DIRECT DISPOSAL OF SPENT LWR-FUEL: STATE OF DEVELOPMENT IN THE FR GERMANY

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ABSTRACT

As stipulated by the German Atomic Energy Act, reprocessing is the reference waste management route for LWR's in the Federal Republic of Germany (FRG). After abandoning the construction of a domestic reprocessing plant, the FRG will mainly rely on foreign reprocessing services for the next decades.

Spent fuel disposal without reprocessing is being developed to technical maturity for those fuel elements for which reprocessing is either technically not feasible or economically not justifiable. An ambitious "R&D Program on Direct Disposal" has been launched which covers the time period 1986 to 1994. A pilot conditioning and encapsulation plant will be built at Gorleben, and repository related demonstration tests will be performed on a 1:1 scale. Moreover, layout and optimization studies for a common repository for both reprocessing waste and spent fuel are under way. All findings will be available before the licensing procedure for the FRG's first repository for heat-generating nuclear wastes will start. This means that direct disposal of spent fuel could be introduced in the future in addition to the reprocessing and recycling waste management route. A safeguards concept for direct disposal still has to be elaborated, but since direct disposal has become more attractive worldwide in recent years, an international consensus on open questions is probably going to be reached soon.

INTRODUCTION

As stipulated by the German Atomic Energy Act, reprocessing is the reference waste management route for LWR's in the Federal Republic of Germany (FRG). After the utilities abandoned the construction of a domestic reprocessing plant in mid-1989 for economic reasons, the FRG will rely on reprocessing contracts with France and Great Britain for the next decades (1, 2). The waste stemming from those reprocessing contracts will be transported back to the FRG. The heat-generating waste will be disposed of in a repository located in a salt dome.

Direct disposal of spent LWR-fuel has been studied in the FRG since 1979. After the state of the art was summarized in 1980 (3), a technical concept for direct disposal was developed and a comprehensive comparison of the two waste management options with and without reprocessing was performed between 1981 and 1984. Based upon the results of those studies (4), the Federal Government decided in January 1985 that, in addition to the realization of the fuel cycle via reprocessing, spent fuel disposal without reprocessing be developed to technical maturity for those fuel elements for which reprocessing is either technically not feasible or economically not justifiable (5). For this reason, an ambitious "R&D Program on Direct Disposal" has been launched which covers the time period 1986 to 1994.

Up to 1984, the funds for R&D in the direct disposal area were provided almost exclusively by the Federal Ministry of Research and Technology (BMFT). After the technical feasibility of direct disposal had been demonstrated in principle by the Systems Study Alternative Spent Fuel Management Technologies (4), the responsibilities were newly assigned. The contribution by the Federal Government is primarily limited to the area for which the Federal State is responsible under the Atomic Energy Act, namely repository related activities. Cask and spent fuel conditioning technologies are to be developed by the industry, represented by DWK, the German reprocessing company.

Activities concerning the repository are coordinated on behalf of the BMFT by the Alternative Spent Fuel Management Technologies Project Group (PAE) established at the Karlsruhe Nuclear Research Center. In addition, the Karlsruhe Nuclear Research Center participates in the advancement of direct disposal within the framework of its basic funding. The Deutsche Gesellschaft zum Bau und Betrieb von Endlagern für Abfallstoffe (DBE) has been commissioned to carry out the demonstration experiments and to plan the repository. The Gesellschaft für Strahlen - und Umweltforschung (GSF) makes available the former ASSE salt mine for some of the planned tests and participates in conducting those tests. Also the Federal Institute of Geoscience and Natural Resources (BGR) contributes to these advanced efforts. All projects are coordinated with the Bundesamt für Strahlenschutz (BfS) which acts as future owner of the repository on behalf of the Federal Government.

This paper presents an assessment of the state of the art of direct disposal at the present time, i.e., after roughly half the term of the "R&D Program on Direct Disposal" has elapsed.

TECHNICAL CONCEPTS

Two variants are being pursued in the continued activities concerned with direct disposal of spent LWR-fuel elements:

- POLLUX casks for disposal in drifts (reference concept).
- POLLUX canisters for disposal in boreholes (back-up concept).

The spent fuel conditioning and encapsulation techniques for both variants are schematically shown in Fig. 1.

