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Comparison of Selected Foreign Plans and Practices for Spent Fuel and High-Level Waste Management

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April 1990

Prepared for the U.S. Department of Energy
under Contract DE-AC06-76RLO 1830

Pacific Northwest Laboratory
Operated for the U.S. Department of Energy
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ABSTRACT

This report describes the major parameters for management of spent nuclear fuel and high-level radioactive wastes in selected foreign countries as of December 1989 and compares them with those in the United States. The foreign countries included in this study are Belgium, Canada, France, the Federal Republic of Germany, Japan, Sweden, Switzerland, and the United Kingdom. All the countries are planning for disposal of spent fuel and/or high-level wastes in deep geologic repositories. Most countries (except Canada and Sweden) plan to reprocess their spent fuel and vitrify the resultant high-level liquid wastes; in comparison, the U.S. plans direct disposal of spent fuel. The U.S. is planning to use a container for spent fuel as the primary engineered barrier. Use of other barriers such as matrix materials, overpacks, or buffers, common in foreign disposal concepts, is not presently planned in the U.S. The U.S. has the most developed repository concept and has one of the earliest scheduled repository startup dates. The repository environment presently being considered in the U.S. is unique, being located in tuff above the water table. The U.S. also has the most prescriptive regulations and performance requirements for the repository system and its components.

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TABLES

GLOSSARY

AEB	Atomic Energy Bureau (Japan)
AEC	Atomic Energy Commission (Japan)
AECB	Atomic Energy Control Board (Canada)
AECL	Atomic Energy of Canada Limited
AERE	Atomic Energy Research Establishment (United Kingdom)
AFR	away-from-reactor
AGNEB	Interagency Working Group on Nuclear Waste Management (Switzerland)
AGR	advanced gas-cooled reactor
ALARA	As Low As Reasonably Achievable
ANDRA	National Radioactive Waste Management Agency (France)
AR	at-reactor
AVM	Atelier de Vitrification de Marcoule (vitrification plant at Marcoule, France)
BFS	Office for Radiation Protection (FRG)
BMFT	Bundesministerium fuer Forschung und Technologie (Federal Ministry for Science and Technology, FRG)
BMU	Bundesministerium fuer Umwelt, Naturschutz und Reaktorsicherheit (Federal Ministry for Environmental Protection and Reactor Safety, FRG)
BMWI	Federal Ministry for Economics (FRG)
BNFL	British Nuclear Fuels, Ltd
BWR	boiling water reactor
CANDU	Canadian deuterium uranium
CEA	Commissariat à l'Energie Atomique (French Atomic Energy Commission)
CEC	Commission of European Communities
CEGB	Central Electricity Generating Board (United Kingdom)
CEN/SCK	Centre d'Étude de l'Énergie Nucléaire/Studiesentrum voor Kernenergie (Nuclear Energy Research Center, Belgium)
CLAB	Central Spent Fuel Storage Facility (Sweden)
CRIEPI	Central Research Institute of the Electric Power Industry (Japan)
CSIN	Interministerial Committee on Nuclear Safety (France)
CSSN	Supreme Council for Nuclear Safety (France)
DBE	Deutsche Gesellschaft zum Bau und Betrieb von Endlagern fuer Abfallstoffe mbH (Company for Construction and Operation of Waste Disposal Facilities, FRG)
DOE	United States Department of Energy
DOT	United States Department of Transportation
DWK	German Fuel Reprocessing Company
EARP	Environmental Assessment and Review Panel (Canada)
EDI	Federal Department of the Interior (Switzerland)

EIR	Federal Institute for Reactor Research (Switzerland)
EIS	Environmental Impact Statement
EPA	United States Environmental Protection Agency
EVED	Department of Transport, Communications and Energy (Switzerland)
FBR	fast breeder reactor
FRG	Federal Republic of Germany
GCHWR	gas-cooled, heavy water-moderated reactor
GCR	gas-cooled, graphite-moderated reactor
GNS	Company for Nuclear Service (FRG)
GSF/IFT	Company for Radiation and Environmental Research/Institute for Underground Storage (FRG)
GW(e)	GigaWatts (electric)
HLLW	high-level liquid waste
HLW	high-level waste
HMIP	Her Majesty's Inspectorate of Pollution (United Kingdom)
HRL	hard-rock underground research laboratory
HSK	Nuclear Safety Inspectorate (Switzerland)
HTGR	high-temperature gas-cooled reactor
HWR	heavy water reactor
IAEA	International Atomic Energy Agency
ICRP	International Commission of Radiological Protection
ILW	intermediate-level waste
INWAC	IAEA International Waste Management Advisory Committee
IPSN	Institute for Nuclear Protection and Safety (France)
ISVL	in situ site validation laboratory
JAERI	Japan Atomic Energy Research Institute
JNFI	Japan Nuclear Fuel Industries Company, Inc.
JNFS	Japan Nuclear Fuel Service Company, Ltd.
KBS	Kärn-Bränsle Säkerhet (Study Project on Management of Radioactive Wastes, Sweden)
KSA	Swiss Federal Commission for the Safety of Nuclear Installations
LLW	low-level waste
LMFBR	liquid metal fast breeder reactor
LWR	light water reactor
MITI	Ministry of International Trade and Industry (Japan)
MOX	mixed (plutonium-uranium) oxide
MRS	Monitored Retrievable Storage
MW(e)	MegaWatts (electric)

NAGRA	National Cooperative for the Disposal of Radioactive Waste (Switzerland)
NAS	National Academy of Sciences (United States)
NEA	Nuclear Energy Agency
NFWMP	Nuclear Fuel Waste Management Program (Canada)
NII	Nuclear Installations Inspectorate (United Kingdom)
NIREX	Nuclear Industry Radioactive Waste Executive (United Kingdom)
NNC	United Nuclear Corporation (United Kingdom)
NRC	Nuclear Regulatory Commission (United States)
NSB	Nuclear Safety Bureau (Japan)
NSC	Nuclear Safety Commission (Japan)
NTL	Nuclear Transport, Ltd.
NWPA	Nuclear Waste Policy Act of 1982 (United States)
OCRWM	U.S. Department of Energy/Office of Civilian Radioactive Waste Management
OECD/NEA	Organization for Economic Cooperation and Development/Nuclear Energy Agency
ONDRAF/NIRAS	Organisme National des Déchets Radioactifs et des Matières Fissiles (National Institute for Radioactive Wastes and Fissile Materials, Belgium)
PAGIS	Performance Assessment of Geologic Isolation System
PNC	Power Reactor and Nuclear Fuel Development Corporation (Japan)
PNC	Nuclear Fuel Development Corporation (Japan)
PNTL	Pacific Nuclear Transport, Ltd. (United Kingdom)
PRA	spent fuel conditioning plant at Gorleben (FRG)
PSI	Paul Scherrer Institute (Switzerland)
PTB	Physikalisch-Technische Bundesanstalt (Federal Science/Engineering Institute, FRG)
PWR	pressurized water reactor
QA	quality assurance
R&D	research and development
RSK	Federal Reactor Safety Commission (FRG)
RWMAC	Radioactive Waste Management Advisory Committee (United Kingdom)
RWMC	OECDINEA Radioactive Waste Management Committee
SCPRI	Central Service for Protection against Ionizing Radiation (France)
SCSIN	Central Service for the Safety of Nuclear Installations (France)
SFL	Central Repository for Spent Fuel (Sweden)
SFR	Central Repository for Reactor Wastes (Sweden)
SGHWR	steam-generated heavy water reactor
SKB	Swedish Nuclear Fuel and Waste Management Company

SKI	Statens Kaernkraftinspektion (Nuclear Power Inspectorate, Sweden
SKN	National Board for Spent Nuclear Fuel (Sweden)
SIN	Swiss Institute for Nuclear Research (Switzerland)
SS	stainless steel
SSEB	South of Scotland Electricity Board (United Kingdom)
SSI	National Institute of Radiation Protection (Sweden)
SSK	Radiation Protection Commission (FRG)
STA	Science and Technology Agency (Japan)
TAC	Technical Advisory Committee (Canada)
TBD	to be determined
THORP	Thermal Oxide Reprocessing Plant (United Kingdom)
THTR	thorium high-temperature reactor
TN	Transnuklear (Germany)
TRU	transuranic
UKAEA	United Kingdom Atomic Energy Agency
U.K.	United Kingdom
URL	underground research laboratory
U.S.	United States
WAK	pilot reprocessing plant at Karlsruhe (FRG)

SUMMARY

This report describes the major parameters for management of spent nuclear fuel and high-level radioactive waste in selected foreign countries and compares them with those in the United States. The foreign countries included in this study are Belgium, Canada, France, the Federal Republic of Germany (FRG), Japan, Sweden, Switzerland, and the United Kingdom (U.K.). The major waste management parameters in foreign countries and the United States (U.S) are summarized in this section and discussed in subsequent sections for each country.

LEGISLATIVE/INSTITUTIONAL BASES

The institutional aspects of spent fuel and high-level waste management are summarized in Table S.1. The current percentage of electricity produced by nuclear power ranges from a low of about 20% in Canada, the U.K. and the U.S. to highs of 65% and 70% in Belgium and France, respectively. France and Japan have aggressive nuclear power programs. Belgium, Canada, the FRG, and the U.K. are experiencing a slowdown in growth and growth of nuclear power has essentially stopped in Switzerland and the U.S. Sweden plans to phase out all nuclear power by 2010, and a phase-out referendum is scheduled to be held in Switzerland in 1991. In the FRG, Japan, Sweden, and Switzerland, development or operation of nuclear power plants hinges on finding a solution to management of the spent fuel and high-level radioactive wastes.

Local opposition to the development of repositories for disposal of spent fuel and high-level radioactive wastes is common and appears to be growing. During the repository siting process, public hearings are held in most countries. In some cases, the local governments have some authority over repository development, and in others they provide only opinions, but local and public opinions are taken into consideration in all cases. In most countries,

TABLE S.1. Institutional Aspects of Spent Fuel and High-Level Waste Management

Country	Current Nuclear Power Policy	Nuclear Power Waste Stipulation Law	HLW Management Organization	State/Local Role In Repository Development	Waste Fund Manager
Belgium	65% of electricity; no growth in 1990s	No	Government (ONDRAF)	Review only and opinions	Government
Canada	20% of electricity; slow growth	No	Government (AECL)	Review only and opinions	Government
France	70% of electricity; continuing growth	No	Government (ANDRA)	Public inquiry and opinions	No fund; government supported
Fed. Republic of Germany	35% of electricity; growth has slowed	Yes[a]	Government (BfS) & Industry	Public hearing; local approval; state - licensing decision	Government
Japan	32% of electricity; continuing growth	Yes[a]	Government (STA)	Public hearing; consensus approach	Government
Sweden	50% of electricity; phase-out by 2010	Yes	Industry (SKB)	County siting veto; public comments	Government
Switzerland	40% of electricity; phase-out referendum to be held in 1991	Yes	Industry (NAGRA)	Consultative only	Industry
United Kingdom	20% of electricity; Magnox phase-out, PWR phase-in	No	Government-policy; Industry-implementation	Public inquiry; local veto (can be overridden)	Government
United States	20% of electricity; no new orders since 1978	No	Government (DOE)	State siting veto; public comments and hearings; state monitors programs; review committees	Government

(a) Applies to management or storage of spent fuel.

local vetoes of repository siting may be overruled by the federal government, but Japan and the FRG would prefer to attain acceptance by the local institutions.

The federal government is primarily responsible for long-term management of spent fuel and high-level wastes in all countries except for Sweden and Switzerland, where industry has this responsibility. In all cases, including the U.S., the waste producers pay waste disposal costs. In Canada and France, the major waste producers, the utilities, are government-owned. The federal governments manage a waste fund to pay for long-term management of the wastes in most countries, including the U.S.

