi-monolithic block.

The following steps are carried out above ground:

1. Granulation acording to specifications of the untreated LLW-ILW (slurries, powders etc.) at the site of origin. The granulate has a grain size range of 0.3 to 5 mm and must be allowed to harden for 7 to 10 days during which time the heat of hydration disperses.
2. Transport of the granulate from the waste producer to the cavern site in special containers with loading and unloading devices.
3. Unloading of the granulate at the above ground mixing plant where a liquid mixture of equal proportions of granulate and cement paste is produced. Besides water, tritiated waste water with appr. 5.5×10^{12} Bq/m^3 (150 Ci (H3)/M^3) can be used for making the cement mixture.
4. Underground operations consist of:
   Vertical transport of the mixture from the suface through a fairly narrow pipe (50 – 60 mm diameter) into a salt cavern (1000 m below the surface) with a volume of 75,000 m^3. Supply is purely by gravity which, with a flow rate of appr. 0.5 m/s, balances the frictional resistance in the pipe.
5. The mixture spreads out in layers on the cavern floor and solidifies in-situ, i. e. at the disposal site. Each layer corresponds to an emplacement phase.

Fig. 23 gives a schematic representation of the cavern field concept. The disposal chambers can be prepared by drilling or blasting, by two-slice system or by a leaching procedure.

Fig. 20:
View into a salt cavern

Fig. 21:
Schematic representation of a cavern

Fig. 22:
Schematic representation of the in – situ concept

ILW
LLW

Granulate production

T – Effluent

Feed tank

Cement

Mixer

Waste gas

Filter stack

Overburden

Salt

Borehole lining

Supply pipe

In – situ – product

Overburden Salt 300 m

Borehole seal Supply borehole

Main shaft

Backfilled cavern

Dam Cavern in operation

Cavern under construction

by overhead stoping

222 m 900 m 170 m

980 m 1000 m

Fig. 23:
Planning of a cavern site (KfK / GSF)

Important results from research into disposal of high level wastes (HLW)

High level waste consists of concentrated fission product solutions from reprocessing of spent nuclear fuel. Compared to low and intermediate level waste it has a much higher activity level which can reach 10^{13} Bq per litre. In order to achieve high solidification before disposal, these solutions are mixed with molten glass and poured into 150-l high grade stainless steel canisters. The high activity of the solidified glass gives a dose of approximately 2.5×10^3 Gy/h (10^5 rad/h) at the canister surface and a heat output of up to 2.5 kW per canister, depending on the age of the waste. For the host rock salt this can result in a maximum increased temperature of up to 200°C.

According to the PTB's current plans for a repository, the high active waste containers are to be emplaced in boreholes some 300 m deep. These will be drilled down from the emplacement level at intervals of around 50 m in parallel drifts.

As a part of the research and development work of GSF on the final disposal of high level radioactive wastes, their most fundamental properties, namely heat production and radiation, are being investigated with respect to their effects in in-situ tests in the salt itself subsequent to calculational and experimental investigations carried out in the laboratory in advance.

For this purpose, heater tests have been and are being carried out in boreholes up to 25 m deep. The heat output of the specially developed electric and hydraulic heaters matches that of the waste to be emplaced.

Rock temperatures in the repository develop as a function of the heat output of the radioactive waste. For various reasons the maximum temperature in salt stipulated by the PTB was 200°C. This means that the vitrified waste intended for disposal must be prepared and packaged accordingly (maximum waste concentration, maximum canister diameter, sufficiently long above ground interim storage to allow cooling). The heater tests at Asse investigate all relevant properties and alterations of the heated salt: heat migration and temperature gradients, convergence movements in the salt (particularly of borehole walls) thermal liberation of water (present in very small quantities in salt) and gases and their release into the borhole as a function of porosity and permeability of the salt rock.

An important aim of all experimental investigations is to examine computer programs and their input data to allow predictions to be made on the long-term behaviour of repositories. It is always noted in these tests that the plastic properties of the salt which are important for the tight enclosure of the wastes are activated in a particular way by rock pressure and heat. It also appears that the water present in the rock salt horizons (in quantities less than 0.05 weight percent) is liberated in the immediate vicinity of the emplacement boreholes and migrates into the borehole. This means that under normal disposal circumstances there is insufficient moisture to allow corrosive attack on the waste canisters.

In a joint German-American in-situ test, the so-called "Brine Migration Test", besides heat, a large quantity of radioactive isotopes was also introduced into a salt test field for the first time. Cobalt-60 sources with an overall output of 1.3×10^{13} Bq (36,000 Curies) were used. The test was running without any particular problems on the 800 m-level of the mine from December 1983 to November 1985. To date no previously unknown radiation effects have been detected. The radiation absorbed by the borehole wall was about 300 Gy/h (30,000 Rad/h).

The first test disposal of high level radioactive sources in the Asse salt mine is planned for 1987. It is intended to fill the lower third of six 15 m deep boreholes with glass-filled canisters with a diameter of 300 mm and a height of 1200 mm. The glass canisters will contain Sr-90 and Cs-137 as radioactive emitters. There will be two additional boreholes equipped only with electrical heaters, allowing comparative measurements to be made to determine radiation effects other than heat in the boreholes. To ensure that the waste simulates can be retrieved at any time the boreholes have to be cased but this is basically no different to the conditions in a future repository for high-level radioactive waste. The test will continue for five years and will allow an extensive measurement program to be carried out over a comparitively long period. Fig. 29 showes the planned layout of the two disposal galleries.

For above and below ground transport as well as for handling and emplacement of high level radioactive sources a system had to be developed which, with certain modifications, could be taken to be exemplary for a Federal repository. This complex system is to be tested in the course of the test disposal. An extensive calculational and laboratory program will accompany the test and the results will be made available to the PTB for the planning of a repository.

Further heater tests in the Asse salt mine to investigate special queries regarding disposal will run parallel to the main assignment during the coming years. The Research and Development Program of the GSF envisages that all problems of the high-level waste disposal will have been solved and the corresponding techniques developed and tested by the time a repository is to be constructed.

Fig. 24:
Filling of canister with vitrified high level waste

Fig. 25:
Test equipment and data registration of a temperature test field

Fig. 26:
Heater prior to installation into the test field

Fig. 27:
Transport of Cobalt-60 sources on the 800 m-level

Cable trench **Sealing** **Porous measuring volume**

External heater

Heater and
^{60}Co-Sources

Fig. 28:
Layout of the HLW test with Cobalt – 60 sources

electric radioactiv

Reference borehole without heater

AVR - gallery

Workshop

Measurement station

ASSE
Arrangement of the high level radioactive waste test
- planned test area, 800 m - level -

- Emplacement boreholes
- Maximum temperature reached in salt (= Borehole Edge)
- Workings of the 750 m - level

Fig. 29:
Layout of the HLW test field

Repository safety

The aim of disposal of radioactive waste in geological formations is the long-term protection of man and the environment from damage caused by the ionising radiation emitted by waste nuclides. To demonstrate that this aim is achieved, a site-specific safety analysis must be carried out in accordance with the Federal Ministry of the Interior's "Safety criteria for disposal of radioactive wastes in a mine".
A safety analysis for a radioactive waste repository consists of the two parts
- scenario analysis
- consequence analysis.

The scenario analysis identifies and describes events which can result in a release of radionuclides into the biosphere. Using model concepts of transport and retention of radionuclides, consequence analysis then estimates the possible radiation dose to man.

Selection of possible scenarios

Significant release of radionuclides from a geological repository can occur only if water or brine is present as a transport medium. In contrast to a repository in rock magmatic such as granite, this is not normally the case with a repository in salt: the distance to ground water-bearing strata is several hundred metres and the salt rock is impermeable to flow.

The task of scenario analysis is to identify on this basis accident scenarios which, although improbalbe, are still conceivable. An accident scenario is understood to be the consequence of events during the course of which, for example, the following two processes occur:
- contact between water/brine and the emplaced waste
- nuclide transport into the biosphere.

For example, supposing that the salt were penetrated by formations of more brittle and therefor possibly more fissured rock, such as anhydrite, and if there were sufficient mechanical stresses to cause opening of flow paths, a connection could be set up between groundwater-bearing strata and the repository. The waste canisters could then corrode, leading to contact between the brine and the waste.
Nuclide transport into the biosphere could be caused by convergence, i. e. the gradual closing of underground cavities. With the pressure prevailing underground, the salt rock behaves similar to a very viscous liquid and creeps into available cavities. The contaminated brine would thus be forced along flow paths into the overburden. The nuclides would then be taken up by ground water-bearing strata and would finally reach the biosphere through the various groundwater formations.
The final stage of the type of scenario description outlined as an example here forms the boundary conditions for consequence analysis. In particular it must be made clear when and where brine could enter into the repository. Furthermore the chemical composition of the brine and its rate of inflow also play a part.
In knowledge of site-specific conditions, many imaginable scenarios can be ruled out. Geological and hydrogeological conditions are investigated in extensive above and below ground reconnaissance programs. Laboratory and in-situ experiments such as have been carried out at the IfT for many years permit rejection of certain accident scenario hypotheses.

Considering the consequences

If, however, a scenario remains credible for a particular site, a consequence analysis is carried out, i.e. the radiation dose resulting form the scenario is estimated. This means that important individual effects which can affect nuclide release must be reproduced experimentally or theoretically, described in mathematical models and incorporated into computer programs. Nuclide transport from the waste matrix to the biosphere is impeded by a series of barriers. The behaviour of such geological (salt dome, overburden) or technical (waste matrix, canister, seals and backfilling of caverns) barriers are to be investigated in-situ, experimentally or by analogue studies.

Thus, for example, rock samples, backfill materials and sealing agents are tested with regard to properties which are relevant from a technical and safety aspect. The relationship between differing temperatures and pressures which can affect the materials and the stability of the above materials is assessed by hydraulic pressure tests. The results are used to derive a general material law for salt and backfill materials as well as for making mathematical prognoses. An extensive rock observation program in the Asse salt mine checks on the accuracy of existing computer programs for describing the overall rock mechanical behaviour of a salt mine and to practically standardize them.
Further experiments are aimed at clarifying the processes occurring in the case of "water or brine inflow". Before, during and after flooding of an old potash mine, the dissolution behaviour of salt, setting-up of density stratification in brine- and water-filled shafts, occurrence of advective flows and the stability of a sealing structure were all measured. Individually observed effects are simulated in the laboratory in order to increase understanding and to allow corresponding mathematical models to be prepared.