The process starts with disassembling the fuel assemblies. In the reference concept (shown on the left-hand side) intact consolidated rods together with the compacted skeletons of the fuel assemblies are placed into the fuel disposal cask which is surrounded by a shielding overpack. Such a package is called a POLLUX cask, more details of which are shown in Fig. 2.

The POLLUX cask is an advanced version of the con-

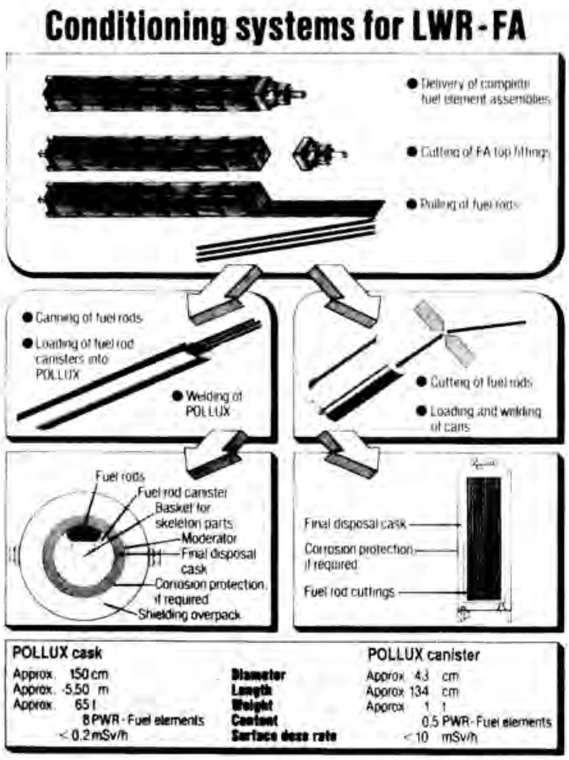

Fig. 1. Conditioning techniques for LWR fuel assemblies.

Fig. 2. POLLUX cask.

cept pursued from 1981 to 1984, which constituted the basis of the comparison of the two back-end fuel cycle concepts with and without reprocessing. The POLLUX cask which is designed for transport as well as for interim storage and disposal fulfills the IEAE requirements (Type B(U)). The multiple shell structure of the cask (disposal cask, corrosion protection coating - if required - and shielding overpack) permits flexible adaptation to the requirements ultimately imposed upon the package system by the repository. The shielding overpack is designed such that the personnel in the conditioning and encapsulation plant as well as in the repository can handle the package hands-on. The POLLUX cask, which may accommodate, e.g., the fuel rod and structural parts of a total of eight PWR fuel assemblies, is approx. 5.5 m long, has a diameter of approx. 1.5 m, and weighs some 65 MT in its loaded condition.

The handling and emplacement procedure for POLLUX casks at the repository is shown in Fig. 3. After having been shipped by rail or road to the surface facilities, the cask passes through the entrance control in the transfer hall. Next, it is placed on a railroad car which is conveyed to the shaft and pushed into the shaft cage. After shaft hoisting the loaded railroad car is pulled from the hoisting cage at the shaft landing. There, the loaded railroad car is taken over by a mine locomotive and transferred along the pilot heading to the emplacement drift. At the emplacement position, the cask is lifted from the railroad car by an emplacement device which is designed for lifting and depositing the cask on the floor after removal of the wagon. The railroad car is pulled out of the emplacement drift all the way back to the surface for reloading. The drift section accommodating the emplaced POLLUX cask is backfilled immediately with crushed salt.

Since handling such heavy (65 MT) and large (5.5 m long) casks in a repository mine is not state-of-the-art, a back-up solution for spent fuel conditioning and encapsulation resulting in smaller spent fuel packages is being pursued. This concept is shown on the right-hand side of Fig.

