SRP REPRESENTATIVE TO THE FEDERAL REPUBLIC

OF GERMANY (FRG)

Donald E. Clark

Monthly Report for June 1987

Summary

A number of U.S. visitors were in the area for meetings and tours of the German facilities during this reporting period. The Karlsruhe Nuclear Research Center (KfK) was visited and field testing planned in connection with the program for direct disposal of spent fuel was discussed. A meeting on radiation effects on salt was held in Braunschweig and plans were made for testing of irradiated Asse Salt Mine samples from the Brine Migration Test (BMT). The May 1987 construction accident at Gorleben remained a subject of discussion in the FRG as the investigation into its cause and the development of a recovery plan for further shaft sinking were continued.

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.

Karlsruhe Nuclear Research Center (KfK)

A visit was made to the Karlsruhe Nuclear Research Center (KfK) primarily to discuss field testing planned in connection with the program for direct disposal of spent fuel (SF). At the KfK, the responsible group for managing the direct disposal R&D program is the Projektgruppe Andere Ensorgungstechniken, PAE, headed by Dr. K.-D. Closs. A copy of the R&D program plan for the PAE is included as Attachment 1. An update of this plan is currently in preparation, however, the attached version contains the essential aspects of the program, including the schedules for various work elements. All of this work is of some interest to the SRP, however, the thermal simulation test appears to have greatest potential for providing data that is directly relevant to U.S. repository conditions.

The PAE R&D program is structured as follows:

1. Demonstration Tests of Direct Disposal of SF

   - thermal simulation of tunnel emplacement
   - handling experiments for tunnel emplacement
   - simulation of shaft transport
   - borehole emplacement of SF canisters

2. Planning of Dual-Purpose Repository (SF and HLW)

   - systems analysis
   - in-depth planning of two selected repository concepts

3. Experimental Program

   - container material corrosion testing
   - retention of fission gases in backfill and seals
   - SF leach testing
   - combined leach testing (glass/SF/corrosion products)

The thermal simulation test (TSS) will be conducted in the Asse Salt Mine. This test was discussed in detail at the US/FRG workshop on geotechnical instrumentation held at the Asse Salt Mine in May 1987 (see DE Clark's monthly report for May). The conditions that will be simulated in this test are similar to what may prevail in a U.S. salt repository. Large containers (cylinders with a length of 5.5 meters and diameter of about 1.6 meters) will be emplaced in tunnels, backfilled with salt and heated internally to obtain a surface temperature of ca. 200 degrees Centigrade. Currently, two different backfilling techniques are being tried in the Asse Salt Mine using dummy containers. Thus, pneumatic stowing is being compared with a slinger (high-speed belt) stowing technique.

There is interest in having U.S. participation in the thermal simulation test. At a minimum, information exchange will be possible under the U.S./FRG bilateral agreement. Also, there may be opportunity to test U.S. geotechnical instrumentation under repository conditions in the TSS. Further discussions will be held concerning the TSS and possible opportunities for direct participation in the test.

Discussions were also held with the corrosion testing group at KfK. The most recent report of their work is KfK 4265 (May 1987) entitled "Corrosion Behaviour of Container Materials for the Disposal of High-Level Waste Forms in Rock Salt Formations" by E. Smailos, W. Schwarzkopf, and R. Koester. Data obtained in in-situ testing in the Asse Salt Mine are presented in this report. Copies of this and other pertinent reports will be sent back to SRP as they become available.

Dr. Closs is interested in reconvening the U.S./FRG Corrosion Workshop in 1988 during the week immediately following the Waste Management '88 meeting in Tucson, AZ. Many Germans plan to attend the Tucson meeting and it is felt that several special U.' -FRG meetings could usefully be held in conjunction with that e' .it.

While at KfK, the library facilities were visited and arrangements made to receive regular listings of all KfK publications. The PAE Semi-Annual Report for July-December 1986 will soon be available in English translation and copies will be sent back to the SRP. Other reports are beginning to be published in English as well as German, making them generally more useful to U.S. workers.

Attachment 2 is a list of papers that have been published to date on the PAE program.

Radiation Effects Meeting in Braunschweig

A meeting on the effects of radiation on salt was held at the Asse Salt Mine and in Braunschweig. Technical experts involved in the testing of Asse salt for radiation effects produced during the Brine Migration Test (BMT) in the Asse Salt Mine met and discussed the joint testing efforts for salt from the BMT site. A tour was made of the Asse Salt Mine and the participants were able to observe the BMT site firsthand. After much discussion, agreement was reached on a harmonized test plan for the BMT salt. A draft version of the minutes of this meeting is included as Attachment 3, however, it should be noted that this is subject to revision as it is currently being reviewed by the participants.

The appearance of the irradiated BMT salt differs from that of other salt that has been similarly irradiated (e.g., in the Lyons, Kansas salt mine). In particular, there is considerable variability in the coloration of crystallites, ranging from dark blue colored (due to presence of colloidal sodium) to that of little or no change in appearance from the unirradiated salt. This is so even for crystallites that are in intimate contact with one another and would have had the same radiation exposure. Two different views have been put forth to account for this observation. One view is that the coloration is related to trace element/mineral content; in this case, one should be able to correlate the coloration with chemical/mineralogical content of the crystallite. The other possible explanation is that the lack of coloration (i.e., decrease in colloidal sodium content) is due to recrystallization of the salt. Salt recrystallization, which should be directly related to water content of the crystallites, thus may serve as a "healing" process when rock salt is irradiated in the natural condition. The testing plans are intended to help clarify the observed condition of the irradiated salt, and lead to a better understanding of the effects of radiation on salt under actual repository conditions.

Discussions were also held with Paul Levy concerning salt radiation effects work at Brookhaven National Laboratory (BNL) that will be performed under direct funding by the FRG. It is intended that a 2-1/2 year program be performed at BNL for the Germans as a joint program under the US/FRG Bilateral Agreement. When the FRG program has been finalized, the terms of such a joint program will have to be negotiated with U.S. counterparts.

U.S. Visitors in June

Much of the remainder of this period was spent with meetings and tours with other U.S. visitors. In all cases, very productive interactions were had with the various FRG companies and institutes, and a considerable amount of useful information was exchanged. Summaries of these visits are contained in the trip reports submitted by the individuals who are listed below along with dates and places of their different visits.

- June 9, Jerry Saltzman (USDOE) and Robert Browning (USNRC) Gorleben candidate salt repository
- June 10, Robert Browning (USNRC), Asse Salt Mine and the Konrad Iron Ore Mine
- June 10, Paul Levy (BNL) and Larry Pederson (PNL), Asse Salt Mine

June 11-12, Paul Levy (BNL) and Larry Pederson (PNL), GSF-IfT, Braunschweig
June 22, Manny Comar (SRPO) and Ron Helgerson (ONWI), Asse Salt Mine
June 23, Manny Comar (SRPO) and Ron Helgerson (ONWI), Konrad Iron Ore Mine
June 24, Manny Comar (SRPO) and Ron Helgerson (ONWI), BGR, Hannover
June 25, Manny Comar (SRPO) and Ron Helgerson (ONWI), Gorleben candidate salt repository
June 26, Manny Comar (SRPO) and Ron Helgerson (ONWI), DWK, Hannover
June 29, Lynn Tyler (SNL) and John Stormont (SNL), GSF-IfT, Braunschweig
June 30, Lynn Tyler (SNL) and John Stormont (SNL), Asse Salt Mine
July 1, Lynn Tyler (SNL) and John Stormont (SNL), Gorleben candidate salt repository
July 2, Lynn Tyler (SNL) and John Stormont (SNL), BGR, Hannover

Status of the Gorleben Project

Shaft excavation activities were halted at the Gorleben candidate salt repository site during this reporting period as a result of the construction accident of May 12, 1987. Another press release was issued by the Federal Ministry for Research and Technology (BMFT) on June 9, 1987 (see Attachment 4). While the early press coverage of this accident had appeared to be fairly objective, the trend now seems to be otherwise. There now is increasing criticism in the media, probably in response to the more active role being assumed in this matter by anti-nuclear and opposing political factions. No new developments have occurred during this period. The accident is still under investigation by the competent authorities, and a technical review and recovery action plan is being developed by the Germans. The freezing plants continue to operate at both shafts. In the case of shaft #2, the continued cooling is necessary before further shaft sinking can occur, as was previously planned for the present stage of its development. As noted in the attached press release, the schedule for sinking of shaft #1 (where the accident occurred) will probably be delayed for at least several months until the investigation and technical reviews have been completed.

Planned Activities for July 1987

A major objective will be to continue to stay in close touch with the situation at Gorleben. A consultative visit to the U.S. is planned and this will provide an opportunity to brief SRPO on the current status of the FRG repository program.