Table S.2 summarizes the regulatory aspects of spent fuel and high-level waste management facilities. The federal governments in nearly all countries are responsible for repository licensing. In several countries, local government permits are required, and in the FRG the states have major licensing responsibility (although the federal government can override decisions). In all countries, public and local government input is considered in the licensing process.

Most countries, unlike the U.S., have established only general performance requirements for repositories; some of these currently do not believe that detailed performance requirements and regulations for waste repositories are necessary, and do not plan on developing them. All believe that a combination of engineered and natural barriers should be used to achieve safe disposal, and that the overall safety of the disposal system as a whole should be considered.

OVERALL WASTE MANAGEMENT SYSTEM

The overall spent fuel and high-level waste management systems of the countries are summarized in Table S.3. All the countries reviewed except the U.S., Canada, and Sweden plan to reprocess spent fuel and vitrify the resultant high-level liquid wastes. Canada, however, has not completely ruled out reprocessing, although its CANDU fuel cycle does not encourage recycling. Sweden plans to phase out nuclear power by 2010, and therefore, has elected to

TABLE S.2. Regulatory Aspects of Spent Fuel and High-Level Waste Management Facilities

Country	Approval Steps for Repository Licensing	Licensing Required by Federal/Local Agencies	Public Role in Licensing	Repository Performance Requirements	Status of Regulations
Belgium	Local and regional review; Ministry approval; Government decree	yes/no	Review and comment only	General only	Details to be developed
Canada	Expert and public review; ministry approval	yes/no	Review and comment only	General only	Under development
France	Public inquiry; safety review; Government degree	yes/no	Public inquiry; local veto (may be overruled)	General only	Under development
Fed. Republic of Germany	Initial - local; site specific - federal, state, expert review; public hearing; state licensing decision	yes/yes	Public hearing; local veto	General for total system	Regulations complete
Japan	Agency reviews; public hearing; local consensus desired	yes/no	Public hearing; local consensus desired	Not yet established	Not yet established
Sweden	Siting (local and federal); construction (local and federal); R&D plan (federal); operation (federal)	yes/yes	Public hearing; local veto	General for total system	Regulations complete
Switzerland	Federal approval; canton-nonnuclear licensing	yes/yes	Consultative	Total system objectives	Regulations complete
United Kingdom	Not yet defined	yes/no	Local veto on siting (government can override)	General only	Defined for spent fuel management; deferred for repository
United States	Siting (state and federal); construction (federal, permits - states); operation and closure (federal)	yes/yes	Public hearings; state veto on siting; regulatory advisory committee	Specific for system and individual components	Complete, although some are undergoing revision

TABLE S.3. Overall Spent Fuel and High-Level Waste Management Systems

Country	Interim Storage of Spent Fuel	Fuel Reprocessing	Extended Interim Storage of HLW or Spent Fuel	Transport to Repository	Geologic Disposal and Waste Age at Disposal
Belgium	AR only until reprocessing	Yes; in France and U.K.	HLW 30-50 yr in dry AFR	Rail offsite; truck onsite	Yes, HLW; ~50 yr
Canada	AR only until disposal	No	Spent fuel in water and air ~50 yr	Truck	Yes, spent fuel; ~50 yr
France	1 yr AR; 2-3 yr at reprocessing plant	Yes	HLW 20-30 yr in dry AFR or at disposal site	Rail; truck for short hauls; ship from other continents	Yes, HLW; ~30 yr
Fed. Republic of Germany	1-10 yr AR; AFR planned	Yes; recent redirection to foreign suppliers (France and (U.K.)	Spent fuel - wet AR, dry AFR; HLW - dry AFR	Rail; truck	Yes, HLW and spent fuel; ~20 yr
Japan	2-3 yr AR	Yes; in Japan, France and U.K.	HLW 30-50 yr in dry AFRs at reprocessing plants	Ship; truck over land	Yes, HLW; 30-50 yr
Sweden	1-5 yr AR/30-40 yr AFR	None (early contracts being traded)	Spent fuel wet AR and AFR	Ship; truck over land	Yes, spent fuel; ~40 yr
Switzerland	~10 yr AR; dry AFR planned	Yes; in France and U.K.; direct disposal option open	HLW 40 yr in dry AFR	Rail; truck	Yes, HLW and spent fuel ~40 yr
United Kingdom	AR and AFR for Magnox and AGR - short term; LWR - AR	Yes; U.K. and foreign fuel (U.K. PWR deferred)	HLW - dry AFR; LWR spent fuel - 18 yr AR	Rail and truck - U.K. fuel; Ship - foreign fuel	Yes, HLW; 50-100 yr
United States	AR (wet) and extended AR (dry); one small wet AFR until disposal; dry federal AFR proposed	None (small amount done before 1972)	Spent fuel up to ~30 yr AR; Some dry storage AR and at proposed AFR; small amount of HLW in dry AFR	Rail and truck; possibly some barge	Yes, spent fuel and HLW; 5-40 yr

Note: AR = At-Reactor
AFR = Away-from-Reactor

trade its early contracts for reprocessing a small amount of spent fuel with other countries. However, some countries (Switzerland and the FRG) are re-examining the direct disposal option.

The spent fuel or solidified high-level waste will be 20 to 100 years old before disposal (although it could be as young as 5 years in the U.S.), and thus, extended interim storage for decades will be needed. Most of the countries that intend to reprocess are planning for interim storage of their spent fuel in the reactor storage pools (typically for up to 10 years) until the fuel is shipped to the reprocessing plant, which will also provide some interim storage (all in storage pools). Canada plans to interim store its spent fuel at reactors (using both wet and dry storage facilities) until disposal. Sweden is storing its spent fuel at reactors for a few years, followed by interim storage in its central away-from-reactor wet storage facility. Switzerland and the FRG are planning for some dry interim storage of spent fuel at one or more central locations (i.e., at away-from-reactor facilities) to supplement their at-reactor storage.

Because no repository is planned to be operating for at least the next 20 years, even those countries that are having their spent fuel reprocessed are planning to interim store their solidified high-level waste for 10 to 50 years. In the U.K., repository development is being deferred in favor of long-term interim storage, which may continue for as long as 100 years. All interim storage of solidified high-level waste in the countries reviewed in this document will use dry storage concepts.

Transport of spent fuel to reprocessing plants and transport of spent fuel and solidified high-level waste to disposal facilities will generally be done using truck or rail. Transport of spent fuel and solidified high-level waste by ship will be carried out for transport between continents or when the facilities are near sea ports (for example, in Sweden and Japan).

Most countries use single-purpose casks for spent fuel and high-level waste transportation. Interest in dry storage is increasing and some countries (the FRG and Switzerland) are using dual-purpose (transportation and storage) casks. Triple-purpose (transportation and storage and disposal) casks are being developed in the FRG and Canada.

Some supporting spent fuel and high-level waste management system considerations are shown in Table S.4. Most countries plan to dispose of certain intermediate-level wastes and alpha-bearing (or transuranic [TRU]) wastes in geologic repositories. The FRG, Sweden, and Switzerland will dispose of all radioactive wastes (including low-level waste) in geologic repositories. Japan is the only country seriously evaluating the feasibility of fractionation of high-level waste and transmutation of long-lived radionuclides to non-radioactive nuclides or to short-lived radionuclides. Japan is pursuing a major research and development program on this approach in parallel with its plans for geologic disposal. Several countries have participated in studies of seabed disposal, including Canada, the FRG, Japan, the U.K., and the U.S. (as a remote option), and several countries retain interest in seabed disposal. Switzerland would prefer to dispose of its high-level waste in a foreign or multinational repository, and Belgium and Canada have also expressed interest in this concept.

GEOLOGIC WASTE REPOSITORY

Table S.5 provides summary information on the geologic disposal system planned in each country; all countries are presently planning for disposal in deep geologic repositories. The U.S. concept is the most advanced, having a preliminary design. The FRG, Sweden, and Switzerland have well-developed concepts, and Belgium and Canada have developed concepts. Of the countries discussed here, France, Japan, and the U.K. currently have the least-defined concepts, but they are following the progress of repository development in other countries. The U.K. does not plan to pursue repository development for a number of years and has carried out only preliminary conceptual studies.

Three of the eight foreign countries have determined the host rock for their repositories (Canada, granite; Belgium, clay; and the FRG, salt); the others have yet to narrow their choice from several candidates. Candidate host rocks in other countries are granite, gabbro, gneiss, schist, salt,

TABLE S.4. Supporting Spent Fuel and High-Level Waste Management System Considerations

Country	Other Wastes in Geologic Disposal	Transport Cask Type	Interest in Other Disposal Concepts	Interest in Multi-National Repository
Belgium	ILW; alpha wastes	Single-purpose	None identified	Yes
Canada	Uncertain	Triple-purpose	Yes - seabed	Yes
France	Alpha wastes	Single-purpose	Possibly seabed	No information
Fed. Republic of Germany	LLW/ILW	Dual- and triple-purpose	None identified	None identified
Japan	TBD - alpha wastes	Single-purpose	Partitioning/transmutation and seabed	None
Sweden	LLW/ILW	Single-purpose	None identified	None identified
Switzerland	LLW/ILW - Type B HLW/TRU - Type C	Dual-purpose	None	Yes
United Kingdom	ILW; some LLW	Single-purpose	Yes - seabed	None identified
United States	Defense HLW; ILW	Single-purpose	None; seabed is remote alternative	None

TBD = To be determined

TABLE S.5. Geologic Disposal Systems

Country	Status	Repository Concept	Host Rock and Water Environment	Approximate Year of Repository Startup	Package Retrievability Planned	Post-Closure Monitoring
Belgium	Conceptual	12 canisters in each borehole in floor of drifts; alternate is emplacement in drifts	Clay; dry	2030	No	No
Canada	Conceptual	1 canister of 72 assemblies in borehole in floor of drifts	Granite; saturated	2030	No	No
France	Pre-conceptual study	18 canisters in cooled boreholes or 1 canister in uncooled hole in floors of drifts	Granite (first choice), salt, clay or schist; TBD	2010	TBD	~300 yr
Fed. Republic of Germany	Reference concept	HLW stacked vertically in boreholes; SF horizontally in drifts	Salt dome; dry	2008	No	None identified
Japan	Pre-conceptual study	No information	Crystalline or sedimentary; TBD	after 2030	TBD	TBD
Sweden	Reference concept; unchanged since 1983	Single packages in boreholes in floors of drifts	Granite, gneiss or gabbro; saturated	2020	None required but possible with concept	Not required technically but TBD
Switzerland	Reference concept, may change	Single packages horizontally in drifts	Granite, anhydrite or clay, saturated	2020	No	No
United Kingdom	Deferred disposal, interim storage pursued	Preconceptual; packages in vertical boreholes or horizontal drifts	Granite preferred; TBD	after 2040	Strong public support for retrievability	TBD
United States	Preliminary design	Single packages in boreholes in floors (or multiple packages in walls) of drifts	Tuff; unsaturated	2010	Retrievability for 50 years after start of emplacement	None planned

TBD = To be determined

anhydrite, or clay. The U.S. is the only country presently considering tuff and disposal above the water table.

The most popular repository arrangement is for emplacement of waste containers in boreholes in the floors of mined tunnels (drifts). Other concepts include emplacement in boreholes in the tunnel walls or directly in the mined tunnels.

Most of the countries plan to begin repository operations between the years 2010 and 2040. The FRG and the U.S. are planning for the nearest-term startups, about 2010. None of the countries, other than the U.S., is currently planning to include provisions for retrievability of the waste package after emplacement in the repository. Post-closure repository monitoring requirements are yet to be determined in most cases but are being considered in some countries.

