SWISS PROJECTS FOR GEOLOGICAL DISPOSAL OF LLW/ILW IN MINED CAVERNS

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ABSTRACT

In Switzerland, repositories of differing protection grade are foreseen and the wastes are allocated to an appropriate disposal facility in accordance with their radionuclide content. One possible concept involves the combination of low- and intermediate-level waste in one repository, but the finally chosen concept will depend upon the characteristics of the specific site selected for the facility. Following a lengthy site selection procedure, three sites have been retained for more detailed characterization, one each in anhydrite, marl and alpine gneisses.

A data set derived from the marl site has been used as a basis for a model repository project which was required by the Swiss safety authorities for showing feasibility and safety for LLW/ILW. A mined cavern system was chosen with horizontal access. The safety assessment of the multibarrier system gave satisfactory results.

WASTE CATEGORIZATION AND DISPOSAL CONCEPT

A comprehensive disposal concept for all types of radioactive waste was developed and published by the Swiss utilities and the Nagra Cooperative in 1978. Subsequently, responsibility for all development and project work concerning final disposal was allocated to Nagra, which presented in 1983 an interim report documenting a more advanced form of the concept.

This concept is based on various types of geological repositories with differing degrees of protection. For each of them, the acceptable waste sorts are defined by fixing the maximum permissible concentrations and quantities of the critical radionuclides. The maximum permissible concentrations (MPC) are derived by comparing the results of simplified safety assessments (considering a system of multiple safety barriers) with the allowable radiation exposures specified in the protection guidelines. Before being accepted into a repository, the waste has to pass through a quality assurance system by which, inter alia, its composition is determined. Waste with radionuclide concentrations greater than maximum values set for each specific waste type and specific repository type cannot be disposed of in the repository in question and must be assigned to a repository type with a higher degree of protection.

To quantify this concept, Nagra has divided the radioactive waste in a first approximation into three categories similar to those defined by IAEA: high level waste (HLW), intermediate level waste (ILW) and low level waste (LLW)[1]. With regard to the total quantity, the present planned capacity of the repositories assumes doubling of the current nuclear power production in Switzerland and that the repositories must be able to accomodate the maximum accumulated waste from seven decades of radioisotope application in medicine, industry and research, as well as all waste from nuclear energy use during that period. This corresponds to 240 GW-years (electrical) and results in 1200 m^3 HLW and roughly 200'000 m^3 ILW and LLW. In the project submitted last year to the Government with a view to guaranteeing feasibility and safety of final disposal, the latter two categories have been lumped together and allocated to one single type of repository. In the following I will describe in some detail the work already done and the planning for realization of this repository.

GUIDELINES FOR FINAL DISPOSAL

The Nagra Cooperative is the common undertaking of those organizations required by law to take care themselves of the disposal of radioactive waste which they have produced or collected. The members are the Federal Government (represented by the Office of Public Health) and the six utilities operating or intending to operate nuclear power plants. It is intended in the first instance for research, development and project engineering work. Organisational adaptations will be necessary for constructing and operating the disposal facilities.

Of course, governmental authorities will review the disposal projects and will also undertake the necessary control and supervision measures. It is their responsibility therefore to define the objectives to be reached.

The concept of final disposal is quantitatively defined by protection objectives in official guidelines issued in 1980 by the Federal Commission for Safety in Nuclear Installations and the Federal Office of Energy (Nuclear Safety Department). Protection objective No. 2 specifies that:

> A repository must be designed in such a way that it is capable of closure at any time within a few years. After closure of a repository it must be possible to dispense with safety and supervision measures.

In other words, the waste must be so safely disposed of in a repository that it is not necessary for future generations to concern themselves with it. The quantitative objective for the protection of man and the environment from harm through ionising radiation from the waste material is given in protection objective No. 1:

> Radionuclides from a sealed repository, which reach the biosphere as a result of realistically conceivable processes and events, may at no time lead to individual doses which exceed 10 mrem per year.

The dose restriction postulated applies to the sum of all possible radiation effects i.e. in particular also to that part which could occur in the human body through the intake of materials with short range radiation. The dose limit was set to be less than 5 % of the average continuous radiation exposure from all natural sources in Switzerland.

There are no further criteria for disposal defined by the authorities. It is the responsibility of the waste producers, in practice of the Nagra cooperative, to develop detailed projects and subordinate criteria and to prove that, by observing them, the overall protection objectives of the guidelines are achieved.