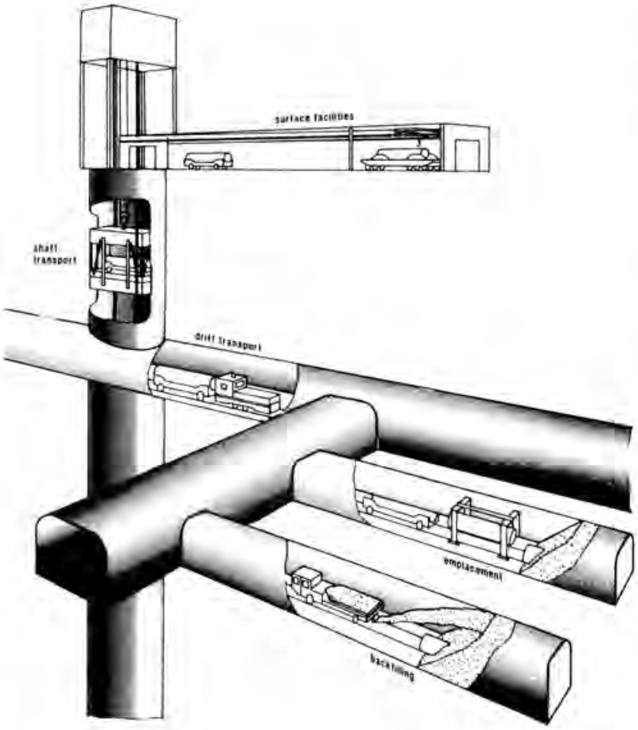

Fig. 3. Emplacement procedure for POLLUX casks.

1. The POLLUX canisters have the same outer dimensions as the canisters for vitrified reprocessing waste. In this case the disassembled fuel pins have to be cut into pieces of about 1 m in length. A POLLUX canister can hold the chopped fuel rods of half a PWR fuel assembly. Handling and, if necessary, intermediate storage of the POLLUX canisters requires additional shielded casks designed for this purpose. The advantage of this concept is that the handling and emplacement techniques in the repository are the same as for vitrified reprocessing waste, namely the emplacement of the canisters into 300-m-deep boreholes, drilled in the floor of the disposal drifts. The technique for handling such canisters in a repository has been developed and will be tested soon (6). This paper will not deal with this topic, but will mainly concentrate on such aspects associated with the POLLUX casks.

R&D PROGRAM ON DIRECT DISPOSAL

The "R&D Program on Direct Disposal," which lasts from 1986 to 1994, combines both the work of DWK on conditioning and package development and the repository related projects funded by the BMFT. In order to develop direct disposal of spent LWR-fuel to technical maturity, some DM 500 million will be spent altogether. Work on conditioning and package development will be financed by DWK. The funds required for this activity, including the construction of the pilot conditioning plant, are currently estimated at approximately DM 400 million. The activities associated with the repository will cost approximately DM 100 million, of which some DM 75 million will be special funds made available by the BMFT, while approximately DM 25 million will come from the basic funding of KfK.

Conditioning and Package Development

The atomic license application for a pilot plant for spent fuel conditioning and encapsulation was submitted by DWK to the licensing authority in May 1986. The site selected is at Gorleben in Lower Saxony and is directly adjacent to the 1500 MT AFR interim storage facility of DWK, which is licensed for storage of spent fuel in special transport/storage casks, e.g., CASTOR casks. A perspective view of the planned pilot plant (PKA) as well as the existing AFR storage building is shown in Fig. 4.

The pilot plant is designed as a multipurpose facility, in which the conditioning and encapsulation technology for various forms of radioactive waste will be demonstrate. Due to the pilot function of the facility, the throughput is limited to 35 MT of spent fuel per year which is sufficient for the demonstration of the technology. The process building is about 60 m long, 50 m wide and 21 m high; its total volume of 80,000 m^3 includes a hot cell volume of 2,000 m^3.

The first partial construction permit was granted in late January 1990. In case construction of the plant will start soon, hot commissioning would be possible in late 1994.

The main components of the pilot conditioning plant have been optimized on the basis of non-radioactive tests. The basic functions of the following process steps were tested on a 1:1 scale in test rigs and demonstrated:

- Disassembling fuel assemblies;
- Compacting structural parts;

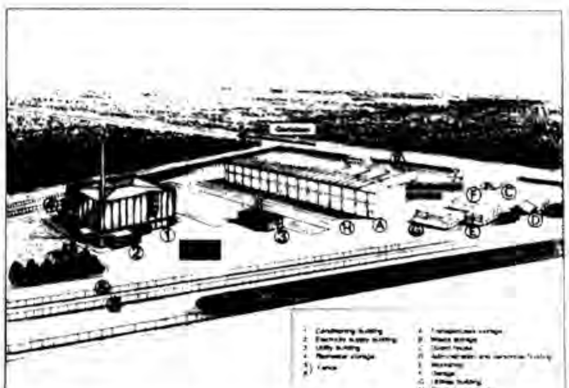

Fig. 4. Pilot plant for spent fuel conditioning.