Attachments

1. R&D Program for Direct Disposal of Spent Fuel
2. List of Publications on the PAE Program
3. Draft Minutes of US/FRG Technical Meeting on Testing BMT Salt for Radiation Effects
4. Press Release on Gorleben Accident by the BMFT

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ATTACHMENT 1

R&D Program Direct Disposal of Spent LWR Fuel

Compiled by

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Project Group Alternative Entsorgung
(KfK-PAE)
P.O. Box 3640
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Federal Republic of Germany

April 1986

Introduction

In its decision of January 23, 1985, the government of the Federal Republic of Germany ruled that direct disposal of spent fuel be further developed for fuel for which reprocessing is neither technically feasible nor economically justifiable. By that the government emphasized its intention to dispose of directly spent fuel from the German HTGR (high temperature gas cooled reactor)-program and, in addition, to bring to technical maturity direct disposal of spent LWR fuel. This program description will deal only with direct disposal of LWR fuel.

Federal funds will be used almost exclusively for the repository development program which is, according to the German atomic law, the responsibility of the federal government. The fuel cycle industry, namely DWK (Deutsche Gesellschaft zur Wiederaufarbeitung von Kernbrennstoffen, German Reprocessing Company) and utility companies, are responsible for the advancement of conditioning technology.

On behalf of the Federal Ministry of Research and Technology, PAE (Projektgruppe Andere Entsorgungstechniken, Project Group Alternative Entsorgung) has been installed at KfK (Kernforschungszentrum Karlsruhe, Karlsruhe Nuclear Research Center) and will co-ordinate all tasks within the framework of underground disposal of spent fuel. In addition, KfK contributes funds to the further development of spent fuel disposal. DBE (Deutsche Gesellschaft zum Bau und Betrieb von Endlagern für Abfallstoffe, German Repository Company) is in charge of both the demonstration tests and the repository design. GSF (Gesellschaft für Strahlen- und Umweltforschung, Radiation and Environmental Research Company) operates the Asse underground laboratory and will take part in the planned demonstration tests. BGR (Bundesanstalt für Geowissenschaften und Rohstoffe), the German equivalent to the USGS (US Geological Survey) will be involved in the experimental and the systems analytical program. Finally, with PTB (Physikalisch-Technische Bundesanstalt), the German equivalent to the US National Bureau of Standards, agreements will be sought with respect to the overall objective of the program as well as the documents required for each single task.

I. Demonstration Experiments of Direct Disposal of LWR Fuel

I.1 Thermal Simulation of Tunnel Emplacement

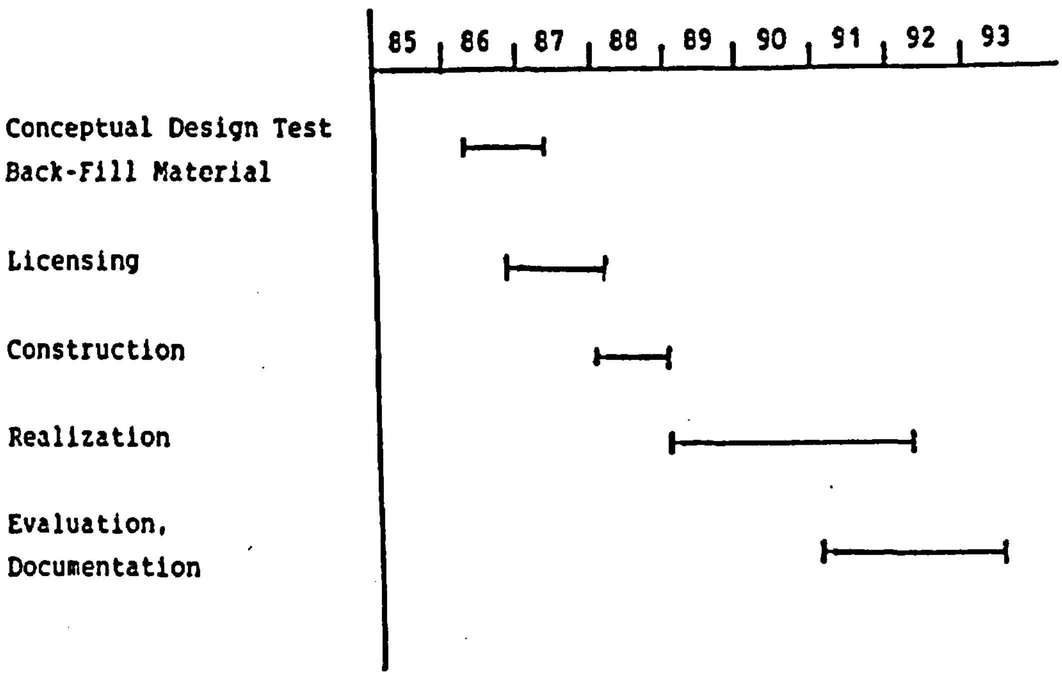
Objectives:

The reference concept for direct disposal of spent fuel consists in emplacement of large and heavy disposal containers on the floor of storage tunnels in a repository. Contrary to emplacement of reprocessing waste in boreholes, high temperatures are reached in the tunnels at an early point of time after emplacement of spent fuel packages. Licensing for direct disposal of spent fuel requires investigations of the behavior of both the salt and the back-fill material and comparison with results of model calculations. Within this experiment back-filling will be demonstrated and mathematical calculations will be verified by comparison with geothermal and geomechanical quantities measured in the undisturbed rock salt and in the back-fill material.

Realization:

In a screening test two methods of back-filling will be investigated (blowing, centrifuging) under realistic conditions. The more qualified method will be employed in the demonstration experiment, in which two parallel tunnels will be excavated in the ASSE underground laboratory, the tunnels having the same dimensions as the future disposal tunnels. In each tunnel, 2 to 3 electrically heated container dummies will be emplaced and backfilled. These dummies will have the same dimensions, weight, and heat production as the spent fuel disposal containers. For measurements of temperature profiles, stresses, deformations, moisture, and porosity instrumentation will be provided for containers, backfill material, and rock salt. The measured data will be used for comparisons with results of predictive calculations, which will be performed with computer codes developed for the layout of the repository. Special concern will be given to verification of input data for material laws of crushed salt.

Time Schedule:



Activities:

- Screening test: DBE (GSF)
- Handling: DBE
- Container dummies: DBE
- Scientific program
  . Layout calculations: KfK
  . Detailed test field layout: BGR (GSF, DBE)
  . Temperature measurements: GSF (BGR, DBE)
  . Thermomechanical measurements: BGR, GSF (DBE)
  . Back-fill material properties: BGR, GSF

. Corrosion tests: KfK

. Model validation: BGR (KfK, GSF)

- Planning, documents for licensing procedure, construction, realization, documentation: DBE
- Filing of licensing documents: GSF

I.2 Handling Experiments for Tunnel Emplacement

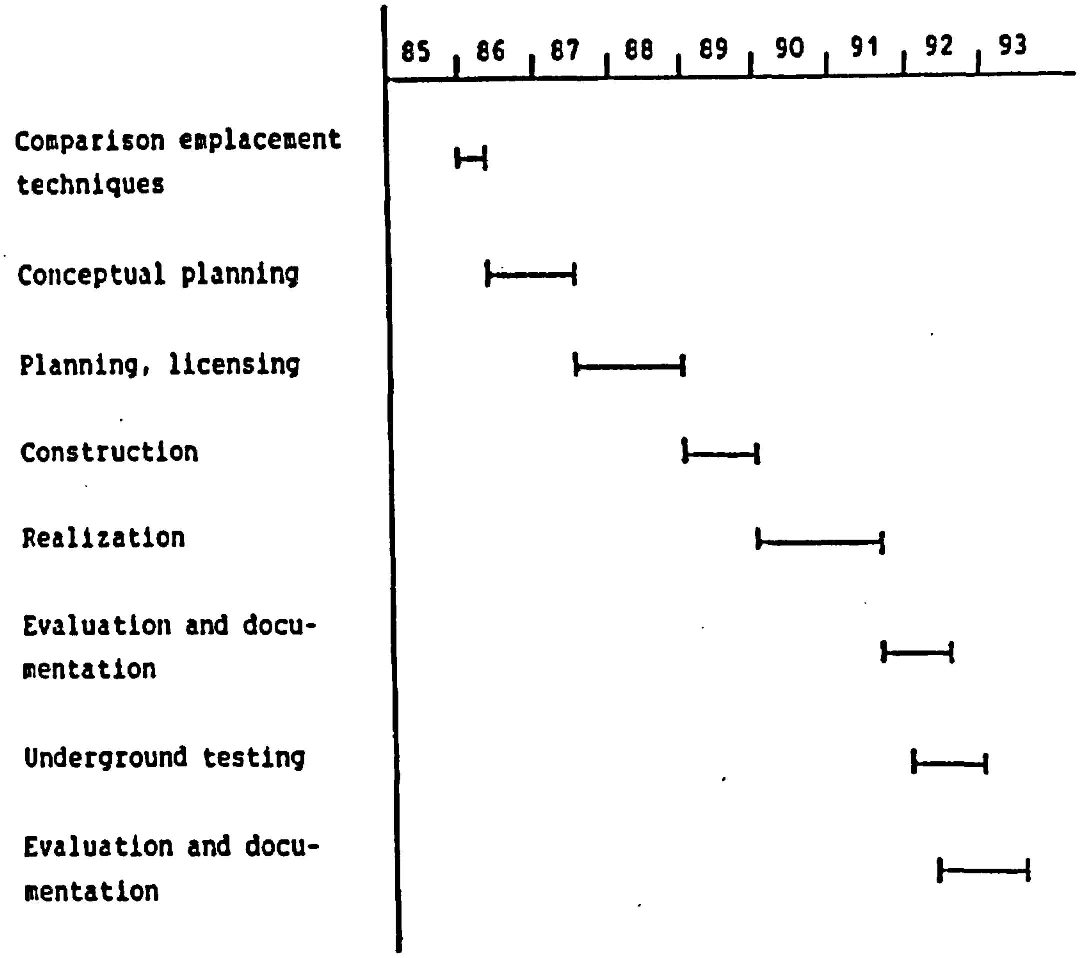
Objectives:

Licensibility of direct disposal of spent fuel in heavy containers is contingent upon safe handling in the repository. Equipment for both emplacement and transport to the emplacement tunnel has to be developed. Underground transport of an (inactive) full size package of original size and related handling steps have to be demonstrated. Optimization of tunnel geometry (radii, cross-sections) and of emplacement operations (waste handling personnel, occupancy distance and time) will be carried out.

Realization:

Based on actual DBE-planning for the emplacement vehicle, a concept optimization will be performed. The selected equipment will be constructed and tested above ground employing a full size container dummy. These tests will contribute to the optimization of tunnel geometries and emplacement operations. After successful demonstration above-ground, underground testing in the ASSE will be performed if deemed necessary.

Time-Schedule:



Activities:

- Design of transport and emplacement equipment (DBE)
- Planning, realization and documentation of the tests (DBE)
- Provision of test field and licensing procedure (if underground testing in the Asse): GSF

I.3 Simulation of Shaft Transport

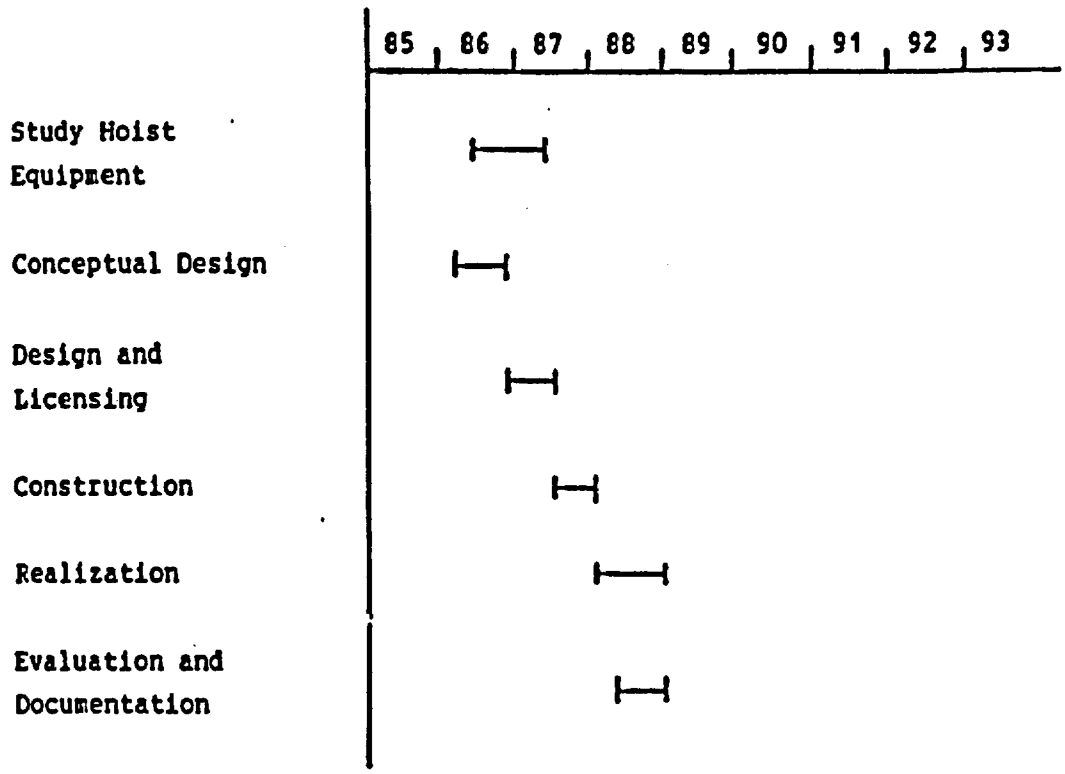
Objectives:

Licensing of direct disposal of spent fuel in large containers requires demonstration of safe handling and transport of big and heavy components in the hoist system. Since loading of the hoist cage with loads of up to 80 Mg has not been realized in existing facilities, loading and unloading of the hoist cage have to be tested in a demonstration experiment. For this purpose the components shall be ccnstructed and tested. Lateron handling in the hoist system shall be demonstrated under realistic circumstances.

Realization:

At first a study will be performed in which the state of technology of the equipment and facilities required for shaft transport of heavy components will be reviewed. The demonstration experiment will be run above ground by use of a hoist cage mock-up. A facility will be constructed and tested, which enables a controlled charge and discharge of the hoist cables. In these tests, which will be run as long-term tests, all possible operational conditions will be simulated. The results will be used as a basis for the designing of the entire hoist.

Time-Schedule:



Participant: DBE (Planning, construction, realization and documentation)

I.4 Borehole Emplacement of Spent Fuel Canisters

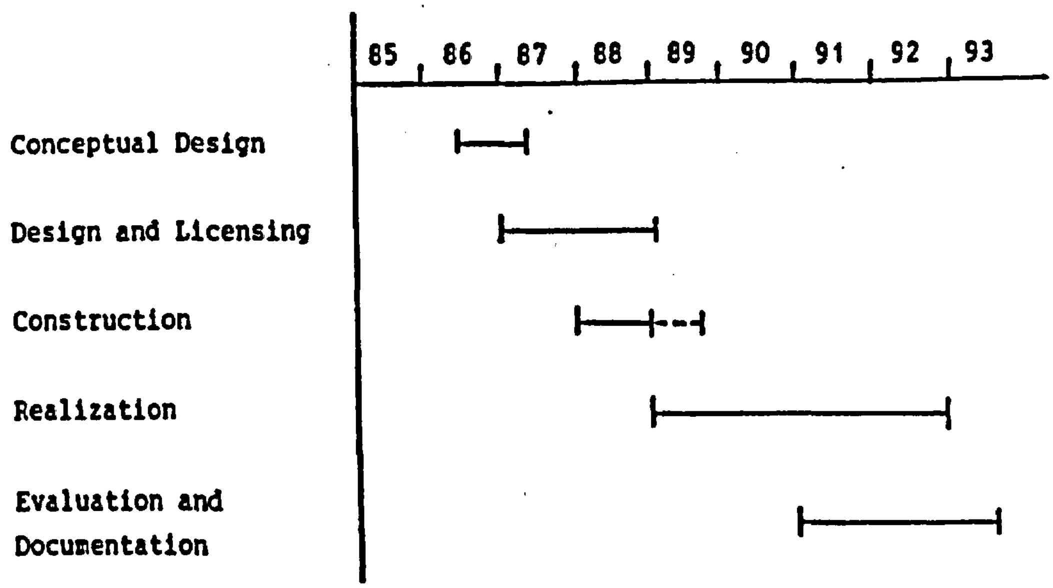
Objectives:

In addition to the reference concept of spent fuel conditioning, an alternative conditioning process is discussed that consists in cutting the fuel pins and encasing in canisters the size of HLW-canisters. Supplementary to the trial emplacement of HLW, an experiment will be performed with such a spent fuel package in the ASSE underground laboratory using the same technical equipment. The goal is to demonstrate that canisters filled with HLW or spent fuel can be handled and disposed of with the same handling equipment. Furthermore, it has to be demonstrated that no fission gases are released from spent fuel canisters. As spent fuel emits neutrons, the impact of neutrons on both radiation protection measures and the ambient rock salt could be investigated.