CHARACTERISATION OF WASTES ACCEPTED IN ILW-LLW REPOSITORY

At present, all nuclear fuel used in Switzerland is under contract for reprocessing in France (COGEMA) or the United Kingdom (BNFL). For fuel discharged after 1993, the option of direct disposal is also kept open. From the reprocessing abroad, 6 different waste sorts (as shown in Table I) are returned to Switzerland. They would be replaced by one single waste sort, the spent fuel, if the reprocessing option were to be abandoned.

TABLE I
Reprocessing Waste

Nature of waste	Matrix	Container	Total activity MCi	Reference age	Total volume m^3
High level waste	Glass	SS	β/γ 1120 α 9.4	40 a	1120
Precipitates and concentrates	Bitumen	SS	β/γ 4.2 α 0.02	3 a	4320
Ion exchange resins	Cement	Asbestos-cement	β/γ 0.3 α 0.000003	5 a	860
Fuel element hulls and end-caps	Cement	Steel	β/γ 7.9 α 0.03	40 a	5600
Technological waste low α	Cement	Asbestos-cement	β/γ 0.14 α 0.0045	4 a	27800
Technological waste medium α	Cement	Asbestos-cement	β/γ 1.4 α 0.055	4 a	13900
TOTAL GROSS VOLUME					52500 m^3

The operation of the nuclear power plants yields 7 waste sorts, the characteristic data of which are summarized in Table II.

TABLE II
Operational Waste

Nature of waste	Matrix	Activity β/γ in kCi	Activity α in Ci	Gross volume m^3
Ion exchange resins	Cement, resins	6400	6.2	35100
Concentrates, slurries	Cement, bitumen	2.4	0.002	1150
Filters	Cement	4.7	0.3	100
Air ventilation filters	Cement	0.02	$100 \cdot 10^{-3}$	150
Non-incinerable solid waste	Cement	1.1	$0.6 \cdot 10^{-3}$	5430
Incinerated waste	Cement	2.1	$4.4 \cdot 10^{-3}$	560
Fuel element casings	Cement	1500	5.6	1750
200 l STEEL DRUMS, TOTAL GROSS VOLUME				44000 m^3

Decommissioning of the power plants according to present plans is expected to produce 8 waste sorts with characterisation as shown in Table III.

TABLE III
Decommissioning Waste

Nature of waste	Activity β/γ in kCi	Activity α in Ci	Gross volume m^3
IL activated steel	481000	1.5	9480
IL activated concrete	3.9	0	1440
LL activated steel	23	0.1	5910
LL activated and contaminated concrete	7	0	10010
IL contaminated steel	1	1	10220
LL contaminated steel	0.04	0.05	39870
LL contaminated materials and secondary wastes	0.08	0.08	7510
Secondary wastes	105	105	12660
CEMENT MATRIX, TOTAL GROSS VOLUME			97000 m^3

The wastes from radioisotope application and nuclear research are subdivided into 5 sorts according to Table IV.

TABLE IV
Waste from Medicine, Industry and Research

Nature of waste	Activity α in kCi	Activity β/γ in Ci	Gross volume m^3
Beta-/gamma-emitters (without tritium)	0	9940	3000
Tritium-containing waste	0	686000	600
Alpha-emitters (without radium)	1300	250	1870
Radium-containing waste	60	0	230
Alpha- and beta-/gamma-emitters	420	13500	1800
35000 STEEL DRUMS OF 200 l, CEMENT-MATRIX			7500 m^3

The waste characteristics given in the 4 Tables have to be compared with the maximum permissible concentration (MPC) profile for the combined ILW-LLW repository. As a test case, a first specific set of MPC values has been derived from preliminary safety analyses using available geological and hydrogeological data from a marl formation used as a model site for which a suitable repository engineering design was developed, as described in further detail below. Some typical MPCs resulting from this first calculation are compared in Table V with MPCs for "Shallow Land Burial" as published in the U.S. Federal Register.[2]

TABLE V
Examples of MPCs in Ci/m^3

Nuclides	US shallow land burial	CH LL/ILW repository
C-14	0.8	3700
Sr-90	700	-
Zr-93	700	6 Mio
Tc-99	0.3	15
Pd-107	700	12
Sn-126	700	4
I-129	0.008	0.0001
Cs-135	84	0.27
Cs-137	4600	-
Ra-226	0.02	630
Pu-239	0.02	4.6

Detailed comparison of MPC values and waste characteristics indicates that, with the exception of vitrified high level waste from reprocessing (or alternatively the spent fuel), all waste sorts could be allowed in the combined ILW-LLW repository as modelled within the feasibility study. However two sorts, namely bituminised precipitates and concentrates from reprocessing and fuel element hulls and end-caps, come near to the postulated MPCs. To ensure flexibility, the corresponding storage volume is therefore also allowed for within the Nagra HLW repository design, which is not discussed in this paper.