Measurement in the Asse salt mine serve to provide information on the behaviour of waste canisters, boreholes and boreholes seals under accident conditions.

Fig. 30:
Compression and tensible strenght tests on rock salt samples

The combination of such barriers is known as the multiple barrier system. Its objective is to sufficiently delay if not quantitatively to prevent the liberation of nuclides, even if individual barriers should fail. Consequence analysis estimates whether the remaining barriers provide sufficient retardation of nuclide transport to ensure repository safety.
For example, as a result of consequence analysis, "effective dose equivalents" can be calculated for the relevant nuclides. These quote the dose of the particular nuclide which would be received by a person drinking, for example, 0.8 m^3 of contaminated water per year (corresponding more or less to the average annual individual fluid requirement of a person).
These results can then be compared with the boundary values set by legislators (or the licensing authorities).

Safety of the Asse salt mine

In considering the long-term safety of the Asse salt mine, the question of possible water or brine inflow into the mine is of prime importance as, in the 130 years of salt mining in Germany, many mines have been flooded during operation. Flooding was frequently caused by mining of potash close to the water-bearing salt table and encounter with water-bearing anhydrite strata which are connected to the salt table. Investigations of the geological and hydrogeological conditions in the vicinity of the Asse salt mine have shown that:

- further salt strata around 40 m thick are to be found in the adjoining strata on the flanks of the salt anticline in the Mittlerem Muschelkalk,
- there are no water channels at the salt table of the Asse 2 and Asse 4 shafts,
- the main anhydrite and the grey salt pelite are present as small isolated formations only not as connected strata,
- the distance between the uppermost chambers and the salt table is over 200 m,
- brine infiltration into the mine in the past originated from isolated brine occurrences within the salt deposits.

If, despite the density of the neighbouring rock series on the southern flank, the possibility of water flow in deeper formations is assumed, then the rock mechanical behaviour of the salt and the neighbouring rocks is of significance for a hypothetical water inflow into the mine. Damage to the environment caused by the emplaced waste can only be ruled out given sufficient long-term stability of the geological disposal medium. Therefore a stability demonstration for the mine is of considerable importance. Such a demonstration involves making predictions on future rock behaviour, using rock mechanical model calculations.

Such calculations give an increasingly better representation of measured or observed mine performance and therefore allow continually more accurate prognoses to be made. In addition, in contrast to most rock types, rock salt has both elastic-plastic and viscous properties. The "creep" characteristic of rock salt is a distinct advantage with respect to the stability of cavities in the rock salt: peak loads on the rock are decreased to some extent during their production phase already. A cavern constructed according to safety regulations thus remains stable even over a long period of time.

Overall assessment of the geological, hydrogeological and rock mechanical investigations to date shows that water inflow into the Asse salt mine is highly improbable.

Perspectives

The techniques and methods for the disposal of radioactive waste developed by the Institut für Tieflagerung of GSF are all aimed at the long-term protection of man and the environment.
At the Asse salt mine, suitable procedures for the disposal of non-heat-emitting radioactive waste in geological formations have been developed and tested.
Current research therefore concentrates mainly on disposal methods for heat-producing intermediate and high-level radioactive wastes. Emplacement and confinement procedures based on the results of research done to date are being developed in further detail in large-scale experiments. This work is being done in cooperation with over 30 institutes from the fields of major research, industry and universities. Moreover, with respect to safety analysis, the models used for describing repository systems are being continually refined and their long-term safety is being checked in accompanying calculations.
The results of the research and development work are recorded annually in numerous publications and reports. They form an important source of information for the Physikalisch-Technische Bundesanstalt (PTB) which, according to the Atomic Act, is the Federal authority responsible for the disposal of radioactive waste, with respect to the planning, construction and operation of Federal repositories.

Explanation of technical terms

(The original German version of this glossary was compiled from Lueger, Lexikon der Technik, Volk. 4: Lexikon des Bergbaus, Stuttgart 1962, and Taschenbuch der Geologie "Entwicklungsgeschichte der Erde", Hanau 1962)

Accident –
Interruption of the normal relationship between rock formations, produced by tectonic of atectonic processes.

Activity –
see Radioactivity.

Additional liner –
a cylindrical support introduced into a shaft in front of an already existing support.

Aerosol –
very finely distributed liquid and solid particles suspended in a gas.

Aller Series –
the youngest series of the Zechstein, also known as Zechstein 4.

Anhydrite –
rock-forming mineral, $CaSO_4$.

Blind shaft –
a vertical working which does not lead to the surface but connects individual underground levels.

Brine –
a more or less concentrated salt solution.

Buntsandstein –
Lower Division of the Triassic, which is the oldest formation of the Mesozoic era.

Carnallite –
salt mineral, $KCl \cdot MgCl_2 \cdot 6\ H_2O$, found particularly in the Stassfurt deposit (K2).

Carnallitite –
salt rock consisting mainly of carnallite and rock salt. Other components are, e.g. kieserite or anhydrite.

Cavern –
a rock cavity produced by leaching (dissolution with water) or by mechanical methods such as drilling or blasting.

Contamination –
introduction of impurities in the form of radioactive materials.

Convergence –
decrease in distance between two opposite points of an underground cavity.

Deposit –
natural accumulation of exploitable mineral ores in the earth's crust.

Dose –
amount of radiation resulting from the interaction of radiation with a material.

Dose rate –
dose observed per unit of time, e.g. per hour or year.

Drift –
see Gallery.

Drilled shaft –
a shaft constructed by drilling.

Excavation chamber –
a cavity produced by the systematic mining of exploitable minerals or mineral ores.

Floor –
the strata underlying the formation or deposit under consideration.

Ion, ionisation –
electrically charged atomic or molecular particle produced from a neutral atom or molecule by separation or deposition of electrons.

K2 –
symbol for the older potash deposit of the Stassfurtlager which, in the Asse anticline, consists mainly of carnallite, rock salt and kieserite.

Keuper –
Upper Series of the Triassic which is the oldest formation of the Mesozoic era.

Kieserite –
salt mineral, $MgSO_4 \cdot H_2O$.

Leaching residues –
residues left after chemical alteration of a rock by inflowing salt or fresh water.

Leine Series –
younger series of the Zechstein, also known as Zechstein 3.

Level (floor) –
a) a horizontal level forming part of the mine installations, with all mine workings,
b) the lower level or tilted boundary of a mine working.

Men hoisting system –
shaft hoisting system in shafts and blind shafts for the transport of personnel.

Mine –
(synonimous with pit), incorporating all above- and below ground workings.

Muschelkalk –
Central series of the Triassic which is the oldest of the Mesozoic era.

Nuclide, radionuclide –
a nuclide is a nucleus characterized by its number of protons and neutrons (e. g. U-235, U-238). If a nuclide is unstable, i. e. if it changes spontaneously through no extend influence with emission of radiation, then it is known as a radionuclide.

Particle radiation –
see Radiation.

Permanent support –
collective term for all materials used fqr keeping mine workings open and secure.

Pillar –
a symmetrical section of rock left standing between adjacent chambers, with the aim of preserving the original cohesion of the rock and preventing collapse.

Plasticity –
the ability of solids to behave elastically under stress or compression up to the elastic boundary, but to "flow" under greater stresses or compression. With plastic bodies, flow occurs before the fracture boundary is reached. Deformation through flow is generally irreversible.

Quaternary –
the youngest of the geological formations.

Radiation –
radiation which is emitted upon radioactive decay of atomic nuclei (see radioactivity). The different forms are:

α-radiation	(particle
β-radiation	radiation)
γ-radiation	(electromagnetic radiation)

Radioactivity, activity –
the ability of certain atomic nuclei to change without any external influence with emission of radiation (radioactive decay). Activity is a parameter which gives the number of nuclei decaying per second and thus the amount of radioactive material.

Radiological units –
Becquerel (Bq)
The unit of measure for activity.
1 Bq = 1 decay/second
1 Ci = 3.7×10^{10} Bq).
Gray (Gy)
The unit of measure for radiation energy absorbed by material (1 rd = 0.01 Gy).
Sievert (Sv)
The unit of measure for the biological radiation effect. For β- and γ-radiation, 1 Sv corresponds to 1 Gy. Fpr α-radiation, e. g. the value incrases tenfold, i. e. 1 Gy corresponds to 10 Sv.

Radionuclide –
see Nuclide.

Rock salt –
salt mineral, NaCl, also known as halite.

Roof –
Layer always overlying the formation or deposit under consideration.

Safety pillar –
protected area in a mine where workings exist only under exceptional conditions.

Salt table –
upper, more or les even boundary of a salt deposit, produced during the course of geological time by the dissolving effect of seepage water und groundwater.

Salt rock –
rock formed by salt minerals of which rock salt (NaCl) is the most important and most common. Salt rocks often contain impurities in the form of anhydrite, clay or other salt minerals, e. g.

Seismics –
geophysical methods of examining the construction of underground formations by emitting elastic waves produced by blasting or mechanical vibration.

Sinking –
construction of a shaft or blind shaft

Stassfurt Series –
older series of the Zechstein, also known as Zechstein 2.

Stope –
a regularly formed section of rock left between two vertically arranged excavation chambers, the aim being to seal the chambers at the top and to perserve the original rock cohesion to prevent collapse.

Suface shaft –
a vertical working which leads from the earth's surface to a deposit.

Tectonics –
science of the formation of the crust the earth and its movements.

Tubbing –
cast-iron or -steel segment of a cylindrical shaft support ring.

Underground workings –
collective term for all mined cavities in a mine, such as shafts, blind shafts, driftes, chambers etc.

Upper Cretaceous –
Upper series of the Cretaceous which is the most recent formation of the Mesozoic era.

Weather –
mining term for the air circulating in a mine, but also for other gas mixtures.