Realization:

One spent fuel canister with a high neutron source strength will be fabricated. The transport of the canister will be accompanied by extensive neutron dose rate measurements in differently sized rooms. These measurements will enable estimation of radiation doses of the operational personnel for various other emplacement concepts. The canister will possibly be emplaced retrievably in a vertical hole beneath the floor of a specially excavated tunnel. Borehole, tunnel, and ambient rock salt are equipped with monitoring devices as in the HLW-experiment in order to (1) investigate the impact of neutrons on the rock salt and the borehole atmosphere and (2) validate geothermal and geomechanical model calculations at elevated temperatures ($\sim 100^{\circ}C$). The long-term borehole emplacement and the rock salt investigations are still under discussion.

Time-Schedule:



Participants: DBE, GSF, BGR, KfK

DBE: Procurement of spent-fuel canisters; planning and realization of the experiment with emphasis on handling the spent-fuel canister; provision of licensing documents; overall documentation

GSF: In-depth planning, realization, evaluation, and documentation of salt-specific experiments; licensing procedure; excavation of experimental field

BGR: Thermomechanical calculations

KfK: Neutron dosimetry; thermomechanical calculations

II. Planning of the Repository for Spent Unreprocessed Fuel (SURF) and High Level Waste (HLW) (Dual-Purpose Repository)

II.1 Systems Analysis Dual-Purpose Repository

Objectives:

For a fixed ratio of HLW and SURF, an optimized waste management scheme will be developed. Beside different repository concepts, variations of spent fuel treatment and HLW-treatment (pre-emplacement, storage, and conditioning) will be subject to systems analysis.

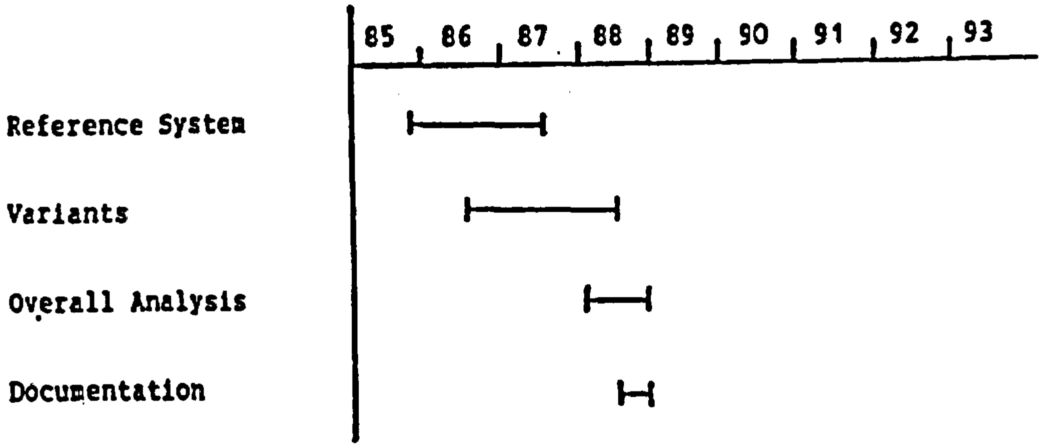
In order to select an optimized waste management scheme, the following criteria will be quantified: Radiological safety during the operating phase, long-term safety of the repository, cost, expenditure to reach technical maturity, geothermal and geomechanical characteristics, repository areal requirements.

The results will enable PTB to select an appropriate disposal concept as soon as characterization of the Gorleben salt dome and quantification of the waste streams will be concluded.

Realization:

Based on preselected amounts of spent fuel waste and package designs, various repository concepts will be investigated with respect to geothermal and geomechanical characteristics. Subsequently, the repositories will be conceptually designed yielding logistics and operating personnel requirements. Radiological safety and the expenditure for technical realization will be quantified for the entire path. For some selected repository variants far-field calculations will be performed and long-term safety will be assessed. Finally, the complete system variants (including pre-emplacement treatment and disposal) will be assessed on the basis of the different criteria.

Time-Schedule:



Participants: DBE, KfK, BGR, GSF, DWK

DBE: Conceptual design of repository variants, quantification of radiological safety and provision of basic cost data

KfK: Near-field geothermal calculations,

BGR: Far-field geothermal and geomechanical calculations,

GSF: Repository long-term safety assessment

DWK: Spent fuel/waste flow sheet, conditioning, packages, interim storage

PAE, DBE, DWK: Evaluation of variants

PAE: Compilation of "Systems Study Dual-Purpose Repository"

II.2 In-Depth Planning of Two Selected Repository Concepts

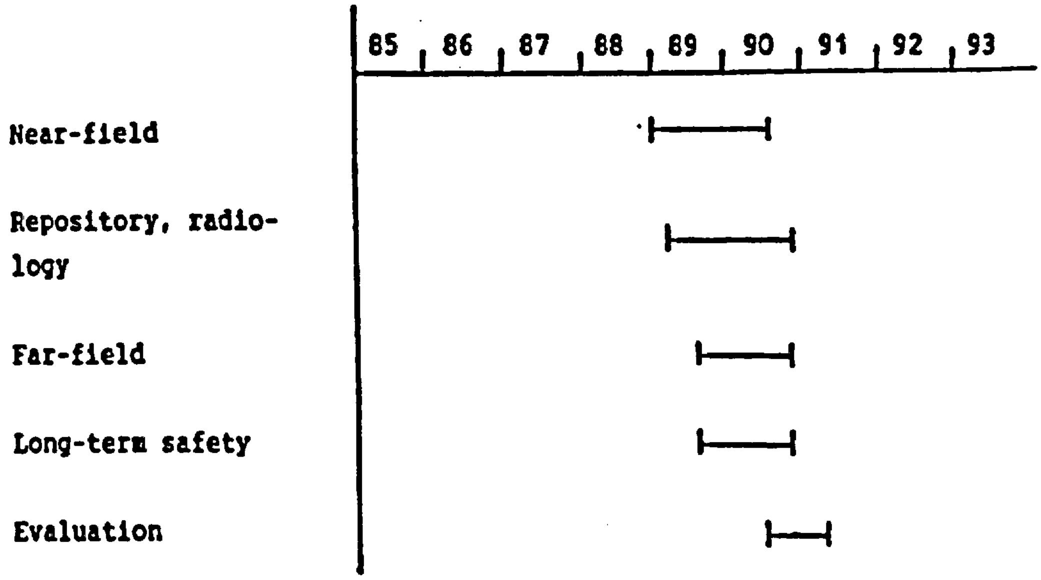
Objectives:

Subsequent to the analysis of numerous different disposal concepts (II.1), two candidate concepts will be investigated in detail. Special emphasis will be laid upon emplacement procedures, safety assessment as well as thermomechanic and rockmechanic analyses. Results gained during exploration of the Gorleben salt dome will be taken into account as far as possible. This study will be a basis for planning of the repository in Gorleben.

Realization:

As the amount of waste generating only negligible heat constitutes a major part of the overall waste, this type of waste will be included in the detailed analysis of two candidate concepts. Safety investigations will be performed, including emissions from secondary waste and incidents during handling of the waste packages. Furthermore, the influence of geometrically complex repository configurations on the operation of the repository, on thermal and thermomechanical aspects and on long-term safety will be investigated.

Time-Schedule:



Participants: DBE, KfK, GSF, BGR

Activities:

- Repository layout, logistics, operational personnel, radiology operational phase: DBE
- Thermal and thermomechanical calculations near-field: KfK
- Thermal and thermomechanical calculations far-field: BGR
- Long-term safety analysis: GSF

Activities will be co-ordinated with PTB and DWK in the technical committee.

III. Experimental Program

III.1 Canister Material Corrosion

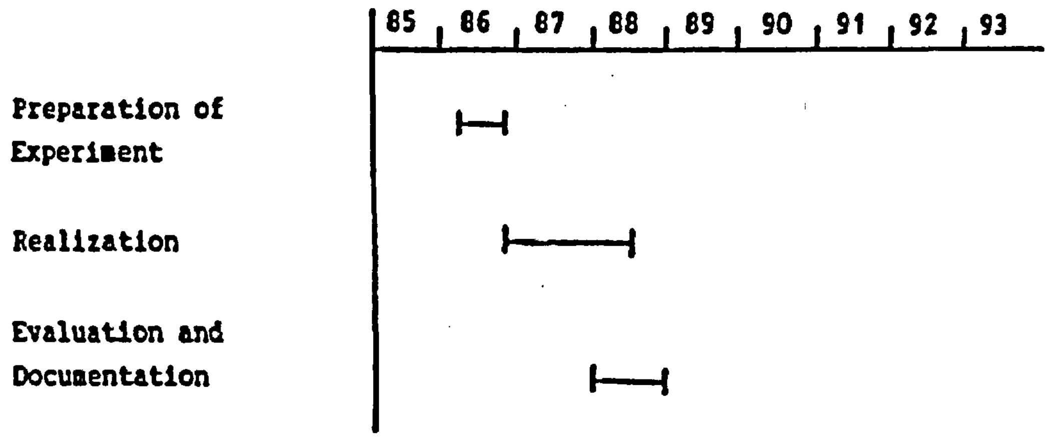
Objectives:

The containers provided for direct disposal of spent fuel are designed to ensure long-term containment of the radionuclides. An extensive corrosion test program is necessary in order to demonstrate the effectiveness of the container as a barrier even in the case of brine intrusion into the repository. Investigations on individual container materials were performed in a former program (1981-1984) in different laboratories. In the current program, qualification of the disposal container will be carried out by DWK. In addition, material combinations encountered in the disposal container will be subject to corrosion testing at KfK.