The division of radioactive waste into readily specifiable sorts and the estimation of quantities for final disposal reflects present-day technology. It can be assumed that the future will bring new improved conditioning procedures, which means that the definition and specification of individual waste sorts and their nuclide inventories can alter. The sorting method described and the quality control system suggested ensure that each waste sort can be located in the appropriate repository. The total inventory of fission and activation products is physically determined by the extent of fuel burn-up and reactor construction and can not be increased by any reactor accident, although the total conditioned waste volume of LLW and ILW could be considerably greater.

MULTIBARRIER REPOSITORY SYSTEM, ENGINEERED BARRIERS

All low- and intermediate level wastes with radionuclide concentrations not exceeding the maximum permissible values shown in Table V are foreseen for disposal in a mined cavern system. To satisfy the Government's demand for guaranteeing feasibility and safety of disposal, this model project has been worked out as shown in Fig. 1. It has the following design and operation characteristics:

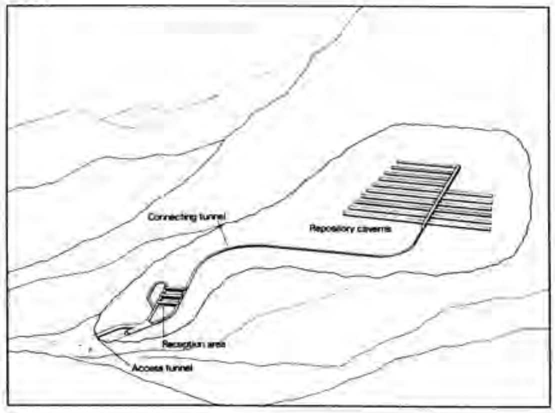

Fig. 1. Mine Cavern System Model.

- the final disposal caverns are under several hundred meters of overburden;
- access is through horizontal tunnels;
- the reception area is also underground but nearer to the surface;
- successive emplacement and backfilling allows long-term, in-situ experiments for safety evaluation before final closure of the repository;
- retrieval of the waste after closure is only possible at considerable technical expense, as no concessions to safety barriers are allowed to make it easier;
- the waste is delivered in a conditioned state (in the solidification matrix) and additional technical barriers are provided during construction, operation and closure of the repository; see Fig. 2

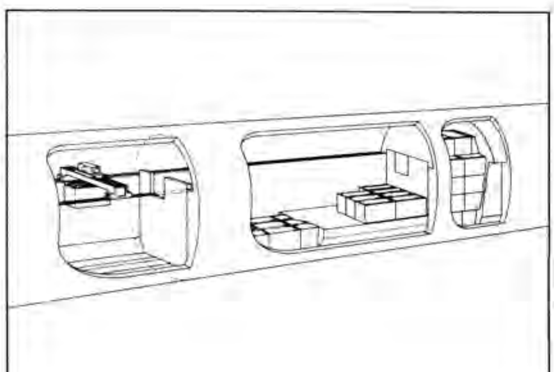

Fig. 2. Additional Technical Barriers.

- all waste is emplaced in standard containers, into which the smaller waste canisters are transferred in the reception area and immobilized with liquid cement; see Fig. 3

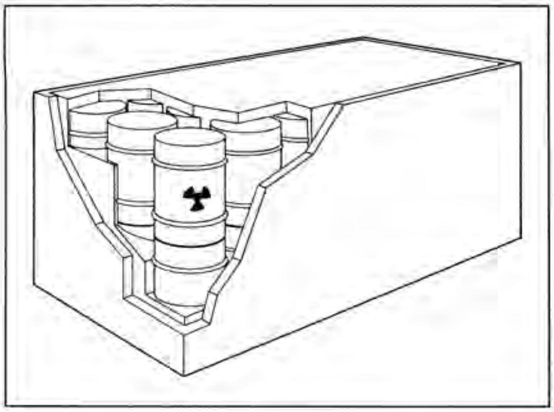

Fig. 3. Standard Container.