Information on the development and safety of the repository is summarized in Table S.6. Most countries will perform detailed site characterization for confirmation of site suitability of one site only, unless that site proves unacceptable. Japan and Sweden plan to characterize at least two sites prior to making the final site selection. Underground research laboratories (URLs) exist or are planned in all countries except for the U.K., which is presently deferring repository development. These laboratories will provide for detailed research and development of earth science technologies and measurement techniques, which are necessary for repository development.

The barriers for prevention of radionuclide escape from the repository include the repository host rock as a major barrier in all countries except Sweden (where it is considered to be a back-up to the engineered barriers). The disposal concept in Sweden is based on a long-life (up to about 1,000,000 years), thick-walled copper canister for waste isolation and radionuclide retention. The Canadian titanium canister is also considered to be a major engineered barrier. The borosilicate glass waste form is considered a major barrier in all countries planning for reprocessing rather than disposal of spent fuel.

TABLE S.6. Repository Development and Safety

Country	No. of Sites to be Characterized	Underground Research Laboratory (URL)	Major Safety Barriers	Proving Safety	Major Use of Natural Analogues	Peer or Foreign Review of Progress
Belgium	1	URL exists at the site	Host rock	Deterministic and stochastic	No	Yes, indirectly
Canada	Undecided	URL exists	Host rock, canister	Deterministic and stochastic	Yes	Yes, indirectly
France	1	URL exists; ISVL to be constructed at the site	Host rock, waste form	Deterministic	Some use	National advisors; no foreign reviews
Fed. Republic of Germany	1 (ongoing)	URL exists	Waste form, geologic formation	Deterministic; conservative	Some use	Some, through licensing process
Japan	2	URL planned	Host rock and engineered barriers (waste form, canister)	Stochastic	Some use	No information
Sweden	2-3	URL planned near the site	Canister and matrix	Conservative; deterministic, some stochastic	Some use	National peers; IAEA and NEA
Switzerland	1	URL exists (not at site); URL to be constructed at the site	Host rock, bentonite overpack, waste form	Deterministic; conservative	Some use	None identified
United Kingdom	TBD	None	Waste form, others TBD	Conservative; deterministic, and stochastic; time-dependent simulation modeling	Some use	None to date
United States	1	URLs have been used; URL planned at the proposed repository site	Canister, host rock	Stochastic and deterministic; detailed; extensive model validation	Some use	Several national peer groups; no foreign reviews

TBD To be determined

The methods being used to evaluate repository safety in most foreign countries are both deterministic and stochastic. Most countries are participating to some degree in natural analogue studies. Sweden uses extensive international and national peer review of its repository program, while most other countries receive international peer reviews indirectly through participation in multinational activities. The countries with the more developed programs generally have specific processes for internal peer review.

Table S.7 summarizes the overall spent fuel and high-level waste package systems planned for geologic disposal. All countries that are reprocessing spent fuel are planning to vitrify their high-level waste as monolithic borosilicate glass. Of those countries planning for direct disposal of spent fuel, Canada and Sweden are planning to incorporate their spent fuel within a matrix material of sand and copper or lead, respectively; the FRG and U.S. are not planning to use matrix materials. Belgium, France, Japan, the FRG, Switzerland, and the U.K. all plan to use the French-type canister for their borosilicate glass (stainless steel, with 5 mm wall thickness). The stainless steel canister planned for use in the U.S. has a slightly thicker wall (1 cm). Thick-walled canister concepts include the Swedish 10-cm-thick copper canister and the FRG steel Pollux cask for spent fuel disposal. The FRG is planning to use a triple-purpose package for spent fuel disposal, which includes a disposable transportation overpack. For solidified high-level waste, Switzerland and the U.S. are planning to use overpacks, and the U.K. will consider overpacks in the future. Most of the countries, although not the U.S., are also planning to utilize a buffer material (typically a clay such as bentonite or cement) surrounding the disposal package in the repository.

COMPARISONS OF FOREIGN PLANS WITH U.S. PLANS

A summary of some of the comparisons between foreign and U.S. plans is given in Table S.8. The FRG and the U.S. have the most developed repository concepts and are planning the earliest repository startup dates (about 2010). All other countries are planning to start up their repositories at least one decade later.

TABLE S.7. Spent Fuel and High-Level Waste Package Systems

Country	Status	Waste Form	Matrix	Canisters	Disposal Container	Buffer (or packing) Material
Belgium	Reference concept defined by French canister	Borosilicate glass	None	5 mm wall Z15 CN 24-13 SS (French canister)	None	Possibly clay or cement
Canada	Reference concept	Intact spent fuel	Sand	4.76 mm wall titanium	None	Clay-sand mixture
France	Canisters in use; other components are conceptual	Borosilicate glass	None	5 mm wall cylinder Z15 CN 24-13 SS	TBD	Possibly clay
Fed. Republic of Germany	Reference concept well-developed	Borosilicate glass; some spent fuel (may be intact, consolidated or chopped)	None	French design for HLW (SS); Pollux for spent fuel (steel and hastelloy coating)	None for HLW; Pollux and overpack for spent fuel	None
Japan	Conceptual	Borosilicate glass	None	304L SS for Japan vitrification; French-type canister for French or U.K. vitrification	TBD	Possibly clay or cement
Sweden	Reference concept; others considered	Intact spent fuel	Lead or copper	10 cm-thick copper; provides long-life up to 1,000,000 yr	None	Pressed bentonite
Switzerland	Reference concept	Borosilicate glass	None	5 mm wall Z15 CN 24-13 SS (French and U.K. canisters)	Yes, cast steel	Compacted bentonite
United Kingdom	HLW glass in French-type canister; other components TBD	Borosilicate glass	None	French-type canister	Overpack of thin titanium or thick cast iron or steel will be considered	Bentonite or cement backfill
United States	Preliminary design for site characterization; others considered	Intact or consolidated spent fuel; some borosilicate glass	None	1-cm-thick SS for HLW; no "canister" for spent fuel (container only)	1-cm-thick SS for spent fuel; 1-cm-thick SS for HLW (both may change)	None presently; considering clay

TBD To be determined
SS = Stainless steel

TABLE S.8. Comparison of Major Waste Management Parameters

Country	Approx. Repository Startup Time	Repository Host Rock and Environment	Waste Management Organization	Regulation and Performance Requirements	Approach to Proving Long-Term Safety	Waste Package and Engineered Barriers
Belgium	2030	Dry clay; below groundwater	Government (ONDRAF)	General requirements only to date; specific requirements TBD	Deterministic and stochastic	Glass in SS canister; clay or cement buffer
Canada	2030	Granite; below groundwater	Government (AECL)	General only	Deterministic and stochastic	Spent fuel in titanium; canister filled with sand; clay-sand buffer
France	2010	Granite (first choice), clay, salt or schist	Government (ANDRA)	General requirements only; specific requirements under development	Deterministic	Glass in SS canister, possible clay buffer
Fed. Republic of Germany	2008	Salt dome; dry	Government (BfS) & Industry	General for total system	Conservative; deterministic	Glass in SS canister; spent fuel in steel/ hastelloy and overpack;
Japan	After 2030	Sedimentary or crystalline	Government (STA)	Not yet established	Stochastic	Glass in SS canister, possible clay or cement buffer
Sweden	2020	Granite, gneiss or gabbro; saturated	Industry (SKB)	General for total system	Conservative; deterministic; some stochastic	~1,000,000 yr copper canister; bentonite buffer
Switzerland	2020	Granite, anhydrite, or clay, saturated	Industry (NAGRA)	Total system objectives	Deterministic; conservative	Glass in SS canister, cast steel overpack, bentonite buffer
United Kingdom	2040	Granite preferred; TBD	Government - policy; Industry - implementation	General only	Deterministic and stochastic codes; time-dependent simulation modeling; conservative	Glass in French-type canister; other details TBD
United States	2010	Tuff in mountain; unsaturated	Government (DOE)	Highly prescriptive for system and components	Stochastic and deterministic; detailed; extensive model evaluation	Spent fuel in SS or other corrosion-resistant container; Glass in SS canister and SS or other corrosion-resistant overpack container

TBD = To be determined
SS = Stainless steel

The preferred host rocks for most other countries are granite (or other crystalline rock), salt or clay, and all are below the water table. The site presently being considered for the U.S. repository is unique, in that it is located in welded tuff in a mountain in an unsaturated zone above the water table.

The primary engineered barrier in the reference U.S. design is planned to be a container into which the spent fuel or the canister of vitrified high-level waste will be placed. The U.S. is also planning to use stainless steel canisters for the small quantity of vitrified civilian high-level waste, similar to the canister planned in most other countries. However, use of other barriers, such as matrix materials and overpacks for spent fuel, or buffers, is not presently planned in the U.S. Matrix materials will be used in the containers of spent fuel in two other countries, and buffer materials around the emplaced containers are planned in most other countries.

The U.S. has the most prescriptive regulations and detailed repository performance requirements. These require both deterministic and stochastic modeling approaches to evaluate the performance of the major repository components as well as the total repository system.

1.0 INTRODUCTION

Radioactive waste management is a sensitive issue throughout the world, particularly in countries with nuclear-generated power. Management of spent fuel and high-level wastes, especially the disposal of these wastes, is particularly sensitive. No country is yet disposing of these wastes, but most countries with nuclear power, including the United States (U.S.), are pursuing disposal in deep geologic formations.

Policies, strategies, institutional structures, public participation, and technical considerations in radioactive waste management in one or more countries can affect those of other countries. An international consensus has developed within the technical waste management community that deep geologic disposal is the preferred concept for disposal of spent fuel and high-level radioactive wastes. In addition, a general consensus on the approach to managing these wastes is developing. However, differences in specific aspects can have a significant influence upon the public response, timing, costs, and specific design of a waste management system.

Trends in foreign plans and activities have the potential to impact the U.S. civilian spent fuel and high-level waste (HLW) management program. Such impacts could result from national public and/or international community pressures.

This report describes the major parameters for management of spent nuclear fuel and HLW in selected foreign countries and the U.S as of December 1989. A comparison of the major parameters with those in the U.S. civilian spent fuel and high-level waste management program is also presented.

This study is intended to help provide a basis for the U.S. Department of Energy's Office of Civilian Radioactive Waste Management (OCRWM) to compare foreign practices with those of the U.S. to identify possible impacts that trends in foreign programs may have on the U.S. program. The study will also be used by OCRWM as a basis to improve or support the U.S. selection of major program parameters.

The major parameters are discussed under the following general categories of information:

- primary legislative bases and institutional/organizational approach for managing and funding the waste management program
- overall waste management system strategy, description and plans, including interim storage, reprocessing, transportation, and disposal
- description, status and rationale for spent fuel and HLW storage and transportation subsystem activities
- description, status and rationale for spent fuel and HLW disposal subsystem activities.

Included in the study are non-Communist countries that have been relatively active in developing a spent fuel and/or HLW management program. These countries are Belgium, Canada, France, the Federal Republic of Germany (FRG), Japan, Sweden, Switzerland, the United Kingdom (U.K.), and the U.S.

Much of the information presented was supplied by the respective countries to the Organization for Economic Cooperation and Development's Nuclear Energy Agency (OECD/NEA) for use by its Radioactive Waste Management Committee. Other information was obtained from the files of Pacific Northwest Laboratory's International Program Support Office. There is some inconsistency in the level of information included among the various countries because of the variations in the level of information available. Where no information on a particular subject is available, that fact is so noted.