- Loading and handling fuel rod canisters and baskets with compacted skeleton parts for POLLUX casks;
- Cutting fuel rods;
- Loading and handling POLLUX canisters.

Extensive development work for the POLLUX system has been carried out over the past few years, primarily on the optimization of welding techniques. An automatic welding station has been developed for producing the leaktight lid closure of the inner POLLUX container and the POLLUX canister, and an automatic system has been designed and tested for the production of the corrosion protection layer. A thermal load test was conducted with a POLLUX cask in 1987, which fully confirmed the thermal design basis calculations for the cask. Prototypes of a POLLUX cask and a POLLUX canister are currently manufactured.

Positive comments from official authorities are available for the POLLUX cask and POLLUX canister with respect to their principle suitability as disposal packages. Moreover, no difficulties are expected with respect to the suitability of the POLLUX cask as transport and interim storage cask. An official application will be filed soon with the competent authority by DWK.

Work Pertaining to the Repository

This part of the program, which is supported by the BMFT, is made up of the following projects:

- Demonstration tests of direct disposal;
- Planning activities for a dual-purpose repository concept which will accommodate both reprocessing waste and spent fuel;
- Experimental laboratory investigations.

Demonstration Tests of Direct Disposal

As it is a fundamental principle of the German atomic licensing practice to license only demonstrated technology, demonstrations of the most important steps in the transport and emplacement of POLLUX casks have become a key item of the program. The demonstration tests must show that:

- Payloads of up to 85 MT (65 MT for the POLLUX cask plus 20 MT for the in-plant railroad car) can be safely transported in the shaft = "Simulation of Shaft Transport" demonstration tests (SST);
- Repository packages weighing 65 MT can be handled and deposited safely below ground = "Handling Tests for Drift Emplacement" demonstration test (HHV);
- The thermal and thermomechanical behavior of the rock salt surrounding the disposed packages can be predicted with the same accuracy in the case of drift emplacement as in borehole emplacement = "Thermal Simulation of Drift *Emplacement*" demonstration test (TSS).

As no shaft hoisting equipment with a payload of 85 MT has so far been built anywhere in the world, a shaft hoisting system was designed for this very purpose in the first stages of the "Simulation of Shaft Transport" demonstration test (SST). Requirements were defined and design parameters determined for the most important components and plant systems. In order to find out which of these components and plant systems can be documented to exist in reference plants, 93 existing shaft hoisting systems were analyzed in detail, 47 of these in the Federal Republic of Germany. The conclusion of this investigation was that the readiness for application of almost all major components and plant systems of a shaft hoisting unit for 85 MT payload can be demonstrated by reference plants. Only for some components, such as the cage, the diagonal arrangement of the guide rails, intermediate cage bottom locking during loading and unloading, mechanical overrun protection, and a charging unit for push-on and withdrawal of the railroad car, this demonstration still needs to be produced (7). This

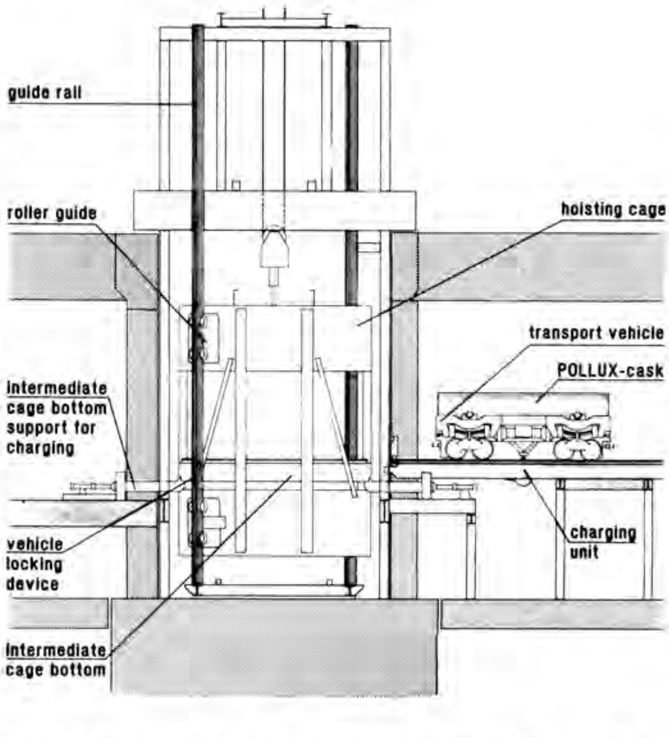

Fig. 5. Test rack for the "Simulation of Shaft Transport" demonstration test (SST).