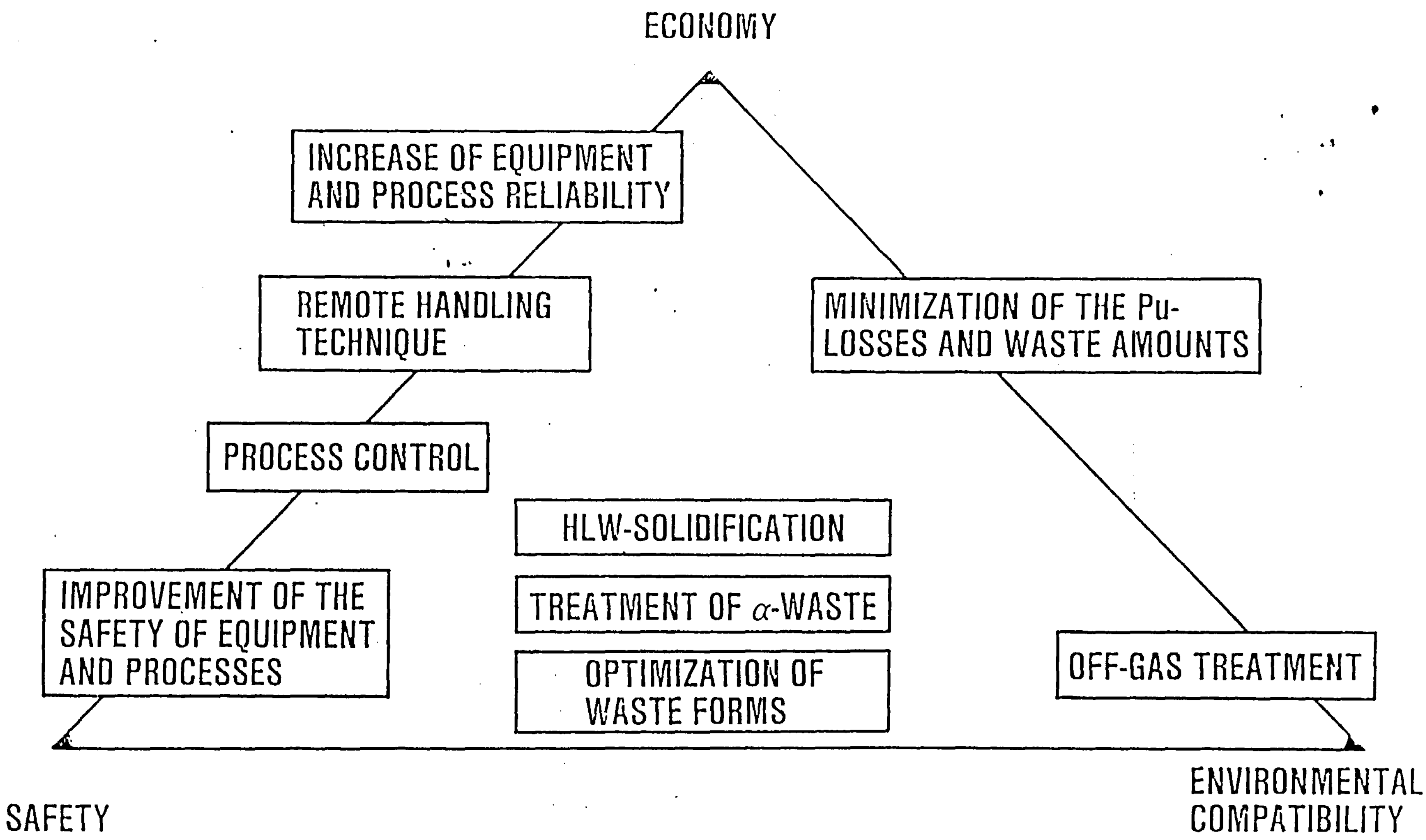
Working –
a systematically mined cavity.

Zechstein –
Upper series of the Permain which is the most recent formation of the Palaeozoic era.

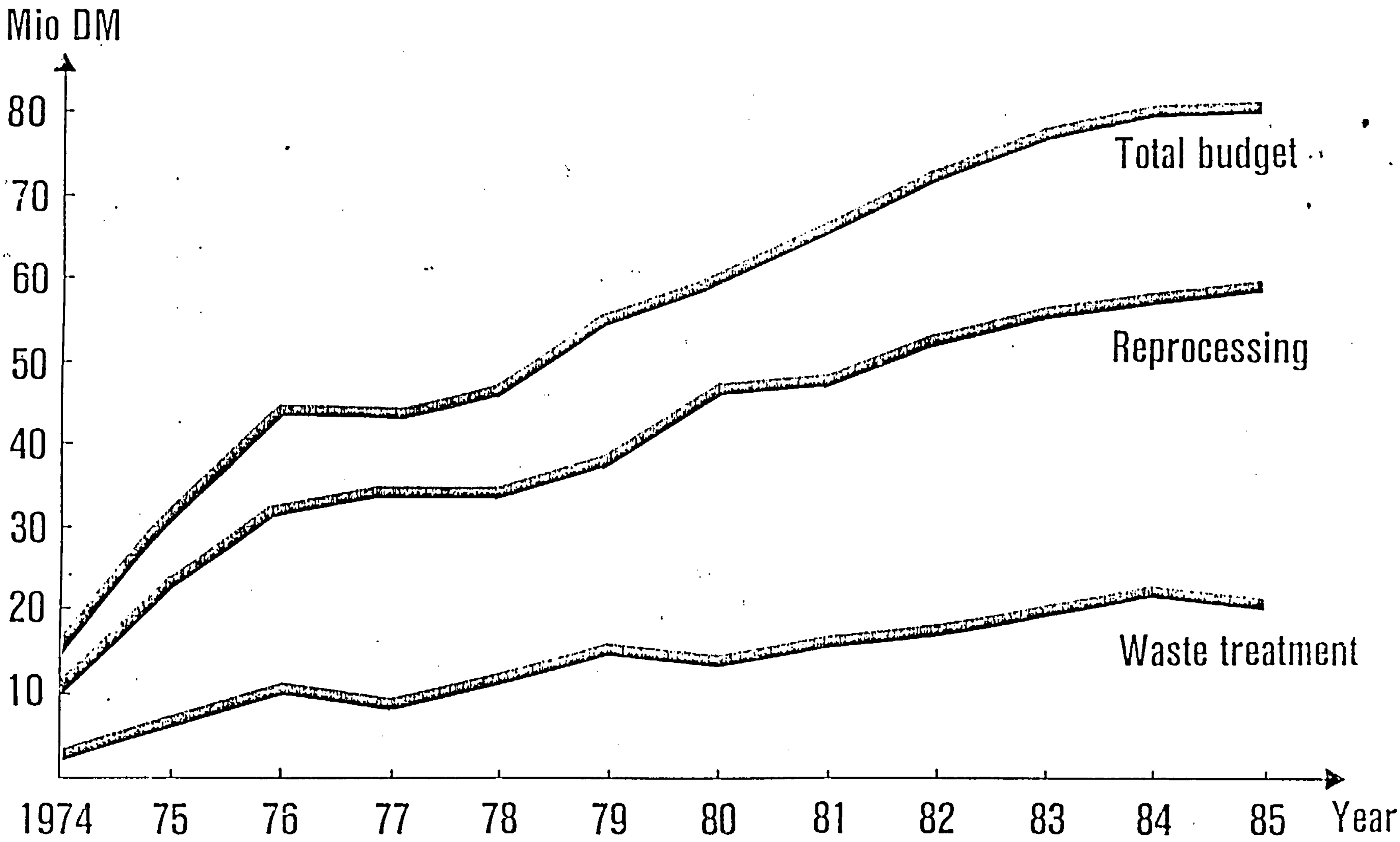
1960	Beginning of KfK activities in spent fuel reprocessing and waste treatment
1967	Beginning of waste disposal in the asse repository
1967-69	Construction of MILLI
1967-70	Construction of WAK
1971	Commissioning of WAK Foundation of KEWA Foundation of URG
1974	Establishment of PWA
1975	Foundation of PWK
1977	Foundation of DWK
1979	KfK-DWK cooperation agreement signed Gorleben hearing
1983/84	Beginning of licensing procedure at Dragahn and Wackersdorf for plant construction on one of these sites
1985 (Nov.)	First construction permit for Wackersdorf reprocessing plant

KfK

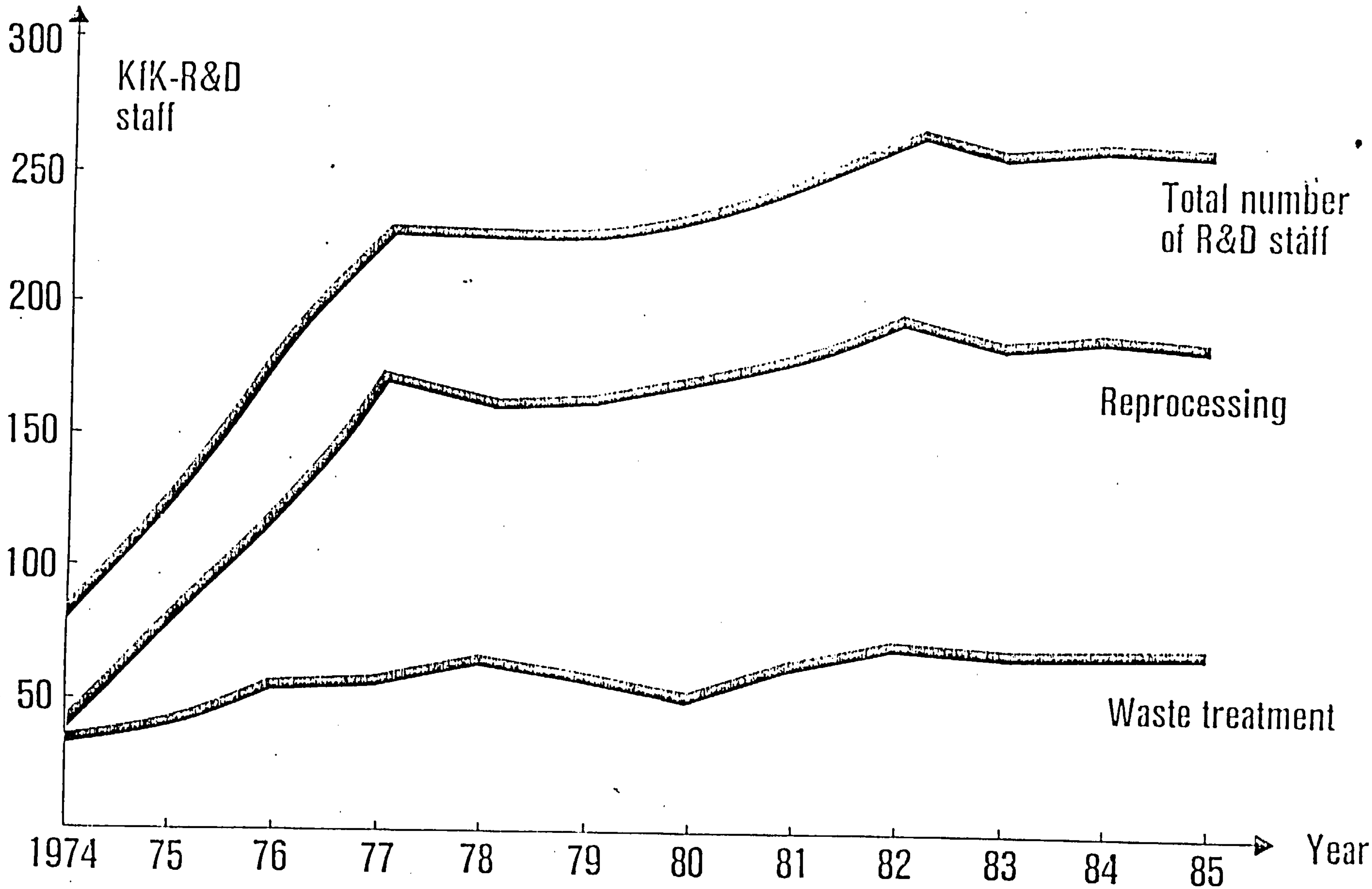
Reprocessing and Waste Treatment Project (PWA)