The report is divided into four sections: 1) an abstract; 2) a summary, with summary tables of the major parameters; 3) this introduction; and 4) specific information for the foreign countries, with a subsection for each foreign country and the U.S. References are given at the end of the text for each country. The headings and subheadings for each country are identical to assure that information on the same subjects is provided.

2.0 BELGIUM

2.1 LEGISLATIVE/INSTITUTIONAL BASES

2.1.1 Nuclear Power Policy (Detilleux et al. 1989; Leigh 1989; Schneider et al. 1988; NEI 1989)

Belgium, with a dense population (10,000,000) and a highly industrialized economy, relies heavily on nuclear energy; its seven pressurized water reactors (PWRs) produced 65% of its electric power needs in 1988. Nuclear power growth has slowed recently, with cancellation of the proposed DOEL 5 PWR reactor and withdrawal from participation in European breeder reactor programs. Belgium's fragile ruling party coalition is dependent upon support by the anti-nuclear Flemish Socialists, leading to a decision by the Belgian Cabinet to opt exclusively for non-nuclear energy sources through the 1990s (NUKEM 1989).

Belgium has a well-rounded capability in the nuclear fuel cycle. In addition to operating its power reactors, Belgium is a shareholder in the Eurodif enrichment operations, fabricates light water reactor (LWR) and mixed (plutonium-uranium) oxide (MOX) nuclear fuel, has its spent fuel reprocessed in France and the United Kingdom (U.K.), and supports considerable nuclear research at the Nuclear Energy Research Center (CEN/SCK) at Mol, Belgium.

2.1.2 Major Legislation (IEAL 1985, 1987; OECD/NEA 1988b)

Under the Act of March 29, 1958, the federal government was given authority to impose conditions to protect the public and environment from ionizing radiation. The government was also authorized to regulate the disposal of radioactive materials and establish levies to cover the costs of implementation. The Act also gave the government authority to undertake measures to protect the public and environment from unforeseen events and to prescribe corrective action in the event of accidental contamination.

This authority was implemented by the Royal Decree of February 28, 1963, and subsequent addenda. The Royal Decree established general regulations for the protection of the public and workers from damage by ionizing radiation, for transporting radioactive materials, for licensing nuclear installations, and for releasing radioactive materials. In 1981, a department was established within the Ministry of Employment and Labor (Service for the Technical Safety of Nuclear Installations) and another within the Ministry of Public Health (Service for Protection Against Ionizing Radiation). The former department is concerned with worker safety and health, the latter, with licensing of facilities and monitoring of compliance with regulations.

Also in 1981, an independent federal agency was established under the Ministry of Economic Affairs to manage all radioactive wastes. This agency, called the National Organization for Radioactive Waste and Fissile Materials (ONDRAF/NIRAS), is financed by the power utilities and other producers of radioactive waste.

2.1.3 Linkage Between Nuclear Power and Waste Management

There are no legislative acts linking the development or operation of nuclear power to assurance that radioactive wastes can be managed satisfactorily.

2.1.4 Policy on Spent Fuel and Waste Management (OECD/NEA 1988b)

Presently, Belgian utilities are sending spent fuel to France and the U.K. for reprocessing. The high-level waste (HLW) will be returned from reprocessing and will be interim stored, then disposed of along with transuranic (TRU) waste in a geologic repository being developed at the Mol site. Alternative concepts for disposing of non-HLW are under study. To provide the long-term storage required for all wastes before disposal facilities become available, engineered storage facilities will be built at a central location.

2.1.5 Organization/Responsibilities for Waste Management (Leigh 1989; Schneider et al. 1988; OECD/NEA 1988b; IEAL 1987)

ONDRAF/NIRAS reports to the federal Ministry of Economic Affairs and is the principal organization responsible for definition and implementation of policies for managing radioactive wastes and fissile materials. This responsibility includes enforcement of regulations and conduct of research and development (R&D). The technical tasks are carried out either by ONDRAF staff or by subcontractors. ONDRAF/NIRAS is also authorized to create or assume interest in subsidiary companies. One of these is Belgoprocess, a company in charge of the waste management operations at the Mol site and 100% owned by ONDRAF/NIRAS.

The Nuclear Energy Research Center (CEN/SCK) at Mol is a federal organization reporting to the Ministry of Economic Affairs and has the responsibility for performing basic and applied R&D in the nuclear energy field. CEN/SCK also had responsibility for the waste management operations and related R&D at the Mol facility until the recent (1989) transfer of all waste management responsibilities to ONDRAF/NIRAS.

The Belgian Nuclear Fuel Company, SYNATOM, manages the supply of enriched uranium for Belgian utilities and the reprocessing of spent fuel in foreign facilities. It is jointly owned by the Belgian utilities and the government through the Ministry of Economic Affairs.

Engineering services related to radioactive waste management are provided by Belgonucleaire, an organization jointly owned by the utilities and the government through the Ministry of Economic Affairs.

Transportation of radioactive materials is the responsibility of TRANSNUBEL, a subsidiary of Belgonucleaire and the French firm, Transnucleaire S.A. of Paris. TRANSNUBEL also represents Britain's Nuclear Transport, Ltd., in Belgium.

The Radiation Protection Service, which forms part of the Public Health Department in the federal Ministry of Public Health, prepares and enforces regulations for the protection of the public and workers against ionizing radiation. Applications for licenses to handle radioactive materials are reviewed by the Radiation Protection Service, as well as the Technical Safety Service for Nuclear Installations within the Ministry of Employment and Labor.

The following advisory bodies operate under the auspices of the Belgian government regarding nuclear matters.

- The National Council for Science Policy makes recommendations on behalf of the scientific research and academic community.

- The Interministerial Commission for Nuclear Safety and State Security in the Nuclear Field serves as a coordinating vehicle for nuclear activities of the departments within the Ministries of Employment and Labor, Public Health, Justice, Foreign Relations, Interior and National Defense.
- The Higher Council for Public Health is an advisory body under the Minister of Public Health that submits its opinions regarding the protection of the public from ionizing radiation to the appropriate authorities.

2.1.6 Repository Licensing Process, Regulation Status, Major Regulations, Safety Bases

ONDRAF/NIRAS is responsible for developing the criteria for handling, treating, transporting, and disposing of all radioactive wastes. These criteria must be consistent with the public health criteria established by the Ministry of Public Health (IEAL 1987).

For regulatory and administrative purposes, Belgian nuclear facilities are classified into four categories, depending on the nature of the installation, the characteristics of the operating equipment, the presence of nuclear materials, and the quantity and radiotoxicity of the isotopes in the material.

Facilities handling HLW are expected to be considered Class I, along with power reactors (IEAL 1987). The licensing procedure for Class I facilities in Belgium requires that a general review and opinion be obtained from local and regional authorities before conducting any technical reviews. Licensing requests are filed with the governor of the local province. After local review, the request is passed on to the Provincial Council for its opinion. It is then passed on to a federal ministerial-level Special Commission for a detailed technical review. This body issues a provisional opinion, together with the proposed rules and conditions for operation. A license is granted in the form of a royal decree signed by both the Federal Minister of Employment and Labor and the Federal Minister for Health and Environment. Licenses may be granted for a fixed or an unlimited period (IEAL 1987).

The final approval for operation and the continuing regulation of Class I facilities is in the hands of experts approved by the ministries responsible for licensing. The experts, selected from non-profit control and supervisory associations, assure that the installations and operations conform to the description given in the safety reports prepared by the applicant or operator (IEAL 1987).

The radiation exposure limits for waste storage and disposal facilities in Belgium are identical to those required of all Belgian nuclear facilities (IEAL 1987). These limits, specified in the Royal Decree of February 28, 1963, and subsequent amendments, are as follows:

- radiation workers: 5 rem/yr
- dose rate at building wall: 2.5 rem/yr
- population limit: 500 mrem/yr from all nuclear facilities combined.

A specific time period for application of these radiation protection limits to a final repository has not been specified but will be before the licensing procedure for disposal begins (IEAL 1987). (Time periods as long as 1,000,000 years have been considered [CEC 1988].)

No official radioactive waste classification has been adopted in Belgium. However in practice, a distinction is made between "non-geological" and "geological" waste. Wastes destined for a deep geologic repository, the "geological" wastes, are those that arise essentially from fuel reprocessing and MOX fuel fabrication. These include vitrified HLW, spent fuel cladding hulls and spent fuel hardware, bituminized sludges, and other alpha-bearing (TRU) wastes (IEAL 1987).

"Geological wastes" are categorized in terms of both surface dose rate and heat release. They are considered to be low-, intermediate-, or high-level wastes if the surface dose rates are less than 1 rad/h, less than 1000 rad/h, and greater than 1000 rad/h, respectively. Low- and high-heat release waste packages are separated at the level of 5 watts per package (IEAL 1987).

2.1.7 Roles of the Public, Local Organizations, Multinational Organizations

The Belgian public approval process is characterized by a high degree of centralism. The ultimate decision-making authority is at the national level, although a full range of opinions is solicited at the local and regional levels. These opinions are included with the siting request as it passes from the local to national level, as described previously. In addition, Belgium uses close contact with the media to give the central decision-making process wide public exposure (IEAL 1987).

Belgium is a member of the International Atomic Energy Agency (IAEA), the Organization for Economic Cooperation and Development Nuclear Energy Agency (OECD/NEA) and the Commission of European Communities (CEC) and actively participates in their waste management programs. It has a cost-sharing contract with the CEC for conducting the R&D work at the underground research laboratory (URL) located at Mol. It also has bilateral agreements with the French Atomic Energy Commission (CEA) and Japan's Power Reactor and Nuclear Fuel Development Corporation (PNC) on specific aspects of geologic disposal (OECD/NEA 1988b).

Belgium has not requested multi-national peer review of its HLW management plans or activities.

2.2 OVERALL WASTE MANAGEMENT SYSTEM (OECD/NEA 1988b; Detilleux et al. 1989)

The overall waste management plan for Belgium is to store all radioactive wastes until disposal in central engineered storage facilities on the Mol site. The expected date for initiating geologic disposal of HLW and TRU waste is about 2030, based on the assumption that the clay formation under the Mol site will be acceptable for a geologic repository. Alternatives for disposal of low-level waste (LLW) are under study. The total volume of wastes to be disposed of in the geologic repository is presently estimated to be 30,000 cubic meters (5000 of HLW, including cladding hulls from spent fuel, and 25,000 of intermediate-level waste [ILW] and alpha-bearing [TRU] wastes). The total volume of LLW to be disposed of by other means is presently estimated to be 150,000 cubic meters.

Presently, facilities exist on the Mol site for storage of LLW, ILW, and HLW. These facilities are being expanded as needed.

2.2.1 Reprocessing (OECD/NEA 1988b)

Currently, spent fuel assemblies are stored temporarily at the power station sites prior to being shipped to France or the U.K. for reprocessing. The reprocessing contracts stipulate that all waste from the reprocessing of spent fuel will be returned to Belgium after conditioning. Reprocessing continues to be the basic option for the back end of the nuclear fuel cycle, based upon the premise that the recovered plutonium will eventually be recycled for power generation.

2.2.2 Interim Storage Before and After Reprocessing (OECD/NEA 1988b)

Spent fuel assemblies are stored in fuel pools at the reactor sites until shipment to France or the U.K. for reprocessing. Construction of a new storage facility was started in 1987 for vitrified HLW to be returned from the reprocessor beginning in the early 1990s. Its completion is expected in 1993.

Liquid HLW resulting from the past reprocessing activities of Eurochemic on the Mol site are being vitrified in the PAMELA facility and stored in air-cooled storage pits on the Eurochemic site, now operated by Belgoprocess.

2.2.3 Geologic Repository (OECD/NEA 1988b; Bonne and Detilleux 1989; Manfroy et al. 1985)

Investigation of the geology under the Mol site has been underway since 1974, including construction of a URL in the clay media. Site confirmation studies will continue for several years, with construction of the repository to begin in 2025 and operations in 2030.