will be done in a test rack above ground to be established in the turbine hall of a power plant at Landesbergen, Lower Saxony. A diagrammatic sketch of the planned test rack which is now under construction is given in Fig. 5. The tests will be performed in 1991.

Within the framework of the "Handling Tests for Drift Emplacement" (HHV), the railroad car and the emplacement device will be developed and tested first above ground (1991/92) and, later on, below ground (1993/94). The construction of the components is under way. In Fig. 6, the emplacement device and the railroad car for the POLLUX cask are shown.

The "Thermal Simulation of Drift Emplacement" demonstration test (TSS) serves to study the properties and the behavior both of the backfill material, crushed salt, surrounding the large and heavy POLLUX casks, and of the rock salt under the influence of heat and the resultant temperatures leading to elevated stresses and salt creep. For this purpose, two parallel drifts have been excavated in the Asse mine (see Fig. 7). In each drift, three electrically heated dummy containers of the dimensions and weights of the real POLLUX casks will be installed and backfilled with crushed salt (8). Basically, this test of drift emplacement is a reiteration of what was done in the 1970's on borehole emplacement in the heater tests conducted in the Asse mine, namely the validation of computer codes and constitutive laws (6). Another purpose was to select in preparatory tests a suitable method of backfilling to be employed in the main test.

The preparatory tests were run successfully in the Asse mine in the spring of 1987 (9). In the autumn of 1987, excavation of the drifts for the main test was begun. Meas-

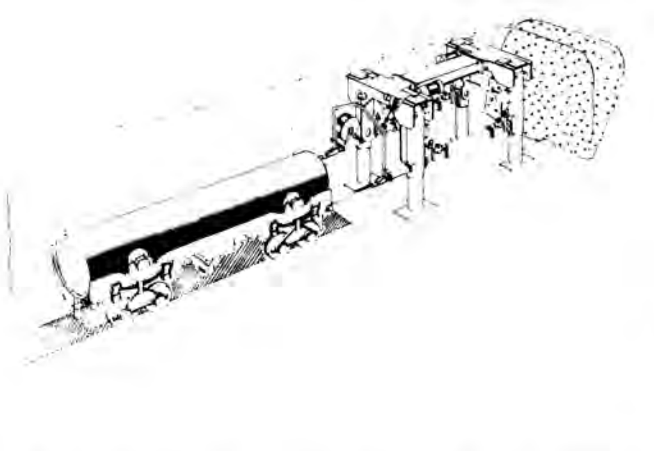

Fig. 6. Drift emplacement for POLLUX casks (HHV).

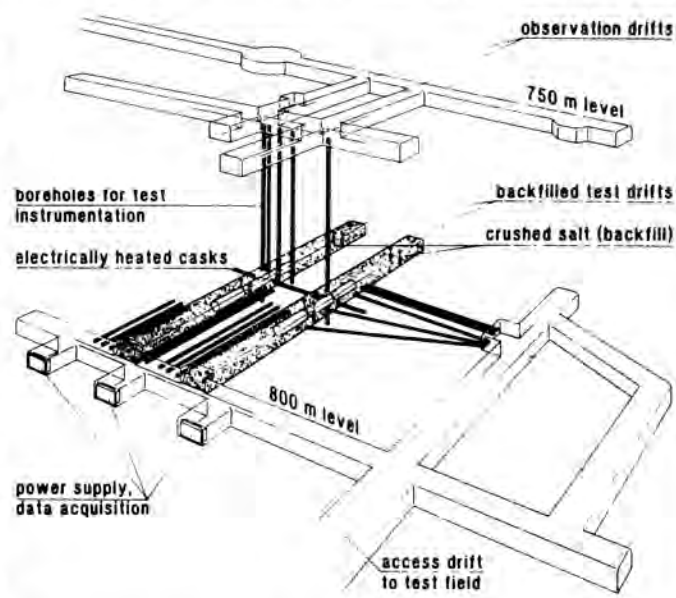

Fig. 7. Test field for the "Thermal Simulation of Drift Emplacement" simulation test (TSS).

urements of the rock condition have been run since late 1988. A total of 2,700 m of boreholes were sunk for the instrumentation. The heaters will be actuated in May 1990. The text will take at least three years. However, the technical components were designed in such a way that the test could be extended by another three years, if necessary.

