SYSTEMS APPROACH AND RESULTS FOR A CONCEPTUAL DESIGN OF A REPOSITORY FOR SPENT FUEL ELEMENTS IN SALT FORMATION BASED ON THERMOMECHANICAL AN RADIOLOGICAL SAFETY

H. J. Engelmann
Deutsche Gesellschaft zum Bau und Betrieb von
Endlagern fur Abfallstoffe mbH (DBE)
Woltorfer Str. 74, D-3150 Peine
Federal Republic of Germany

K. D. Closs
Kernforschungszentrum Karlsruhe GmbH (KfK)
Postfach 3640, D-7500 Karlsruhe
Federal Republic of Germany

ABSTRACT

In the framework of the R&D-program "Alternative Spent Fuel Management and Disposal Techniques" efforts in the repository development are focused on the adaption of existing plans for the repository at Gorleben salt dome to the specific requirements of disposal of spent fuel disposal in galleries with self-shielded containers should be the reference concept. This concept considers a mining design with three sectors. The eastern wing accomodates the spent fuel packages, the western wing all other wastes from the conditioning facitity, from nuclear power stations and other sources and the central area with two shafts, the infrastructure etc. In this approach we will discuss the safety-related aspects of the mining design and lay-out of a repository, namely the long term stability of the salt dome, radiological exposure of the plant personnel during normal operation and after accidents as well as the radiological impact of future generations during the post operational phase.

BACKGROUND

Spent fuel reprocessing is the reference concept for the back end fuel cycle in the Federal Republic of Germany. Due to a political decission in 1979, disposal of spent fuel is also being investigated. In the framework of the R&D-program "Alternative Spent Fuel Management and Disposal Techniques" efforts in the repository development sub-program are focused on the adaption of existing plans for the repository at the Gorleben salt dome, which is designed to the emplacement of reprocessing wastes, to the specific requirements of disposal of spent fuel.

In phase 1 of the R&D-program, which lasted up to mid - 1982, several disposal techniques were investigated and one concept was selected based on criteria such as safety - related aspects, technical feasibility and costs. It was recommended that disposal in galleries with self-shielded containers should be the reference concept.

This concept considers a mining design with the following lay-out:
The repository features two shafts. Shaft 1 is for material and personnel transport and air inflow. Shaft 2 is designed to accommodate radwaste transportation and the outflow of the exhaust air. The diameter of the shafts is 7.5 m allowing the transport of the disposal packages in the cage in a horizontal position. The distance between the two shafts is 400 m. Two parallel access galleries are driven at a distance of 770 m from each other. Every 200 m, drifts are branching off at right angles. Each of these connetion drifts belongs to an actual emplacement field comprising fifty parallel disposal tunnels, 5 m wide and 4 m high, which are separated by pillars 10 m wide. At an areal heat load of 180 kW/ha (about 70 kW/acre) the longitudinal spacing (center to center) between the spent fuel packages in the disposal tunnel is 7.5 m.

The whole repository is subdivided into three sectors:
The eastern wing accomodates the spent fuel packages. Secondary waste from the conditioning and encapsulation facility, from nuclear power stations of 26 GW capacity, from industry and other sources which cause only negligible thermal loading are disposed of in the western sector. In between those two wings is the central area which accomodates the two shafts, the infrastructure, etc. A general lay-out of the mine is give in Fig. 1.

In this approach we will discuss the safety-related aspects of this lay-out, namely the long term stability of the salt dome, radiological exposure of the plant personnel and the inhabitants during the operation of the repository and the radiological impact of futur generations after a very unlikely brine instrusion during the post operational phase.

LONG-TERM STABILITY

The long-term stability of the repository depends - among other things - on the local and spatial temperature increase in the salt dome. Therefore the temperature distribution in the geological formation was analyzed as well as the consequences of the heat load on the stability.

The specially designed containers with a self-shielding (6.0 length, 1.4 m diameter and 55,000 kg weight) is able to contain three spent fuel elements The container will be placed in series in parallel drifts of the east-wing which after disposal will be backfilled and sealed.