2.2.4 Other System Considerations

Belgium participated in the 1987 OECD/NEA study assessing the use of international waste repositories, which concluded that there are no insurmountable safety, technical, economical, or institutional reasons why such a concept could not be seriously considered (RWMC 1987). It is thus assumed that Belgium would be interested in an international repository. Belgium also participated in the OECD/NEA's study on the feasibility of sub-seabed disposal of HLW. An interim report on the first eleven years of research on this topic concluded that sub-seabed disposal could be a very safe option in radiological terms (OECD/NEA 1988a).

Nothing is known regarding Belgium's interest in transmutation or partitioning as a means of managing HLW.

2.3 WASTE STORAGE AND TRANSPORTATION

2.3.1 Spent Fuel Storage (Harmon and Johnson 1984)

The water pools, along with expansions, have been providing adequate spent fuel storage capacity at the nuclear power station sites.

2.3.2 High-Level Waste Storage (OECD/NEA 1988b)

Intermediate- and high-level reprocessing wastes, which will be returned from France and the U.K., will be stored in a facility to be built by ONDRAF/NIRAS on the former Eurochemic site at Mol. The project, which started in 1987, will be ready for operation in 1993 and will accommodate all the waste from existing reprocessing contracts. It can be expanded as new contracts are concluded. The facility incorporates the following features:

- a hall for receiving the incoming waste
- a shielded cell for unloading the shipping casks and handling the waste packages
- a transfer hall and storage cell for very radioactive HLW (80 cubic meters)
- a transfer and storage cell for moderately radioactive HLW and ILW (460 cubic meters).

Further information on the storage facilities could not be found.

Liquid HLW resulting from the former pilot-scale reprocessing activities of Eurochemic are currently being vitrified in the PAMELA plant, and the vitrified wastes are stored in air-cooled pits in a bunker building erected on the Belgoprocess site at Mol.

2.3.3 Transportation (OECD/NEA 1988b)

Truck, rail, and ship transport are used for moving spent fuel to reprocessing plants in France and the U.K. and to transport other wastes between Belgium and other countries. Various types of casks are used for shipment of spent fuel. Trucks are used for transport over short distances, and rail is used for transport to the French reprocessing plant at La Hague. Truck and rail transport are used to ship other wastes between Belgium and other countries. Ships are used for transport of research spent fuels abroad.

For off-site shipping, casks are provided and shipping is done by TRANSNUBEL under contract to SYNATOM. In general, the IAEA regulations for the safe transport of radioactive materials are followed, but the transport license may have special requirements, such as escorts for certain types of shipments. The nuclear industry is free to select its transport system, provided that relevant national and international regulations are satisfied.

2.4 GEOLOGIC WASTE REPOSITORY

2.4.1 Safety Requirements and Approach

All nuclear activities are subject to the Royal Decree of February 28, 1963, related to the "General regulation for the protection of the population and the workers against the hazards of ionizing radiation." This decree, updated several times, does not include any specific performance requirements for waste disposal. It is the responsibility of ONDRAF/NIRAS to propose new regulations in this area to the competent authorities in due time. However, the general licensing procedures as described in the decree (see Section 2.1.6) are applicable to the waste repositories as well as other nuclear facilities (OECD/NEA 1988b).

Preliminary safety analyses of a geologic waste repository in clay have been made by CEN/SCK (Dejonghe et al. 1986), and a general performance assessment of a geologic repository in clay has been conducted as part of the CEC's Performance Assessment of Geologic Isolation System (PAGIS) effort (CEC 1988). These analyses indicate that such a repository can provide adequate retention of the radionuclides if properly sited, designed, and constructed.

ONDRAF/NIRAS uses a quality assurance program in which standards and regulations existing at the national level are strengthened by adding specific requirements for the application under consideration. Implementation and oversight of quality assurance on a specific project is delegated to the project organization (OECD/NEA 1988b).

2.4.2 Siting (Bonne and Detilleux 1985, 1989; Manfroy et al. 1985; OECD/NEA 1988b)

In 1975 when the first nuclear power plants were put into operation in Belgium, the Nuclear Research Center began an inventory of deep geologic formations in Belgium that might be used for disposal of radioactive waste. Based upon selection factors established in agreement with the National Geologic Service and considering those suggested by working groups of the IAEA and the CEC, a site in the tertiary clay formation (called Boom clay) in the Mol region was selected. The Boom clay is part of the Oligocene clay formation covering northern Europe and extending into the North Sea. Data available at the time indicated the clay layer at the site had a minimum thickness of 100 meters and lay at an average depth of 230 meters. The layers above and below the clay were water-bearing sand.

During the ensuing five years, deep drilling, borehole logging, aerial photography, and seismic measurements were used to characterize the site. At the same time, concepts of the facility design were developed and an initial safety assessment was completed. This effort was followed by construction of an underground laboratory at the site. Research and development is underway at the site and will continue for several years to develop design features and characterize the clay media.

2.4.3 Design Concept(s) (Bonne and Detilleux 1985; OECD/NEA 1988b; INWAC 1989; Schneider et al. 1989; Heremans et al. 1989)

Current design bases for the repository conceptual designs and evaluation studies are as follows:

- The waste to be disposed of in the repository originates from an installed nuclear power generation of 5600 MW(e) over the next 30 years.

- The waste to be produced in this program amounts to:
  - 850 cubic meters of vitrified HLW
  - 2200 cubic meters of spent fuel hulls fixed in cement
  - 6560 cubic meters of ILW in cement
  - 5200 cubic meters of ILW in bitumen.
- The age of the waste at disposal is 50 years.
- Maximum allowable temperatures are 500°C for the vitrified waste and 100°C for the clay.

Evaluations of various design concepts for the repository layout are underway. Design studies for a repository in the horizontal formation have resulted in two generic concepts using parallel disposal drifts in the mid-plane of the formation. Two or more service galleries, to be sealed by the clay after completion of disposal operations, will provide access from the entrance shafts to a series of parallel disposal galleries located perpendicular to the connecting service galleries.

In the "Radial Disposal Concept," the bottom half of a disposal gallery would be filled with canisters of ILW fixed in cement. After backfilling around the ILW, HLW canisters with overpacks would be placed on the ILW. The top half of the disposal drift would then be backfilled. Three of the disposal galleries would be dedicated to ILW fixed in bitumen, and one gallery would be dedicated to cladding hulls. This concept is considered "non-retrievable."

In the "Axial Disposal Concept," waste packages would be placed in holes drilled in the disposal drift floor at an angle of 45 degrees from the horizontal. The holes for HLW would be 20 meters apart, and no liner between the canister and hole wall is planned, though the space may be filled with sand. Each hole would contain twelve canisters. This concept is considered "retrievable."

The overall area required by the repository is governed by the amount of heat-generating waste (about 850 cubic meters of vitrified HLW) and the maximum thermal loading that can be accepted for that site. Present assessments indicate thermal loadings up to 2.5 watts per square meter are acceptable. The maximum temperature of the host rock and rock/canister interface is 100°C to prevent degradation of the clay. The maximum estimated lithostatic pressure on the waste package is 50 atmospheres.

2.4.4 Retrievability and Monitoring (IEAL 1987; Schneider et al. 1989)

No retrievability requirements have been established for waste placed in a repository. Although it is not expected that any such requirements will be established, a demonstration phase of operation with actual waste and several years of observation will precede routine operation of the repository. Likewise, no monitoring provisions are planned.

2.4.5 Waste Package System (Schneider et al. 1989)

The waste package design under consideration is conceptual only and includes the vitrified HLW and a French-type stainless steel canister, with no buffer material other than a clay backfill in the drifts. Sand backfill of disposal boreholes is being considered. No plans for a disposal container (overpack) or a borehole sleeve have been developed. The French-type canister is a stainless steel cylinder measuring 0.43 meters in diameter by 1.335 meters in height, including the filling neck and nozzle. The canister has a reverse head on the bottom, which allows for a stacking height of 1.285 meters; the gross canister internal volume is 170 liters; the net canister volume, filled, is 150 liters; and the total filled weight is 480 kilograms.

2.4.6 Research and Development (Bonne and Manfroy 1989; Manfroy et al. 1985)

Research and development has been underway for several years and is continuing. Studies not requiring in situ conditions are underway in surface laboratories to characterize the clay and waste forms and to assess the repository performance.

In the underground research facility near Mol, an ambitious study program is being carried out on clay-related phenomena such as corrosion, migration, heat dissipation, local hydrology, etc. These studies are in addition to those conducted to optimize placement and handling in the repository, such as constructing, handling, backfilling, and sealing methods.

2.4.7 Approach to Proving the Safety of the Repository (OECD/NEA 1988b; IEAL 1987)

The safety of the repository will have been "proven" before an operating license is granted. (See Section 2.1.6.)

The final approval will come in the form of a "Royal Decree" issued by the government. This decree will be prepared by the Ministry of Economic Affairs and have the approval of other involved ministries. In preparing the decree, the Ministry will rely on a group of experts, selected from non-profit control and supervisory associations, who will verify that the installation and operation conform to the description given in the safety analyses prepared by the applicant. In addition, the Ministry (and experts) will have access to the extensive data base prepared in the years of repository development and the results of local, regional, national, and international reviews of the proposed project.

Since the mid-1970s Belgian activities in repository research have been centered on the Mol site, a clay formation in the northern part of Belgium near the town of Boom, from which the clay derives its name. A preliminary and generic total system performance assessment suggested that a repository in undisturbed clay was likely to be very safe for very long time periods. Another generic case in a previously disturbed formation was also evaluated (i.e., a faulting scenario in which a preferential upward path of groundwater to a well was created). A Monte Carlo simulation of the possible consequences of this scenario resulted in estimated radiation doses to humans which ranged over five orders of magnitude. It was concluded that site-specific evaluations were needed. Performance assessment methodology development is being pursued as an integral part of the in situ testing/demonstration project at the underground research facility at Mol.

Belgium recently completed another comprehensive performance assessment, including both deterministic and probabilistic calculations, as part of the CEC's PAGIS exercise (CEC 1988). Calculations were carried out for millions of years. Perturbations from altered evolution scenarios were included. The perturbing scenarios, selected by expert opinion, included another faulting case and a drastic climatic change that involved a long period of extreme drought. The drought scenario proved to be the worst in terms of dose increases, but those increases were on the order of four times the undisturbed case doses and did not exceed 0.1 mrem/yr for individuals drinking water from a well at the clay/aquifer boundary.

Because of the sensitivity of performance assessment results to properties of the host clay rock, the Belgian plan is to continue to define and refine the properties of the Boom clay formation. Radionuclide/host rock interaction tests, percolation tests, and diffusion experiments are ongoing.

The next phase of the performance assessment for the potential Belgian waste repository in clay is an important part of the Belgian research program. In the new performance assessment, more attention will be given to human intrusion scenarios and to scenarios in which the engineered barrier will malfunction. The computer code used for performance assessment is LISA, which was developed at the CEC's research center at Ispra, Italy.

2.4.8 Peer Review Activities (OECD/NEA 1988b)

Belgium has not specifically sought foreign peer reviews of its programs. However, it obtains defacto peer reviews through open participation in the activities of the OECD/NEA, the IAEA, and the CEC. Belgium's CEN and ONDRAF/NIRAS have implemented cost-sharing contracts with the CEC or various R&D topics, such as waste characterization and safety analyses. Bilateral research agreements on specific aspects of geologic disposal exist between the CEN and both the French CEA and Japan's PNC. The program also receives extensive review through a process described in Section 2.1.6.