- decommissioning waste which is already solidified and immobilised in large containers at the reactor site undergoes only quality control checks in the reception area and is then transported directly to the disposal caverns;
- the possibility exists of allocating the waste to individual caverns in accordance with toxicity classification in order to make best use of longer potential migration paths to the biosphere;
- organisational arrangements provide for an inventory and continual overview of accumulated quantities of radionuclides;
- the design allows construction and emplacement periods to overlap to a certain extent.

The design concepts result in a repository in which the radionuclides are isolated from the biosphere by a multiple barrier system as summarized in Fig. 4. From the waste matrix outwards to the lining of the caverns, all barriers are man-made and require careful quality control. The outer, geological barriers, on the other hand, are defined by the site selection and characterization.

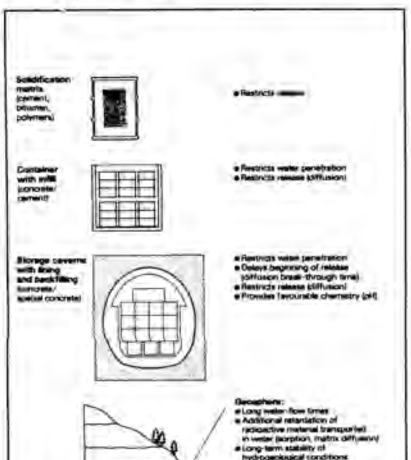

Fig. 4. Safety Barrier System for Low- and Intermediate-level Waste.

GEOLOGICAL BARRIERS AND THE PRE-SELECTION OF THE REPOSITORY SITE

The structure of the engineered barriers will change with time, although deterioration of the barrier quality will be slow. The bulk of the barrier material will remain in place but its permeability may tend to increase. Thus water might reach the waste matrix where radionuclides could be leached out. The geological barriers then play an important role with regard to the transport with water of leached nuclides through the rock formations to the biosphere. The geological surroundings also define the groundwater chemistry which, in turn, influences corrosion of the waste matrix and the leaching process.

Host rock formations must therefore meet a number of requirements. The main ones are:

- high impermeability
- low water flow
- good sorption properties for nuclides in aqueous solution
- sufficiently large extent of host rock
- geometric predictability
- geological stability of the siting area over the waste isolation times required
- suitability for tunnel construction.

Of the criteria listed, the first two are the most important. From this point of view, host rocks where the dryness of the disposal caverns is due to the natural impermeability of the rock itself or of its overburden are suitable. Geological conditions in Switzerland led to concentrating investigations on the following host rocks:

Anhydrite

i.e. anhydrous calcium sulphate which in its pure form is dry and stable. On uptake of water it alters to gypsum, with a 60 % volume increase, which can result in re-closure of water-bearing cracks. Anhydrite deposits are, however, rather small and, in the presence of water, calcium sulphate corrodes metals and concrete.

Alpine marls and clay shales

In the Alps there are large deposits of marls and clays which are distinguished by their very low permeability and good sorption properties. The plasticity of these rocks renders them relatively insensitive to tectonic stresses but, on the other hand, presents problems for tunnel construction.

Jura Opalinus clay (nothern Switzerland)

In the Tabular Jura there is a layer of Opalinus clay with thickness extending up to 110 m. This rock is practically impermeable and has excellent sorption properties. However, from the point of view of tunnel construction, Opalinus clay is mechanically unstable and undergoes plastic deformation.

Crystalline rocks

Crystalline formations occur on a large scale. They are largely impermeable, their sorption properties are good and they have practically no chemically aggressive components. They are also suitable from a construction point of view. Both the crystalline bedrock below the sedimentary cover in northern Switzerland and alpine mountain formations can be considered.

Shielded limestone

The only permeable rock to be considered was limestone which is very suitable for tunnel construction. It is considered only when occurring in an anticlinal form which is overlaid by a protective, watertight layer of clays and marls, and when the planned disposal zone can safely be located above the highest expected groundwater level.

These five types of host rock formation were used in a first step as a basis for screening geological formations in Switzerland. About one hundred potential sites were identified and evaluated in the years 1978 through 1981 using safety and engineering criteria and the geological and hydrogeological information available at that time.

This resulted in the selection of twenty sites, (see Fig. 5) appearing promising enough to justify further non-licensable investigations which were undertaken in 1982/83. For further ranking, criteria from consideration of safety, engineering and environmental planning were used. Three sites were chosen for characterisation with first priority. Five further exploratory areas are being kept in reserve and the remaining 12 sites have been deferred from further processing.

