412.21
received letter
dtd 2/26/88

ORGANISATION FOR ECONOMIC
CO-OPERATION AND DEVELOPMENT

NUCLEAR ENERGY AGENCY

RWM/DOC(88)6

RESTRICTED

Paris, drafted: 11th February 1988

distr.: 26th February 1988

Engl. Text Only

COMMITTEE ON RADIOACTIVE WASTE MANAGEMENT

Conclusions and Proposal from the RWMC Consultant Group Concerning the Preparation of Status Reports on National Waste Management Programmes

1. A small RWMC Consultant Group was convened in Paris on the 27th-28th October 1987 as a result of an action placed upon the Secretariat by the RWMC at its eighteenth session on the 11th-12th June 1987. A list of participants at the Consultant Group meeting is attached as Annex 1. Specifically, the Consultant Group was to give further consideration to a US proposal for an NEA initiative in documenting the status of waste disposal programmes for high-level waste and spent fuel in Member countries [SEN/RWM(87)8], and to prepare recommendations for consideration by the RWMC.

2. The following key points emerged from the discussions of the proposal by the Consultant Group:

(a) Support was given for the preparation of a set of documents which would provide a concise, factual description of the status of national waste management programmes.

(b) Ad-hoc meetings of small groups were seen as a useful means for the timely discussion of incidents and significant decisions and policy changes in Member countries, beyond fora such as provided by the RWMC. In particular, full discussion of the rationale underlying national strategies and policies, while originally part of the US proposal, should preferably be addressed by such meetings rather than in the proposed status reports.

(c) Bilateral contacts remain a useful form of timely information exchange, although there exists a need for a better identification of the key personnel, or contacts, in each country's waste management programme.

10542

412.21
NHXA 1/1

(d) Although the issue being discussed primarily involved the distribution of information to a relatively restricted technical and policy audience, there was seen to be a need for the NEA to distribute "positive statements" on the status of waste management issues to a broader audience. [See point 10(b) of the RWMC Agenda for the March 1988 meeting].

3. The Consultant Group agreed to recommend to the RWMC that a set of restricted NEA documents be prepared to provide information on the status of national waste management programmes. These documents would have the general objectives of describing the basic elements of the radioactive waste management programmes in Member countries, and of providing information for use by national authorities in discussing their own strategies in light of the approaches taken in other countries.

4. The documents, prepared by each Member country, would present concise, factual information, and would address all waste types and all waste management systems, although emphasis would be placed on repository systems and on high-level waste and spent fuel management. Regulatory aspects would be included in the documents, including strategies for site characterisation/selection and long-term safety criteria.

5. The reports would be collated by the NEA Secretariat, and made available for distribution in an appropriate published form. The documents would not be intended for broad, public distribution, but rather would be restricted (i.e., control of its distribution would be the responsibility of RWMC Members, using their own judgement and requirements). The documents would need to be updated periodically to reflect significant changes in programmes. Use of a modular format for the documents would help to facilitate such revisions without the need for complete redrafting of the documents.

6. The Consultant Group noted that a common format for preparation of the individual documents would be preferable, and have recommended that the documents be prepared in general accord with the annotated outline attached as Annex 2. Furthermore, in order to illustrate the type of information to be provided and to examine to which extent the format proposed for the status reports was suitable, the consultants agreed, on a trial basis, to undertake the preparation of preliminary status reports concerning their respective countries.

7. After a review of the prepared reports by the Group and the RWMC Bureau during the beginning of 1988, it was recognised that the attempt to prepare self-contained reports for each country was difficult and that the format led to a certain degree of repetition in the information presented. In addition, it was recognised that useful information might still be lacking in the reports, and it was considered that feedback from readers from other countries would be desirable in order to clarify or supplement the various sections of the reports. Finally, flexibility was advocated in the use of the proposed format, particularly for the description of high-level and low-level waste management strategies and practices (which could be treated separately or together), as well as for the inclusion of the references, and/or short summaries on regulatory matters (particularly concerning main regulatory criteria). It was therefore recommended that national authorities should use their own judgement in preparing their contributions and adapting the proposed format to their own particular situations.

8. After consultation with the RWMC Bureau, it was decided that the preliminary contributions prepared by the consultants should be presented to the RWMC as examples of possible national status reports, for the purpose of facilitating discussions at the Committee meeting, and on the understanding that they only represent draft documents (see comments in paragraph 7 above). Draft reports from France, Sweden and Switzerland have, therefore, been attached as Annex 3 to this document. Contributions from the Federal Republic of Germany and the United States will be distributed shortly in a separate Addendum to this document.

9. Based on the above considerations, the Committee is invited to discuss the proposal of the Consultant Group, and to agree with the compilation and distribution of national reports prepared in accordance with a common format.

ANNEX 1

List of Participants

Consultant Group Meeting on the Status of National Waste Management Programmes; Paris, 27-28 October 1987

FRANCE

Mr. J. LEFEVRE, Directeur Délégué pour les Effluents et Déchets Radioactifs, Commissariat à l'Energie Atomique (CEA), Centre d'Etudes Nucléaires, B.P. No. 6, F-92265 Fontenay-aux-Roses Cedex.

Mr. A. FAUSSAT, Directeur Adjoint, Agence Nationale pour la Gestion des Déchets Radioactifs (ANDRA), B.P. No. 510, 29-33, rue de la Fédération, F-75752 Paris Cedex 15.

F.R. of GERMANY

Mr. H. GEIPEL, Bundesministerium für Forschung und Technologie, Heinemannstrasse 2, D-5300 Bonn 2.

SWEDEN

Mr. N. RYDELL, National Board for Spent Nuclear Fuel (SKN), Sehlstedtsgatan 5-9, S-115 28 Stockholm.

SWITZERLAND

Mr. E. KOWALSKI, Société Coopérative Nationale pour l'Entreposage de Déchets Radioactifs (NAGRA), Parkstrasse 23, CH-5401 Baden.

UNITED STATES

Mr. C. COOLEY, Office of Civilian Radioactive Waste Management, US Department of Energy, Forrestal Building, RW-40, Washington D.C. 20585.

Mr. T.H. ISAACS, Deputy Associate Director of the Office of Geologic Repositories, Office of Civilian Radioactive Waste Management, US Department of Energy, Forrestal Building, RW-43, Washington D.C. 20585.

NEA SECRETARIAT

Mr. J.-P. OLIVIER, Head, Radiation Protection and Waste Management Division, OECD Nuclear Energy Agency, 38, boulevard Suchet, F-75016 Paris, France.

Mr. L. CHAMNEY, Radiation Protection and Waste Management Division, OECD Nuclear Energy Agency, 38, boulevard Suchet, F-75016 Paris, France.

ANNEX 2

ANNOTATED OUTLINE
WASTE MANAGEMENT SYSTEM DESCRIPTIONS

A. GENERAL STRATEGY

1. Overall Waste Management Strategies and Systems

- general strategy and system description for storage, transport and disposal for radioactive waste

2. National Policy/Implementing Laws

- general policies, laws and procedures pertinent to radioactive waste management (including public consultation, and safety)

3. Organisational Structure

- regulatory and operational responsibilities

4. Policy on Spent Fuel Management

- in particular, specific policies related to fuel reprocessing

5. Overall Schedule

- schedule for implementing waste management programme

6. Total System Costs and Funding

- expected costs and planned budgets for implementing waste management programme
- source and management of funds for implementing waste management programme
- R&D budgets and funding

7. Quality Assurance Considerations

- specific guidelines for implementing quality assurance requirements (either in general or specific sections)

8. International Co-operation

- description of co-operative programmes with other countries, and a listing of bilateral agreements

B. STORAGE SYSTEM

1. National Policy

   - national policy for storing spent fuel and radioactive waste

2. Requirements

   - selected requirements for storing spent fuel and radioactive waste

3. Description

   - description of facilities and location

4. Schedule

   - schedule for constructing or expanding storage facilities

5. Costs and Funding

   - expected costs and planned budget for storing spent fuel and radioactive waste

   - source of funds for implementing storage plans

6. Experience/Status

   - how long has spent fuel and radioactive waste been stored in what type of facility

7. Safety Considerations

   - what is planned to assure operational safety and performance

8. Decommissioning Considerations

   - what plans have been made to decommission the storage facilities

9. Quality Assurance Considerations

   - how is the quality assurance programme for storage being implemented

C. TRANSPORTATION SYSTEM

1. National Policy

   - general policy for transporting spent fuel and radioactive waste

2. Requirements

. selected requirements for transporting spent fuel and radioactive waste

3. Description

. description of transport casks plus the major characteristics of truck, rail and ship transport

. description of the methods for transferring spent fuel and radioactive waste between facilities, and the transport modes

4. Schedule

. schedule for transporting spent fuel and radioactive waste to storage and disposal facilities

5. Costs and Funding

. expected cost and planned budget for transporting spent fuel and radioactive waste

. source of funds for implementing transportation plans

6. Experience/Status

. how long has spent fuel and radioactive waste been transported, and in what type of casks

7. Safety Considerations

. what is planned to assure operational safety and performance

8. Decommissioning Considerations

. what are the decommissioning plans for the transport casks and any other special facilities or equipment related to the transport of spent fuel and radioactive waste

9. Quality Assurance Considerations

. how is the quality assurance programme related to transportation being implemented

D. DISPOSAL SYSTEMS (REPOSITORIES)

1. National Policy

. national policy for disposal of spent fuel and radioactive waste in a repository (including public consultation)

2. Requirements

. selected requirements for disposal of spent fuel and radioactive waste

3. Regulations

. specific regulations governing the disposal of spent fuel and radioactive waste in a repository

4. Site Selection

. approach and methods used for site selection

5. Underground Research

. plans for constructing and using an underground research facility to understand the characteristics of a future repository

6. Repository Concepts

. Description

* specific description of repository concept
* provide references

. Waste Receipt and Handling

* description of spent fuel and radioactive waste handling methods at the repository

. Waste Package

* description of the waste package

. Emplacement

* process of drilling emplacement holes and placing waste packages inside, or of placing waste packages in prepared tunnels/silos/caverns/etc

- Buffer, Backfill and Seals

  * description of the buffer, backfill and sealing materials that will be placed around the waste packages, and in galleries/ tunnels

- Temperatures and Pressures

  * designed internal and external temperatures and pressures of the waste package

- Radiation Protection and Safety Considerations

  * design goals for radiation protection for routine and non-routine events
  * what is planned to assure operational and post-operational safety and performance

- Decommissioning

  * shaft sealing technology/criteria
  * decommissioning plans for the surface facilities of the repository

- Post-Closure

  * requirements for monitoring and retrievability

7. Schedule

. expected schedule for site selection, underground research, construction, operation and decommissioning

8. Costs and Funding

. cost estimates for equipment and labour for site selection, research, construction, operation and decommissioning

. source of funds

9. Experience/Status

. experience in underground research and siting activities, construction and operation

10. Quality Assurance Considerations

. how is the quality assurance programme being implemented

ANNEX 3

DRAFT EXAMPLES OF NATIONAL STATUS REPORTS

France

Sweden

Switzerland

DRAFT

RADIOACTIVE WASTE MANAGEMENT IN FRANCE

A/ GENERAL STRATEGY :

1. Overall waste Management Strategy and system :

The current French nuclear capacity is 44 GWe, primarily PWR, and expected to increase to 57.3 GWe in 1990 and 71 GWe by the year 2000. The nuclear industry is geared to a closed fuel cycle, with reprocessing and use of the recovered plutonium in breeders and light-water reactors. Spent fuel is stored first at the reactor site then in a pool-type AFR facility at the reprocessing plant until it is reprocessed.

All short-lived low and intermediate-level wastes are currently being emplaced in engineered surface disposal facilities at the La Manche site adjacent to the La Hague Plant near Cherbourg. Capacity of the La Manche installation will be reached about 1990. An additional site (Centre de stockage de l'Aube) is in the construction stage at SOULAINES, in Central France, South west of Paris. Commissioning should occur in 1990.