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3.0 CANADA

3.1 LEGISLATIVE/INSTITUTIONAL BASES

Canada has an active nuclear power program supported by the public and the major political parties. The program is built around the production of uranium (Canada is a major supplier) and the use and export of the Canadian deuterium uranium (CANDU) reactor system. Uranium mining is dominated by private firms, while electricity is produced by provincial utilities, the largest being Ontario Hydro. Development of nuclear technology, including that for radioactive waste management, is the responsibility of the quasi-commercial federal agency, Atomic Energy of Canada Limited (AECL). The industry is regulated by the federal organization, Atomic Energy Control Board (AECB). Public input to the nuclear program is obtained through the ballot box and through a federally-mandated public hearing process. Canada cooperates extensively with foreign entities through participation in the IAEA, OECD/NEA, and CEC and through bilateral and multilateral agreements with other countries.

3.1.1 Nuclear Power Policy

Beginning in 1962, Canada developed significant electricity generation capacity using the natural-uranium, heavy-water CANDU reactor system. Canada currently operates 18 heavy-water reactors providing about 20% of its electricity needs, with an additional four units scheduled to go on line between 1989 and 1992. Until recently, the nuclear option has enjoyed strong public support, and export of the CANDU system has been encouraged. Because of disappointing acceptance of the CANDU system in foreign markets, government support has waned, and efforts are underway to shift the development and marketing burden from the government to private sources. Despite this pressure, both the federal and provincial governments (Ontario contains the bulk of nuclear activities) want to maintain, and expand if possible, a viable nuclear program (Leigh 1989; Nucleonics Week 1989a, 1989b; Nuclear Waste News 1988c).

3.1.2 <u>Major Legislation</u>

Nuclear activities in Canada are guided by the Atomic Energy Control Act of 1946 and subsequent amendments. These activities are controlled through the federal Ministry of Energy, Mines, and Resources. Canada has no nuclear weapons program. There is no national legislation specific to nuclear waste management (IEAL 1987).

3.1.3 <u>Linkage Between Nuclear Power and Waste Management</u>

Canada has no legal or regulatory stipulation that makes reactor licensing contingent upon having a radioactive waste disposal plan. Ontario's Royal Commission on Electric Power Planning did recommend to the Ontario government in 1980 that if R&D on HLW disposal does not make sufficient progress by 1990, a moratorium should be declared on building additional nuclear power plants in the province; the recommendation was not clear as to the criteria for ascertaining whether progress had in fact been sufficient (IEAL 1987). In 1987, Manitoba's provincial legislature passed an act banning disposal of radioactive waste in the province (<u>Nuclear Waste News</u> 1987).

In early 1988, the Canadian government was asked to respond to a House of Commons committee report calling for a moratorium on construction of nuclear power plants in Canada until the people of Canada have agreed on an acceptable solution for the disposal of high-level radioactive waste. The Canadian Energy Minister rejected this recommendation but agreed to submit the entire question of HLW management to a panel, referred to as the Environmental Assessment and Review Panel (EARP).

To facilitate evaluation of the scientific and technical matters, a subgroup of this panel, called the Scientific Review Group, was established. The EARP panel will examine a broad range of topics, including social, economic and environmental implications of long-term HLW management. The panel's work will guide AECL in determining what to include in its Concept Assessment

Documentation, expected to go to the federal government in early 1991[a] (IEAL 1987; Nuclear Waste News 1988a, 1988b).

3.1.4 Policy on Spent Fuel and Waste Management

While no definitive policy decision has been made on reprocessing, there are no plans to reprocess, and Canada has no significant reprocessing experience. Treatment followed by long-term interim storage of wastes, including spent fuel, is planned until disposal means become available (OECD/NEA 1988b).

A generic research and development program, the Nuclear Fuel Waste Management Program (NFWMP), was initiated in 1978 to assess AECL's reference concept of deep geologic disposal of nuclear fuel waste in a disposal vault 500 to 1000 meters deep, in plutonic rock of the Canadian Precambrian Shield. The findings and recommendations of the NFWMP will be reviewed in depth by the AECB and other regulatory and scientific bodies and the general public in a concept assessment process expected to take place between 1991 and 1993. Site selection for the disposal vault and its construction will likely follow if the proposed concept is found acceptable. Methodologies for safe transportation of the nuclear fuel waste are also being developed as part of the NFWMP (OECD/NEA 1988b).

3.1.5 Organization/Responsibilities for Waste Management

Under the federal Ministry of Energy, Mines and Resources, two organizations for implementing and regulating nuclear activities have been established. The Atomic Energy of Canada Limited (AECL), a Crown corporation, is responsible for developing nuclear reactor and waste management technology; it functions as a commercial company. The Atomic Energy Control Board (AECB) is responsible for regulating all stages of the nuclear fuel cycle.

Currently, AECL shares responsibility for managing spent fuel and high-level radioactive wastes with the provincial government of Ontario, where the bulk of the spent fuel and HLW is located. According to an agreement

(a) "Nuclear Waste Disposal Concept and Waste Management Issues Referred for Environmental Review." News Release from Canadian Ministry of Energy, Mines and Resources, No. 88/217, September 28, 1988.

negotiated in 1978, the government of Ontario, via the publicly-owned provincial utility, Ontario Hydro, is responsible for interim spent fuel storage and waste transportation. The federal government, via AECL, is responsible for immobilization and disposal of HLW and for the NFWMP. Ontario Hydro provides some technical support to AECL. Two other provincial utilities, Hydro-Quebec and New Brunswick Electric Power Corporation, operate one nuclear station each but to date have allowed Ontario Hydro to take the lead in cooperation with the federal government on radioactive waste management matters (IEAL 1987).

Although the federal AECB is the lead agency in the disposal concept assessment process for the HLW disposal program, the federal Department of Environment and the Ontario Ministry of Environment assist it. Together the three agencies constitute the Interagency Review Committee, the official government review group for the NFWMP. The sequential review process will be tripartite, consisting of a safety and environmental review, a public hearing process, and a final joint federal-provincial decision on concept acceptability. AECL is to submit a Concept Assessment Document for approval by the AECB, which is to take into account the advice of the federal and provincial environmental authorities and prepare a report on the document in making its recommendation.

AECL, shortly after initiating the NFWMP, established the Technical Advisory Committee (TAC), a group of experts selected from major scientific and engineering societies in Canada, to provide a continuing review of the program (IEAL 1987; TAC-6 1985).

In addition to the IRC and TAC, there are several other federal/provincial coordinating committees in the nuclear waste management area. The Canada/Ontario Nuclear Spent Fuel Management Policy Committee makes recommendations on policy issues. The Canada/Ontario Nuclear Fuel Waste Management Coordinating Committee coordinates activities involving radioactive waste management field research in Ontario. There is also a Canada/Ontario Program Integration Working Party that identifies program objectives, assures requisite information flow, and ensures that the R&D program plan reflects program objectives (IEAL 1987).

The provinces of Quebec and New Brunswick, having limited reactor capacities, are not involved, other than to be kept informed, in the formal institutional framework managing the nuclear program.

3.1.6 Repository Licensing Process, Regulation Status, Major Regulations, Safety Bases

All nuclear operations and facilities are regulated by the AECB under the Canadian Atomic Energy Control Act of 1946. The AECB regulations are based upon recommendations of the International Commission of Radiological Protection (ICRP) and utilize the "ALARA" (As Low As Reasonably Achievable) principle. Regulations are issued as Regulatory Documents and cover all spent fuel and waste management activities, including disposal.

Because repository development and site selection are in the early stages, applicable regulations are general in nature. Current regulations applicable to repository siting and development are provided in AECB's regulatory documents R-72 and R-104.

The requirements for an acceptable site for geologic disposal are specified in AECB's regulatory document R-72. These are as follows: 1) the host rock and geological system should have properties such that the release of radioactive material from the disposal vault and its subsequent transport is retarded; 2) there should be little likelihood that the host rock will be exploited as a natural resource; 3) the geological system should be capable of withstanding stresses without significant structural deformation or fracturing; and 4) the dimensions of the host rock should be such that the disposal vault can be deep underground, as well as removed from major geological discontinuities (OECD/NEA 1988b).

AECB's regulatory document R-104 deals with the long-term performance of radioactive waste disposal. It specifies the acceptable radiological risk to individuals in the most exposed group from a waste disposal facility and states that the timeframe of concern for the repository is 10,000 years (OECD/NEA 1988b).

3.1.7 Roles of the Public, Local Organizations, Multinational Organizations

Other than through use of their voting power, the public and local organizations generally have no direct input to the planning and regulation of nuclear programs. The federal government, the provincial government of Ontario, and AECL expend considerable effort, however, in providing information to the public through displays, literature, films, tours, and public speakers. Also, the review process for the Concept Assessment Documentation for the repository program includes public hearings. AECL has established the TAC, a group of experts selected from the major scientific and engineering societies, to provide a continuing review of the waste management program (Frech 1982; Burge 1988; Delbridge 1989; TAC-6 1985).

Similarly, Canada has not asked multinational organizations for direct input to its nuclear program, but Canada participates actively in the nuclear programs of the IAEA and the OECD/NEA. Canada also receives indirect input through bilateral technology exchange agreements with the U.S., Sweden, and the CEC and through contacts maintained with appropriate organizations in Japan, the U.K., Switzerland, and the FRG (OECD/NEA 1988b).

3.2 OVERALL WASTE MANAGEMENT SYSTEM

The Canadian strategy has not included fuel reprocessing, though some research has been conducted in anticipation of the possibility of eventual reprocessing. All radioactive wastes are being placed in interim storage pending future selection of disposal systems. Both wet and dry technologies for storage of spent reactor fuel are being used.

Surface or near-surface structures are used for storage of LLW and ILW. Extensive R&D is underway to improve ILW storage systems, to develop a deep geologic disposal system for spent nuclear fuel and/or HLW, to develop a transport system for safe handling of spent fuel waste, and to develop an engineered structure for disposal of short-lived wastes (OECD/NEA 1988a).

3.2.1 Reprocessing

No decision has been made as to reprocessing or direct disposal of spent nuclear fuel. The decision, when made, will consider economics and the need for recoverable byproducts. No time scale for the decision has been identified. Mini-scale reprocessing flowsheet studies, hot and cold, have been conducted on the uranium-plutonium and uranium-thorium fuel cycles at AECL's Whiteshell Nuclear Research Center (Schneider et al. 1988).

3.2.2 Interim Storage Before and After Reprocessing

At-reactor storage, wet and dry, of spent nuclear fuel will be continued until a decision is made in favor of reprocessing or until a disposal system is commissioned. Assuming that the review of the NFWMP is favorable (completion expected in 1991-93), it is expected that a transportation and disposal system will be in place about 2015-2025 (OECD/NEA 1988b).

3.2.3 Geologic Repository

The reference concept being assessed for disposal of spent fuel or HLW is burial in a vault 500 to 1000 meters deep in plutonic rock of the Precambrian Canadian Shield. It is expected that there will be one central disposal facility located in Ontario, which will serve the nation's needs for most of the 21st century.

A multiple-barrier approach, which includes both engineered and natural barriers, is being developed for isolating the waste. The waste form, durable containers, buffer, backfill, and the rock mass between the disposal vault and the biosphere are the barriers.

3.2.4 Other System Considerations

While generally moving toward a fuel cycle with no reprocessing but with deep geologic disposal of spent nuclear fuel, Canada has shown an openness to other options. This is evidenced by its participation in the recent Nuclear Energy Agency (NEA) evaluation of international waste repositories and in the NEA's study on the feasibility of disposal of HLW into the seabed. Canada is also a significant producer of radioisotopes for industrial and medical uses; reprocessing would enlarge its supply of radioisotopes and byproducts for marketing (OECD/NEA 1987, 1988a)

3.3 WASTE STORAGE AND TRANSPORTATION

Canada is storing all radioactive waste, including spent fuel, at the generation site. Most spent fuel is stored in water basins, but use of dry storage is increasing. Transportation casks are under development for use when transfer of spent fuel from the generation site to a repository begins.