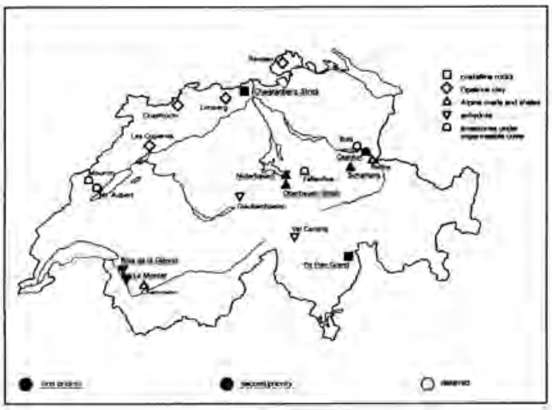

Fig. 5. Selected Sites.

For the three sites of the first priority group - one each in anhydrite, marl and alpine gneisses - an extensive geological, geophysical and hydrogeological exploration programme has been set up. The necessary licence applications were prepared and submitted to the authorities at the end of 1983.

In September 1985 the Swiss federal government authorized all planned exploration work except construction of experimental access galleries.

Detailed geological mapping as well as observation and monitoring of wells and springs is underway. Seismic investigations and test drilling will soon start. Only when comparable knowledge of the three sites has been accumulated in a couple of years will the withheld authorisation of construction of test galleries be considered by the federal government.

SAFETY ASSESSMENT OF THE MODEL REPOSITORY

In the meantime, a geological data set corresponding to one of the three first priority sites has been chosen for elaborating the model project already discussed and illustrated in Fig. 1. The location of the repository within an extended marl formation known as a result of a motorway tunnel construction is shown in Fig. 6. The aim of this model project is to prove feasibility and safety of final disposal. Technical feasibilty has been demonstrated by numerous similar underground constructions in our country. Thus the safety assessment remains the crucial issue.

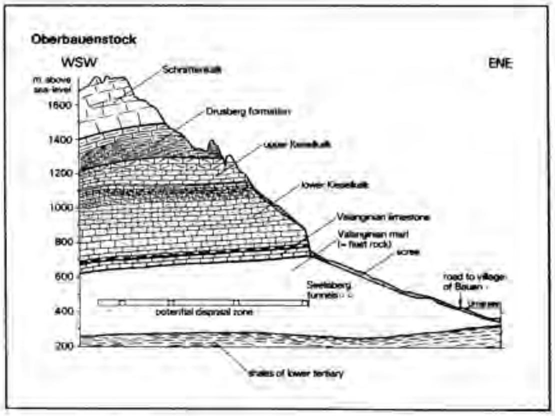

Fig. 6. Geological Data of Potential Disposal Zone.

The ultimate aim of safety analysis of a repository is to quantify the extent and probability of possible radiation exposure to human beings caused by release of radionuclides at any point in the future. Three steps are necessary. The first is complete specification of the repository system and the contained radionuclide inventory as summarized above and given in detail in Nagra reports on Project Gewähr 1985[3]. The second step consists of a scenario analysis i.e. consideration of all events and processes which can affect the transport of radionuclides from the repository to man. The third step involves converting effects within selected scenarios into predictions of radiation doses to the most exposed persons in the population.

Two main scenarios have been retained for deterministic consequence calculations. The first involves dissolution and transport of radionuclides to the biosphere by groundwater; a wide range of parameter variations has been studied here. The second scenario assumes massive erosion of the mountain in which the repository is placed so that, after 100'000 years, the entire rock mass has been converted into farmland in which the remaining radionuclides are homogeneously distributed.

The chain of calculation models for the groundwater release scenario describes the groundwater movement, release from the near-field, geosphere transport and uptake in the biosphere. A realistic-conservative data-set yields the results shown in Fig. 7.

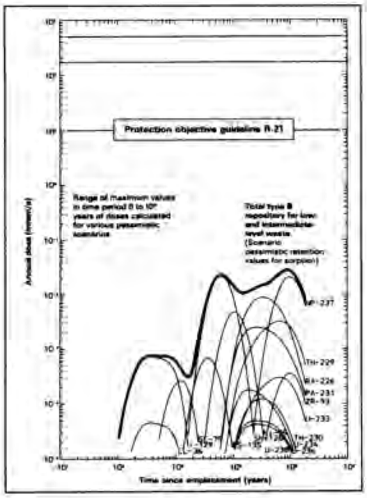

Fig. 7. Natural Radiation Exposure in Switzerland.