High-level waste is to be vitrified (by the AVM process in use at the Marcoule Centre) and stored in a vault until the canisters can be placed in a repository (a storage period of 30 years or more is envisioned).

All long lived wastes i.e. TRU and HL glass are to be disposed of by emplacement in a suitable deep geological formation. Four types of formations are being investigated : salt, granite, clay and shale.

A proposal for a site for an underground laboratory will be submitted by the end of 1989, with the intention of investigating the potential of this site to receive an actual long-lived waste repository. Construction of this In situ Validation Laboratory should begin in 1990, the validation test being completed by 1996. If the site proves to be acceptable, the repository would become operational at the beginning of the next century for TRU and around the 2010s for HLW.

Long term radioactive waste management is entrusted to the National Agency, ANDRA, created within CEA in 1979.

2. National Policy/Implementing laws :

Waste management policy has been set out in the CEA report to Government "General Radioactive Waste Management Program" approved by the government and made public on June 19, 1984.

.../...

2.1.

All "basic nuclear installations"* including waste storage and disposal facilities are subject to :

a/ National environment protection laws, and regulations, including the law on "the democratisation of public enquiries and environment protection" dated July 12, 1983 with application decrees of April 23 and 24, 1985 ;

b/ Specific regulations (decree of December 11, 1963 modified April 24, 1985) governing their authorization and control procedures.

c/ National laws and regulations governing man-at-work health protection, and specifically the laws and decrees concerning the protection of workers against "ionizing radiation". The latest of which are the decree of October 2, 1986 (adapting legislation to the Euratom Directive of 1980) and the Departmental Order of September 30, 1987 confirming application of previous procedures to the new legislation.

d/ Fundamental Safety Rules (RFS), which express basic safety principles, procedures and technical specifications, are enacted. The following concern radioactive waste management specifically:

RFS I.2. Safety aims and design bases for surface installations intended for the long term storage of short or medium half-life solid radioactive waste, of low or medium specific activity (revision 1 dated 19.6.1984).

RFS III.2.a. General rules applicable to the production, control, processing, packaging and storage of various types of waste resulting from pressurized water reactor fuel reprocessing (24.9.1982).

RFS III.2.b. Special rules applicable to the production, control, processing, packaging and storage of high activity vitrified waste, resulting from pressurized water reactor fuel reprocessing (12.11.1982).

RFS III.2.c. Special rules applicable to the production, control, processing, packaging and storage of low or medium activity bitumen-solidified waste, resulting from pressurized water reactor fuel reprocessing (5.4.1984).

RFS III.2.d. Special rules applicable to the production, control, processing, packaging and storage of cement-solidified waste resulting from pressurized water reactor fuel reprocessing (1.2.1985).

RFS III.2.e. Prerequisites for acceptance of packages of encapsulated solid waste intended for surface storage (31.10.1986).

---

* Installations where radioactive material is handled in quantities above limits fixed by decree.

.../...

2.2.

Construction authorization is subject to approval of a safety report and to a "Public Enquiry". This entails : 1. making available to the public, in the area of the planned installations, all necessary information concerning the project including the environmental impact statement ; the places, length of time, and conditions of making this information available are stipulated in the April 24 and 25, 1985 decrees ; 2. making at the same time and in the same place registers available for the public to set down in writing comments, reasons for opposition and counter proposals. These can also be mailed direct to the Enquiry Commission.

The Enquiry Commissar or Enquiry Commission is designated by the local administrative Court (Président du Tribunal Administratif). The Commission can request supplementary information from the licence applicant ; it can also request a public information meeting, subject to approval by the local government representative (Préfet).

The Commission reports on the comments received after hearing anyone it judges pertinent and useful to question and the applicant if he (the applicant) so wishes.

The Commission report, conclusions, and recommandations, are sent to the local Administrative Court to the applicant, the Regulatory authorities and each of the local town councils (where the enquiry took place) where there are made available to the public for one year.

There are special stipulations where construction entails pre-emption of land rights, which requires that the installation be declared "of Public Utility".

2.3.

Operation licensing is subject to approval of the final safety report and if relevant, radioactive gaseous and/or liquid effluent release authorization, which entails approval of a preliminary effluent release impact study.

3. Organisational Structure :

3.1.

The broad outline of waste management policy, the national rules, regulations and control, as well as the authorization and licensing of nuclear installations, waste disposal site included, are the responsability of the government. The main ministry concerned is the Ministry responsible for Industry. A special department of this Ministry, the SCSIN (Service Central de Sûreté des Installations Nucléaires) develops and enforces safety regulations, issues construction permits and operating licenses,

.../...

monitors operating safety for all "Basic Nuclear Installations", including nuclear waste repositories.

All concerned ministries are consulted, a special role being played by the Departments responsible for Health and Labor, with a specialised service SCPRI (Service Central de Protection contre les Rayonnements Ionisants), whose approval is compulsory before effluent releases can be authorized. SCPRI monitors and controls releases and monitors radioactivity in the environment apart from its role in health protection proper, as concerns labor in particular.

3.2.

Waste management proper is the responsibility of the waste producer who usually performs all necessary operations to put the waste into a form suitable for disposal (i.e. conforming to ANDRA specifications see below) until it is handed over to ANDRA for disposal. As stated above, ANDRA is responsible for long term management. As such, ANDRA :

. sets the requirements which waste packages must fulfill to be accepted for disposal, it is up to ANDRA to see that these requirements conform to RFS and satisfies the safety authorities and to ensure quality assurance and control (as concerns waste forms, waste packages, waste disposal facilities, etc.).

. plans disposal facilities and their financing according to needs.

. selects, constructs, operates and closes waste disposal facilities.

4. Policy on spent fuel management :

Reprocessing of spent fuel in La Hague (enriched U) and Marcoule (natural U) installations. FBR fuel has been reprocessed both at La Hague and in the pilot plant in the process of being renovated at Marcoule.

5. Overall Schedule :

- Low Intermediate short lived waste :

. Centre de Stockage de la Manche operating since 1969. Due for closure in the 90ies,

. Centre de Stockage de l'Aube now building due for commissionning 1990.

.../...

- Long lived waste :

. Selection of site of In situ Site Validation Laboratory launched in 1987. Due to end 1989 with proposal of one site (salt, granite, shale or clay),

. 1990 : start construction of ISVL,

. Early 2000s : start up of TRU disposal facility,

. 2010 : start up of HL glass disposal facility.

6. Total system costs and funding :

R&D on general management systems and safety is conducted and funded by CEA. Technology R&D (treatment and conditioning, engineered barriers) is also mainly carried out and funded by CEA, with some contribution from the European Community, and from waste producers such as COGEMA. CEA waste R&D annual budget approximates 250 MFF in 1988. Other waste producers (mainly EDF) and the industry carry out research in their own laboratories, as also do various Universities and Institutes.

Work on geological disposal site investigation, ISVL, disposal lay-out, etc. being the responsibility of ANDRA, and either directly carried out or more often contracted out by ANDRA, is financed by the waste producers through ANDRA, which charges them their share of the cost according to their future delivery forecasts. 120 MFF were spent in 1987 - 250 MFF are expected for 1988. The present site investigation program is reckoned to reach a total cost of 1 billion FF. Constructing and operating the ISVL is expected to cost 1.5 billion FF.

Short-lived waste disposal is payed by the waste producer to ANDRA on a cost basis. Annual turn-over is of the order of 160 MFF. To this must be added the Centre de stockage de l'Aube construction cost (800 million FF), also shared between the producers according to the disposal volume they have booked.

7. Quality assurance considerations :

ANDRA ensures that specific measures are established to guarantee the quality of radioactive waste packages received at all disposal centers.

To achieve this goal, ANDRA has a quality assurance system based on the performance of systematic management and control actions throughout the cycle of operations ranging from waste production and packaging in appropriate containers up to and including long-term disposal. This system specifies the required quality, together with the means to achieve and maintain it.

.../...

All actions undertaken for this purpose comply with official French requirements, particularly those set forth in the ministerial order dated August 10, 1984.

8. International co-operation :

France being a member of IAEA, OECD and EEC, is actively involved in their overall waste management activities and in many specific cooperative R&D projects. Moreover, bilateral R&D cooperation agreements have been signed with US DOE, Sweden, Switzerland, Spain and cooperative projects are actively pursued.

B/ STORAGE SYSTEM

1. National Policy :

Storage of spent fuel first at reactor site (approx 1 year) then at reprocessing plant site (2 to 3 years).

Storage of radioactive waste :

LLW : no long storage. Conditioned waste is shipped to disposal site as soon as possible.

TRU : at production site awaiting availability of repository.

HLW : id.

2. Requirements :

Same regulation and procedures as all Basic Nuclear Installations.

3. Description :

LWR spent fuel : Pool storage AR and La Hague.

FBR spent fuel : dry storage.

HLW : . liquid : High integrity storage tanks at La Hague (LW fuel) and Marcoule (Nat U fuel),

. vitrified : air-cooled vault at Marcoule and La Hague.

TRU & LLW : Engineered halls.

.../...

4. Schedule :

Spent fuel storage pools at La Hague :

NPH	: 2000 t U	1981
C	: 2400 t U	1984
D	: 2400 t U	1986
E	: 4000 t U	1988

Glass storage : AVM Storage at Marcoule ; two storage facilities at La Hague, one for each vitrification plant, R7 and T7 ; capacity of first module : 4500 cannisters (reprocessing of 800 t/year for 5 years) to be commissioned 1988.

5. Costs and funding :

The costs for intermediate storing spent fuel and radioactive wastes on the reprocessing site of La Hague are included in the global cost of reprocessing service within the limits of time of intermediate storage described before.

6. Experience/status :

Spent fuel : the first spent fuels have been received at La Hague site in 1973. So far the largest storage period for some fuels has been approximately 10 years.

Radioactive wastes : the eldest wastes (FP solutions, liquid wastes, treatment slurries...) are stored in tanks and vaults since 1966.

7. Safety consideration :

For all the different types of storage, safety reports are submitted to the agreement of the French Safety Authorities (SCSIN).

8. Decommissioning consideration :

General studies for decommissioning the storage facilities are presently being carried out.

9. Quality assurance consideration :

Quality assurance is implemented in the form of the general program of quality assurance of Reprocessing Branch of Cogema.

.../...

C/ TRANSPORTATION SYSTEM

1. National Policy :

There is no national policy as such in France for transporting spent fuel or radioactive waste : the industry is free to undertake transports which might be needed provided that the relevant regulations are met and necessary formalities are fulfilled.

2. Requirements :

A full set of regulations exists regarding transport of spent fuel and radioactive waste, practically all of them modelled upon IAEA recommendations as far as safety is concerned.

3. Descriptions :

Many different types of transport casks are utilized for transporting radioactive materials, most of them being of French design. Concerning more particularly spent fuel and LWRs fuel, Cogema has elaborated a set of standards for the design of casks serving the French reprocessing plants (dry casks, steel body, double containment, large capacity, standard sizes and ancillary equipments). Transport is made preferably by rail from power plants situated in continental Europe and by ship from Japanese plants, with road transports generally limited to short distances between a nuclear site and a rail siding equipped for transfer from truck to rail car.

Short-lived waste are transported to the Centre de stockage de la Manche by either road vehicles, 23 t to 27 tonnes semi-trawlers or standard rail cars (56 t), either equiped for containers or canvas-covered for concrete blocks.

4. Schedule :

Schedules for transporting spent fuel are generally dictated by commercial considerations between utilities and the reprocessor, with the reprocessor being, as a rule, in charge of taking delivery of the spent fuel at the power plant when required by the utilities.

Schedules for transporting LLW are agreed upon between the waste producer, the freight company, and ANDRA.

5. Costs and funding :

Transport of spent fuel and more generally of nuclear materials is considered in France as a normal industrial activity : suppliers and customers agree upon transport prices on a case by

.../...

case basis, prices being established in order to allow the transporter to recover its investments properly and to leave a reasonable margin considering the normal contingencies met in this particular field.