ECONOMY
INCREASE OF EQUIPMENT AND PROCESS RELIABILITY
REMOTE HANDLING TECHNIQUE
PROCESS CONTROL
IMPROVEMENT OF THE SAFETY OF EQUIPMENT AND PROCESSES
HLW-SOLIDIFICATION
TREATMENT OF α-WASTE
OPTIMIZATION OF WASTE FORMS
MINIMIZATION OF THE Pu-LOSSES AND WASTE AMOUNTS
OFF-GAS TREATMENT
SAFETY
ENVIRONMENTAL COMPATIBILITY



Expenditure for PWA



Number of R&D Staff Assigned to PWA

Year	Investments	Operating Costs	Staff employed Manyears
1974	2.4	15.2	85 •
1975	5.6	28.6	185
1976	9.4	35.4	207
1977	7.8	37.0	226
1978	9.0	39.1	226
1979	11.2	42.5	225
1980	8.6	51.1	231
1981	7.6	58.0	249
1982	8.6	65.2	268
1983	10.1	68.8	260
1984	9.9	72.4	262
1985	11.2	71.0	260
1974 1985	101.4	584.3	2684

* by PWA only

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R&D Expenditure by KfK* 1974-1985

Institut für Heiße Chemie	IHCH
Institut für Nukleare Entsorgungstechnik	INE
Hauptabteilung Ingenieurtechnik	IT
Institut für Radiochemie	IRCH
Institut für Datenverarbeitung in der Technik	IDT
Laboratorium für Aerosolphysik und Filtertechnik I und II	LAF
Institut für Material- und Festkörperforschung	IMF
Institut für Neutronenphysik und Reaktortechnik	INR
Institut für Reaktorbaulemente	IRB
Hauptabteilung Datenverarbeitung und Instrumentierung	HDI
Hauptabteilung Kerntechnische Betriebe	KTB
Labor für Isotopentechnik	LIT
Entwicklungsabteilung Kernmaterialsicherung	EKS
Institut für Kernphysik III	IK-III

KfK

In the Framework of the Reprocessing and Waste Treatment Project (PWA) the Following KfK Institutions Cooperate

General objective

Improvement of existing and development and testing of new processes, equipment and products with a view to

- Increasing economy
- Increasing safety
- Decreasing environmental impact

Main Topics

- Head-end and off-gas
- Extraction
- Waste treatment
- Intervention and handling techniques
- Material investigations
- Process control and process data processing

KfK

Objective

- Optimization of the conditions for dissolution of FBR fuels
- Characterization of the fuel dissolution residues
- Development of methods for immobilization of Krypton
- Development of the integral dissolution of FBR fuel

Results Obtained

- The solubility of several Pu fuels has been determined; development of a process for almost complete removal of iodine during the fuel dissolution step (HET testing facility)
- The chemical composition of dissolution residues was analyzed (LWR + FBR)
- For the dissolver off-gas a cleaning system was developed and subsequently tested for retention of Krypton by cryogenic separation (Testing facilities WÄSCHE, PASSAT, REDUKTION, KRETA). The results were used in the conceptual design of AZUR
- The Krypton immobilization processes, zeolite encapsulation and ion implantation, were investigated and the waste form properties was determined; two test facilities were constructed
- Kr storage in high-pressure bottles feasible
- Vessel ventilation system developed (BEATE, PASSAT) and implemented in the off-gas of the PAMELA active facility
- Clarification of aerosol formation and retention (BEATE)

Head-end and Off-gas Treatment

Current and Planned Activities

- Optimization of conditions for dissolution of FBR fuel e.g. integral dissolution
- Iodine and aerosol tests with WÄSCHE/PASSAT with a view to WAW and SBR-WA
- Study of formation of solids in feed solutions after clarification
- Investigations into Kr-retention by fluorocarbon absorption in TED
- Development of accident filters for fire and explosion hazards
- Development of the Krypton encapsulation technique on a semi-technical scale

A. = F. active Operation IA. = F. inact. Operation		Organizational Unit	Period of Construction	Costs Mio DM
Headend-Teststand (HET)	IA.	IHCH	1975/1977	2.5
KRETA + ADAMO	IA.	IT/IHCH	1975/1977	2.5*[1]
PASSAT	IA.	LAF/IT	1978/1980	1.0 + 1.4*[1]
Wäsche	IA.	IT/LAF	1981/1985	2.0
Reduction	IA.	IHCH/KTB	1979/1981	0.8 + 0.7*[1]
BEATE	IA.	LAF	1981/1982	1.0
TED	IA.	IHCH	1983/1986	0.8
				8.1 + 4.6*

*[1]Financed by Other Funds

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PWA Testing Facilities (since 1974); Investments over 1 Mio DM — Head-End

Objective

- Elaboration of chemical flowsheets for reprocessing of LWR and FBR fuels
- Minimization of Pu losses and quantities of waste
- Improvement of safety and reliability of equipment and processes
- Testing of processes for H-3 retention
- Increasing reliability of pulse column operation by study of dynamic behaviour

Results Obtained

- Improvement of the chemical flowsheet by experimental (MILLI, LABEX) and theoretical (simulation analyses) investigations:
  Reprocessing of FBR fuel in MILLI

  Development of electrochemical processes and equipment for plutonium reduction and oxidation, for denitration and for hydrazine oxidation: (ELKE, EMMA, ROXI) PUTE test facility
  Successful testing in WAK of an electro-reduction mixersettler and electrooxidation cell (e.g. EMMA)
- Expansion of the electrochemical process to first extraction cycle (1B-EMMA)
- Application of Hf as a neutron absorber for large pulsed columns
- Modeling of disturbances in the PUREX-process (MINKA)
- Determination of the characteristics and specifications of equipment and process for optimal H-3 retention in the first cycle as well as the construction of 1:1 WAW pulsed column (PUSTA)
- Radionuclide tracer techniques for determination of the fluid dynamics
- Uranium-active facility of 1B-ELKE for the WAW in cooperation with DWK
- Providing know-how for TEKO-ROXI

Current and Planned Activities

- Determination of favorable flowsheets for short cooled FBR-fuel (MILLI)
- Further development of the electrochemical and salt free processes for improvement and simplification of the process as well as minimization of waste arisings
- Theoretical and experimental studies on the kinetics of mass transfer in exctraction equipment
- Development and testing of fast-contactors
- Investigations of the influence of disturbances in the PUREX-process (MINKA, LABEX, MILLI)
- Minimization of the Pu losses in the FBR-reprocessing cycle (PUTE)
- Data collection for scaled-up pulsed columns including the H_3-scrub columns (PUSTA)
- Experimental study and math. simulation analysis on the safety of 1. extraction cycle operation (MINKA)

A. = F. active Operation IA. = F. inact. Operation		Organizational Unit	Period of Construction	Costs Mio DM
UTE	A.	IHCH	1975/1982	2.0
PUTE	A.	IHCH	1976/1986	10.0
MINKA	A.	IHCH	1979/1985	2.0
PUSTA (in TEKO)	IA.	IT/KTB	1979/1985	4.0
MILLI-Modernisierung	A.	IHCH	1983/1986	11.5
				29.5

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PWA Testing Facilities (since 1974); Investments over 1 Mio DM — Extraction

Objective

- Development and testing of solidification processes for HLW
- Development of processes for the solidification of fuel dissolution rediues
- Development of processes for conditioning hulls and structural materials
- Development of processes for the treatment of α-waste
- Development of processes for the treatment of H_3-waste
- Optimization of waste forms
- Development of processes for embedding MLW and LLW
- Development of processes for MLW decontamination
- Development of the "in-situ"-solidification of radioactive waste

Results Obtained

- Vitrification process with ceramic melter developed and tested
- A suitable active facility (PAMELA) has been constructed in mol by DWK and has been in successful active operation since Ooctober 1, 1985
- For conditioning of dissolution residues a process for embedding in a ceramic matrix was conceived and subjected to active testing on a laboratory scale
- Embedding of hulls in concrete and compaction by rolling
- A process for acid digestion of α-waste
- Active operation of ALONA was completed in June 85
- Determination of the repository relevant properties as well as characterization of the waste forms largely completed (VG 98, MEGA)
- Providing know-how for the eurobitum plant of eurochemic for bituminization of LLW/ MLW
- Process for fixation of H_3 in zeolite matrices developed
- Process for enrichment of tritium based on CECE conceived
- The decontamination of MLW was demonstrated in active operation on a laboratory scale
- The in-situ technique was demonstrated in inactive operation by the production of 1300 t granulates

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Waste Treatment

Current and Planned Activities

- Evaluation of experience gathered in active operation of PAMELA and ALONA
- Development of ceramic melter for the Wackersdorf reprocessing plant
- Further development of a process for embedding in ceramic
- Determination of the quality of HLW glass waste forms produced in a technical scale facility (technology of melting and hot pressing)
- Isotopic enrichment of tritium according to the CECE process
- Active demonstration of the MLW decontamination process on 100 l scale
- Large scale test of the in-situ process by vertical conveyance technique
- Development of waste form and process technique for FBR dissolver insolubles
- Development of waste form and process technique for hulls and structural materials for FBR and LWR-fuel elements by mechanical compaction or melting