Near-field maximum temperatures at the surface of the emplaced spent fuel canisters were calculated to be 167°C after 30 years. The results are shown in Fig. 2. The average temperature of the total emplacement area will bei 120°C after 65 years. In the salt dome top about 250 m below surface the temperature will increase by 15°C after 1,600 years.

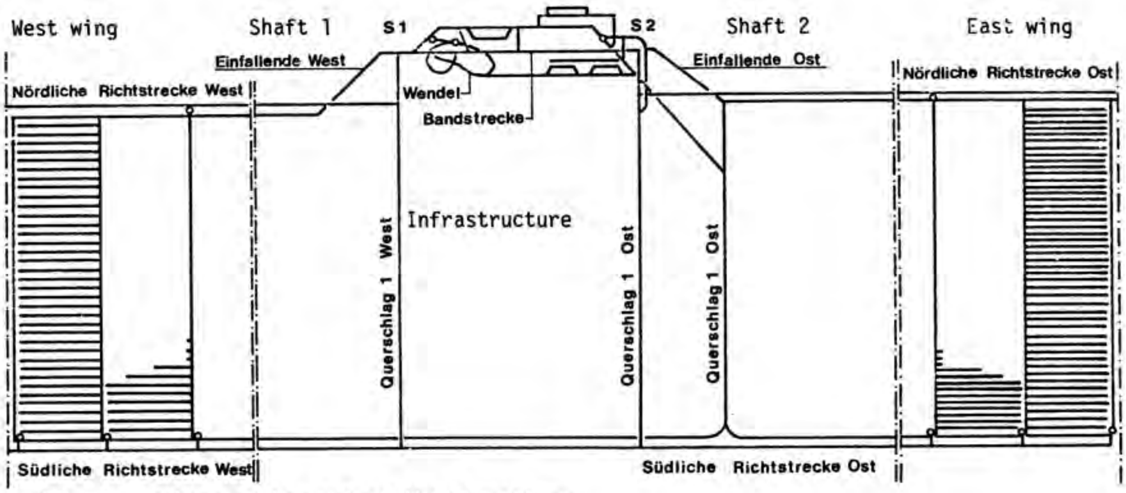


Fig. 1. A General Lay-out of the Repository.