Realization:

Combinations of canister materials will be tested with respect to their general and local corrosion resistance at different temperatures and dose rates. The tests will be carried out with three corrosive brines which possibly could come into contact with the containers in the repository. Comparative tests are planned with pure Hastelloy. It is also planned, but with somewhat lower priority, to quantify the impact of radiolytically and thermolytically formed gases (HCl, H_2S, CO_2) on the corrosion rate.

Time-Schedule:



Investigations will be carried out at KfK. They will be co-ordinated with DWK, DBE and PTB.

III.2 Retention of Fission Gases in Back-Fill Material and Dams

Objectives:

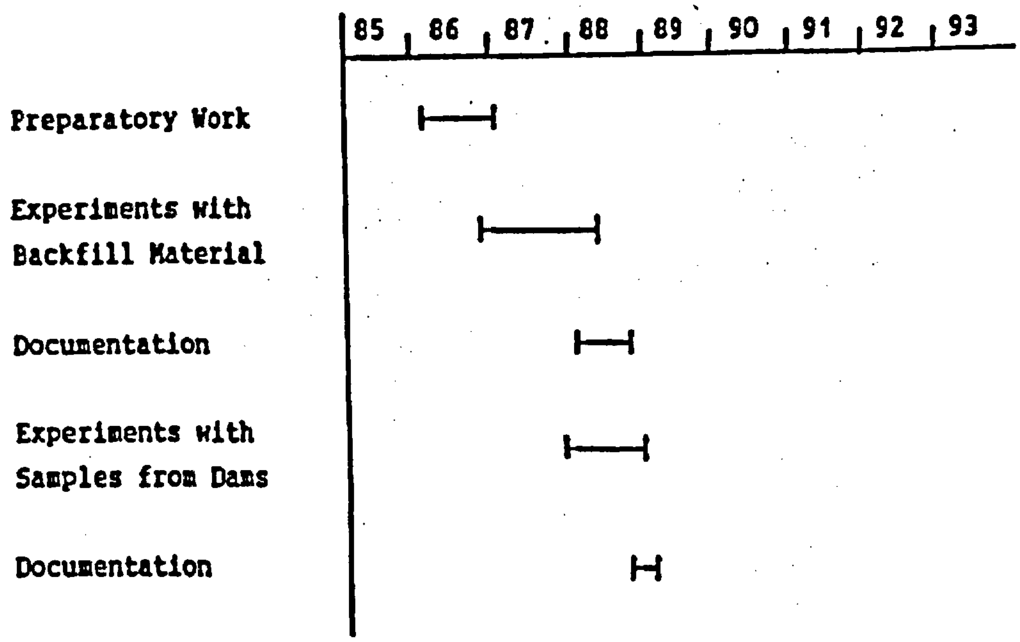
A large portion of the fission gases and volatile fission products generated during reactor operation are firmly bound to the fuel matrix. Only a small percentage ($\sim$0,06 Kr^{85}, $\sim$ 3E-6 H^3, $\sim$ 1E-4 I^{129}) is released into the gap between fuel and cladding. A major release from the back-filled section of the repository during the operating phase is very unlikely due to the presence of several barriers: Fuel rod cladding, welded disposal container, backfill material and dams.

Defendable results exist on the barrier functions of cladding hulls and container under disposal conditions, but knowledge is limited on retention capacities of backfill materials and dams. For confirmation of theoretical studies laboratory tests will be carried out on representative samples. These tests in particular will be focused on the retention capacity as a function of increasing backfill compaction.

Realization:

For to estimate the impermeability of backfill materials and dams against gases the sorption capacity for Krypton, Tritium and Iodine as a function of pressure will be determined on representative samples. For this estimation different methods will be employed: determination of equilibria, determination of diffusion coefficients, adsorption and desorption cinetics.

Time-Schedule:



Participant: ~~KfK~~ KWU

III.3 Leaching of LWR-Fuel

Objectives:

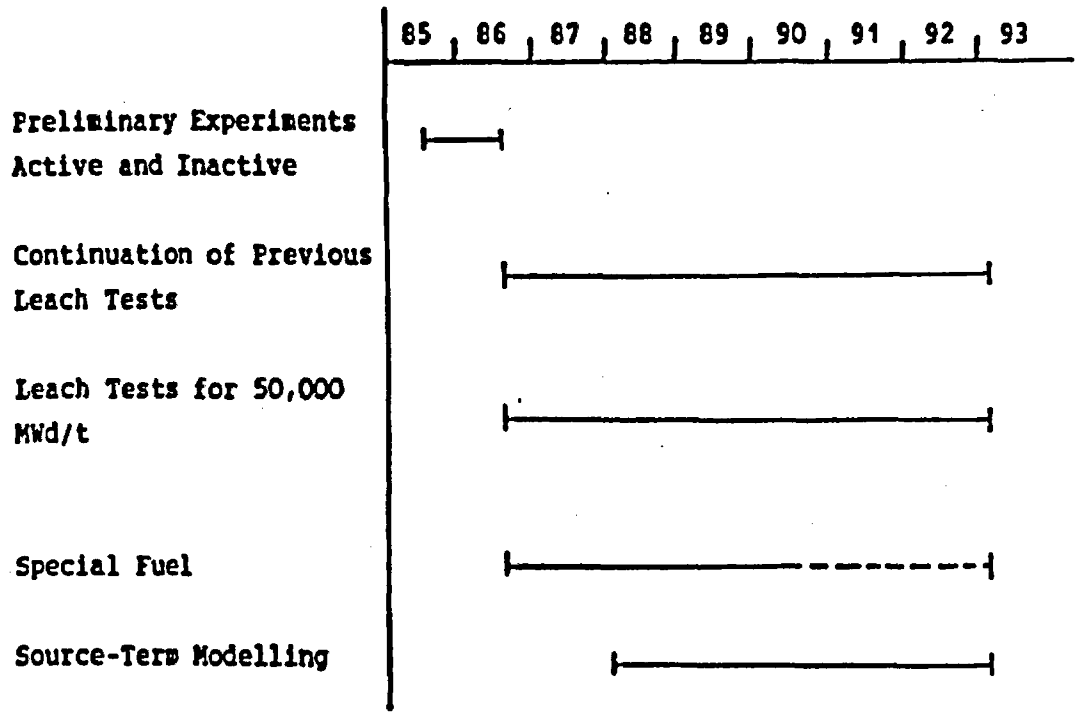
The leaching resistance of spent fuel under salt repository conditions is to be determined. The experimental results can be used for sourceterm modelling for radionuclide release from spent fuel in a repository in the unlikely event of brine intrusion in the post-operational phase.

Realization:

Leach tests of spent fuel have been performed for LWR fuel with a burn-up of up to 36,000 MWd/t. The planned experiments will include higher burn-ups (50,000 MWd/t), various temperatures (100 and 200°C), and other types of fuel. The leachants will be limited to those agreed upon by German expert groups. The effects of an altered redox-potential through canister materials will be simulated by inclusion of iron. In one experiment, deionized water will be used to enable comparison with experiments in Canada, Sweden, and the US. The experimental set-up will be arranged in accordance with the recommendations of the US Materials Characterization Center (MCC).

The leaching experiments will be carried out by KfK/KTB, whereby ten autoclaves will be used. Leachates will be analyzed by KfK/IRCh. For source-term modelling co-operation with external partners will be sought.

Time-Schedule:



Participants: KfK

The experiments will be discussed with external partners such as DBE, DWK, PTB, and the working-group "HLW-Products"; Collaboration with Battelle, HMI, KWU, and GSF/IfT will be directed to source-term modelling.