6. Experience and status :

Spent fuel are transported on a routine basis in France since 1966 (GCR fuel) and 1974 (LWR fuel). Up to 1987 some 14 800 tonnes of uranium in spent fuel have been transported from various power plants around the world to La Hague and Marcoule reprocessing plants in 4 100 different casks movements.

Short-lived waste have been transported in France for over 20 years. Each year an average of 30.000 m^3 are the subject of :

2800 road transports (= 400.000 km),
550 rail transports (= 420.000 km).

7. Safety considerations :

Being in charge of transporting the spent fuel, the industry is responsible for taking all necessary care to assure operational safety, subject to the controls imposed by regulation and with the necessary back-up from the relevant public bodies in case of emergency or accidental circumstances.

8. Decommissioning considerations :

Decommissioning is considered by the companies involved in the transport activity as part of their responsability. The experience is relatively limited as of today, however several units, including casks in the range of 50 tonnes have already been decommissioned, with part of the body and the material being decontaminated for re-use and the rest being cut and conditionned in a form suitable for disposal by ANDRA.

9. Quality assurance considerations :

In accordance with the IAEA recommendations and in full compliance with the instructions of the French Ministry of Industry, the transport industry is aiming to generalize the implementation of Quality Assurance Programmes for the transport (design and operation), specially regarding spent fuel transport.

The freight companies working for ANDRA are submitted to a qualification procedure within the frame of the ANDRA quality assurance program.

.../...

D/ DISPOSAL SYSTEMS (Repositories)

1. National Policy :

No spent fuel disposal. See section A/ 1. and A/ 2. for waste disposal policy.

2./3. Requirements - Regulations :

a/ Low and intermediate level short-lived waste disposal in engineered surface facility.

Ruled by above mentioned RFS 1.2. edicted June 19, 1984, Major provisions :

α content :

α content limit : 370 Bq.g (0.01 Ci/t) averaged over the site.

α content limit per package : 3.7 kBq.g (0.1 Ci/t), can reach, on a case by case basis, 18.5 kBq.g (0.5 Ci/t) for individual packages.

Site selection criteria :

Seismicity, geotechnical and tectonic stability, hydrology (good knowledge of local hydrogeology, low water table, dilution capacity of draining water system).

Intrinsic safety : soil radionuclide retention capacity.

b/ Long lived waste disposal :

No rules yet issued. Report by Ad Hoc Commission on technical criteria for underground disposal, made public November 1987, to be considered by safety authorities.

4. Site Selection :

4.1. Short-lived radioactive waste :

To replace the Centre de la Manche which will be filled to capacity early in the 90ies, ANDRA has looked for another site. Exploration of several areas in France led to the selection of the Soulaines site. This site fits a model concept which was defined in order to add to the engineered barrier safety system. It is made up of a small hill of filtrating sandy-clay material, over a water-tight layer which isolates the water table. The hydrological system is therefore simple to study and control.

.../...

4.2. Long-lived radioactive waste :

Considering the broad variety of possible host rocks in France, it has been decided to investigate the possibility of creating an underground repository in different rock types : sedimentary rock such as clay or salt and hard rock such as granite or schists.

The possibility of having suitable sites in several host rocks provides an interesting flexibility for the final choice. The first step of the site selection process described in the National Program was to compile a national inventory of the possible sites, based on criteria among which the most important were : long-term stability, and favorable hydrogeologic sites with very low permeability and good physico-chemical properties such as nuclide retention. The national inventory was completed at the end of 1983. About 30 zones covering the four main typical geologic settings : clay, outcropping granite, schists, and salt, were identified as possible locations for a future repository. In some cases a combination of layers of the different materials improves the isolation capacity of the site.

Among the 30 zones, a preselection was made of the most attractive ones on which further investigations had to be carried out. The second step now in progress is field investigations to confirm the choice of the preselected sites and to meet the Government's requirement that a candidate location be nominated for the In situ Site Validation Laboratory (ISVL) by the end of 1989. The kind of work performed naturally depends on the type of geologic formations, but in every case it needs geophysical measurements from the surface and several deep drillings with core recovery. This second phase started in 1987 on four zones covering all the different typical kinds of host rock.

5. Underground Research :

After a candidate site has been selected, construction of the In Situ Underground Site Validation Laboratory will start. The laboratory will be the main tool to complete the site selection process by validating the site. Validation means that, with the data collected during this phase, it will be possible to demonstrate the technical feasibility and the economics of the repository, and to prepare a preliminary safety impact report to show that the future repository's consequences for the environment are acceptable.

To achieve this, it is necessary to explore in depth the whole volume of rock involved in the repository construction and to carry out in-situ experiments to confirm thermal and mechanical behaviour of the host rock and to evaluate and model the isolation capability of the whole system of barriers, including backfilling material and the different layers of the geosphere. Construction of the ISVL will probably spread over two years and its implementation will last between two and three years. If the program proceeds normally, site evaluation will be obtained before the end of 1994.

.../...

Prior to the actual setting up of the ISVL, a sizable R&D program has been underway (in situ and in laboratories) for some years in order to improve understanding of basic hydrological and geochemical mechanisms, and to develop instrumention and methodology for the demonstration of long-term safety.

6. <u>Repository concepts</u> : (ANDRA)

6.1. Short-lived waste :

The experience acquired at the Centre de la Manche enabled ANDRA to develop a disposal system which results in protecting the waste packages both from weathering and human intrusion for the length of time necessary for decay (< 300 years).

This system is ensured through the quality of the waste package, that of engineered barriers which protect them from water, a surveillance network and the quality of the site itself.

6.2. Long-lived waste :

ANDRA has not yet decided on a disposal lay-out. The various concepts which can be envisaged for the reception, handling, and lowering of the packages, for the drifts or silos etc, are being examined for various host rocks.

The economic consequences of these concepts, according to the data collected during the exploratory phase, could be one of the factors for site selection.

7./9. <u>Schedules</u> : see A.5

8. Costs and funding : see A.6

10. Quality assurance considerations.

According to the ministerial order dated August 1984, every element concuring to the safety of a Basic Nuclear Installation, is submitted to a Q.A. program.

D R A F T

RADIOACTIVE WASTE MANAGEMENT IN SWEDEN

GENERAL STRATEGY

Overall Strategy and System

Transport, storage and disposal facilities will be designed to handle all radioactive wastes from the nuclear power reactors, the Studsvik Research Center and hospitals, universities and industry as well.

Spent fuel is stored at reactors (which are all on the sea coast) for about 1 to 5 years, then transported by ship to a single national storage facility where it will be stored for 30-40 years. The aging makes the repository design simpler and repository volume smaller. Following interim storage, the spent fuel will be shipped to a geologic repository in crystalline rock (granite, gneiss or gabbro), without reprocessing. Long-term safety depends on multiple barriers: spent fuel, canister, buffer, and host rock. Disposal of long-lived ILW is also planned similarly after interim storage. Short-lived wastes will be disposed of without interim storage. Some LLW will be disposed by shallow land burial at the reactor sites. All decommissioning wastes will be included in the waste management system.

A site license application for the geologic, repository will be submitted about the year 2000. Repository construction will start about 2010.

National Policy/Implementing Laws

The management and disposal of radioactive wastes in Sweden are regulated by:

- The Act on Nuclear Activities (1984)
- The Radiation Protection Act (1958)
- The Act on Financing of Future Expenses for Spent Nuclear Fuel etc.

All waste generated from Swedish nuclear power plants shall be disposed of in Sweden. A solution of the waste problem shall be devised by the same generation that utilizes the nuclear electric power. Nuclear wastes from other countries will not be accepted in Sweden. Disposal of spent fuel will be in a deep geologic formation.

The Stipulation Act of 1977 required that the nuclear power utilities "demonstrate" a safe method for disposal of spent fuel or vitrified HLW before a nuclear power plant due for commissioning could receive a permit to fuel the reactor.

The Act on Nuclear Activities of 1984 amended the 1977 Stipulation Act by requiring the the owner of nuclear reactors shall bear the responsibility for all steps necessary for safe handling and final disposal of all radioactive residues from nuclear energy production, including the costs. The Act also requires a comprehensive plan by waste generators for R & D that will lead to final disposal.

Based on a 1980 referendum on nuclear power, Parliament decided to phase out all nuclear power by the year 2010.

The public must accept and support the waste management system. The local government has veto power over having a waste management facility on land within its jurisdiction as it also has on other industrial installations which may have an impact on environmental quality.

Institutional Framework

The responsibilities of the four nuclear utilities are handled by their jointly owned company, the Swedish Nuclear Fuel and Waste Management Co (SKB). SKB carries out the R&D program for handling and disposal of the waste which by the Nuclear Activities Act shall be submitted to the Government every third year.

The National Board for Spent Nuclear Fuel, SKN, reviews and comments to the Government on the SKB program. SKN also proposes the fee on nuclear electricity that the utilities must pay to a special fund stipulated to cover the costs for present and future handling and disposal of spent fuel and for decommissioning of the nuclear power stations.

The Swedish Nuclear Power Inspectorate, SKI, is the safety authority for nuclear facilities. SKI licenses and inspects repositories with respect to safety. The National Institute of Radiation Protection, SSI, supervises in the same way the implementation of the Radiation Protection Act. The license to construct and operate a repository is given by the Government on recommendation from SKI and SSI. In practice the three authorities act closely together to cover all aspects of final disposal in a consistent manner.

Policy on Fuel Reprocessing

Reprocessing of 140 MTU has been contracted with UK (Sellafield). This quantity has been shipped and will be reprocessed. No waste will be returned to Sweden. About 730 MTU was contracted with France (La Hague). Sweden has shipped 57 MTU, however, these 57 MTU have been exchanged with West Germany who will take care of the Pu and the waste. Sweden in turn will take care of 24 MTU MOX-fuel for Germany. The MOX-fuel is less suitable for reprocessing and will be disposed of directly. The remaining quantities contracted for reprocessing at La Hague will not be used by Sweden. Instead, about 25 % of the contracted amounts have been transferred to other Cogema customers. Efforts are in progress to do the same with the remaining 75 %.

Overall Schedule

- 1977 Start repository R&D and site investigations
- 1985 Start up of CLAB central storage facility for spent fuel
- 1988 Commission SFR for disposal of short-lived radwastes
- 1992 Start up underground research facility
- 1998 Complete site characterization for spent fuel repository
- 2000 Site license application
- 2010 Start construction of spent fuel repository
- 2020 Commission spent fuel repository and start emplacement
- 2050 Complete operational phase of spent fuel repository

System Costs as of 1/87. (Life-Cycle through 2050)

- 3840 MSEK Administrative, R&D, siting, URL
- 7920 MSEK Decommission all nuclear power reactors
- 7760 MSEK CLAB central spent fuel storage
- 16350 MSEK Spent fuel repository with receiving and encapsulation facility
- 1290 MSEK Repository for long-lived LLW and ILW
- 1840 MSEK Repository for short-lived LLW and ILW
- 1790 MSEK Transportation of all wastes
- 4790 MSEK Foreign reprocessing of 640 MTU spent fuel
- 45580 MSEK Total Program

System Funding

Funding is collected at a present rate of 0.019 SEK/kw-hr of nuclear electricity by the National Board for Spent Nuclear Fuel. This rate is reassessed yearly by the Board in the light of cost developments for the planned waste management system. The funding shall cover all costs for spent fuel management off the plant premises and for decommissioning of the plants. The dues are deposited at the National Bank of Sweden. The interest is added to the fund to preserve the real value of the funded capital. The SKB is reimbursed from the fund for its costs for waste management R&D and costs for construction and operation of facilities. Costs for operating wastes from reactors including their disposal are not covered from the fund. They are born directly by the utilities.

System Decommissioning Considerations

Decommissioning of all parts of the system must be done by dismantling to unrestricted use of the nuclear sites.

Q/A Considerations

The main objective is to provide assurance that there are no deviations from the desired quality that can significantly impair the system safety. A systematic program of quality control, as typically done in nuclear activities, will be applied to the design, manufacture and construction of the various parts of the system.