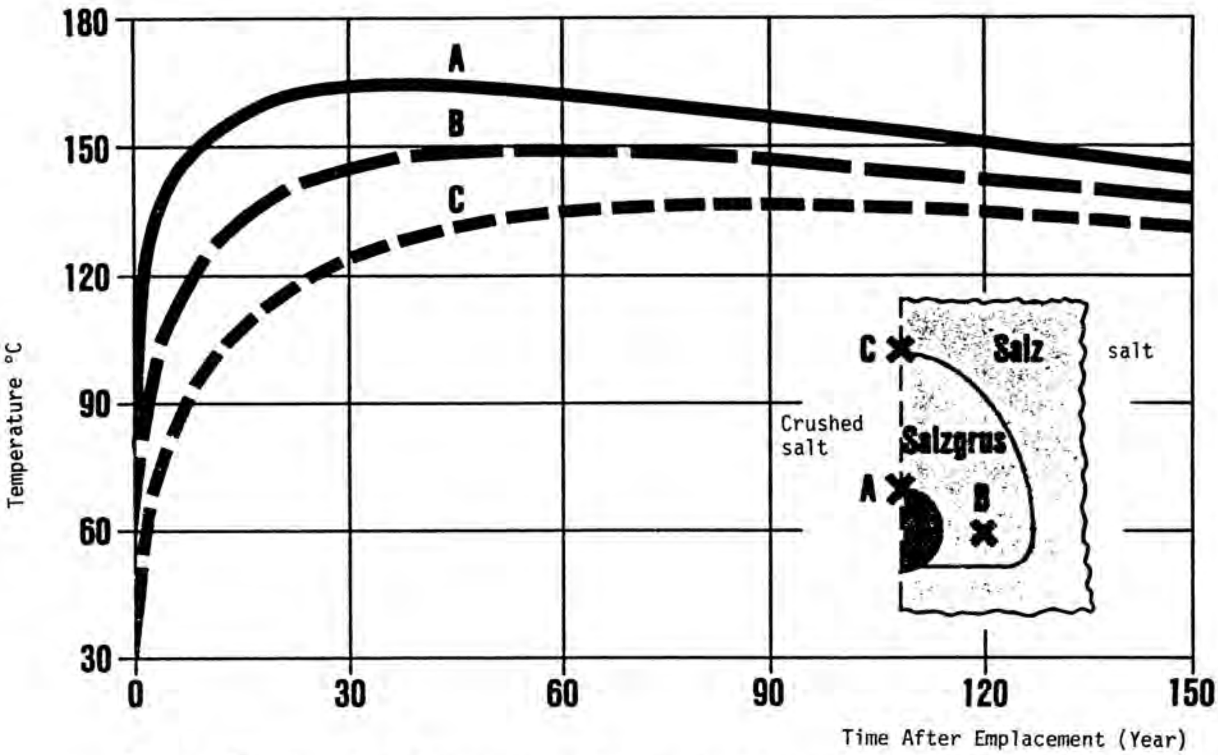
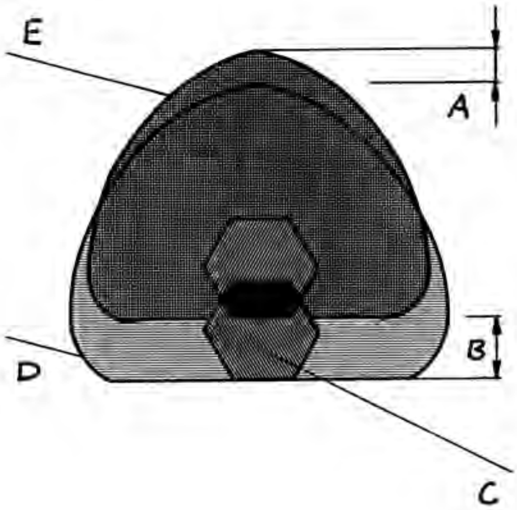


Fig. 2. Development of Temperature at Characterized Points in a Backfilled Drift.

The longtermed thermomechanical effects on the far-field were examined for a time period of 10,000 years. Althoug the calculations were done for a very rough model simplifying the geological structure.

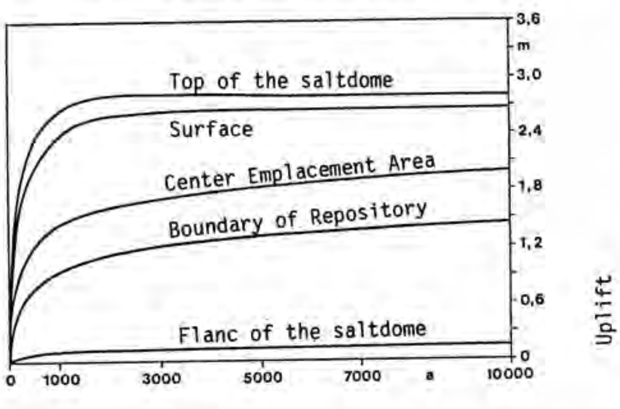
The convergency of backfilled emplacement-drifts after 50 years is demonstrated at Fig. 3.



A: Relative Roof Creep
B: Relative Sole Creep
C: Spent fuel Container
D: Drift at the Time of Emplacement
E: Drift 50 Years after Emplacement

Fig. 3 Convergency of a Backfilled Drift after 50 Years

As a result of the planned heat load of 70 kW/acres, an uplift of the surface above salt dome by 2.7 m after 4,000 years was calculated (figure 4)



Time after Emplacement

Fig. 4 Uplift

The thermomechanical consequences of emplacement of spent fuel elements in the above mentiones lay-out does not cause any critical stress conditions in the geological barrier salt dome.

RADIOLOGICAL EXPOSURE DURING THE OPERATION OF THE REPOSITORY

According to the German radiation Protection Ordiance the dose burden for the staff and the inhabitants has to be as low as reasonable achievable during normal operation and after accidents. The radiation exposure to the staff is attributed to inhalation. The dose burden to the general public is mainly due to inhalation, ingestion, β-and γ-submersion as well as γ-radiation from sedimented and contaminated particles.