III.4 Combined Leach Tests Glass/Spent Fuel/Corrosion Products

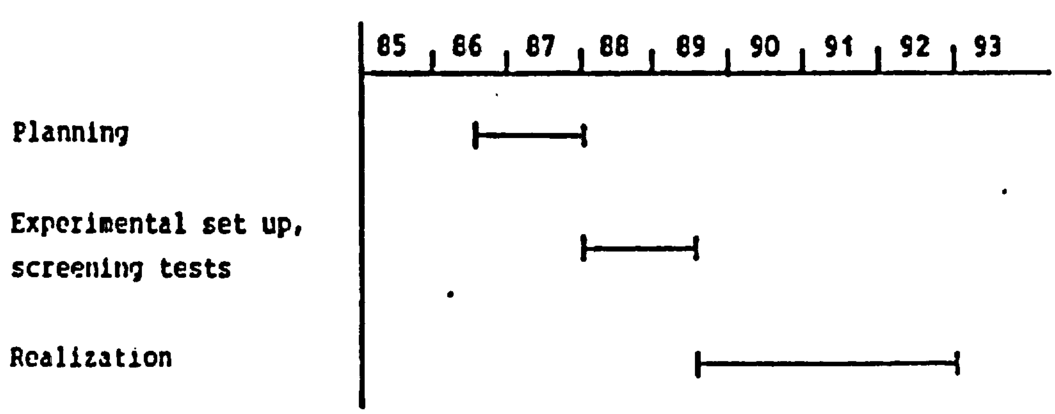
Objective:

In addition to the reference concept of spent fuel conditioning, cutting the fuel rods and encasing in canisters with the same outer dimensions as the HLW-canisters will be studied. Both canisters for either spent fuel or HLW can be disposed of in the same emplacement hole. In order to investigate whether or not leaching of glass or HLW is influenced by the presence of different materials in the same borehole, materials combinations consisting of HLW-glass, spent fuel, and canister corrosion products will also be subject to leach tests.

Realization:

Will be planned in 1986/87.

Time Schedule:



Participants:

The tests will be carried out in KfK. They will be co-ordinated with PTB, DBE and DWK. Collaboration is intended with industry and scientific institutes. Information will be exchanged with USA, Canada and Sweden.

ATTACHMENT 2

Stand: 09.04.1987

Veröffentlichungen über die

DIREKTE ENDLAGERUNG

- zitierfähige Liste -

DBE A. Jacobi, H.-J. Engelmann: "Endlagerkonzepte für die Direkte Endlagerung abgebrannter Brennelemente". Jahrestagung Kerntechnik Berlin (1983), Tagungsbericht, ISSN 0720-9207, S. 484-487

H.-J. Engelmann, A. Jacobi, H. K. Nipp, M. Wallner: "Auslegung des Grubengebäudes eines Endlagers für abgebrannte Brennelemente anhand von thermischen und thermomechanischen Rechnungen". Jahrestagung Kerntechnik Frankfurt (1984), Tagungsbericht, ISSN 0720-9207, S. 341-345

H.-J. Engelmann, K. D. Closs: "Systems Approach and Results for a Conceptual Design of a Repository for Spent Fuel Elements in Salt Formation Based on Thermomechanical an Radiological Safety". Proceedings of the Symposium on Waste Management '85, Tucson, USA (März 1985), Vo. 1, S. 319-323

P. Young, H.-J. Engelmann, K. D. Closs: "Systemanalyse der Planung eines Endlagers für abgebrannte Brennelemente in einem Salzstock anhand thermomechanischer und radiologischer Sicherheitsaspekte". Jahrestagung Kerntechnik München (1985), Tagungsbericht, ISSN 0720-9207, S. 385-388

H.-J. Engelmann, B. Hartje: "Aspects of Radiological Safety as one Design Criterion for the Operation for a Spent Fuel Repository in Salt Formations". Proceedings of the International Topical Meeting on High-Level Nuclear Waste Disposal-Technology and Engineering held, Pasco, Washington, USA (September 1985), S. 425-431

H.-J. Engelmann, M. Wallner, H.-K. Nipp: "Repository Design Calculations for the Alternative Storage of Spent Fuel". Proceedings of the International Topical Meeting on High-Level Nuclear Waste Disposal-Technology and Engineering hold, Pasco, Washington, USA (September 1985), ISBN 0-935470-29-8, S. 433-437

H.-J. Engelmann, B. Hartje, C. Schrimpf: "Demonstration of the Spent Fuel Disposal Techniques Test Program in the Federal Repbulic of Germany". Proceedings der Tagung: "Waste Management '86", Tucson, USA (April 1986), Vo. 2, S. 531-534

DWK H. Pirk: "Entwicklung von Endlagerbehältern für abgebrannte Brennelemente". Referate KTG-Fachseminar "Sicherheitsaspekte bei Transport und Lagerung von bestrahlten Brennelementen", Stuttgart, 3.-4.6.1981, Kerntechnische Gesellschaft (1981), S. 146-163.

K. Einfeld, F. W. Popp: "Dry storage systems using casks for long term storage in an AFR and repository". Third International Spent Fuel Storage Technology Symposium/Workshop, Seattle, USA (1986)

Veröffentlichungen über die

DIREKTE ENDLAGERUNG

- zitierfähige Liste -

DWK K. Einfeld: "Zum Stand der Konditionierungstechnik". Vortrag auf dem Symposium: "Die Zukunft der Entsorgung von Kernkraftwerken", München (15.5.1986)

K. Einfeld, F. W. Popp, U. Knapp: "Material Selection for final Storage of Spent Fuel in a Salt Repository". Vortrag IAEA, Wien (2.-5.9.1986), T1-Tc-580

K. Einfeld: "Pilot Plant for Spent Fuel Conditioning at Gorleben". Beitrag in Nuclear Europe 3-4/1987, S. 21 (1987)

KFA R. Buttler, W. D. Lauppe, E. Pohlen, B. Richter, G. Stein: "Internationale Kernmaterialüberwachung bei einem Endlager der Anderen Entsorgungstechniken". Beitrag zum F+E-Schwerpunkt Andere Entsorgungstechniken des Bundesministeriums für Forschung und Technologie, Jül-Spez-269 (Dezember 1984) ISSN 0343-7639

R. Buttler, W. D. Lauppe, B. Richter, G. Stein: "Safeguardsaspekte der Direkten Endlagerung abgebrannter Brennelemente". Jahrestagung Kerntechnik München (1985), Tagungsbericht, ISSN 0720-9207, S. 391-394

KfK K. D. Closs: "Direkte Endlagerung oder Wiederaufarbeitung?". Umschau in Wissenschaft und Technik, 81. Jahrgang, Heft 10, 1981, S. 290-299

R. Papp, H. Geipel: "Aspects of the FRG Waste Management Policy and the Alternative Fuel Cycle Evaluation". Proceedings der Tagung: "Waste Management Conference", Vol. 1, Tucson, USA (8.-11.3.1982), S. 39-46

R. Papp, K. D. Closs: "The FRG Alternative Fuel Cycle Evaluation". Abstract in den Proceedings der International ENS/ANS Conference, Brüssel (26.-30.4.1982), S. 110-113

K. D. Closs, H.-J. Engelmann, H. Loser, O. Mehling, R. Papp: "Auswahl eines Referenzkonzeptes für die Direkte Endlagerung abgebrannter Brennelemente". Jahrestagung Kerntechnik Berlin, (1983), Tagungsbericht, ISSN 0720-9207, S. 318-323

K. D. Closs: "Direkte Endlagerung abgebrannter Brennelemente". Atomkernenergie-Kerntechnik, Vol. 44, 1984, No. 2 , S. 114-117

R. Papp, K. D. Closs, H.-J. Engelmann, O. Mehling: "The Technical Concepts for the FRG Alternative Fuel Cycle Evaluation". Proceedings der Tagung: "Waste Management Conference", Vol. 1, Tucson, USA (11.-15.3.1984), S. 69-76

K. D. Closs, K. Einfeld, A. Jacobi, H. Pirk: "Technik der Direkten Endlagerung abgebrannter Brennelemente". Atomwirtschaft/Atomtechnik (August/September 1984), S. 459-463

Veröffentlichungen über die

DIREKTE ENDLAGERUNG

- zitierfähige Liste -

KfK K. D. Closs: "Direkte Endlagerung später zur Ergänzung". Energiewirtschaftliche Tagesfragen, 35. Jahrgang, Heft 5 (Mai 1985) S. 314-319

R. Papp: "The Alternative Fuel Cycle Evaluation in the Federal Republic of Germany". Vortrag bei: International. Association for Impact Assessment (IAIA), Annual Conference on Methods and Expriences in Impact Assessment, Utrecht (27.-28.6.1985)

R. Papp, K. D. Closs: "Alternative Fuel Cycle Evaluation in the Federal Republic of Germany". Nuclear Technology, Vol. 72, Mar. 1986, S. 312-320

R. Papp, K. D. Closs: "Results of the German Alternative Fuel Cycle Evaluation and Further Efforts Geared Toward Demonstration of Direct Disposal". Proceedings der Tagung: "Waste Management (WM 86), Tucson, USA, (2.-6.3.1986), Vol. 2, S. 523-526

R. Papp, H. Loser: "Fuel Roprocessing Versus Direct Disposal of Spent Fuel - A Comparison From the Standpoint of Radiological Safety". Nuclear Technology, Vol. 73, May 1986, S. 228-235