STORAGE SYSTEM

National Policy

In-pool storage of spent fuel at reactor sites is used for about 1 to 5 years, followed by central interim storage for 30 to 40 years. Interim storage was proposed by a Government appointed Parliamentary committee in 1975/76 to allow time to resolve final waste management plans. Since then, storage is also desired to allow decay of spent fuel radioactivity to allow for cooler repository temperatures and for more efficient repository design.

Requirements

Central storage is required for total Swedish production of spent fuel (ca. 7800 MTU) and core components of decommissioned reactors until a repository is commissioned. If the repository is delayed, storage for longer than 30 to 40 years may be required at the central facility. Handling capacity is to receive or discharge the total inventory of spent fuel in 25 years.

Storage at a central site was picked over reactor sites based upon economics. Wet storage was selected over dry storage based upon available experience. Wet storage has been well demonstrated all over the world to be safe, it allows for handling and cooling of short-cooled fuel, and it minimizes fuel degradation by keeping spent fuel temperatures low.

Underground storage was selected after cost studies showed costs for above-ground and below-ground storage were a standoff. Also, Sweden has much experience in using underground storage for other materials, and underground storage provides protection from outside activities such as sabotage or war.

Description

The central storage facility for spent fuel (CLAB) is located adjacent to the site of the Oskarshamn nuclear power station. CLAB consists primarily of an above-grade spent fuel receiving and handling facility and an underground, man-made rock cavern in granite about 30 meters below grade at the sea coast site. The rock cavern is completely lined with reinforced concrete, and the ceiling is lined with sheet metal. Handling capacity is 300 MTU/year. Pool water is cooled by exchange with sea water to a normal temperature of 32 C.

The initial facility has a storage capacity of 750 MTY in each of four stainless-steel-lined concrete pools (plus one spare). Each pool has a water depth of about 12.5 m and holds 3,000 cu. m water. Later expansion will be added by excavating more caverns parallel to the first and using the same handling facility.

The spent fuel in transport casks is unloaded and transferred to open, square handling canisters under water in the surface facility. The canisters are lowered to the storage area in a water-filled elevator. In the storage area, they are moved under water by another transport system to the final storage location in storage racks.

Schedule

- 1977 Siting application submitted
- 1978 Siting application approved
- 1979 Construction permit application submitted
- 1980 Construction permit granted and construction started
- 1985 Operating permit granted and facility was commissioned
- 1995 Approximate date for needed expanded CLAB capacity
- 2050 Approximate date for emptying and decommissioning

Costs (as of 1/87)

- 1750 MSEK Capital through 1986
- 250 MSEK Operating through 1986
- 3030 MSEK Total life-cycle capital
- 4490 MSEK Total life-cycle operating
- 240 MSEK Decommissioning
- 7760 MSEK Total life-cycle costs
- 70 MSEK Approximate average operating costs/yr

Funding

Funding is included in the waste management fund.

Experience/Status

Storage of spent fuel in reactor pools since 1973. Interim storage of spent fuel in CLAB since summer 1985. Current capacity of the CLAB is 3,000 MTU, and final capacity will be 8,000 MTU. 650 MTU is stored at CLAB as of 1/88.

TRANSPORTATION SYSTEM

National Policy

Spent fuel and wastes are to be transported in casks placed in the hold of a specially-built ship and transported along the sea cost by ship. The nuclear power stations, CLAB, and repository sites are on the coast. Sea transport is seen to be advantageous over land transport for the very heavy packagings required, and Sweden and other countries have much favourable experience in sea transport of numerous materials, including early use for transporting spent fuel from Sweden to foreign reprocessing plants.

Requirements

Transport the approximately 7,800 MTU of spent fuel from the four nuclear power stations to the CLAB interim storage facility between 1985 and about 2015. Then transport the same spent fuel from CLAB to the SFL deep geologic repository between 2020 and 2050.

Transport the short-lived LLW and ILW from reactor operations to the SFR repository from 1988 to 2010 at which time all reactors will be shut down. Transport the long-lived LLW and ILW from reactor core components and from certain reactor decommissioning activities to the SFL repository between 2020 and 2050.

Description

Spent fuel is shipped in TN 17/MK2 nitrogen-filled casks (under total internal pressure of about 0.5 atmospheres) made of thick steel cylinders with a neutron-shielding layer, copper cooling fins on the surface, and removable impact limiters on each end. Cask capacity is 17 BWR or 6 PWR assemblies (about 3.2 MT fuel), and cask empty weight is about 76 MT. The cask is sized as the maximum capacity that can be handled at all the Swedish nuclear power stations. Larger casks are planned to be used from CLAB to repository. The casks use conventional technology. Similar casks will be used to transport the long-lived core components from the reactors to the deep geologic repository.

Casks are transported from the reactor stations to the CLAB (all facilities are on the sea coast) on a specially-designed ship, MS Sigyn (which can hold 10 transport casks, or about 32 MT fuel). The ship was first placed in service in 1983 to transport spent fuel from Sweden to France. Casks are placed in holds which have double bottoms on top of the double hulls of the ship. The ship is classified for use in ice. The ship is 90 m long, 18 m wide, requires 4 m draft. It weighs about 2,000 MT empty, and can carry a maximum payload of 1,400 MT. Its cruising speed is about 11.5 knots. The ship is designed for roll-on/roll-off or lift-on/lift-off handling. The ship can make 15 to 20 trips/season to CLAB plus 6 to 7 trips/season to France. The same ship will transport the spent fuel from CLAB to the spent fuel repository starting in 2020. The ship will transport the short-lived LLW and ILW to the SFR repository starting in 1988. These latter wastes will be in large, rectangular metal or steel containers.

Interfaces

At the reactor, the cask loaded with spent fuel is lifted out of the pool and placed horizontally on a transport frame with the cask trunnions used for mounting and tie-down. A special one-piece overweight truck is used for driving the casks, mounted on its transport frame, to/from the reactors and on and off the transport ship. On the ship, the trailer wheels are lowered and the cask transport frame and cask are lowered onto the floor of the ship's hold where they are fastened for shipment. Overweight truck is used because of simpler interfaces (relative to rail transport) and the relatively short on-land transport distances. Also, the truck offers special lift-on/lift-off capabilities. At the CLAB port, the casks are removed from the ship by the same process that they were loaded onto the ship. At the CLAB unloading station, the casks are removed from their transport frame by overhead crane; the casks are moved horizontally (casks are oriented vertically) into a pool in the CLAB canyon-type facility. Spent fuel is unloaded under water by conventional techniques.

The short-lived wastes, in their large steel or concrete containers, are handled in a similar manner. The wastes in their respective containers can also be lifted on and off the transport ship by crane after removing the cover blocks for the ship holds.

Schedule

- 1981 Order the transport ship M/S Sigyn from France
- 1983 Commission transport ship M/S Sigyn and transport spent fuel to France
- 1984 Cease transporting spent fuel to France
- 1985 Initiate transport of spent fuel to CLAB
- 1988 Start transporting short-lived reactor wastes to the SFR repository
- 2015 (Approximate time) Complete shipment of all spent fuel to CLAB and short-lived reactor wastes to SFR
- 2020 (Approximate time) Start transporting spent fuel and long-lived reactor wastes to the SFL respositories for long-lived wastes
- 2048 (Approximate time) Complete transporting spent fuel and long-lived reactor wastes to the SFL repositories for long-lived wastes

Costs (as of 1/87)

- 130 MSEK Capital through 1986
- 240 MSEK Operating through 1986
- 590 MSEK Total life-cycle capital
- 1200 MSEK Total life-cycle operating
- ca 0 MSEK Decommissioning
- 1790 MSEK Total life-cycle costs
- 15-35 MSEK Approximate average operating costs/yr

Funding

Funding is included in the waste management fund.

Experience/Status

Transport of approximately 60 MT spent fuel to France. Transport of 650 MT of spent fuel and 16 MT of MOX-fuel to CLAB as of 1/88.

Q/A Considerations

IAEA transport standards have general Q/A requirements, which are being followed.

REPOSITORIES FOR SPENT FUEL AND LONG-LIVED WASTES

National Policy

The long time required for some of the transuranics in the spent fuel to decay, emphasizes the importance of long term isolation of the fuel in a stable predictable environment. Containment of the fuel in long life canisters and disposal at depth in the Pre-Cambrian crystalline rock was selected at the outset as the best way to satisfy the 1977 legislative requirement to show how and where spent fuel or vitrified high-level waste could be safely disposed. This policy has been maintained while studies have progressed on the design of the repository including variants of the initial design.

The site will be sought so that all fuel can be disposed of in one location.Other long-lived wastes from nuclear industry and research will be disposed at the same location in a separate repository.

Multiple barriers will be used to provide redundancy and diversity of barrier function. The adequacy of the planned design of the repository and of the R&D programme on disposal is being assessed from time to time. In these assessments advice has been sought from the international reseach community including expert panels set up by IAEA and NEA. A high level of safety is considered essential for public acceptance of the disposal.

Requirements

Disposal is required for approximately 7,800 MTU of spent fuel, approximately 19,000 cu. m (volume as emplaced) of long-lived wastes from reactor internals (including fuel channels from BWR fuel assemblies and poison rods from PWR fuel assemblies), and approximately 6,000 cu. m (volume as emplaced) of long-lived wastes from other activities in Sweden.

Site Selection

General reconnaissance surveys for potential site started 1976. A total of about 900 sites have been reviewed using data from aerial and satellite photographs, geological and geophysical mapping. Since 1977, geologic investigations have been carried out on a total of 14 study sites, with limited investigations (i.e., mostly surface work, sometimes with a borehole) at 6, and more extensive investigations at 8 sites (includes geophysical tests, several small boreholes and corings down to and below disposal depth, water injection tests, modelling).

Studies to date have focused on fractures in bedrock, hydraulic properties of bedrock, chemical composition of groundwater, and chemical properties of rock types and fracture minerals. These studies will be continued through the 1980's. A procedure for successive narrowing down of the inventory of potential sites will be implemented through the early 1990's. A few sites will then be thoroughly investigated as candidates for a site license application. It is expected that one site will suffice for the disposal of all Swedish spent fuel and long-lived ILW.

Underground Research

Current plans for a new underground research laboratory are:

1986-88	Preliminary investigations at Simpevarp site including some borehole drilling and measurement.
1988	Final decision on site.
1988-89	Final investigations from surface and facility layout work.
1989-92	Excavation and related research
1993-	Experimental work at "repository depth" can start. Expected to continue for 15 years.

The underground research laboratory will be used for detailed investigation of the natural barrier in bedrock of a final repository character, for in-situ tests on performance/interaction between the engineered and natural barriers, for validation of models, for development of excavation, construction, and Q/A methods, and for demonstration of the system and its technology.

An underground research facility was established in an old Swedish iron mine at Stripa in granite in 1977. This facility was operated with U.S.A. cooperation until 1980, and with expanded, multi-national cooperation (NEA auspices) starting in 1980. The initial studies were focused on measuring thermomechanical, geophysical and geochemical properties of the granite. Phase II, running between 1983 and 1987, performed geohydrological investigations of the granite, and migration tests in simple and complex fracture systems; chemical investigations of the groundwater; techniques for detecting and characterizing fracture systems; and studies of bentonite clay for use as a backfill and seal material in fractured bedrock. Phase III, the final phase, will run from 1986 until 1991. Phase III will focus on applying past experience to an undisturbed granite rock volume, and coupling field measurement technologqy to mathematical modelling to compare values.

Sweden (SKB) has been cooperating with the Canadian (AECL) underground research laboratory in granite by exchange of information.

Repository Concepts

A reference repository concept was developed in 1977-1978. Refinements and variants of this are being studied. A final concept will be selected for approval around the year 2000. The description relates to the KBS-3 concept used to prove the feasibility of safe disposal as required for the fuelling of the last two reactors in 1984.