Generally the radiation exposure is calculated in Germany as an individual dose for each organ at the human body. Additionally we have calculated a collective dose. This collective dose is a weighted committed dose equivalent according to ICRP (International Commission for Radiological Protection).

We will show the aspects and the results of our analysis for normal operation and accidents. The staff will be burdened by direct radiation and air-borne releases. The exposure due to direct radiation was evaluated in greater detail by estimating dose rates in different parts of the plant as well as the number of workers and the duration of time they spend in radiation field. The radionuclide release must be limited to maximum permissible release so as to observe the quantitative exposure limits of the German Radiation Protection Ordinance. Or on the other hand, the barriers against radionuclide releases must be of such a quality that the maximum releases are not exceeded.

Values for the maximum release were calculated from the exposure limits for the operational phase using model assumptions on the repository and the spent fuel. The calculations were carried out separately for the personnel and the environment for normal operation and accidents under the following conditions:

- known main spectrum of radionuclides in the waste same release fraction for all radionuclides
- only nuclides with higher activity than 1.0 E+6 Bq/container will be take into consideration
- general calculation principles for radiation exposure from radioactive releases from exhaust air or into surface waters
- 20 minutes exposure for the personnel in the case of an accident for inhalation and γ-submersion
- retention of 90 % of the released radionuclide fraction in the repository
- it was assumed, that particles with a diameter smaller than 100 μm are airworthy, particles smaller than 10 μm are able to enter the lungtissue

The collective dose for normal operation for the staff of the repository is shown in table I. E means the personnel for emplacement, S is the staff for the radiation protection, the personnel T is working in the surface facilities and B are the pitworkers for excavation.

To calculate the consequences of accidents all handling procedures were analyzed with the spent fuel casks and the other waste canisters. For the accidents with radiological consequences. Sources of the mechanical and thermical stresses were estimated in order to describe the radionuclide release. With these data

TABLE I

OCCUPATIONAL COLLECTIVE DOSE COMMITMENT OF THE STAFF IN A REPOSITORY

Depart-ment	Number of workers	Part of Collective dose		Total Collective dose manSv/year
		Direct Radiation manSv/year	Inhalation manSv/year	
E	43	2,4 E-1	3,8 E-5	2,4 E-1
S	22	8,0 E-3	7,2 E-6	8,0 E-3
T	119	3,7 E-2	6,6 E-6	3,7 E-2
B	152	-	9,7 E-5	9,7 E-5
Total	336	2,8 E-1	1,5 E-4	2,8 E-1

the accidental collective dose expressed terms of inhalation, γ-submersion and direct radiation. The sum of the collective dose commitment multiplied by the frequency of occurence of such accidents was calculated to be 6.6 E-4 manSv/year for the plant personnel.

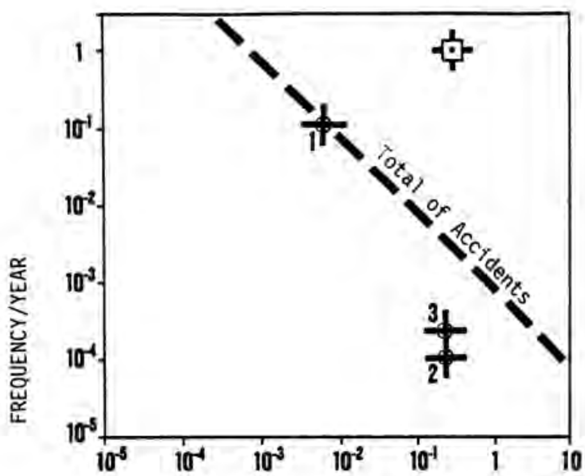
Site specific parameters of Gorleben like population density and local weather conditions were used for calculating the exposure of the general public in the vicinity of the repository due to accidental and routine release. The collective dose commitment was calculated for two regions:

- a local area surrounding 50 km
- a regional area surrounding 1,500 km.