3.3.1 Spent Fuel Storage

In Canada, spent fuel originates from the natural-uranium, heavy-water CANDU reactor system developed by Canada. A CANDU fuel assembly (50 cm long, 10 cm in diameter) is made up of natural uranium dioxide pellets enclosed in Zircaloy-4 tubes, 19 to 37 tubes per assembly. The average burnup is 7500 MWd/MTU, and average heat content at 10 years of age is 218 W/MTU. Spent fuel arisings were about 12,400 MTU in 1987 and are expected to reach 100,000 MTU in 2025. The age at time of disposal will be at least 10 years but is expected to be typically 50 to 75 years (Schneider et al. 1988; OECD/NEA 1988b).

Most spent fuel is stored in water pools at the reactor sites. It is stored horizontally in stacked racks. The water pools at a reactor are normally sized for 10 years of reactor operation (Schneider et al. 1988; OECD/NEA 1988b).

Four dry-storage concepts have been under evaluation for incremental interim storage of spent fuel: convection-cooled vaults, concrete casks, concrete "integrated" casks (for storage, transportation, and disposal) and metal casks. Presently, vertical, steel-lined concrete canisters are being used to store WR-1, Gentilly-1, and Douglas Point spent fuel (Schneider et al. 1988). The New Brunswick Electric Power Commission has placed an order with AECL for dry storage casks to be used at its Point Lepreau nuclear power station (Nuclear News 1989). The first Ontario Hydro concrete integrated cask was loaded in December 1988 in the Pickering Station storage pool and has been placed on an outdoor pad for a two-year demonstration. The concrete integrated cask concept appears to be the choice for extended storage of Ontario Hydro spent fuel. Its outer size is 2.6 meters in diameter and 3.6 meters high, with a 1.64-meter inside diameter. It would contain nearly 400 fuel bundles stacked horizontally on racks. The gross weight would be 70 MT (Schneider et al. 1989).

3.3.2 High-Level Waste Storage

Canada has not yet decided whether or not to reprocess spent fuel, and it has no HLW to immobilize and store.

3.3.3 Transportation

Except for small amounts used in R&D experimentation, all spent fuel is currently stored in water pools at the generating station. It is anticipated that the fuel will eventually be transported from the generating station to a central disposal facility, when such a facility is commissioned in the distant future (OECD/NEA 1988b).

Research has been carried out to investigate road, rail, and water transport (on the Great Lakes) for distances up to 1,600 kilometers.

Transportation will comply with AECB regulations, which are generally similar to the IAEA's "Regulations for the Safe Transport of Radioactive Materials," Safety Series No. 6. The weight and size will comply with existing federal and provincial transport requirements, including the Transportation of Dangerous Goods Regulations, AECB Transport Packaging Regulations, Spills Bill Regulations, and the Canadian Transport Commission Regulations (OECD/NEA 1988b).

A rectangular cask for road transport, of monolithic stainless steel construction, has been licensed by the AECB, and it meets the requirements for an IAEA Type B(U) transport license. This cask is about 1.8 meters high, and the dimensions of its base are 1.6 meters by 1.9 meters. The gross weight when placed upon the transport trailer will be 34 MT. It will hold 192 fuel assemblies, or 3.8 MT of spent fuel (OECD/NEA 1988b; Schneider et al. 1988).

Canada also has under development a rail cask similar to but larger than the road cask and a concrete "integrated" cask for transport, storage, and disposal of spent fuel (OECD/NEA 1988b; Schneider et al. 1988).

The wall thickness of these casks will be such that the dose rate from the spent fuel will be well below regulatory limits. A 10-year cooling period has been assumed for design of the reference cask. Overhead cranes will be used for loading and unloading casks (OECD/NEA 1988b; Schneider et al. 1988).

3.4 GEOLOGIC WASTE REPOSITORY

The concept being assessed for disposal of spent fuel is emplacement in a vault 500 to 1000 meters deep in plutonic rock of the Canadian Precambrian Shield. It is expected that there will be one central disposal facility located in Ontario, which will serve the nation's needs for most of the 21st Century. The reference design envisions a vault holding 191,000 MT of uranium (OECD/NEA 1988b).

3.4.1 Safety Requirements and Approach

A multiple-barrier safety approach, which includes both engineered and natural barriers, is being developed for isolating the spent fuel. The waste form, durable containers, buffer, backfill, and the rock mass between the disposal vault and the biosphere are the barriers. The performance of the entire disposal system is being assessed by predictive computer modeling and by a study of uranium ore bodies, which represent large inventories of radioactive materials in a natural setting similar to that proposed for the vault. Safety requirements have been defined in AECB's regulatory documents R-72 and R-104 (OECD/NEA 1988b; IEAL 1987), as follows:

- Regulatory Document R-72. This guide applies to siting of an underground repository and specifies that a successful disposal system will:
  - isolate and retain radioactive substances to allow for more complete radioactive decay

- restrict the movement of radionuclides that may escape from the repository, thus prolonging the time that passes prior to their return to the accessible environment and during which further radioactive decay can take place

- restrict inadvertent or deliberate human contact with the waste.

- Document R-72 also states five criteria for acceptable geological characteristics:

  - The host rock and geologic system should have properties such that their combined effect significantly retards the movement or release of radioactive material in the waste. (This criteria is affected by the groundwater chemistry.)

  - There should be little likelihood that the host rock will be exploited as a natural resource.

  - The repository site should be located in a region that is geologically stable and likely to remain stable.

  - Both the host rock and the geologic system should be capable of withstanding stresses without significant structural deformation, fracturing, and breach of the natural barriers.

  - The dimensions of the host rock should be such that the repository can be deep underground and well removed from structural boundaries.

- Regulatory Document R-104. This document provides the objectives, requirements and guidelines for radioactive waste disposal, which include thé following:

  - The burden on future generations shall be minimized 1) selecting disposal options for radioactive wastes that, to the extent reasonably achievable, do not rely on long-term institutional controls as a necessary safety feature; 2) implementing these disposal options at an appropriate time, taking into account technical, social, and economic factors; and 3) ensuring that there are no predicted future risks to human health and the environment that would not be currently accepted.

  - Radioactive waste disposal options shall be implemented in a manner such that there are no future impacts on the environment that would not be currently accepted and such that the future use of natural resources is not prevented by either radioactive or non-radioactive contaminants.

- The predicted radiological risk to individuals from a waste disposal facility shall not exceed 1 x 10^{-6} fatal cancers and serious genetic effects in a year, calculated without taking advantage of long-term institutional controls as a safety feature.

- The individual risk requirements in the long term should be applied to a group of people that is assumed to be located at a time and place where the risks are likely to be the greatest, irrespective of national boundaries.

- The period for demonstrating compliance with the individual risk requirements using predictive mathematical models need not exceed 10,000 years. Where predicted risks do not peak before 10,000 years, there must be reasoned arguments that beyond 10,000 years the rate of radionuclide release to the environment will not suddenly and dramatically increase and acute radiological risks will not be encountered by individuals.

- The probabilities of exposure scenarios should be assigned numerical values either on the basis of relative frequency of occurrence or through best estimates and engineering judgments.

- Calculations of individual risks should be made by using the risk conversion factor of 0.0002 fatalities/rem and the probability of the exposure scenario with either 1) the annual individual dose calculated as the output from deterministic pathways analysis; or 2) the arithmetic mean value of annual individual dose from the distribution of individual doses in a year calculated as the output from probabilistic environmental pathways analysis.

3.4.2 Siting

The Geological Survey of Canada (GSC) was requested by the AECL in 1975 to evaluate the suitability of geological formations in Canada for disposal of spent fuel. A primary consideration in this evaluation was that because the province of Ontario was, and would continue to be, the principal region for nuclear power development, the first disposal vault would most likely be located there. Plutonic rock is the predominant geologic formation in Ontario (Dormuth et al. 1989).

Geologic procedures for characterization of a site for the disposal facility, which would ensure that the above-listed AECB requirements are satisfied, have been established by AECL as a result of scientific research carried out under the NFWMP. It is expected that a very small number of sites will initially be selected for characterization and that each of these will be centered around a pluton of approximately 1000 square kilometers. The initial characterization, which will consist of reconnaissance-scale geological and geophysical surveys, will map the boundaries of the pluton, estimate its depth, locate large-scale faults or fractures within it, and estimate the general rock quality. The general groundwater regime of the areas will also be determined at this stage. Site characterization will result in the selection of smaller areas, up to 100 square kilometers in extent, for more detailed site evaluation.

Detailed geological and geophysical mapping and drilling will be carried out to determine the geometry of the natural groundwater pathways in the pluton. Extensive hydrogeological testing in the boreholes will determine the pattern of groundwater flow. As a result of this second phase of site characterization and evaluation, a specific site technically suitable for the disposal vault could be selected. No firm time schedule exists for the siting process (OECD/NEA 1988b).

3.4.3 Design Concept(s)

A major study was conducted by the AECL between 1980 and 1990 to develop a reference concept for a Used Fuel Disposal Center. Its main feature is an underground disposal vault at a depth of 1000 meters in plutonic rock having a single-level room and pillar configuration. This vault would have a plan area of 4 square kilometers and be capable of accommodating 191,000 MT of uranium in the form of spent fuel. In addition to the vault, the disposal center would have facilities for receiving the spent fuel, containerization of the fuel bundles, and transfer to the vault (OECD/NEA 1988b).

The incoming spent fuel would be off-loaded from transportation casks and temporarily held in a receiving storage pool. It would then be placed in disposal canisters in a facility on the disposal site and transferred to the subsurface vault in shielded transfer casks through a dedicated waste handling shaft. The design throughput is 8100 canisters per year, and the filled repository would accommodate 191,000 MTU (OECD/NEA 1988b).

The spent fuel containers would be placed in vertical boreholes drilled in the floors of rooms inside the disposal vault. The holes would be 1.2 meters in diameter and 5 meters long and spaced 2 meters from adjoining boreholes. Prior to receipt of the container, a clay-buffer material would be compacted into the hole and then drilled to form a 4.2-meter-long hole 0.7 meters in diameter in which the container would eventually be seated. The annular gap between the canister and buffer material would be filled with loose sand and the top of each hole filled with loose buffer material that is compacted in place (OECD/NEA 1988b; Schneider et al. 1989).

A rectangular array of 60 parallel disposal drifts would be constructed in a total plan area of 2 kilometers by 2 kilometers; each disposal drift would be about 250 meters long by 7.5 meters wide by 5.0 meters high; each drift would accept 240 waste canisters. Backfill used in the emplacement rooms would likely consist a mixture of lake-bottom clays and granitic aggregate. Seals would be used to form gaskets around bulkheads and fill exploration holes. Grouts would be used to limit water flow in naturally occurring fracture zones and excavation damage zones (Schneider et al. 1989; OECD/NEA 1988b).

The zone of rock overlying the vault, which undergoes extension due to thermal expansion of the rock mass, must be no more than 100 meters from the surface. The sustained long-term, far-field temperature must be no higher than 75°C, while the maximum buffer temperature has been set at 100°C (Schneider et al. 1989).

3.4.4 Retrievability and Monitoring

The AECB policy on retrievability is that capability for retrieval of the waste must be maintained during repository operation. Presently, no plans exist for post-closure retrievability. Requirements and procedures for post-closure monitoring are now being formulated (IEAL 1987; OECD/NEA 1988b).