A. = F. active Operation IA. = F. inact. Operation		Organizational Unit	Period of Construction	Costs Mio DM
Verglasungsanlage I	IA.	INE	1974/1975	1.0
Verglasungsanlage II	IA.	INE	1979/1980	2.5
Verglasungsanlage III	IA.	INE	1981/1982	5.5
Verglasungsanlage WA-350	IA.	INE	1979/1982	2.0
MAW-Boxenlinie	A.	INE	1979/1982	6.0
ILONA	IA.	INE	1979/1980	1.0
ALONA	A.	INE (Mol)	1980/1982	3.0
Keramiklinie	IA.	INE	1984/1986	1.5
				22.5

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PWA Testing Facilities (since 1974); Investments over 1 Mio DM — Waste Treatment

Objective

- Development and testing of remote handling systems and components
- Investigation of material properties
  Determination of the best suited construction material

Results Obtained

- Development of the remote handling systems for PAMELA
- Definition of time schedules for remote maintenance
- Corrosion studies on construction materials for dissolver and process equipment
- Investifation of Hf as material for critically safe equipment
  Hf approved as construction material by TÜV (Technical inspectorate)

Current and Planned Activities

- Testing of remote handling concepts and components for the WA reprocessing plant and the HEZA hot cells
- Corrosion tests on materials for components
- Development and testing of modular technique for FBR reprocessing

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Intervention and Handling Technique
Material Investigations

A. = F. active Operation IA. = F. inact. Operation		Organizational Unit	Period of Construction	Costs Mio DM
3 Fernbedienungsteststände	IA.	IT	1976/1986	3.5
Fernbedienbare Kupplung	IA.	IT		0.85
Fernbedienbarer Schweißstand	IA.	IT		0.2
PAULA	IA.	IT		0.2
				4.75

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PWA Testing Facilities Remote Handling Technique

Objective

- Development of in-line instruments
- Automation of analytical laboratory systems
- Computer based process control and operating procedure
- Nuclear safeguards

Results Obtained

- Instruments developed up to prototype level
  - — fuel element monitor
  - — waste drum monitor
  - — α-monitor
  - — non-dispersive XRF
  - — γ-absorptiometer
  - — density/conductivity measuring device
  - — K-edge densitometer
- Automated laboratory equipment
  - — α-preparation box
  - — γ-sample changer
  - — sample distribution system
  - — ICP-AES assembly
- Laboratory-scale tested neutron measuring systems
  - — gadolinium monitor
  - — HLW monitor
  - — accumulation monitor
  - — hafnium monitor
- Computerized process control
  - — PRODES open-loop process control system at WAK
  - — DIANA information system for laboratory
  - — test evaluation in HLW vitrification
  - — waste accounting system at HDB (KADABRA)
- Process modeling
  - — fissile material inventory system for 1000 t/a plant

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Process Control and Process Data Processing

Current and Planned Activities

- Development of equipment
  - — neutron monitor for hulls and dissolver residues
  - — fiber optical laser photometer
  - — hybrid K-edge/K-XFA solution assay system
  - — improved α-preparation box
  - — in-line XFA monitor

- Development of new measuring methods
  - — Pu concentration profile in pulsed column
  - — laser-Raman spectroscopy
  - — uranium determination by laser fuorimetry
  - — glass level measuring in HLW canisters
  - — determination of Tc at very low concentration

- Application of computer based operating system
  - — expansion of PRODES and DIANA for testing of NRTA
  - — process control system for the VW 1
  - — process data acquisition system on IHCH test facilities
  - — process control in TEKO

- Strategical concepts
  - — WA process simulation
  - — NRTA methods for fissile material balancing
  - — verification of measuring systems

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Process Control and Process Data Processing

A. = F. active Operation IA. = F. inact. Operation		Organizational Unit	Period of Construction	Costs Mio DM
Heiße Zelle f. Analytik	A.	IRCH	1978/1982	1.8
Teststand für n-Meßverfahren	A.	IRCH	1982/1985	0.4
ESKA + SIMS	IA.	IRCH	1981/1982	1.5
Laser-Raman-Spektrometer	A.	IRCH	1980/1986	0.7
PRODES DIANA-KALAU	A. A.	IDT IDT	1980/1986	0.5 + 3.0 BMFT
Brennelementmonitor	A.	INR/DWK	1980/1986	0.5 + 2.5 DWK
Faßmonitor	A.	INR/IT	1983/1985	1.0
Hülsenmonitor	A.	INR/IT	1984/1986	0.2
				6.6 + 5.5

PWA Testing Facilities; Investments over 1 Mio DM — Analytics and Process Control

B. International

The european commission grants financial aids to 7 programs of PWA

EUROCHEMIC (acid digestion of α-containing waste, bituminization, PAMELA)

USA, KfK as partner in the waste treaty of BMFT-US/DOE with PNL, HEDL, SRL, ORNL, SANDIA, INL

PNC-Japan (agreement PNC-KfK) in the field of waste treatment since febr. 1981, reprocessing since Nov. 1982

Brazil-Agreement KfK-NUCLEBRAS

India-HLW-vitrification

KfK

Partners Cooperating with PWA

A. National

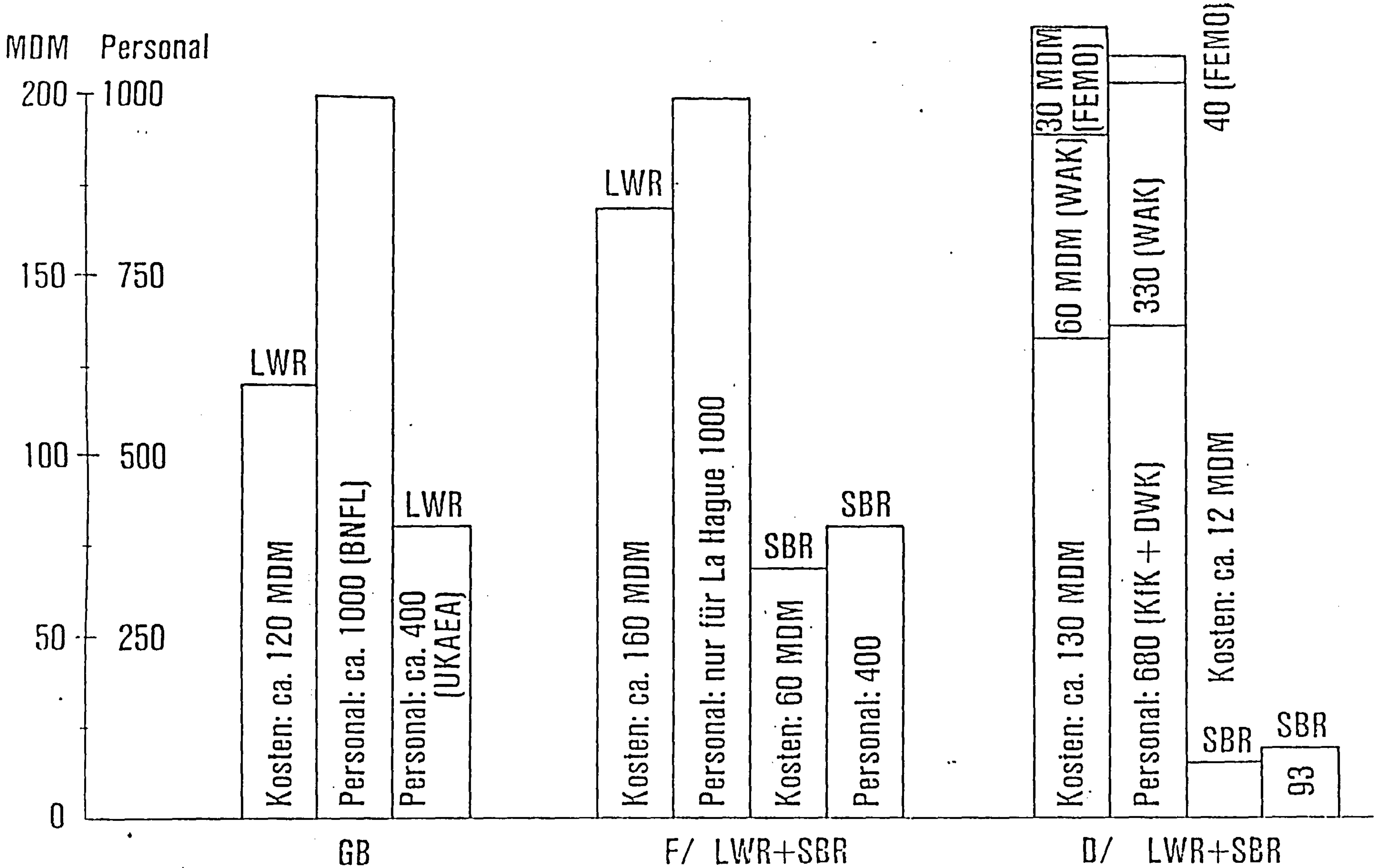
(exept for DWK, with which the joint program was set up)

KFA, HMI, GSF
NUKEM, KAH, UHDE/LURGI (deleg. engineers)

13 universities with 28 teams
(70 persons and 7.5 mio dm/year)

DWK R&D partly conducted by its own staff (e.g. FEMO-bench), partly by WAK (e.g. TEKO). The operation of WAK includes testing of flowsheets and equipment for a largi seale plant

Partners Cooperating with PWA



KfK

Abb. 6

Aufwand und Personaleinsatz für F+E zur Entsorgung von LWR/SBR in 1983

Quelle: ATAG-Studie Jan. 1983

Attachment 5

INTERNATIONAL ATOMIC ENERGY AGENCY
OECD NUCLEAR ENERGY AGENCY

INTERNATIONAL SYMPOSIUM ON THE BACK-END OF THE NUCLEAR FUEL CYCLE – STRATEGIES AND OPTIONS

Vienna, Austria, 11–15 May 1987

IAEA-SM-294/ 32

CONCEPTS FOR DIRECT DISPOSAL OF SPENT LWR AND HTR FUEL IN THE FEDERAL REPUBLIC OF GERMANY

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Abstract IAEA-SM-294/32

CONCEPTS FOR DIRECT DISPOSAL OF SPENT LWR AND HTR FUEL IN THE FEDERAL REPUBLIC OF GERMANY

In a comparative study initiated by the German Ministry for Research and Technology (BMFT) which has been conducted in the period from 1981 to 1985, direct disposal of spent LWR fuel was compared with the traditional fuel cycle based on reprocessing and thermal recycling. The results of the study did not exhibit decisive advantages of direct disposal over fuel reprocessing. The German Federal Government concluded it would continue to adhere to fuel reprocessing as the reference scheme of "Entsorgung". Under existing German atomic law, direct disposal of spent fuel is only permissible for fuel for which reprocessing is neither technically feasible nor economically justified. Accordingly, direct disposal was selected as the reference technology for spent fuel from the German high-temperature reactor (HTR) program. In the case of LWR fuel, direct disposal serves as a supplementary option. A program has been launched by BMFT to further develop direct disposal of spent fuel to technical maturity. Thereby, emphasis is laid upon canister development, in-situ tests in the Asse experimental mine, and systems analysis, the latter being geared toward identification of repository configurations that accommodate both spent fuel and reprocessing waste in an optimal way.