Description In the reference concept, spent fuel is received at the surface facility at the repository site where it is packaged with several fuel assemblies into a high-integrity canister. The spent fuel in high-integrity canisters is lowered through a large shaft to a repository in granite or gneiss about 500 meter below the surface. From there, the canistered spent fuel is transported through tunnels to the emplacement area where one canister is emplaced in each vertical hole (about 1.5 m diameter and 7.5 m deep) in the floor of a horizontal emplacement tunnel. The individual holes (about 6 m apart) are then backfilled with a buffer material. After all holes are filled in an emplacement tunnel, the emplacement tunnel is backfilled and sealed. There are a large number of parallel emplacement tunnels with a total length of about 38 km.

The other long-lived wastes (reactor core materials) will be received at the surface receiving facility for spent fuel disposal. There they will be encapsulated into long-life containers and transferred to a separate facility at some distance. The containers will be lowered through shafts to the repository level about 300 m below grade. At the repository level, the waste containers will be deposited into troughs in large rooms or stacked in tunnels, depending on the waste type. Backfill will be with concrete or a buffer material.

Waste Receipt and Handling Spent fuel is received from the CLAB in large rail transportation casks if the repository is not located on the coast. The casks are placed in a large pool and unloaded vertically while under water. The spent fuel assemblies are placed in an adjacent pool for lag storage. For BWR spent fuel, the fuel channels are removed under water and moved to the separate facility for encapsulating these wastes. For

PWR spent fuel, any poison rods are also removed under water and moved to the separate facility. The spent fuel is then moved to the dry hot cells where encapsulation is done. Two lines of waste receipt and handling are provided.

Waste Package The basic package concept involves encapsulation of whole spent fuel assemblies in a metal matrix that provides shielding as well as a high level of corrosion resistance (life expectancy is a few million years). Each package contains about 1.4 MTU.

The canister is copper, 10-cm thick, 0.8 m in diameter and 4.5 m long. Total weight of the filled canister is 22 MT (2.0 MT fuel assemblies, 10.5 MT lead, and 9.5 MT copper). The canister material is expected to last 1,000,000 years; it, with the internal fill, provides high-crush resistance; the thick canister shields ground water from a high radiation dose that might cause electrolytic decomposition of ground water. Copper is the present choice because it is the noblest of the common metals and is highly corrosion resistant. However, other materials, e.g., carbon steel, ceramics, are continuing to be studied.

The copper canister with spent fuel (8 BWR assemblies or 2 BWR plus 2 PWR assemblies) held by spacers is filled with molten lead. An alternate concept involves filling the canister (containing 9 BWR assemblies) with powdered copper followed by hot isostatic pressing into a solid body of copper. The alternate is considered to be easier to implement and circumvents the toxicity problem accompanying the use of lead.

No overpack is planned except for the buffer material around each canister in its emplacement hole.

Fuel channels from BWR assemblies and poison rods from PWR assemblies are removed before encapsulation of the spent fuel. These pieces that are received separately are placed in a rectangular concrete box (1.25 m square by 5.3 m long), that is backfilled with concrete. These boxes are to be disposed of in a separate geologic repository at the same site.

Emplacement Drilling of each emplacement hole is preceded by drilling a small core hole from which a judgement is made about the suitability of the location. After suitability is confirmed, the larger emplacement hole is drilled.

Before emplacing a canister in its hole, the compressed rings of the buffer material are added to the hole and a temporary funnel-like steel guide inserted, using a special truck. Each canister is picked up by a deposition vehicle at the bottom of the elevator at the repository level. The deposition vehicle with a canister is moved in the horizontal position over the emplacement hole. The vehicle tilts the canister to vertical, then lowers the canister into the hole. After the canister is in place, the vehicle leaves and additional buffer material is placed in the hole using the buffer emplacement truck.

Buffer and Backfill The buffer material is pressed bentonite "donuts" with a radial thickness of 0.35 m between the canister and rock and dish-shaped cover blocks above and below the canister. Bentonite swells or exerts a swelling pressure when in contact with water and is highly impermeable to water; it has very good ion exchange properties; and it provides plasticity in the event of rock movements.

Backfill of the tunnels is to be with 90 % sand/10 % bentonite mixture, which has been shown to be practical to use. It provides good sorbent characteristics and good structural support of the excavation. The top of each emplacement hole is also to be filled with this material; gaps in emplacement holes are not backfilled because bentonite will provide good sealing. Main tunnels, shafts, and fracture zones are to be plugged with pressed bentonite blocks to minimize water infiltration.

The lower part of each horizontal access tunnel is filled in layers by dumping and compacting. The upper half of each tunnel is filled pneumatically. This combination of techniques provides good structural support of the excavations and is straightforward to implement.

Temperatures and Pressures The external pressure on the canister will depend on hydrostatic pressure (depth) and on the swelling pressure of the bentonite. The internal pressure from helium gradually builds up to about 15 MPa.

Rock or barrier material temperature limit is set at 80. The limit was picked to provide chemical stability and durability of the bentonite, which is expected to be stable for one million years at 100. Also, lower temperatures minimize thermal and structural effects on the rock and bentonite and reduce thermal convection of ground water to insignificant levels.

Radiation Protection Regulatory guidelines and radiation protection criteria are being developed by SKI och SSI. The pressently applied criterion is that the contribution to the radiation dose to the most highly exposed public group shall constitute only an insignificant portion of the dose from natural background and shall lie within the natural range of variation.

SSI has in its assessment of the KBS-3 repository design made a distinction between radiation dose criteria for the first thousands of years after repository sealing or at most up to the next glaciation, and for more distant times. Conventional dose calculations are appropriate for the first time period and the requirement is that the calculated doses shall be below 0,1 mSv/year to the most exposed group. For more distant times SSI is developing new concepts and policies for judgement of the acceptability of a repository system. SSI has thus proposed that criteria are developed which are based on comparisons of calculated releases of waste radionuclides to the biosphere with natural releases of radionuclides from weathering of the bedrock.

Occupational exposures have to be kept within IAEA and ICRP limits (50 mSv/y). The maximally - exposed individuals are not expected to receive more than 5 mSv/y (most of this will be from transport cask handling).

<u>Decommissioning</u> Final sealing of shafts is to be done with the same 90 % sand/10 % bentonite as used to backfill the emplacement tunnels. The shafts also will have several large plugs of compacted bentonite blocks. At the top of each shaft, which is covered by soil to grade level, will be a large concrete plug below which will be compacted moraine to a depth of about 100 m.

Boreholes will be plugged with perforated metal tubes filled with pellets of compacted bentonite.

<u>Post-Closure</u> Monitoring is not expected to be needed for long-term safety.

In crystalline rock, the waste will in fact be retrievable for a very long time - it is a matter of cost and keeping records. Exception is very deep borehole disposal at a depth of several thousand meters.

<u>Schedule</u>

- 1976	Start reconnaissance surveys for repository sites
- 1977	Start repository R&D and site investigations
- 1977	Initiated research in Stripa mine
- 1988	Site and underground research laboratory
- 1993	Start up underground research laboratory
- 1993	Start detailed characterization of final 2 or 3 sites for spent fuel repository
- 2000	Submit license application for repository
- 2000	Decision to be made on final repository concept and design initiated
- 2010	Start construction of spent fuel repository
- 2020	Commission spent fuel repository and start emplacement
- 2050	Complete operational phase of spent fuel repository
- 2060	Complete decommissioning of spent fuel repository

Costs (as of 1/87)

- 7470 MSEK Total life-cycle capital for spent fuel repository
- 6080 MSEK Total life-cycle operating for spent fuel repository
- 2800 MSEK Total decommissioning for spent fuel repository
- 16350 MSEK Total life-cycle costs for spent fuel repository
- 200 MSEK Approximate average operating costs per year for spent fuel repository
- 900 MSEK Total life-cycle capital for long-lived ILW repository
- 220 MSEK Total life-cycle operating for long-lived ILW repository
- 170 MSEK Total decommissioning for long-lived ILW repository
- 1290 MSEK Total life-cycle costs for long-lived ILW repository
- 8 MSEK Approximate average operating costs per year for long-lived ILW repository
- 17640 MSEK Total life-cycle costs for spent fuel and long-lived ILW repository

Funding

Funding is included in the waste management fund.

Experience/Status

Sweden has much experience in underground storage of other materials such as oil. Research has been performed in Stripa mine since 1977. Detailed analyses and conceptual design for a repository have been completed. Repository siting studies are done since 1977. Research is done on hydrology, rock characteristics and performance, waste package and engineered barriers design and performance, repository performance assessment and modelling, and natural analogues.

SKB has bilateral agreements with AECL (Canada), IVO and TVO (Finland), CEA (France), Nagra (Switzerland), US DoE and Euratom (CEC) and multilateral agreements on the NEA Stripa Project, on glass leaching and on natural analogues.

SKI has initiated international intercomparison and validation studies of safety assessment codes (Hydrocoin, Intraval) and participates with ANSTO (Australia), JAERI and PNC (Japan), UK DoE and US NRC on the Alligator Rivers Project.

SSI has initiated international cooperation on development of biosphere transport models, Biomovs.

SKI and SSI cooperate with authorities in the Nordic Countries, Switzerland and UK on development of criteria for HLW disposal.

DISPOSAL OF LLW AND ILW

National Policy

LLW and ILW from operation of the Swedish reactors will be disposed of in a central repository (called SFR). Some LLW will be disposed by shallow land burial at the reactor sites.

The waste in the central repository will also include similar types of radioactive waste from other industries, research and medical activities.

Requirements

Disposal volume is required for 90,000 m^3 of waste packages. This is the calculated amount of operational LLW and ILW in Sweden until the year 2010.

The total activity content is calculated to be 10^{16} Bq (mainly Co -60 and Cs -137).

The environmental impact shall be very low. The design goal is that calculated dose to most exposed individual shall be below 10^{-4} Sv/y.

Regulations

In Sweden no specific regulations have been given from the authorities for the design of a repository for radioactive waste. A preliminary safety report was prepared by SKB based on a preliminary design of the repository.

The application for a license to construct and operate the facility was submitted to the Government. The preliminary safety report was reviewed by the authorities and they recommended that a license should be granted with specified conditions.

It was stated that the construction and operation of the facility mainly shall be in correspondence with specifications in the application. The Swedish Nuclear Power Inspectorate was authorised to give the necessary additional instructions needed for safety reasons. SSI prescribes the radiation protection measures.

The license is subject to certain stipulations. The most important are:

- SKB shall furnish information to the public on the progress of the project and measures against the release of radioactivity from the repository.

- A comprehensive quality control program backed up by a test and verification program shall be carried out during the construction phase.

- Further studies of gas production reactions shall be conducted, as well as a study of the gas transport capacity of the rock.
- The Swedish Nuclear Power Inspectorate shall scrutinize the design and construction work and, if necessary, issue further requirements.
- Before commissioning, SKB has to apply for a license to operate the facility. The application shall be based on a final safety report.
- Final sealing will require a special license based on a reevaluation of the safety assessment.

Site Selection

Due to geological and hydrological conditions in Sweden it was early decided that the repository should be located under ground in rock caverns. It was also a primary requirement that it should be located adjacent to one of the five nuclear facilities: Barsebäck, Forsmark, Oskarshamn, Ringhals or Studsvik.

The sites were evaluated on the basis of available data concerning the geological situation and other information of importance for site selection. This work indicated that the host rock was best at the sites on the east coast. In the next phase geological surveys were carried out at Oskarshamn, Forsmark and Studsvik. Different locations of the repository on the sites were also considered during this phase. The surveys included geophysical tests and geological mapping of the environment.

The geological and hydrogeological situation at Forsmark and Oskarshamn was found to be suitable for siting of the planned type of repository. When all aspects were considered, Forsmark came out as the best site for SFR.

Repository Concepts

Description

The repository is situated in the bedrock under the Baltic Sea, with a rock cover of about 60 m. The host rock is gneiss-granite with dykes of pegmatite and amphibolitic compositions that occur quite frequently. From an engineering geology point of view, the rock mass has been as good as expected on the basis of the results of the geological surveys.