The dose burden will be 1.4 E-4 manSv/year, respectively. The highest accidental collective dose commitment multiplied by the frequency of occurence was estimates to be 1.7 E-5 manSv/year for the general public.

Risk assessments which have been performed for the repository showed that frequencies and consequences of accidents are not very high. Therefore, the accidental risk as the product of consequence and frequency of occurence is not very high either and, in fact, much lower than the risk from routine operation.

This is demonstrated in the following figures 5 and 6 where the risk from the routine operation compared with its accidental risk. The ordinate gives the frequency of occurence of different events and the abszissa the collective dose commitments due to the radioactive effluents resulting from those events. The dashed line in this diagram is the envelop of all accidents and represents the risk. The square on the right characterizes the risk from routine operation having a probability of 1 per year and a collective dose commitment due to routine releases of about 2.8 E-1 manSv for the staff and 1.4 manSv for thegeneral public. As can be seen, it is almost higher than the accidental risk.



COLLECTIVE DOSE/ACCIDENT manSv

1 Drop of canisters in galleries

2 Collision of means of transportations with fire at the pit bottom (rail-guided)

3 Collision of means of transportation with fire in the emplacement-galleries

-ф- Routine operation

Fig. 5. The Accidental Risk in Comparison to the Risk from Routine Operation

LONG TERM SAFETY ANALYSIS

The third area to proof the correct mining design and lay-out of a repository is by means of the results of the long term safety analysis. Under normal conditions no release of radionuclides is expected from a salt repository during the post operational phase. If, however, a brine inrush into the backfilled repository is postulated, a release of radionuclides into the biosphere cannot be excluded. In this case an inflow of an unlimited amont of brine is assumed via an anhydrite layer situated in the central area of the repository.

For the chosen scenario the radiological consequences in the biosphere were calculated. The analysis was done for three subsystems like salt dome, overlaying strata and biosphere. For each subsystem computer codes are available to take into account the essential effects.

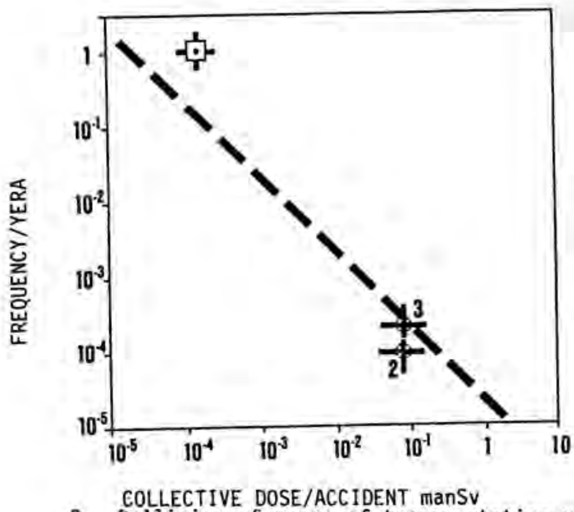
In the West-wing, where no heat generated waste is disposed, it will take 350 years for filling up the void volumen left in the backfill. In the East-wing, where spent fuel elements is emplaced, the process of convergency will be so quick, that the backfill material is compacted very early, and no intrusion of brine is possible. Therefore radionuclide release is only possible from the West-wing, where the secondary waste from the encapsulation plant, waste from the power stations, industries and research centres is disposed of.

The following equivalent doses for the maximum exposed individuals habe been calculated:

TABLE II

Equivalent Doses for the Maximum Exposed Individuals In Dependence of Time of Maximum Concentration for the Different Radionuclides

Nuclide	C 14	Se 79	Tc 99	I 129	Np 237	U 234
Time of max. Concentration-year	6,700	7,600	8,700	6,800	22,000	9,200
Dose Sv/year	4.3E-5	2.4E-6	1.0E-6	8.5E-6	5.4E-5	4.7E-6



2 Collision of means of transportation with fire at the pit bottom (rail-guided)

3 Collision of means of transportation with fire in the emplacement-galleries

Routine operation

Fig. 6. The Accidental Risk in Comparison to the Risk from Routine Operation for the General Public