1. INTRODUCTION

The way the FR Germany is dealing with the back-end of the nuclear fuel cycle can best be characterized by two salient facts: The commitment to closing the LWR fuel cycle through fuel reprocessing and the ongoing site investigation of the salt dome at Gorleben. All steps at the back-end of the fuel cycle are referred to as "Entsorgung" in German and this term will be used frequently in this presentation.

To illustrate how direct disposal came into the picture, one has to go back to the end of the 1970s. In May 1979, the project of a large-scale, 1,400 t/yr fuel-reprocessing plant had been cancelled for political reasons. In the aftermath of this cancellation, the heads of both the federal government and the states agreed on the following approach to the Entsorgung question: Reprocessing should go forward but on a smaller scale. It was also decided that there was to be a major comparative study of fuel reprocessing and direct geologic disposal of spent LWR fuel, enabling a judgment by 1985 whether or not direct disposal might offer decisive advantages.

This comparative assessment was conducted by the Karlsruhe Nuclear Research Center; it was based on the quantification of the following criteria: technical feasibility, radiological safety, economics, and safeguards [1]. The overall evaluation was considerably affected (1) by the fact that, in contrast to fuel reprocessing, conditioning of spent fuel, i.e., consolidation and sealing in large metal canisters, has not been performed at a large scale in a dry environment; (2), disposal of HLW and spent fuel have not been demonstrated. To give an example, neither high-precision dry drilling of parallel 300-m-deep, 0.7-m-diam boreholes - the reference emplacement concept for vitrified HLW from reprocessing - nor lowering packages with weights up to 1 t into these boreholes are state-of-the-art. On the other hand, the lowering through the transport shaft and the underground handling of the heavy spent fuel package - more than 60 t in the current reference concept for spent fuel disposal - have never been demonstrated either. Therefore, demonstration has become the key element in the new R&D program that was started after the governmental evaluation in early 1985.

There were two major constraints to be observed by the federal government when evaluating the findings of the Entsorgung comparison: First, in the German Entsorgung Policy it is clearly spelled out that beginning January 1, 1985, initial operating licenses of new nuclear power plants will be granted only under the stipulation that the site selection process for facilities of either one of the Entsorgung variants has been concluded. Second, the priority of recovery of fissile material over removal of spent fuel is embodied in the FRG Atomic Energy Act. Therefore, implementation of direct disposal as the reference scheme of waste management for LWR fuel would make mandatory an amendment of the Act. It is obvious that, based on direct disposal of spent fuel alone, evidence of ensured Entsorgung could not be given during the period of time required to amend the Act.

In its decision of January 23, 1985, the federal government felt that it best complied with the legal requirements by adhering to reprocessing and recycling as the reference scheme of Entsorgung. The government was vindicated in its decision by the findings of the aforementioned Entsorgung comparison which did not exhibit decisive advantages of direct disposal over fuel reprocessing.

In its new policy statement, the cabinet also ruled that direct disposal was permissible only for fuel for which - from the present point of view - reprocessing was either technically not feasible or economically not justifiable. On the basis of that policy, the federal government decreed to stop further work aiming at reprocessing of spent fuel from the German high-temperature reactor (HTR) program, and direct disposal has become the reference Entsorgung scheme for that sort of fuel. But this decision also implied to carry on work with respect to direct disposal of spent LWR fuel. A new R&D-program for direct disposal of LWR fuel was started in mid-1985. The timing of both programs, LWR and HTR spent fuel disposal, is such that their results can be included in the license application for the Gorleben salt dome in the early 1990s.

2. DIRECT DISPOSAL OF SPENT LWR FUEL

Goals of the LWR spent fuel disposal program have always been low dose burdens for the working personnel during packaging and emplacement, minimum amounts of secondary waste arising from the packaging process, and integrity of the canister barrier over a design lifetime of 500 years.

2.1. Technical concepts

Therefore, technical concepts have focused on self-shielded canisters holding unmodified fuel assemblies (or at the utmost, individual, close-packed fuel rods), and the canisters have to withstand the lithostatic pressure at a depth of about 800 m as well as the corrosive attacks of salt brines over the design lifetime. Consequently, emplacement on the floor of emplacement drifts has become the reference concept of these packages whose heavy weight ensued from the self-shielded design. As an example, the reference concept developed in the first phase of the direct disposal R&D program prior to 1985, is illustrated schematically in Fig.1. The main components of the package are a (MnMoNi)-steel cask with a wall thickness of 15 cm that provides the required strength and an outside layer of Hastelloy-C4 against corrosion. Three intact fuel assemblies are inside a

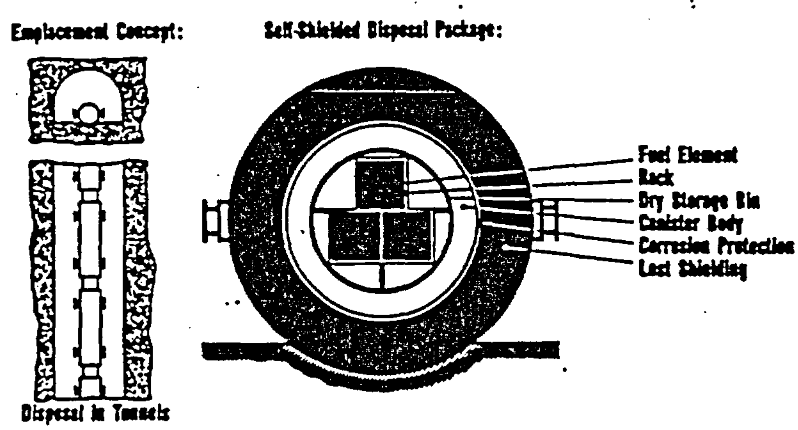

FIG.1.Former spent LWR fuel disposal concept

thin-walled, gas-tight steel cask referred to as a dry storage bin. With respect to both the design strength of 30 MPa and corrosion resistance, no credit is taken from the about 20-cm-thick shielding which dominates the total package weight of more than 50 t. The outside diameter is 1.4 m, the total length 6 m. During emplacement a surface temperature of 50°C and 0.1 mSv/h 1 m off the package are guaranteed. An extra shipping cask has to be provided for shipments on public routes [2].

The age of the spent fuel in the packages and the spacing between consecutive packages on the drift floor have to be such that rock salt temperatures never exceed 200°C. The empty space in the drift is backfilled with crushed salt immediately after emplacement. While the repository together with the underground transportation system was only designed conceptually, detailed engineering was done for the disposal cask and a prototype was manufactured.

The current cask development program is focused on improved economics. DWK's (Deutsche Gesellschaft zur Wiederaufarbeitung von Kernbrennstoffen) newly developed cask dubbed POLLUX [3] is suitable for storage, shipment, and disposal of spent fuel (Fig. 2). Contrary to the old design, the POLLUX holds the rods from 8

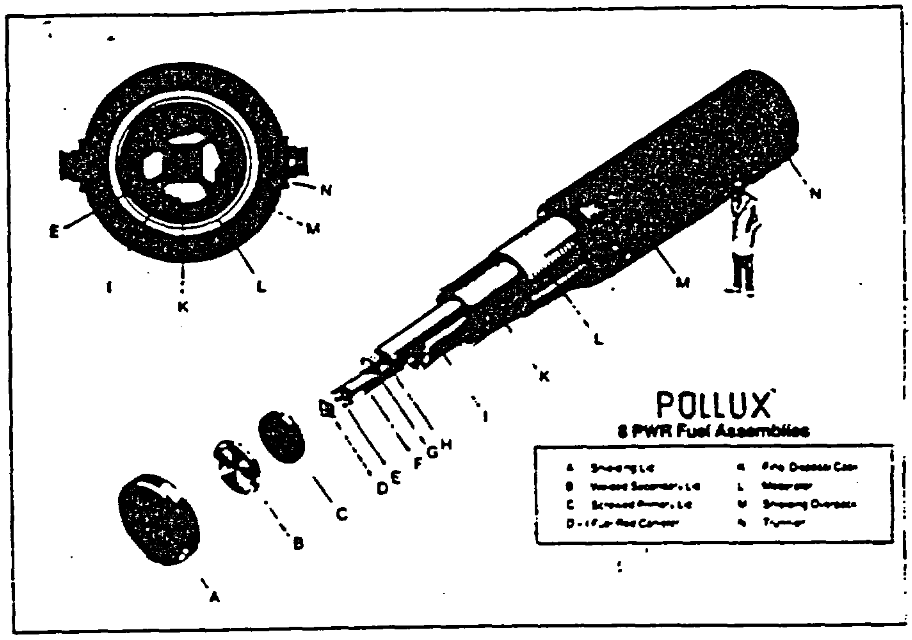

FIG.2.POLLUX spent fuel disposal canister

spent fuel assemblies. The internals of the canister are divided into four equal quadrants and a void space in the center. Each quadrant is large enough to accommodate the rods from two PWR fuel assemblies while the compacted skeleton of the fuel assemblies is placed in the center void. The cask is designed to accept spent fuel three years after discharge from the reactor with a maximum heat load of 20 kW. The dimensions are similar to those of the old design, only the total weight is higher by a margin of about 20 % and totals 64 t. Two main differences deserve mention: gas-tightness is now assured by a thick welding seam connecting the secondary lid with the cylindrical body; second, through the cast iron overpack the cask complies with transport regulations and is therefore qualified for surface and underground transport.