Another demonstration test to be performed in the Asse mine will deal with the behavior of neutrons from disposal packages during handling in a repository, mainly with the effect of neutron back-scattering in drifts. This test will use a Cf-252 neutron source. It is called "Active Handling Demonstration Test with Neutron Source" (AHE). The design of this test is under way, and it will take place in 1992/93.

Planning Activities for a Dual-Purpose Repository Concept

The second project, "Planning Activities for a Dual-Purpose Repository Concept," primarily comprises systems analysis work. While so far only repository concepts have been developed in the FRG in which either waste arising in reprocessing was emplaced in boreholes or spent fuel elements were emplaced in drifts, several repository variants are now to be investigated in which waste from reprocessing and spent fuel will be accommodated. Besides a number of emplacement variants (such as pure borehole emplacement, pure drift emplacement or mixed borehole/drift emplacement), also variants of fuel element conditioning (POLLUX cask or POLLUX canisters), of HLW packaging and the conditioning of head-end waste (hulls, structural materials, and feed clarification sludge) are to be considered.

The first phase of this project, the "Systems Analysis Dual-Purpose Repository" (SYSDUR), is finished now, and the main results will be presented in another paper at this conference (10). The second phase, the "Systems Analysis Repository Concepts" (SYSCON), is under preparation. It will last until 1992 and will mainly concentrate on repository lay-outs in complex salt dome structures. Moreover, the consequences of disposing of high burn-up fuel, spent fuel from reprocessed and re-enriched uranium as well as spent mixed oxide fuel will be investigated in more detail. Special safety related aspects, i.e., hydrogen production and behavior will also be incorporated into the analysis. The outcome of these two systems analyses may be incorporated by BfS in detailed planning for the repository after the underground exploration of the Gorleben salt dome has been completed.

Experimental Laboratory Investigations

The purpose of this project is to make available to the plans approval procedure data on the effectiveness of technical barriers, such as the container material and the fuel. Such data are required for the analysis of the long-term safety of the repository.

Extensive corrosion studies of materials relevant to POLLUX indicate that the corrosion resistance of the Hastelloy C4 material so far planned for corrosion protection is very much influenced by the way in which the anticorrosive coating is manufactured. Whether corrosion protection with prolonged life is really required for the disposal packages, still needs to be analyzed in detail. The results of the leaching studies conducted so far with spent UO_2 fuel with a burnup of 52 GWd/MTU show the leaching behavior to depend very much on burnup and on the irradiation history. After six months, roughly 1% of the inventory of U, Pu, Am, Cs and Sr has been dissolved. From 1990, the leaching studies will be extended to include also reprocessed and re-enriched uranium and mixed oxide fuels.

CONCLUSION

Figure 8 presents a survey of the timetable covering the necessary R&D activities and projects of direct disposal specifically planned or envisaged for the future. The date of implementation of direct disposal in the FRG is determined by the commissioning of the repository planned for construction on the Gorleben site. This is not going to happen before the year 2008. The underground exploration of the Gorleben salt dome is likely to be finished between 1995 and 1999 (11). This will be followed by the plans approval procedure (licensing procedure).

According to the present timetable, all findings will be available in time before the beginning of the plans approval

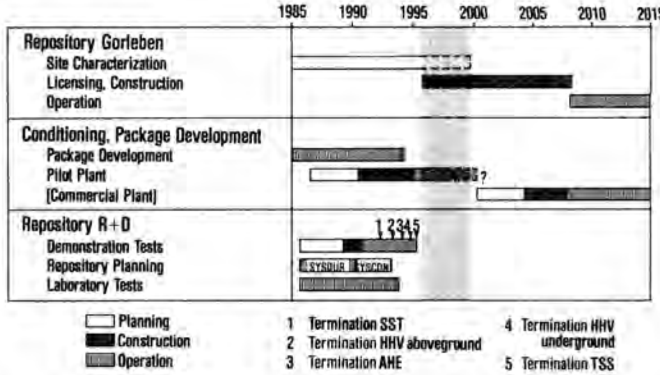

Fig. 8. Time schedule for direct disposal activities.

procedure so that the direct disposal strategy can be taken into account in the licensing procedure. This applies to hot operation of the pilot conditioning plant, which is able to produce all repository packages currently discussed, as well as tot he R&D work required in connection with the repository. All demonstration tests are to be completed by 1994.