The location of the SFR under the bottom of the sea ensures that the hydraulic gradient and thereby the groundwater flow is very low in the repository area. It also ensures that no well will be drilled in the vicinity of the repository for at least 1000 years while the area is covered by the sea. (The rate of land uplift is at present 6 mm/year.) The seawater also provides a recipient with a high dilution capacity.

The SFR has various storage chambers with different barriers, depending on the waste to be disposed of. The function of the engineered barriers is to limit the release of radioactivity to the groundwater. 40% of the waste volume contains most of the activity (90%) and will be deposited in large concrete silos situated in 70 m high cylindrical rock caverns.

The concrete silos will be surrounded by a clay barrier with low permeability. This ensures very low release rates from this part of the repository to the groundwater, since release of nuclides has to take place by diffusion through the concrete and the clay barriers.

160 m long rock caverns will be used for the less active waste. The design of these storage chambers is dependent on the type and dose rate of the waste packages. The release of activity from these caverns will mainly be governed by the exchange rate of groundwater inside the caverns.

Waste Receipt and Handling

Most of the operating waste will be transported by sea on M/S Sigyn. Low-level waste that does not have to be radiation-shielded will also be transported by road, in ordinary freight containers.

The gross weight of the shielded transport container will be limited to 120 tonnes. The limiting factor is the maximum payload of the ship. These shielded containers have to be designed to accept the additional loading and acceleration associated with transport on a ship in rough seas. Different materials have been investigated for construction of the containers. Steel was found to be the best material.

The containers are handled with a specially designed terminal vehicle with one diesel engine and one electric motor. The electric drive will be used in the tunnels and caverns. The total weight of the vehicle with load will be about 155 tonnes.

The ILW-packages will be unloaded by remote controlled handling machines. LLW will be handled with an ordinary forklift truck.

Waste Packages

The reactor waste that will be stored in the SFR consists mainly of ion-exchange resins and filter material from different water treatment systems. Other waste categories are contaminated components and material, trash, and ash from the incineration of combustible waste. Before transport to the SFR, the waste is conditioned and stored at the reactor plants.

The main part of the waste is enclosed in the following types of packages with maximum surface dose rate as follows:

		LxWxH(m)	Dose rate mSv/h	Max.weight (tonnes)
-	Concrete containers with resins solidified in cement	1.2x1.2x1.2	< 30	4
-	Drums with bitumenized resins	Ø0.6x0.9	<500	0.5
-	Drums with resins solidified in cement	Ø0.6x0.9	< 30	0.5
-	Drums with trash and metal scrap	Ø0.6x0.9	0.3	0.5
-	Steel containers with resins solidified in cement	1.2x1.2x1.2	<500 x)	4
-	Concrete tanks with dewatered resins	3.3x1.3x2.3	< 10	20
-	20' or 10' containers		< 0.15	20

x) Pending approval by the authorities

The wet waste contains most of the activity and is normally solidified in cement or bitumen at the reactor stations. Cement solidification is done in concrete cubical containers known as moulds 1.2 m on a side. Bitumen solidification is done in standard 200 l drums. Low active resins from the condensate clean-up system are merely dewatered and stored in large transportable concrete tanks.

Buffer, Backfill and Seals

The use of buffer, backfill and seals differs for the various parts in SFR. The most extensive system is used in the silo repository.

The concrete silo is surrounded by a clay barrier with low permeability. This ensures very slow release rates from this part of the repository to the groundwater, since release of nuclides has to take place by diffusion through the concrete and the clay barriers.

In the concrete silo the waste packages are subsequently backfilled with concrete.

When the silo repository is filled with waste a concrete lid will be cast on top. The buffer will be completed with a layer of sand and bentonite clay over the lid. The space above will be backfilled with sand.

Concrete is used for backfilling of ILW in the other rock caverns.

Concrete plugs will be used to seal the tunnel entrances to the various caverns. The access tunnels will be sealed with concrete plugs in combination with bentonite.

Radiation Protection and Safety Considerations

Measures have been taken to reduce doses to the operational staff. The handling of ILW is carried out by remote controlled machines. The terminal vehicle can also be operated by remote control.

The conservatively calculated dose for routine operation is 25 mMan Sv/y.

Decommissioning

Buildings on ground level are planned to be demolished and deposited on site.

Post-Closure

There are at present no decisions on post closure surveillance.

Schedule

The construction work started in the summer 1983 when the license was granted. Testing and commissioning will be finished in March 1988.

The first waste can then be placed as soon as the final safety report is reviewed and approved, and a license to operate the facility has been issued by SKI.

Costs and Funding

The total cost for the first construction phase is calculated to be 740 MSEK. The costs for site selection and preliminary safety report are not included (13 MSEK).

The costs for the first phase can be divided as follows:

Geological surveys, other research and FSR	20	MSEK
Administration	30	"
Engineering and design	115	"
Civil engineering works (and provisional facilities on site)	420	"
Mechanical and transport systems	60	"
Auxiliary systems	50	"
Electrical and control systems	45	"
Total (current prices)	740	MSEK

The costs for the second construction phase is calculated to be around 200 MSEK (price level 1987).

The calculated costs for 25 years of operation is 360 MSEK and additional 100 MSEK for closure and sealing (price level 1987).

For the expansion of the repository for the decommissioning waste additional 440 MSEK is required for construction and operation.

85% of the costs are covered directly by the reactor owners. Remaining part comes from the waste management fund.

Experience/Status

The experience from construction of the repository is very good. The excavation of tunnels and caverns went very well. The results from the geological surveys complied very well with the real conditions during excavation.

The construction work was finished by the end of 1987. Commissioning will take place in the beginning of 1988. The first waste is planned to be emplaced in April 1988.

Quality Assurance Considerations

A traditional quality assurance program has been conducted for the construction work, with emphasis on:

- Excavation and reinforcement of the rock.

- The bentonite material and performance of the buffer around the silo.

- The construction work for the concrete silo (slipform technique).

- Manufacturing of handling machines and terminal vehicle.

For the quality assurance of waste packages there is a separate program. The waste packages are divided into groups. For each type of waste package the following steps are performed:

- detailed description of the waste package with regard to raw waste composition. container, solidification materials, solidification process and process variations.

- analysis of the planned handling sequence for the waste package type and identification of the functional requirements on the waste package in each step of the handling sequence, e.g. surface dose rates, stacking capability and long-term integrity. Determine the limiting functional requirements.

- conversion of the limiting functional requirements to desired waste package properties, e.g. activity content, compressive strength and water resistance.

- compilation of tests and calculations performed on the waste type to verify that the packages have the desired properties. Identify what complementary investigations are needed and perform these.

- based on the results of the tests and calculations acceptable waste and process variations are established.

- definition of quality control measures; e.g. process control and product sampling.

When this assessment has been done it is reported in a "waste type description" which will have a status similar to a safety report. The "waste type description" will also be used as a reference in the final safety report for the disposal facility, SFR, and must be approved by SKI and SSI before that waste type can be disposed of in SFR.

DRAFT

Nuclear Waste Management in Switzerland

(Status Overview January 1988)

A. General Strategy

A.1 Overall Waste Management Strategies

(NTB 83-02, NTB 83-03, NGB 85-09) Spent fuel elements are reprocessed abroad and the resulting vitrified HLW and conditioned LLW/ILW (including TRU) are returned to Switzerland. Centralised interim storage is foreseen for HLW, spent fuel elements and some LLW/ILW from reprocessing. In Project Gewähr 1985, the total interim storage time (= removal of fuel elements to final disposal) for HLW is taken as 40 years.

All radioactive wastes are to undergo final disposal in repositories situated in suitable geological formations. The principal goal is ensuring long-term safety after closure of the repository - no special provisions for later retrieval of the waste are foreseen since these could prejudice the long-term safety goals. The repositories must fulfil the protection objectives of the Guideline R-21 (cf. chapter A.2).

Two repository types are foreseen, one for HLW (type C repository) and one for LLW/ILW (type B). Waste sorts are defined with regard to maximum allowable radionuclide concentrations for the individual repository types as derived from the R-21 protection objective requirements. TRU waste from reprocessing will be disposed of in either the B or C type repository, according to the maximum allowable radionuclide concentrations for the type B repository as derived from safety analyses based upon the actual site data. In the repositories, the waste is isolated from the human environment by a series of safety barriers. Various engineered (technical) and natural (geological) barriers are employed, depending upon the waste sort and its toxicity.

The responsibility for waste management is divided between the utilities operating NPP's and a special waste disposal company called Nagra (cf. chapter A.3).

A.2 National Policy - Implementing Law

Reprocessing of spent fuel is foreseen, but the option of disposing of non-reprocessed fuel elements is kept open. Equally, the option of disposing of high level waste abroad within a framework of international cooperation is kept open, since this would be preferable from an economic point of view. However, because political factors make full preparation for disposal in Switzerland necessary, it is planned to continue the high-level waste research programme at least up to the stage where selection of a repository site is possible. For the low- and intermediate-level waste, a final repository shall be constructed in Switzerland in any case.

The Federal Government Ruling of 6th October 1978 on the Atomic Act designates the guaranteeing of "permanent safe management and final disposal" of radioactive waste as a prerequisite to future development of use of nuclear energy in Switzerland. Because the 5 nuclear power plants already existing (in total about 3'000 MW(el), 40 % of Swiss electricity production) are outwith the scope of the Federal Government Ruling, the Federal Department of Transport, Communication and Energy (EVED) demanded a project which offers a guarantee of feasibility and safety of final disposal as a prerequisite to the extension of operational licences beyond the year 1985. This project - the so-called "Project Gewähr" - was submitted to the Federal Government by Nagra on 23rd January 1985. At present, the project is still formally under review, the operational licences having been extended provisionally. The reviews by the Swiss Nuclear Safety Authorities of Project Gewähr which have been published are generally positive in that the safety analyses for the HLW repository (performed for a model-site with data based on real data from test sites) are accepted as a demonstration that safe disposal is generally achievable. However, debate has arisen over the availability within Switzerland of a suitable real repository site in the crystalline bedrock. For LLW there is full acceptance of the Project by the government experts. The Federal Government has indicated that a final decision will be published in early summer 1988.

The safety conditions which the final repositories must satisfy are defined in the Guideline R-21 (October 1980) of the Federal Commission for Safety in Nuclear Installations (KSA) and the Nuclear Safety Department of the Federal Office of Energy (HSK). The Guideline states two objectives: 1. Radionuclides which escape into the biosphere must not at any time lead to individual doses exceeding 10 mrem per year; 2. A repository must be designed in such a way that it can at any time be sealed within a few years. After it has been sealed, it must be possible to dispense with safety and surveillance measures. A further Guideline, R-14, regulates the conditions for interim storage of radioactive wastes.

The siting and construction of a repository and all preparatory work (i.e. mainly geo- and hydrogeological investigations) is regulated by the Federal Law. However, the local authorities of the community and state (in Switzerland Canton) involved, as well as the population concerned, must accept the repository. Without the consensus of the population, long delays will result, even if a formal federal licence has been granted.

A.3 Organisational Structure

According to Swiss law, the producers of nuclear waste are responsible for waste management (for all waste categories). Hence, the electricity supply utilities involved in nuclear power and the Swiss Confederation (being responsible for the waste from medicine, industry and research) joined together in 1972 to form the "National Cooperative for the Storage of Radioactive Waste" (Nagra). Nagra is responsible for the final disposal and possible final conditioning of wastes, as well as for the preceding controls; the responsibility for spent fuel reprocessing and transport, for the waste conditioning and for the interim storage remains with the utilities.

The Federal Government is supported in its decisions on waste management topics by the "Federal Interagency Working Group on Nuclear Waste Management" (AGNEB), by the "Federal Commission for Safety in Nuclear Installations" (KSA) and by the "Nuclear Safety Department" (HSK) of the Federal Office of Energy.

Several other federal offices and scientific institutions are involved in the regulatory, control and supervisory work.