Beside drift emplacement of spent fuel, disposal in 300-m-deep boreholes, to be done as with vitrified HLW from reprocessing, is being pursued as the back-up concept. For this purpose the spent fuel package has to have the same outer dimensions as canistered HLW and holds fuel rods from half a fuel assembly cut into 1-m-long pieces (Table I). However, in this case, medium-level heat-generating waste arises through the compaction of the fuel assembly skeleton (~ half a 400-l-drum per t of spent fuel) and has to be disposed of apart from spent fuel.

2.2. Demonstration program

Conditioning of spent fuel and canister development are the responsibility of DWK. Last year, DWK submitted the license application for a pilot conditioning plant at a site at Gorleben adjacent to the existing interim storage facility for spent fuel. On the other hand, R&D pertaining to disposal of spent

TABLE I. CHARACTERISTICS OF LWR DISPOSAL PACKAGES

	Spent Fuel Package Large	Spent Fuel Package Small	HLW Package
Length (m)	5.35	1.33	1.33
Diameter (m)	1.57	0.43	0.43
Weight (t)	64	1.2	0.5
Number of fuel assemblies	8	0.5	2.2
Heat load (kW)[a]	6.5	0.4	2.0
Surface dose rate (mSv/h)[a]	<0.2	$<10^4$	$\sim 10^6$
Emplacement	drift	borehole	borehole

[a] 10 years after discharge from LWR

fuel is funded by the Ministry for Research and Technology (BMFT). This R&D-program is coordinated by KfK; major contributors are BGR (Bundesanstalt für Geowissenschaften und Rohstoffe), DBE (Deutsche Gesellschaft zum Bau und Betrieb von Endlagern für Abfallstoffe), GSF (Gesellschaft für Strahlen- und Umweltforschung), and KfK.

Points of concern where further development is deemed necessary are indicated in Fig.3. They are mainly determined by the heavy load which has to be lowered through the access shaft and handled in the underground tunnel system. To be more specific, the following three tests have reached the stage of advanced planning:

- Up to now, shaft transport for weights up to 80 t (POLLUX+ transport vehicle) has not yet been demonstrated but existing technology can be adapted. Loading and unloading of the hoisting cage will be simulated in an above-ground test rig.

- Safe handling of the disposal casks in the tunnel system has to be demonstrated. Machinery for transport in the tunnel and emplacement on the tunnel floor will be developed. Both above-ground and underground testing will be done employing a full-size canister dummy.

- Improved knowledge of both salt and back-fill behavior under the influence of heat is an essential prerequisite

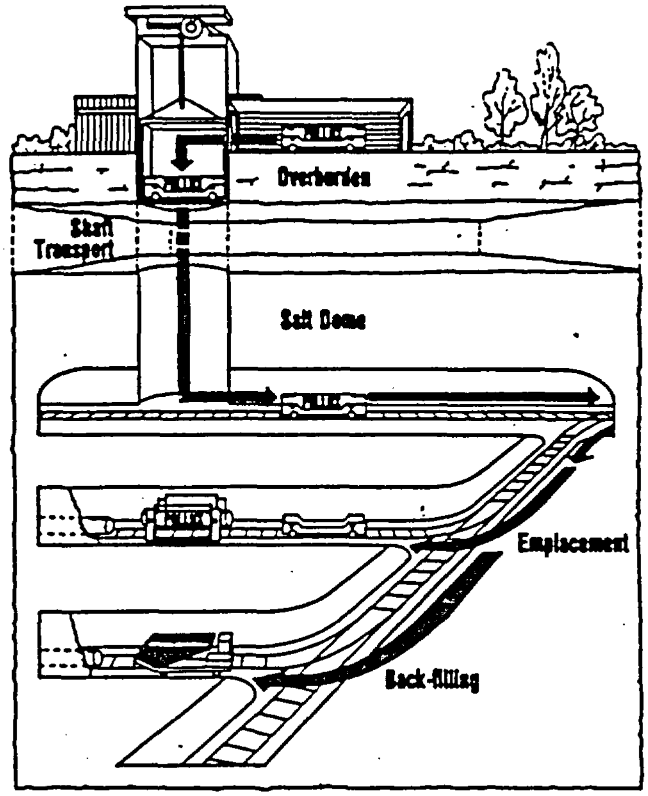

FIG.3.POLLUX underground transport and emplacement

for a licensing application. For that purpose a test field that includes two parallel tunnels will be excavated in the Asse underground laboratory, each tunnel accommodating three electrically heated, original-size canisters surrounded by crushed salt. Temperatures, pressures, room closure rates, heat flow, compaction of the back-fill etc. will be recorded and compared with results of model calculations.

As alluded to earlier, beside the large spent fuel package, the back-up concept of a small canister is to be pursued, identical in size with the HLW package and earmarked for emplacement in 300-m-deep vertical boreholes. A test with such a 43-cm-diam canister is planned in the Asse mine in order to demonstrate proper functioning of the components. The test canister will be filled with Cf-252 emitting neutrons with similar spectrum as those from spent fuel. This will allow measurements of neutron dose rates with special emphasis on back-scattering effects.

Completion of these four tests is scheduled for the beginning of the 1990s so that their results can be included in the license application for the Gorleben repository.

2.3. Experimental program

In the introductory section of this presentation it was already pointed out that a comparative study dealing with fuel reprocessing and direct disposal had been conducted prior to 1985. In this study a major issue was the long-term safety of the repository and, consequently, the performance of different barriers in the repository, especially the canister material and the waste form. Both corrosion and leaching behavior were under investigation in this first phase of the spent fuel disposal program.

As to spent fuel leaching, LWR fuel with a burnup of about 35,000 MWd/t was subjected to salt brines at 100 and 200 °C. The results revealed the well known phenomena of instantaneous gap release of the fission products, followed by incongruent element release until the easily accessible grain boundaries are cleansed, leading into the final phase of congruent release of matrixbound radionuclides. Among the two brines tested, the quinary brine - a brine with a high content of $MgCl_2$ - proved to be more aggressive than the NaCl brine unless considerable amounts of iron were included. With iron in the near-field - as is the case in the reference canister concept - element concentrations in solution are markedly reduced.

In the newly started program, higher burnup fuel of about 50,000 MWd/t will be studied. At least partial credit will be taken from canister lifetime, leading to lower test temperatures (~ 150 °C) and, hopefully, greater experimental ease.

3. DIRECT DISPOSAL OF SPENT HTR FUEL

3.1 Technical concepts

For direct disposal of spent HTR fuel two principal technical solutions exist which, according to the present state of knowledge, both comply with preliminary safety and operating requirements:

- packaging of the fuel spheres in standard 400-l-waste-drums and emplacing the drums in 300-m-deep boreholes,

- packaging of the fuel spheres in special canisters (modified POLLUX) and emplacing in drifts as back-up concept.

Based on the concept developed in the early 1980s for direct disposal of spent LWR fuel, packaging in large canisters had been judged to meet regulatory requirements for licensing power operation of the 300 MW prototype thorium high-temperature reactor, the THTR. A modified POLLUX cask can accommodate the contents of 4 THTR storage cans each holding 2,100 spherical fuel elements. This type of packaging is included in DWK's planned multi-purpose pilot conditioning plant at Gorleben so that this option is kept open.

Packaging of spent HTR fuel spheres in standard waste drums and disposal in 300-m-deep boreholes, comparable to the procedure for medium-level reprocessing waste, is more economic and benefits from special HTR fuel properties such as:

- low specific activity,
- high leach and corrosion resistance,
- the multiple barrier concept of coated particles that enhances fission product retention.

Therefore, this concept has been defined as the reference concept for disposal of spent HTR fuel. In such a 400-l-drum with 1,800 fuel elements, heat generation 10 years after discharge from the THTR amounts to about 90 Watt (Table II). The surface temperatures of the drums within the boreholes will be

TABLE II. CHARACTERISTICS OF HTR DISPOSAL PACKAGE

	HTR-Package
Length (m)	1.08
Diameter (m)	0.71
Weight (t)	0.4
Number of fuel elements	1800
Heat load (kW)[a]	0.09
Surface dose rate (mSv/h)[a]	4×10^4
Emplacement concept	borehole

[a] 10 years after discharge from HTR

about 70-80°C. Leaching of fission products (Cs, Ba, Eu, Ce, Sr-isotopes) in salt brines in case of water intrusion into the disposal area seems to be negligible for intact coated particles. Corrosion-caused damages of coatings under disposal conditions can be excluded for at least 500 years. Results to date suggest that releases of gaseous fission products (H-3, Kr-85) from intact fuel spheres under repository operating conditions will be acceptably low.

The further development of the technical concept concentrates on the influence of mechanical loads on the integrity of the fuel spheres. These loads are due to drum stacking and lithostatic pressures at depths of 800-1,000 m. If significant mechanical damage of coated particles occured during the operating phase or after sealing of the boreholes, causing unacceptable activity releases (H-3, Kr-85, enhanced leachability), backfilling of the hollow spaces between the fuel spheres with suitable materials such as sand or crushed salt might become necessary.

3.2 Demonstration program

Direct disposal of spent HTR fuel will be demonstrated during the course of an R&D-program sponsored by the BMFT and to be carried out by the Kernforschungsanlage Jülich (KFA) in cooperation with other German organizations and firms [4]. The main objective of the program is to develop disposal techniques for heat generating medium-level waste and spent HTR fuel in unlined vertical boreholes in rock salt. Prototype components, e. g., the loading machine and the sealing technique will be developed. These techniques are to be tested during a 5-year retrievable disposal experiment in the Asse salt mine under real operating conditions.

The program, which started in 1983, is intended to demonstrate by 1993:

- safe underground handling of waste containers in removable shielding,
- safe emplacement in boreholes up to 50-m-deep with a diameter of 1 m,
- retrievable disposal of active material in three 10-m-deep boreholes (6 MLW-drums, 4 stainless steel cans of AVR[1]-TL-type, see Fig.4),
- agreement of measured and calculated gas releases (H-1, H-3, Kr-85), temperature distributions, borehole convergence, and waste-salt interactions,
- gas-tightness of borehole sealings.

Furthermore, before emplacement the drums will be characterized with respect to heat generation, gas release and dose rates.