K. D. Closs: "Direkte Endlagerung ausgedienter Kernbrennelemente". Chemiker Zeitung, 110. Jahrgang, 1986, Nr. 6, S. 251-256

K. D. Closs: "Technische und sicherheitstechnische Aspekte der Direkten Endlagerung abgebrannter Brennelemente". Kernenergie, 29. Jahrgang, Heft 7, 1986, S. 268-273

K. D. Closs: "Vergleich der Wiederaufarbeitungsanlage mit Direkter Endlagerung". Tutzinger Studien "Wiederaufarbeitungsanlage Wackersdorf", 2/1986, S. 71-77

K. D. Closs: "Stand und Entwicklung der Forschung für direkte Endlager". Symposium der Friedrich-Ebert-Stiftung "Zukunft der Entsorgung von Kernkraftwerken", München, 15.5.1986

K. D. Closs, K. Einfeld: "Overview of the FRG activities on spent fuel disposal". Proceedings of the 2nd Internat. Conf. on Radioactive Waste Management, Winnipeg, CDN (September 1986), ISBN 0-919784-08-9, S. 26-32

K. D. Closs: "Überblick über die weiterführenden Arbeiten zur Direkten Endlagerung". Vortrag beim Seminar "Projekte zur Endlagerung radioaktiver Abfälle". KTG Fachgruppe Chemie und Entsorgung. Braunschweig, 24.-25.9.1986

K. D. Closs: "Spent Fuel Characterization with Respect to the German Disposal Concept". Proceedings Finnish-German Seminar on Nuclear Waste Management, Espoo, Finnland (23.-25.9.1986)

Veröffentlichungen über die

DIREKTE ENDLAGERUNG

- zitierfähige Liste -

KfK H. Loser, R. Papp: "Wiederaufarbeitung und Direkte Endlagerung, ein Vergleich aus radiologischer Sicht". Atomkernenergie-Kerntechnik, 49 (1986) S. 89-95

NUKEM P. Arntzen, O. Mehling, F. W. Popp: "Spaltgasfreisetzung aus Brennelement-Endlagergebinden im Endlager". -Kurze Beiträge zur Kerntechnik-, atomwirtschaft-atomtechnik, Januar 1986, S. 32-33

H. Pirk, K. Einfeld: "Packaging Techniques for Different Types of Spent Fuel with Respect to Long Term Storage and Final Disposal". Proceedings der Tagung: "Technology Symposium/Workshop" Seattle, USA (8.-10.4.1986)

O. Mehling, P. Arntzen, F. W. Popp: "Kritische Betrachtung der gewählten Anforderungen am Endlagergebinde für die Direkte Endlagerung". Jahrestagung Kerntechnik München (1985), Tagungsbericht, ISSN 0720-9207, S. 389-390

P. Arntzen, K. Einfeld, H. Lahr, H. Pirk, F. W. Popp: "Entwicklung von Endlagergebinden und Konzept einer Konditionierungsanlage für die Direkte Endlagerung von ausgedienten Brennelementen". Jahrestagung Kerntechnik München (1985), Tagungsbericht, ISSN 0720-9207, S. 381-384

ATTACHMENT 5

10 July 1987

DRAFT VERSION

Minutes of the US/FRG technical meeting on testing of Asse salt for radiation effects produced during BMT

Gesellschaft für Strahlen- und Umweltforschung mbH München
Institut für Tieflagerung (GSF-IfT), Braunschweig, FRG, June 10 - 12, 1987

Participants:

D. E. Clark	OHWI, USA	
P. W. Levy	BNL, USA	
L. R. Pederson	PNL, USA	
A. Garcia-Celma	RUU, Netherlands	
J. B. M. de Haas	ECN, Netherlands	
R. B. Helmholdt	ECN, Netherlands	(June 11 - 12, 1987)
L. H. Vons	ECN, Netherlands	(June 11 - 12, 1987)
H. K. Fröhlich	Battelle-Frankfurt, FRG	(June 10, 1987)
O. Schulze	BGR, FRG	(June 11, 1987)
Y. Goeksu-Oegelman	GSF-ISS, FRG	(June 10, 1987)
D. Regulla	GSF-ISS, FRG	(June 10, 1987)
H. Gies	GSF-IfT, FRG	(June 11, 1987)
W. Hild	GSF-IfT, FRG	
H. Hinsch	GSF-IfT, FRG	
N. Jockwer	GSF-IfT, FRG	
J. Moenig	GSF-IfT, FRG	
T. Rothfuchs	GSF-IfT, FRG	
R. Stippler	GSF-IfT, FRG	

Introduction

A meeting of technical experts involved in the testing of Asse salt for radiation effects produced during the Brine Migration Test (BMT) was held at the GSF-IfT on the dates of June 10 - 12, 1987. The group observed firsthand the irradiated rock salt at the BMT site in the Asse mine, and discussed the technical approaches that should best contribute to the characterization and understanding of the radiation effects. An outline of the meeting minutes is given below.

First Day - Wednesday, June 10

(Asse mine tour; discussion of BMT and HLW-test disposal; dosimetry requirements and development)

The meeting started with an overview of the activities at the Asse mine and the present status of the BMT and HLW-test disposal sites. This overview was given by T. Rothfuchs and represented an introduction for the following Asse mine tour. After an extended tour, problems related the dosimetry of high radiation fields at the high temperatures that will be encountered in the HLW-test were discussed. H. Hinsch presented a brief description of the planned dummy containers, which will be placed on top of the simulated waste containers in six of the eight boreholes. The use of these dummy containers will allow irradiation testing of samples during the course of the HLW-test disposal. The locations where dosimetry will be carried out in the HLW test and the expected dose rates were discussed. J. de Haas presented some results of dose rate calculations at various locations inside the dummy containers. H. Fröhlich described the ionization chamber dosimeter which is to be provided by Battelle-Institut, Frankfurt/M., and which is small enough to fit in the 1-cm gap between the steel tube and the salt. Prior to its use at Asse, the dosimeter will be calibrated at Battelle-Frankfurt under high temperature and dose-rate conditions.

D. Regulla and Y. Goeksu-Oegelman discussed the results of their investigation into the problem of high dose measurements at high temperatures. During this discussion, it became clear that no material is presently known which does not show some healing effects at elevated temperatures. However, some materials, e. g., doped glasses, do look quite promising and proper calibration will now be carried out at several temperatures in order to allow corrections to be made. P. Levy pointed out in this context that Ce-doped glasses can be used successfully over various dose-rate ranges, depending on their Ce-content. Generally, it was agreed that the alumina (Al_2O_3) beads used to backfill the emplacement sites could be employed for dosimetry purposes if the samples can be prepared appropriately. A. Garcia-Celma will try to prepare some samples. The possibility of using the irradiated salt directly for dosimetry was also discussed.

The following deadlines were noted by H. Fröhlich and D. Regulla. Battelle-Frankfurt will deliver the ionization chamber dosimetry system by December 1987 on the material selection for solid state dosimetry at elevated temperatures. At the point, an outline for further procedures will also be presented.

Second Day - Thursday, June 11

(Petrochemical effects of radiation on salt; overview of testing at BNL and PNL; the role of recrystallization of halite on the formation or removal of radiation defects; initial results already obtained on the irradiated BMT salt)

O. Schulze discussed the results of his investigations into the petromechanical effects of rock salt irradiation in the BMT. Several parameters, i. e., flow stress and creep rate, were measured for irradiated and unirradiated salt. The results were compared with several theories. These investigations are now nearly finished. Subsequently, the general problems of determining strain in natural salt grains were discussed in some detail.

P. Levy presented an extensive discussion of the fundamental processes underlying the formation of radiation defects in rock salt. The influence of various parameters, e. g., dose rate, dose and pre-irradiations strain, were discussed in terms of theoretical significance, and experimental results for F-center and sodium colloid formation were shown. These investigations were mainly carried out with single crystallites. The irradiation facilities and experimental methodology employed at Brookhaven National Laboratory (BNL) were discussed. The influence of trace elements and/or trace minerals was pointed out in the discussion. It was suggested that impurities may act as recombination centers thus reducing the colloid formation. This effect should be encountered even at very low concentrations of impurities. However, the mechanism is not understood. It was also discussed that the experimental irradiation set-up at BNL corresponds to the one which is presently being installed at the University of Groningen (den Hartog) and that no fundamental differences in the radiation physics are to be expected between the two.

A. Garcia-Celma presented the results of her studies of the recrystallization of irradiated halite and its effect on the extent of radiation defect formation. It was shown that the apparent extent of recrystallization is related to the brine content of the sample. In this investigation, the stored energy in irradiated salt is measured by calometry and DTA. Several methods and mechanisms were suggested to account for the observations. It was genereally concluded that it is very important, albeit very difficult, to distinguish between primary and secondary recrystallization. The question as to why the BMT results with respect to colloid formation differ so much from the results obtained at the Lyons, Kansas mine was raised and discussed. The view that the Lyons salt is much more homogeneous and has a higher brine content was accepted as the most likely explanation.