If direct disposal of spent LWR-fuel would be included in the future FRG waste management concept, a large conditioning and encapsulation plant has to be built for the industrial use of this waste management strategy. To minimize transports of radioactive material, it would be desirable to build that plant on the site of the future repository. Whether the Gorleben salt dome is the proper site for a repository will be known by the turn of the millennium at the latest. By that date, sufficient experience will also be available from the operation of the pilot conditioning plant, which means that at that point in time a decision can be taken about the construction of a large conditioning plant. The plant then will be able to be commissioned at any rate prior to the commissioning of a large conditioning plant, will be necessary.

A still unresolved issue with respect to direct disposal is what a technical concept for safeguarding a repository with spent fuel might look like. The international safeguards authorities are becoming increasingly more engaged in nuclear materials safeguards associated with direct disposal. The same aspect is being taken into account more and more strongly in national development work. As direct disposal has become more attractive worldwide in recent years, an international consensus on open questions is probably going to be reached soon.

It is hoped, that this R&D program enables one to introduce direct disposal of spent LWR-fuel as an additional national back-end fuel cycle strategy in order to reach a high flexibility with respect to waste management in the FRG.

REFERENCES

1. European Concept of Shared Cooperation in Nuclear Power, Press release No. 44/89 by the German Federal Minister for the Environment, Conservation of Nature, and Reactor Safety, June 6, 1989 (in German).
2. Shared Cooperation in Nuclear Power Agreed upon with the United Kingdom, Press release No. 66/89 by the German Federal Minister for the Environment, Conservation of Nature, and Reactor Safety, July 25, 1989 (in German).
3. CLOSS, K. D., et al., "Comparison of Various Spent Fuel Management Options and Assessment of Their Feasibility, Spent Fuel Management Options Study," KfK 3000, Karlsruhe Nuclear Research Center (1980), (in German).
4. Systems Study of Alternative Spent Fuel Management Technologies, Main volume (KWA-2190/1), Karlsruhe Nuclear Research Center (1984), (in German), Translated by Language Services, ORNL/TR-86/31.
5. Bulletin of the German Federal Government No. 8 (1985), January 25, 1985, p. 66 (in German).
6. KUHN, K. and ROTHFUCHS, T., "In Situ-Experiments on the Disposal of High-Level Radioactive Wastes (HAW) at the Asse Salt Mine Federal Republic of Germany," Waste Management '89, Tucson, Arizona, Vol. 1, p. 567-573, Arizona Board of Regents (1989).
7. HARTJE, B., SCHRIMPF, C. and WEBER, W., "Technical Maturity of Shaft Hoisting Facilities for the 65 t-heavy Casks with Spent Fuel," Proc. Conf. on Uranium and Electricity, the Complete Nuclear Fuel Cycle, Saskatoon, Canada (1988), p. 4-39.
8. BECHTHOLD, W., HEUSERMANN, S., SCHRIMPF, C. and GOMMLICH, G. E., "Large-Scale Test on In-Situ Backfill Properties and Behavior under Reference Repository Conditions," Proc. NEA-CEC Workshop on Sealing of Radioactive Repository, Braunschweig, May 1989, p. 111.
9. ENGELMANN, H. J., HARTJE, B., SCHRIMPF, C. and WILDT, K.-H., "Thermal Simulation of Drift Emplacement, Evaluation of the Backfilling Technique," Waste Management '88, Tucson, Arizona, Vol. 2, p. 741-744, Arizona Board of Regents (1988).
10. PAPP, R. et al., "Results of the System Analysis Dual-Purpose Repository," Waste Management '90, Tucson, Arizona, February 25 - March 1, 1990.
11. PTB Info Blatt 2/88, November 29, 1988 (in German).