A.4 Policy on Spent Fuel Management

cf. chapter A.2

A.5 Overall Schedule

No time limits are set in the Federal Government Ruling for carrying out the preparatory work or for the emplacement of waste in the repository. The programming of the work has to take into account the technical and scientific necessities (e.g. site characterization), the operational input (e.g. dates of the waste return to Switzerland) and the political issues (licensing procedures, both federal and local).

With all caveats regarding the licensing procedures and the outstanding decision of the Federal Government on Project Gewähr, Nagra has developed the following time schedule:

	HLW repository	LLW/ILW repository
- 1978	generic studies	generic studies
1979 - 84		pre-evaluation of about 100 potential sites, selection of 3 priority sites
1980 -	beginning of regional investigations of the crystalline bedrock in the northern Switzerland	
1985	***** Submission of Project Gewähr to the Authorities *****	
1985 -	continuation of regional bedrock investigations, desk studies on sediment sites	beginning of investigations of the 3 selected sites, selection of a fourth site for investigations
1990/91	selection of one site (crystalline or sediment) for further investigations by drillings etc.	final site selection, general application for a LLW/ILW repository, thereafter construction
1998	application for an underground rock laboratory on the repository site	beginning of waste emplacement in the repository

	HLW repository	LLW/ILW repository
2010	results of the final site characterization	
2010 - 25	engineering and construction of a Swiss repository - or participation in an international project	

A.6 Total System Costs and Funding

In Project Gewähr (NGB 85-09) the construction costs of the HLW repository have been estimated at 600 million Swiss Francs and the construction costs of the LLW/ILW repository at 320 million Swiss Francs. For the preparatory work (including the geological investigations), the expenses of Nagra up to the end of 1986 amounted to 280 million Swiss Francs, in 1987 to about 30 million. The budget for 1988 amounts to 60 million Swiss Francs.

The costs of waste management are borne directly by the waste producers, i.e. mainly by the electricity supply utilities operating nuclear power plants, according to their power production. They are included in the electricity price. A minor contribution (calculated for a virtual "power equivalent") is made by the Swiss Confederation which is responsible for the management of wastes arising from medicine, industry and research.

The project costs are paid directly by the waste producers; there is no state organization for collecting and redistributing the funds.

A.7 Quality Assurance Considerations

The main objective is to provide assurance that there are no deviations from the desired quality which could significantly impair the system safety. A systematic quality control programme will be applied to the design and construction of the various parts of the system. A special quality assurance programme is under development for waste conditioning, interim storage and transport (cf. NGB 85-02).

A.8 International Cooperation

Nagra is an active participant in international research programmes (e.g. JSS, INTRACOIN, HYDROCOIN, INTRAVAL, BIOMOVS, Poços de Caldas ...). Nagra is also a member of the international Stripa project based at an underground rock laboratory in Sweden.

There are several cooperation agreements with organizations in foreign countries (USA - DOE and NRC, Federal Republic of Germany - PTB, BGR and GSF, European Community, France - CEA and Andra, Finland - TVO and IVO, Great Britain - NRPB, BGS and BNF ...). Intensive contacts are established with NEA/OECD and IAEA. Several foreign countries (Germany, USA, France, Sweden) have also participated in research work at the Swiss underground rock laboratory (Grimsel Test Site, cf. chapter D.5).

B. Interim Storage System

(NTB 83-02, NTB 83-03, NGB 85-09) Spent fuel is stored in storage pools at nuclear power plants, then transported for reprocessing abroad (mainly France, partly GB). The resulting radioactive waste will be returned to Switzerland (after 1992) and stored in a central interim storage facility for a total of 30 - 40 years (HLW).

A project is in preparation which will serve as a basis for the licence application (submission is scheduled for the end of 1988). The project is characterized by the following points:

- Dry storage of fuel elements or HLW in transport containers (CASTOR-Type) in surface halls.
- LLW/ILW will be stored in separate surface halls.
- Storage capacity for interim storage of spent fuel elements and HLW will be sufficient for the present nuclear power plants (3'000 MW). The construction of the store shall proceed in stages. The first stage caters for the capacity requirement of the next 15 to 20 years.

More details can be given only after completion of the above-mentioned project.

C. Transportation System

The transport of the spent fuel elements to the reprocessing plants abroad is performed using standard transport containers on road vehicles. The usual international and national regulations for the transport of radioactive materials are observed. However, due to the small amount of material to be transported and to the relatively short distances, no special "national transportation plan" has been developed to date. The same applies to the transport of wastes.

D. Disposal System - High-Level Waste Repository

D.1 National Policy

Geologic disposal was selected as the best way of meeting the requirements of the Federal Law. In 1978, a concept for final disposal in Switzerland was presented; this leaves several options open with regard to the specific choices of the construction concept and the host rock. According to the general concept, the repository could be, for example, a mined system of tunnels and silos, a fan-like arrangement of deep boreholes from the earth's surface into the host rock, or a combination of both systems with underground tunnels in stable rock and deep boreholes into underlying host rock. Impermeable clays, anhydrite formations, crystalline bedrock and others can be considered as host rocks.

For Project Gewähr 1985, a system of mined tunnels and silos at a depth of around 1200 m in the crystalline bedrock in Northern Switzerland was selected. However, this choice in no way prejudices later planning of a repository project with regard to the host rock, the region of the repository site or the engineering design. The safety analyses in Project Gewähr are based (in the form of a model data-set) on a representative geological situation, the repository being assumed to be located in a stable granite block between two major faults of the crystalline basement overlayed by several hunderd metres of sediments.

D.2 Requirements

In the model repository of Project Gewähr there was provision for disposing of up to 1'120 m^3 of vitrified HLW, corresponding to a spent fuel inventory of 7'900 tU, and up to 53'000 m^3 of TRU. This would be sufficient for 40 years of operation of twice the nuclear power capacity presently installed (so-called 240 GWa scenario).

D.3 Regulations

cf. chapter A.2

D.4 Site Selection

To date, no site selection programme in a strict sense of the word has been performed. The investigations concerned a **region for potential sites**, rather than concrete sites.

After the decision on crystalline bedrock as the host rock of first priority, a regional investigation programme was initiated in 1978 in an area of about 1200 km^2 in Northern Switzerland, with a network of seismic lines and, up to now, 6 deep boreholes, each penetrating through the overlaying sediments and about 1000 m deep into the bedrock. The investigations led to the discovery of a Permocarboniferous trough which is now fairly well outlined in the investigation area. Moreover, the hydrogeological and geochemical characteristics of the bedrock have been investigated thoroughly.

For Project Gewähr, a model-site was chosen using geo- and hydrogeological data mainly from the Böttstein drill-hole. The safety analysis proved that the assumed site characteristics allow the protection objectives of R-21 to be fulfilled, provided there is a sufficiently large area of host rock with the modelled properties (which still has to be established definitively). Further characterization work on sites with crystalline rocks and also on sedimentary alternatives is continuing.

The site (and prior to this the host rock) selection will be made according to the time schedule in chapter A.5.

D.5 Underground Research

Before final site confirmation, there will probably have to be a deep underground rock laboratory at the potential repository site (cf. A.5). In the meantime, especially for testing and development of investigation techniques and tools, an underground rock laboratory has been established at the Grimsel pass in the Swiss Alps. The research programme includes projects within the scope of international cooperation agreements (cf. chapter A.8).

The Grimsel rock laboratory is situated in granite beneath the Juchlistock massif, about one kilometre inside the mountain at an elevation of 1730 metres above sea-level. The rock overburden is around 450 m. The granite there is particularly suitable for rock mechanical, geophysical and hydrogeological investigations as, within a restricted area, dry and impermeable rock areas, damp zones and water-bearing fissures can be found. An extensive research programme has been carried out at the Grimsel Test Site since 1984, including methods for non-destructive rock examination (electromagnetic high frequency borehole radar, underground seismics), rock movement measurements by tiltmeters, various tests regarding rock mechanics (investigation of the decompression zone, rock stress measurements, heat tests) and an extensive hydrogeological experimental programme (fracture system flow tests, migration of radionuclides, ventilation test etc.).

The continuation of the Grimsel Test Site research programme until at least 1990 was agreed upon at the end of 1987.

D.6 Repository Concept

D.6.1 <u>Description</u>

The current reference repository project (Project Gewähr, NGB 85-09) is characterized by the following:

- The repository is foreseen for HLW and TRU wastes. The multiple safety barrier system for HLW consists of a leach-resistant glass matrix, a corrosion-resistant steel canister surrounding the glass cylinder, a layer of highly compacted bentonite surrounding the canister and, finally, the crystalline host rock and its sedimentary overburden. The TRU waste is embedded in a leach- and dissolution-resistant solidification matrix (cement or bitumen), emplaced in a cylindrical concrete silo and surrounded by special concrete. The space between the filled concrete silo and the rock wall of the cavern in the host rock is backfilled with bentonite.

- Disposal of HLW is in horizontal, mined tunnels, while TRU is disposed of in vertical silos. The tunnels and silos are conceived in such a way that they can be positioned to take account of the geometry of disturbed zones in the host rock at the repository depth without compromising the long-term safety. The silo and tunnel areas are spatially separated in order to avoid any undesirable (mainly chemical) interaction.

- Large disturbed zones of the host rock in the repository area are avoided by observing a sufficient safety clearance. Zones of lesser disturbance which intersect the tunnel system are dealt with by storing no waste in their vicinity and sealing the relevant section of the tunnel with backfill.

- Before final closure of the repository, a safety evaluation of a long-term, in-situ experiment will take place (observation of materials used in the repository over several decades). Retrieval of the waste would be technically difficult but not impossible.

- The construction and operational phases of the repository run simultaneously. Appropriate arrangement of the repository installations ensures that continuing mechanical excavation with tunnelling machines is spatially separated from the emplacement operation.

D.6.2 Waste Receipt and Handling

- HLW and TRU waste is received from the interim storage facility in appropriate transportation casks. The waste is prepared for emplacement in the surface reception area, the HLW in particular being encapsulated in corrosion-resistant repository canisters. All waste - the encapsulated HLW and the TRU waste as delivered - is conveyed to the repository in additional transport shielding and work in zones with higher local radiation doses is done by remote control.

- The delivered waste containers and the encapsulated waste for disposal undergo quality control. Facilities for back-up work, replacement of defective HLW canisters, decontamination of container surfaces etc. are foreseen, as well as measuring equipment for monitoring the radioactive inventories of the waste containers. Organisational measures ensure that an inventory is kept of all delivered and stored waste.

D.6.3 Waste Package

- The repository canisters (overpack) for HLW are designed to withstand the chemical, radiological and mechanical conditions in the repository for a minimum lifetime of 1000 years. The chemical milieu is conseravtively established, taking into consideration the groundwater chemistry and the possible radiolysis of water. The heat from radioactive decay will result in a maximum temperature of about 150 °C at the outer wall of the canister. The swelling pressure of the backfill material and the hydrostatic pressure will not exceed a value of 30 MPa. There is no overpack for TRU.

- Cast steel (GS 40) was selected as a self-supporting canister shell for HLW. The canister design selected can be fabricated using technology available today (traditional moulded steel method). The cylinder body and base of the canister have a maximum total length of ca. 2000 mm and a maximum outer diameter of 940 mm. The total weight of the filled and sealed canister is 8.5 t. The body and its hemispherical lid are pressed and welded together.

- Canister quality is guaranteed primarily by the simple production method. In addition to the normal checks done during the production process, the completed components undergo metallurgic inspection for faults. The welded joints (possibly stress-relieved) can be tested using ultrasonics and gas leakage methods.

D.6.4 Emplacement

- The repository is accessible by two independent vertical shafts. One of these handles transportation for the construction operations and, at the same time, serves as a fresh air inlet for the underground mining. The other shaft is for the conveying of radioactive waste and the backfill material and also serves as an air outlet.