The theoretically calculated decrease of dose after one million years is due to depletion of the repository inventory and not to radioactive decay. Introduction of extremely conservative parameter values can result in a dose peak as high as 3 mrem/a, which is however still below the protection objective.

For the erosion scenario, doses resulting from ingestion, inhalation and direct radiation have been calculated to be 13 mrem/a using conservative assumptions. Np-237 contributes 80 % to this dose and a further 10 % is provided by Ra-226. This result should be seen in perspective by comparing concentrations with those of similar radionuclides occurring naturally in the ground, as shown in Table VI.

TABLE VI

Nuclide	Concentration in erosion scenario (Ci/kg)	Natural concentration (Ci/kg)
Ra-226	$1.3 \cdot 10^{-11}$	$3 \cdot 10^{-11} \ldots 5 \cdot 10^{-9}$
Th-232		10^{-9}
Np-237	$1.4 \cdot 10^{-11}$	
U-238		10^{-9}

In fact, erosion of the entire mountain (including the repository residues) after 100'000 years does not result in a significantly higher radiation exposure than that predicted from natural radionuclides following erosion of the same massif without a repository.

We conclude from these consequence calculations using the geological model data set that construction of a repository for low- and intermediate level waste in such a marl formation is feasible and would be acceptable from the point of view of long term safety. However, certain data in the analyses, e.g. the hydraulic conductivity, are known only by extrapolation. Thus, direct exploration and detailed site characterization is needed before final selection of the construction site is possible.

FINAL SITE SELECTION AND CONSTRUCTION PLANNING

Low- and intermediate-level waste exists already in a form suitable for disposal and is at present accumulated in intermediate storage facilities at the power plant sites. It is also planned to construct a central intermediate storage facility in Switzerland for all types of wastes returned from nuclear fuel reprocessing abroad. Such a facility could also provide some buffer capacity for operational waste from the power plants. Nevertheless, final site selection and preparations for construction of a repository for low- and intermediate-level waste are considered the top priority items in nuclear waste management in Switzerland today.

At the three most promising sites mentioned above, site characterisation work is therefore being pushed ahead. After further investigations from the surface, construction of a central gallery and transverse galleries covering the full dimensions of the repository are envisaged. We hope to be able to decide in about two years from now whether such gallery constructions are justified at all three sites.

In addition, at the request of the safety authorities, a further site of different general characteristics, possibly with vertical access, will also be considered. One of the five preselected sites kept in reserve may, in fact, be suitable for a project complying with that request. The inclined marl formation at the site in question may extend far below the bottom of the adjacent valley. Within this formation, subdivision of the cavern system into distinct levels is being considered. Horizontal access and emplacement of low-level, high-volume wastes would then be at the higher level, from which a vertical shaft would provide access to a second cavern or silo system up to 100 m lower. In the latter the waste sorts with higher alpha-content would be emplaced.

Before 1995 it should be possible to formulate a general licence application for repository construction at one of the sites - only one will be needed if the present extrapolations are confirmed. It is expected that licensing procedures up to parliamentary approval may last about five years. Specific construction licensing should take much less time, so that the start of repository construction could be possible around the turn of the century.

Construction of the central gallery and some disposal caverns would take about a further five years using present techniques. Taking into account that it appears quite feasible to provide for simultaneous construction and emplacement phases, final disposal might begin a few years after 2000.

SUMMARY AND CONCLUSIONS

The cooperative organisation responsible for radioactive waste disposal in Switzerland concludes from its investigations to date that it is highly probable that all radioactive waste produced in the country during the coming seven decades, with the exception of vitrified high level waste (or alternatively spent fuel), can be disposed of in one single repository in a single geological formation. A corresponding model repository project is being reviewed at present by federal government authorities and invited experts.

To gain full assurance, further site investigations are under way. They are planned to provide geological and hydrogeological data for the entire volume of the host formation needed for the combined LLW- and ILW-repository. In the course of the work, final selection between four or more different sites and host rocks will be made. In parallel, validation of the mathematical modelling for the safety assessment and further iterations of dose calculation with more precise data will be performed.

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2. U.S. Federal Register, 10 CFR 61 (1982): Licensing Requirements for Land Disposal of Radioactive Waste (New Part 61 to Title 10 of Code of Federal Regulations) Federal Register Notice, December 27, 1982.

3. Project Report NGB 85-01, 85-02, 85-06, 85-07 and 85-09.