The results of the chemical analysis of the BMT salt carried out at Pacific Northwest Laboratories (PNL) were presented by L. Pederson. The measurements encompassed the determination of OCl^-, H_2, and total base. These results were used to calculate the expected corrosion of steel waste packages. It was concluded that at the external dose rates envisaged for the U. S. waste containers, no significant effects are expected.

H. Gies showed results of the chemical analyses of irradiated salt minerals which were separated into two categories according to their color. The blue salt containing colloid had a significantly smaller amount of trace minerals than the white salt minerals. However, the Na/Cl ratio did not deviate significantly from the normal stoichiometric value. This program will be continued on a larger set of samples. After this discussion, a brief introduction of the experimental program envisaged for the BMT salt was given by N. Jockwer.

The question of whether chlorine release can be observed during sample irradiation was addressed by P. Levy. The apparent temperature dependence of chlorine release for salt irradiated at BNL was shown. It was pointed out that chlorine evolution can be measured even without irradiation and that it is at least partly a result of heating the salt. It was therefore suggested to measure chlorine evolution at different temperatures from unirradiated samples as well as irradiated ones (at various dose rates). In addition, the growth of F-centers in natural samples was discussed along with their dependence on the strain applied prior to irradiation. Although increasing strain increased the F-center formation rate, a less certain indication of colloid formation was obtained.

The results of radiation effects calculations were presented by J. de Haas. The radiation damage was calculated using the Jain-Lidiard theory (JLT) taking values for dose rate- and temperature-fields obtained in previous calculations. The overall aim of these calculations is a consequence analysis for the burial concept. The calculations were performed with several variations of variables in the JLT. It was concluded that some improved experimental determinations of parameters are needed, e. g., for the rate of F-center formation. In the ensuing discussion, it was pointed out by P. Levy that the JLT is correct qualitatively in most aspects, but fails to make correct quantitative predictions. In addition, JLT does not allow for the observed induction periods. It was generally agreed that JLT must be modified to provide an improved theoretical treatment. This task can be performed using more accurate sets of data (rate of F-center and colloid formation), which are measured as functions of dose rate at lower dose rates and temperatures. Low dose-rate measurements are problematic due to long experiment times. It was concluded, however, that extrapolations from different dose-rate regions are only permissible if the mechanisms are known accurately. Finally, L. Pederson observed that for a consequence analysis the absolute maximum in radiation damage can be calculated from the band gap and the total absorbed energy values.

Third Day - Friday, June 12

(Agreement on harmonized test plan; final discussion)

On the final day, the participants finalized plans for a full investigation of radiation effects on BMT salt. A general outline of proposed responsibilities for the post-test analysis of salt from the BMT was developed (see Appendix A). Consensus was reached that each laboratory should perform its tasks independently, but that some preference will be given to samples which are also being tested elsewhere. In this context, it was agreed that the experimental program will concentrate on samples from the horizontal borehole. The work is to be completed (and reported to the GSF-IfT in draft form) by the end of this calender year. It was concluded that the necessity of having another meeting of this group will be evaluated at that time.

Conclusion

This meeting proved to be very productive. Frank discussions were held concerning what is currently known about radiation effects in salt, what experimental approaches are viable, and remaining uncertainties. A detailed working plan was produced and the recommended institution to perform each test (see Appendix A). A satisfactory conclusion of the BMT project should result from implementation of the planned working task assignments, and more definitive data will be obtained from the next in-situ radiation effects test at the Asse mine.

Appendix A

Outline of Proposed Tasks for Determining Radiation Effects in BMT Salt

1. Mineralogical-geochemical measurements

   a) determination of the accessory minerals present
   I, II (minor effort), IV (minor effort)

   b) determination of the distribution of accessory minerals
   I, IV (minor effort)

   c) separation of various phases (e. g., blue, white and sulfates) for further analysis
   I

2. Chemical analysis

   a) OCl^- measurement
   IV, I (minor effort)

   b) determination of total base (via Gran-titration); determination of p_cH-values
   I, IV

   c) chemical analysis and measurement of distribution of principal and trace elements (e. g., via ICP, NAA, microprobe of thin sections)
   I, IV, III (minor effort?)

   d) analysis of water content and gases evolved by heating or dissolving the salt
   I, IV, II (minor effort)

3. Radiological work and physical measurements

   a) determination of colloid concentrations by optical absorption methods
   III, II (??? possibly minor efforts)

   b) determination of colloid production rates by optical absorption methods
   III (?)

   c) rock salt dosimetry via comparison with mineralogical equivalent (analogous) mineral crystals
   III (secondary measurement, minor effort)

d) measurement of strain-related effects
III (?)

e) measurement of stored energy (including comparison and calibration of methods to determine stored energy
II, V (minor effort), VI (possible minor effort)

4. Microstructural measurements

a) mapping of a large number (i. e. statistically significant sample size) of thin sections
V

b) re-irradiation of thin slices followed by examination for the occurence of recrystallization
V (minor effort)

c) application of techniques to discriminate between primary and secondary recrystallization
V, VI (possible minor effort)

5. Calculations

a) two-dimensional mapping of radiation fields at the BMT-site(s)
II, VI (possible minor effort)

b) two-dimensional mapping of temperature fields at the BMT-sites with time dependence
II, VI

c) calculation of radiation damage using parameters obtained under a) and b) above according to Jain-Lidiard theory
II

d) extension of Jain-Lidiard theory in accordance with new experimental results
II, III (?)

institutions

I GSF-IfT
II ECN
III BNL
IV PNL
V RUU
VI others (to be determined)

ATTACHMENT 4

Federal Ministry for
Research and Technology (BMFT)

Bonn, June 9, 1987

The Accident in Shaft Nr. 1 of the
Gorleben Exploratory Mine on May 12, 1987

On May 12, 1987, at 9.45 h a steel reinforcement ring (diameter 10 m) dropped from a height of 5 m onto the base of the shaft under construction, which was at a depth of 239 m below surface. At that moment, 7 miners worked on this level. 3 of them were heavily and 3 others slightly injured. One of them died some days later. [illegible] concrete [illegible] based on [illegible]

Technical and geological situation:

Shaft 1 of the exploration mine Gorleben was sunk to a depth of 239 m through a cylindrical set of frozen strata into the interface between the overlying strata and the caprock of the salt dome. [illegible]acement. [illegible] m³ of concrete was brought in from May [illegible] May [illegible], 1987.

The geological formation of the Quarternary (0 - 162 m below surface) including a 3.5 m thick layer of "Lauenburger" clay was passed by shaft sinking without problems. The underlying Tertiary formation (162 to 237 m) consists in principle of 3 layers: an upper clay layer (162 to 210 m), an intermediate layer of sand and sandstone (210 to 229.5 m) and the lower clay layer (from 229.5 m) which overlies the caprock at a depth of 237 m. Due to the proximity of the salt dome, the water content of the lower clay layer is highly saline. The clay formation shows abundant joints and slickensides. Therefore, only in this lower clay layer significant convergencies occured because the freezing body as a result of the shaft sinking freezing procedure could not entirely avoid the movement of rock in this section.

In this critical zone sinking was done more carefully. Additionally, the concrete blocks forming the outer shaft reinforcement were supported by 34 pre-stressed, welded and screwed steel rings. One of these steel rings fell down after a welding had broken.

...

Among the first security actions taken was the emplacement of steel cables to secure the remaining steel rings. During the following days, measurements and visual inspections of these rings showed general deformations due to rock pressure. This means, that the reinforcement rings were too weak to withstand the unexpectedly high rock movement. However, this outer wall is normally not designed to resist to external pressure but to serve as a protection during shaft sinking. Pressure resistance will finally be guaranted by the inner shaft wall consisting of a steel liner and reinforced concrete, both being based on a foundation in the rock salt.

On May 17, it was decided to fill up the shaft in this critical zone of 14 m with poor concrete in order to stabilize the actual situation and to avoid the failure of freezing tubes due to rock displacement. 1.290 m³ of concrete was brought in from May 18 to May 20, 1987.

These measures allow to design, license and fabricate a stronger wall support and to replace the first one successively. The schedule of shaft sinking will probably be delayed by 3 to 6 months.

Apart from the manifestation of convergency, some anomalies have been observed on temperature and water level measurements in the weeks before the accident. The BGR (Federal geological survey) has been asked to evaluate these phenomena and to find out whether they might have had an influence on the deformations of the shaft wall.

Conclusions:

Due to these facts, critics claim again to stop all work at Gorleben. However, it is common understanding within the Government and the companies in charge that this event is only a mining problem and does not effect the suitability of the shaft after its completion and the salt dome as a whole for a radioactive waste repository.