- From an underground central area at the bottom of the shafts, two main tunnels lead to the repository area for HLW. Between the main tunnels, a system of parallel repository tunnels is mechanically excavated. These tunnels have a circular profile of around 3.7 m diameter and lining of the tunnel walls is not envisaged. The HLW canisters are placed in the tunnels axially at regular intervals of 5 m and the remaining space is sealed with bentonite backfill.

- The TRU silos are in a separate area. The silo caverns have a depth of around 55 m and a diameter of 10 m. Each silo consists of a concrete structure standing free of the rock in which the waste is emplaced and immobilised with a special concrete. The space between the rock and the silo wall is backfilled with bentonite.

D.6.5 Buffer, Backfill and Seals

- During emplacement of HLW, the waste canisters are surrounded by compacted bentonite blocks. A specially developed handling machine is used to emplace the prefabricated bentonite blocks. The convective water flow through the bentonite layer is negligible and solute transport to and from the waste canister occurs primarily by diffusion. Bentonite also acts as a chemical buffer and strongly retards diffusion of many important radionuclides by sorption - the retention of many significant radionuclides at the repository location thus being up to a hundred thousand years. Many radionuclides will decay to insignificance before they can penetrate through the bentonite into the surrounding geological formations.

- Before final closure of the repository, shafts and cavities underground are isolated and infilled (i.e. sealed). Selected key zones are sealed using bentonite blocks or a bentonite / quartz sand mixture, the remaining space being infilled by other materials.

D.6.6 Temperatures and Pressures

- The maximum temperature in the axis of the vitrified waste is estimated to be 195 °C 1.1 years and about 115 °C 50 years after emplacement. The respective values for the canister are 153 °C and 100 °C, for bentonite 5 cm from the canister 146 °C and 95 °C, and for the rock wall 76 °C and 73 °C.

- The maximum total presure at the canister wall is estimated to be 30 MPa.

D.6.7 Radiation Protection and Safety Considerations

- The design goal stipulated by the Swiss regulations is that the expected contribution to the radiation dose for the most highly exposed group of public must be less than 10 mrem per year (cf. Guideline R-21, A.2). The safety analyses of Project Gewähr 1985 show that this goal can be achieved. Actually, the doses calculated are generally much - by orders of magnitude - lower than required by the Guideline R-21.

D.6.8 Decommissioning

- Final sealing of shafts is to be done as mentioned under D.6.5 by sealing critical key zones with bentonite and infilling the remaining spaces with a bentonite / sand mixture or other materials (concrete). Also, exploratory boreholes which may potentially compromise the long-term safety of the repository shall be sealed in a similar way.

D.6.9 Post-Closure

- Swiss law requires final disposal of the waste, the final repository being defined as a facility for which "it must be possible to dispense with safety and surveillance measures" once the repository has been sealed. Hence, no special monitoring is expected to be needed for long-term safety. The records about the location of the repository area and other relevant information will be kept by the Federal Government.

- In crystalline rock, the waste will in fact be retrievable from its bentonite surrounding for a very long time - albeit at a rather high cost.

D.7 Schedule

The repository will not be needed before the year 2020 ... 2025. Hence, the programme of further action foresees thorough evaluation of host rock options, with host rock and later site selection and characterization up to the year 2010. Thereafter it will be decided whether a final repository in Switzerland shall be constructed or if an international project can be joined. For a more systematic overview cf. chapter A.5.

D.8 Costs and Funding

cf. chapter A.6

D.9 Experience / Status

Switzerland has much experience in tunnel construction but less in underground mining and deep geological explorations. Nagra has supported the build-up of the necessary experience during the last eight years or so, and today state-of-the-art exploratory, processing and modelling techniques are available. At the Grimsel Test Site, valuable experience in underground testing has been acquired.

E. Disposal System - Low- and Intermediate-Level Waste Repository

E.1 National Policy

Because of the high population density in Switzerland, no shallow land burial has been foreseen for LLW. Geologic disposal in the so-called type B repository was selected as the best way of meeting the requirements of the Federal Law and assuring the necessary public acceptance even for LLW. Certain low- and intermediate-level wastes - e.g. operational waste from NPP's and waste from medicine, industry and research - already exist in a form suitable for final disposal. The type B repository is therefore required more urgently than the HLW repository and it is intended to be operational before the end of this century (1998, cf. chapter A.5). The type B repository will be constructed in Switzerland even if the HLW (and TRU) should be disposed of abroad within the framework of an international cooperative project.

In the type B repository, the waste is isolated from the biosphere by both natural and engineered safety barriers. Key parameters for the realisation of the general multiple safety barrier concept are the choice of the host rock (e.g. clay, marl, anhydrite, crystalline or others), the engineering design of the repository and the specification of the waste categories to be accepted for emplacement in the repository. A mined cavern system with access through a **horizontal** tunnel in an alpine formation of Valanginian marl was selected as the reference repository concept for Project Gewähr 1985. The safety analyses were performed for a model-site based on the present knowledge of one of the potential repository sites (Oberbauenstock, cf. chapter E.4). All LLW and ILW were assumed to be disposed of in the type B repository, including some TRU waste. However, the specific waste acceptance criteria will be defined only later after the repository site has been finally specified, taking into account the Guideline R-21 protection requirements. Hence, depending on the real geosphere situation at the chosen site, more or less of the TRU will have to be emplaced in the HLW repository (or in a special intermediate type repository).

E.2 Requirements

In the reference repository of Project Gewähr 1985 there was provision for disposing of 200'000 m³ waste, corresponding to 40 years of operation of twice the present NPP power output (240 GWa scenario, cf. chapter D.2). The repository to be realised will be somewhat smaller.

E.3 Regulations

cf. chapter A.2

E.4 Site Selection

The site selection for a LLW/ILW repository proceeded in several stages and is described in the Nagra reports NTB 81-04 und NTB 83-15. First of all, Nagra selected possible host rocks according to hydrogeological and geological criteria and evaluated a total of 100 potential sites in the years 1978 - 81. The results led to a selection of 20 potential sites for which additional non-licenced investigations were undertaken.

Evaluation of these 20 sites was performed in 1982 - 83 and led to the selection of three sites - Bois de la Glaivaz (anhydrite), Oberbauenstock (Valanginian marl) and Piz Pian Grand (crystalline) - which are being investigated as a first priority. Five further exploratory areas were designated as second priority; the remaining 12 sites have been deferred from further investigations. For the three top priority sites, Nagra prepared relevant applications for exploratory drillings and tunnels and submitted them at the end of 1983 to the Federal Government.

The necessary licences were granted in September 1985; however, the decision on Phase II of the work - the construction of exploratory tunnels - has been postponed until the results of Phase I have been presented. In the years 1986 - 87, Nagra performed investigations at Oberbauenstock and at Piz Pian Grand and finished Phase I at these two sites. No licensed work has been done at Bois de la Glaivaz due to severe political obstruction at this site. The report on results of Phase I and the application for exploratory tunnels in at least one of the three sites are expected to be ready in spring 1988.

In addition to the three sites mentioned, a fourth has been selected at Wellenberg in Canton Nidwalden, where the geometry of the Valanginian marl could allow the construction of a horizontally accessible LLW/ILW repository combined with a repository cavern for TRU at a depth of 300 m (or so) at the same site. The necessary applications were submitted to the Federal Government in June 1987, the decision being expected in the second half of 1988.

E.5 Underground Research

At the most promising potential repository site, an exploratory tunnel will be constructed (cf. chapter E.4). At the Wellenberg site, the licencsng application includes proposals for a small underground rock laboratory which should allow Nagra to perform marl-specific underground experiments. In addition, at the Grimsel Test Site (D.5) more general methodology development and specific crystalline experiments are taking place.

E.6 Repository Concept

E.6.1 Description

The reference repository project (Project Gewähr 1985, NGB-85-09) is characterized by the following:

- Disposal is in underground rock caverns with access through horizontal tunnels and the reception area is also underground.

- The system of technical safety barriers comprises the waste solidification matrix (cement, bitumen, polymers); possible grouting of the waste drums with liquid cement in a concrete container; backfilling of remaining empty space with special concrete; concrete lining of the disposal caverns and sealing of access tunnels on closure of the repository. The waste is delivered in conditioned form, i.e. in the solidification matrix. All remaining technical barriers are provided during construction, operation and closure of the repository.

- There exists the possibility of dividing the waste into several toxicity classes in order to maximise the barrier potential of the repository by emplacing waste with higher toxicity in areas with longer migration paths to the biosphere.

- The construction and emplacement phases are separated, the whole repository lay-out being complete before the commencement of waste emplacement. (This is assumed for Project Gewähr but, for a real project, construction in phases or simultaneous construction and emplacement will probably be considered).

E.6.2 Waste Receipt and Handling

- The waste is brought to the underground reception area through the access tunnel by road or by rail vehicles. The mechanical condition, surface contamination etc. of the delivered waste is checked in the reception area. The radioactivity inventories of waste units are checked on the basis of the accompanying documents or complementary direct measurements if necessary. Waste with nuclide concentrations which exceed maximum permissible values specified for the repository is transferred to the HLW repository.

- An inventory is kept of delivered and emplaced waste. This provides information on the waste in each drum or container and the respective emplacement positions in the disposal caverns. It also gives a continuous overview of accumulated quantities of significant radionuclides in the waste already emplaced.

E.6.3 Waste Package

- No overpack is foreseen for LLW/ILW.

E.6.4 Emplacement

- In the repository caverns the waste is placed in the emplacement position by remote-handling equipment.

E.6.5 Buffer, Backfill and Seals

- The empty space remaining in the repository caverns after emplacement of the waste will be backfilled with special concrete. After the repository is filled to capacity, backfilling and sealing of the remaining empty space is the final step of repository operation. The method of sealing (concrete ? bentonite ? etc.) is not yet specified; the relevant experiments are foreseen for the marl underground laboratory at Wellenberg.

E.6.6 Temperatures and Pressures

- The heat production of the LLW/ILW considered is so small that there is no significant temperature elevation over the normal underground values. The pressure corresponds to the geological overburden of the repository caverns, there will be provisions for the gas produced to escape without unacceptable increase of the pressure in the repository.

E.6.7 Radiation Protection and Safety Considerations

- The design goal stipulated by the Swiss regulations is that the expected contribution to the radiation dose even to the most highly exposed group of public shall be less than 10 mrem per year (cf. Guideline R-21, A.2).

- The safety analyses of Project Gewähr 1985 indicate that this goal can be achieved. For the model-site, the release of radioactivity into the biosphere was also calculated for an unfavourable transport path to a road tunnel in the neighborhood (assumed to be collapsed) and from there into small springs and surface groundwater. Even then radiation doses lie under the protection objective.

- Release scenarios which take long-term geological changes into account included complete exposure of the repository by erosion after 100'000 years. The calculated radiotoxicity of the soil mixture formed - even for this extreme assumption - is still below natural values.

E.6.8 Decommissioning

- Final sealing of access tunnels is to be done either by concrete plugs or by bentonite, the decision on the method still being open.

E.6.9 Post-Closure

- Disposal is conceived in such a way that no control and supervision is necessary after repository closure and a high level of long-term safety can nevertheless be ensured (in accordance with the Swiss law requirements, cf. chapter D.6.9).

- Quality control during emplacement and subsequent backfilling is ensured. Before final closure of the repository, long-term in-situ experiments will be evaluated from a safety viewpoint. Retrieval of the waste after closure is not envisaged but may be possible, albeit at considerable expense.

E.7 Schedule

The overview of the time schedules has been given in chapter A.5. The LLW/ILW repository is scheduled to start operation in 1998, assuming that there are no unexpected politically motivated delays in the licensing procedures and no geological surprises.

E.8 Costs and Funding

cf. chapter A.6

E.9 Experience / Status

Switzerland has much experience in tunnel construction which is sufficient for the construction of horizontally accessible repository caverns envisaged for the LLW/ILW repository. With regard to the geological explorations, Nagra supported the build-up of the necessary experience during the last eight years or so and today state-of-the-art exploratory, processing and modelling techniques are available.

NUWAMNEA, 14. January 1988