3.3 Experimental program

The current experimental program for direct disposal of HTR fuel spheres includes the following activities:

- evaluation of data stemming from the AVR dry storage facility and from dry storage casks (CASTOR, TN-AVR),

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1 Arbeitsgemeinschaft Versuchsreaktor (the German pebble-bed research reactor)

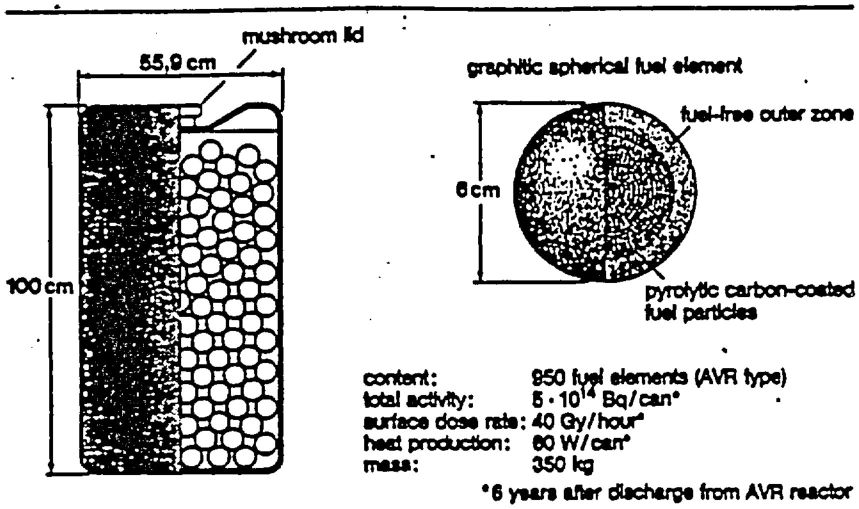


FIG.4.Retrievable disposal test: stainless steel can (AVR-TL-type) with spent HTR fuel elements

- measurements of H-3/Kr-85 release from fuel elements at elevated temperatures (40-400°C),
- leach tests of spent HTR fuel elements with salt brines,
- measurement of radiolysis gas formation under normal disposal conditions and accident scenarios,
- investigation of mechanical interactions between waste packages and salt during borehole convergence,
- full scale testing of waste packages and development of backfilling techniques.

This experimental program is to supplement the demonstration program to gather all relevant data so that borehole emplacement of spent HTR fuel can be established at the Gorleben site. The data accumulated up to now support the economical reference concept for the direct disposal of spent HTR fuel elements in standard waste drums.

4. SYSTEMS STUDY DUAL-PURPOSE REPOSITORY

According to the German Entsorgung policy, direct disposal of spent fuel is mandatory for the entire fuel of the HTR program, but only for certain types of LWR fuel such as recycled uranium, certain MOX, and high-burnup fuel beyond the specifications of the Wackersdorf reprocessing plant. This would amount to less than 100 t/a within a total of approximately 700 t/a discharged from German LWRs by the year 2000. Nevertheless, varying ratios of spent LWR fuel that is to be disposed of directly or reprocessed, respectively, are being considered in the ongoing systems study "dual-purpose repository". In this study a model-repository in salt is to accommodate both reprocessing waste and spent fuel. The goal is to determine an optimized system by chosing from among above-ground and underground concept variants which are generated by varying pre-emplacement cooling times and repository design alternatives [5]. The repository design alternatives that are subject to optimization include a mix of drift and vertical borehole emplacement configurations. In a first step the reference emplacement technique has been employed, namely:

- POLLUX canisters with spent fuel on the floor of placement drifts

- vitrified HLW in vertical 300-m-deep boreholes

for which two different configurations were assumed:

- the vertical boreholes and the emplacement drifts are at separate locations, or

- the vertical boreholes are drilled in the floor of the POLLUX emplacements drifts.

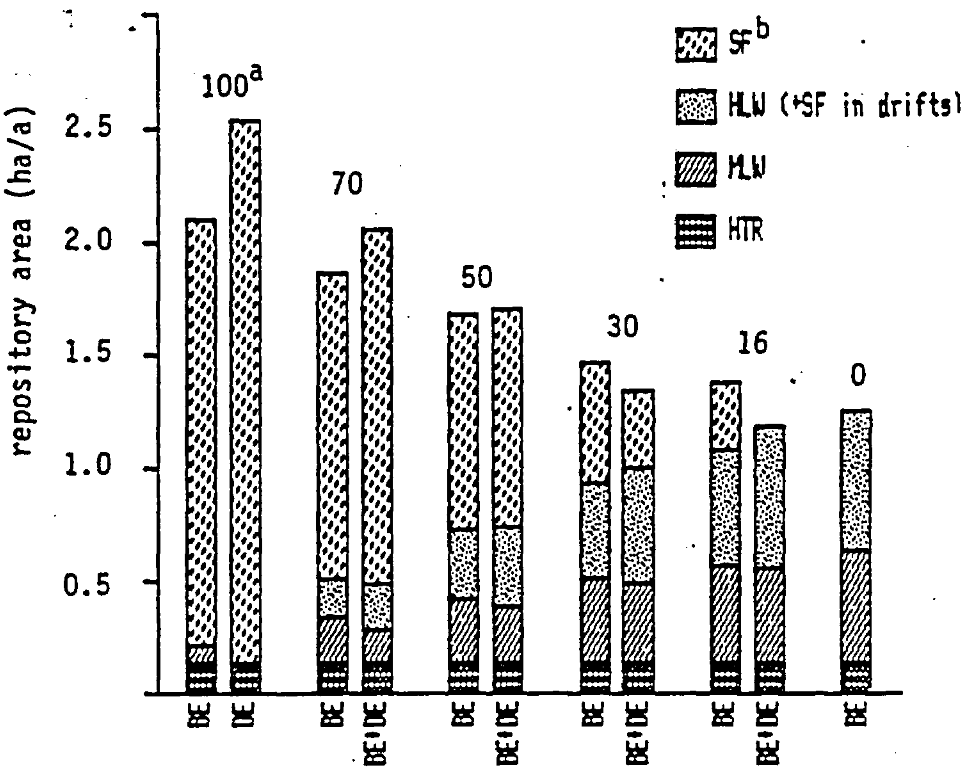
The heat-generating medium-level waste (MLW) and the spent fuel from the HTR-program, both with about equal heat generation rates, are emplaced in a common field in 300-m-deep boreholes.

In a second step other configurations are being considered, either both spent fuel and reprocessing waste only in drifts or only in horizontal boreholes.

Beside a near-field maximum rock salt temperature of 200°C, no binding criteria exist for the design of the repository concepts so far. For instance, permissible stress limits at the boundary between host rock and overlying strata will not be established before the underground exploration of the Gorleben salt dome is terminated.

Taking full advantage of the maximum 200°C rock salt temperature, a result as presented in Fig.5 was obtained. The calculations were based on a total of 700 t of spent fuel from LWRs and 10^6 HTR fuel elements - the equivalent of about two HTR-500s - to be managed annually. The ensuing repository area required each year is shown for various percentages of unreprocessed LWR fuel which is disposed of directly. The bar on the extreme right (0) represents the areal requirement in case all LWR fuel is reprocessed and the resultant vitrified HLW and the heat-generating MLW are emplaced in vertical boreholes. For all other fractions, bars are plotted for two emplacement configurations: one for pure borehole emplacement and the other representing horizontal emplacement of spent LWR fuel in those drifts where the vertical boreholes are drilled; however, for fractions of spent LWR fuel larger ~16%, additional drifts have to be excavated to accommodate LWR fuel in POLLUX canisters.

In the case of reprocessing, heat-generating MLW has to be emplaced in boreholes together with spent HTR fuel but in a section of the repository separate from vitrified HLW. The HTR re-



[a] percentage of spent LWR fuel to be disposed of directly

[b] spent fuel

BE: borehole emplacement
DE: drift emplacement

FIG.5. Annual repository area requirements

quires an area of ~0.13 ha/a. The areal requirement of the borehole field for MLW and HTR fuel is not determined by temperature considerations but by constraints reflecting the state-of-the-art in mining techniques that allows a minimal spacing of 25 m for 300-m-deep boreholes.

According to Fig.5, the areal requirement in the repository will be smaller than 2.5 ha/a. Given a width of 300 m for the emplacement field, a total length of roughly 4 km will not be exceeded even in the least favorable case after 50 years of emplacement.

Beside the repository area required for emplacement, other criteria are being assessed in order to select an optimized system. As there is a trade-off between above-ground pre-emplacement storage time and repository area requirements, the economics of the entire system have to be quantified taking into account costs of canisters, storage, and excavations. Radiological safety of the work force, long-term safety of the repository, and far-field effects are further examples of selection criteria. The schedule for the systems analysis dual-purpose repository is illustrated in Fig.6.

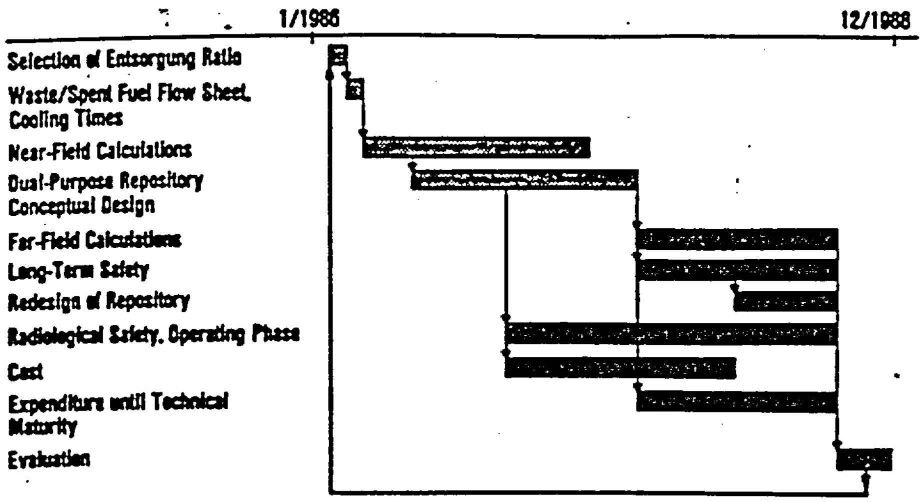


FIG.6.Systems analysis dual-purpose repository

Subsequent to this analysis the two most promising concepts will be subject to more detailed planning past 1988.

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