3.4.5 Waste Package System

In the present reference concept, components of the waste package system include the spent fuel placed in a thin steel basket, a thin titanium shell with the void space filled with particulates, and a buffer material placed between the canister and the host rock. No hole liners are planned. The functions established for the waste canister include 1) handling and sealing during interim storage, transportation and emplacement; and 2) in-repository corrosion protection for 500 to 1000 years. The maximum design temperature and pressure for the emplaced canister have been set at 150°C and 100 atmospheres, respectively (Schneider et al. 1989).

Each canister would contain 72 fuel assemblies (1.4 MTU and 300 W of heat), stacked four high with 18 assemblies in each row. The assemblies would be held in place by a thin mild steel basket placed in the thin-wall titanium (Grade 2 or 12) canister shell. The empty space in the canister would be filled with a particulate material (probably sand) vibrationally compacted. Each canister is 0.63 meters in diameter by 2.25 meters long and has a wall thickness of 4.76 millimeters on the sides and top and 6.35 millimeters on the bottom. The top lid of the canister is tapered and will be pressed into place in the tapered top of the canister and resistance-heated, diffusion-bonded to the shell (Schneider et al. 1989; OECD/NEA 1988b).

A buffer material would be used around the emplaced waste in the repository. The functions of the buffer material are to seal the waste canisters in the repository, to retard radionuclide release from the canisters to the host rock, and to improve heat transfer from the canister to the rock. The reference buffer material is a 1:1 dry mass ratio "Avonseal" sodium bentonite and quartz sand. Avonseal contains 80% sodium-based montmorillonite; 10% illite; and smaller amounts of quartz, feldspar, gypsum, and carbonates. It has a moisture content of about 18% and a final specific gravity of about 1.65 (Schneider et al. 1989).

3.4.6 Research and Development

Canada has an extensive R&D program underway on disposal packages and on a deep underground repository. The R&D on waste packages covers the reference and alternate package designs and materials of construction such as copper, Hastelloy C-276, Inconel 625, ceramics, and ceramic-coated metals. The mechanisms of spent fuel dissolution in a repository are also under study (Schneider et al. 1989; Johnson et al. 1989).

An underground research laboratory (URL) has been constructed by AECL in the granite of the Lac du Bonnet batholith in southeastern Manitoba. This facility was developed in two stages. Initially, the 240-meter-deep research level was developed. Then, in cooperation with the U.S. the shaft was extended to 443 meters, and shaft stations were developed at the 300-meter and 420-meter levels (OECD/NEA 1988b, 1989).

The 3.8-square kilometer site on which the URL is located was leased in 1980. Detailed mapping and drilling between then and 1983 by the Geologic Survey of Canada established the geology and the undisturbed groundwater regime for this area. Shaft sinking began in 1984, and the 240-meter level, with a plan area of about 100 meters by 100 meters, was ready for carrying out experiments in 1986. Hydrogeological data gathered in a network of boreholes in the leased site prior to, during, and after the construction of the URL provided a unique opportunity to study the effect of construction upon the groundwater system and calibrate groundwater flow models. Numerous investigations and experiments related to hydrogeology, rock mechanics, vault sealing, geology, and physics have been carried out in the subsurface, and it is expected that an active experimental program will be maintained until the year 2000 (Boulton 1982; OECD/NEA 1988b; Simmons 1986).

3.4.7 Approach to Proving Safety of the Repository

The emphasis in Canadian regulatory guidelines and requirements is on overall disposal system performance criteria rather than on prescriptive criteria for performance of individual components of the disposal system. Also, it is recognized that the evaluation of the long-term performance of a disposal system must be based on estimates provided by mathematical modeling, because no long-term performance histories of comparable facilities exist. The performance of the entire disposal system is being assessed by predictive modeling and by a study of uranium ore bodies, which represent large inventories of radioactive materials in a natural setting similar to that proposed for the vault (OECD/NEA 1988b).

A continuous program of prediction of system behavior by modeling and comparison of predictions with observations was started in 1981 by AECL and will be continued throughout the construction, operation, and eventual closure of the repository. The predictive models will thus develop progressively over the 60 to 70 years required to site, construct, operate, and close a disposal facility. They will be used to predict performance at every stage and be modified as necessary to represent new information. Thus, a high level of confidence is expected in the validity of the final predictions (OECD/NEA 1988b).

Canada's repository performance assessment program is one of the first to use performance assessment models using probabilistic risk assessments for the total system--the Systems Variability Analysis Code (SYVAC). The SYVAC code system was demonstrated in Canada's first generic comprehensive performance assessment, completed in 1981, which involved Monte Carlo simulations of 1730 parameter value realizations. Of these realizations, only 1000 parameter value combinations led to any calculated releases from the repository. Doses were typically less than 1% of background radiation levels and never greater than 20%. The scenarios involving use of water wells yielded estimates with the greatest doses, especially where domestic wells pumped low-flow rate contaminated groundwater. These results, in combination with estimates of the likelihood of their occurrence, identified evaluation of these scenarios as being necessary for a real site (Lisle and Wright 1986).

The second comprehensive Canadian generic risk assessment, completed in 1985, used a new version of the SYVAC code, called SYVAC2-C, to probabilistically estimate doses to the maximally exposed individual from a repository containing CANDU spent fuel. This version contained three submodels covering the disposal vault, the geosphere, and the biosphere. Ranges and distributions for the major parameters were estimated based upon experience in granitic rock formations. No disruptive events were considered (Lisle and Wright 1986).

The results of the second assessment indicated no significant consequences for tens of thousands of years after disposal. Out of 1000 scenarios analyzed, only 7% resulted in doses greater than 1.8 mrem/yr (1% of background). The geosphere was the most important and effective barrier. The highest doses resulted from using a contaminated waterwell as the source of household water (Lisle and Wright 1986).

Canada is continuing to improve its assessment models using data obtained from the URL and other sources, through participation in the OECD/NEA's expert groups on modeling, and through use of the results of studies on natural analogs, such as the one at Cigar Lake in Saskatchewan, Canada (Wright 1989; OECD/NEA 1989).

3.4.8 Peer Review Activities

AECL has the prime responsibility for executing Canada's program on radioactive waste management. The program is under continual review, however, through a number of agencies, internal and external to AECL. The TAC was established by AECL, using experts selected from scientific and engineering societies, to provide a continual review of the effort.

The AECB, with the prime responsibility for regulating nuclear activities, has established a review process consisting of a safety and environmental review, a public-hearing process, and a final federal-provincial decision process on the acceptability of the plan. It has been joined by the Federal Ministry of Environment and the Ontario Ministry of Environment in this review, working as the Interagency Review Committee. In addition, the Canada/Ontario Nuclear Spent Fuel Management Committee makes recommendations on management issues, and the Canada/Ontario Nuclear Fuel Waste Management Coordinating Committee coordinates activities involving field research in the province of Ontario. The Canada/Ontario Program Integration Working Party identifies program objectives, ensures that information flow needs are being met, and ensures that the R&D program reflects program objectives.

Canada participates with other countries through bilateral agreements and the activities of the international agencies in the area of radioactive waste management. That participation, along with open publication of progress on its activities, provides an unofficial peer review of its programs.

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4.0 FRANCE

4.1 LEGISLATIVE/INSTITUTIONAL BASES

4.1.1 Nuclear Power Policy

The French government is strongly supportive of nuclear power. In France, over 70% of the electricity is generated by nuclear power; this is the highest percentage of nuclear electricity production of any country in the world. The 1988 French nuclear capacity was 51 GWe, primarily supplied by PWRs, and is expected to increase to 71 GWe by the year 2000. The French nuclear industry is geared to a closed fuel cycle, with reprocessing and use of the recovered plutonium in breeders and LWRs (Leigh 1989; OECD/NEA 1988; Schneider et al. 1988).

4.1.2 Major Legislation

All nuclear installations, including waste storage and disposal facilities, are subject to the following:

- national environmental protection laws and regulations, including the law on "The Democratization of Public Inquiries and Environmental Protection," dated July 12, 1983, with application decrees of April 23 and 24, 1985
- specific regulations (decree of December 11, 1963, and modified April 24, 1985) governing authorization, licensing, and control of nuclear installations
- national laws and regulations governing worker health protection against ionizing radiation
- Fundamental Safety Rules (RFS) that express basic safety principles, procedures and technical specifications, including rules applicable to the production, control, processing, packaging, and storage of wastes resulting from PWR fuel reprocessing (RFS III.2) (OECD/NEA 1988; Schneider et al. 1988).

4.1.3 Linkage Between Nuclear Power and Waste Management

No stipulation law exists with requirements for radioactive waste disposal before nuclear power can be implemented (Schneider et al. 1988).

4.1.4 Policy on Spent Fuel and Waste Management

The French radioactive waste management policy is described in the report of the French Atomic Energy Commission (CEA), Commissariat à l'Energie Atomique, entitled "General Radioactive Waste Management Program," approved by the government and made public June 19, 1984. Spent fuel is stored first at the reactor site for about one year and then in an away-from-reactor (AFR) facility at the reprocessing plant until it is reprocessed. High-level waste is vitrified and then stored for 30 years or more in a vault until the canisters can be placed in a repository. All long-lived wastes are to be disposed of by emplacement in a suitable deep geological formation. There are no plans for direct disposal of spent fuel without reprocessing (OECD/NEA 1988).

4.1.5 Organization/Responsibilities for Waste Management (OECD/NEA 1988; Schneider et al. 1988; IEAL 1987).

Waste management policy, regulations, and control, as well as authorization and licensing of waste disposal sites, are the responsibility of the French federal government. The overall coordination of nuclear matters within the French government is the responsibility of the CSIN (Interministerial Committee on Nuclear Safety), which is chaired by the Prime Minister and includes the Ministers for Health, Interior, Defense, and Industry.

The Ministry of Industry, Telecommunications and Tourism has jurisdiction over all nuclear activities.

The CEA (which reports to the Ministry of Industry) manages fuel cycle and waste management activities through subsidiary government agencies and government corporations. Long-term radioactive waste management is the responsibility of the National Radioactive Waste Management Agency, ANDRA, created within the CEA in 1979. The responsibility of the waste producers (i.e., Electricite de France, EdF, the French national electric utility) is to perform all necessary operations to produce a waste form suitable for disposal (conforming to ANDRA specifications). ANDRA's responsibilities include the following:

- setting the requirements for waste package acceptance for disposal, ensuring that these requirements conform to the Fundamental Safety Rules and satisfy the safety authorities, and controlling compliance through a quality assurance system

- planning for disposal facilities and their financing

- siting, constructing, operating, and closing waste disposal facilities.

A department of the Ministry of Industry, the SCSIN (Central Service for the Safety of Nuclear Installations), develops and enforces safety regulations, issues construction permits and operating licenses, and monitors operational safety of nuclear installations, including nuclear waste repositories. The IPSN (Institute for Nuclear Protection and Safety), part of the CEA, provides technical support. The SCPRI (Central Service for Protection against Ionizing Radiation), under the Secretary of Health, is responsible for regulations concerning radioactive effluent releases and monitors radioactivity in the environment. The SCPRI has the right to veto any construction or operating license.

The CSSN (Supreme Council for Nuclear Safety), which includes ministerial department heads, Parliament members, and trade union and environmental association representatives, advises in all aspects of nuclear safety and reports to the Minister of Industry. Within the CSSN is the Working Group on Spent Fuel Management, also known as the Castaing Commission, which has produced reports dealing with spent fuel management, reprocessing, and nuclear waste disposal.

The French Transport Ministry is responsible for certifying transportation packaging (e.g., spent fuel casks), and the Ministry of Industry issues transport licenses and carries out required inspections.

Cogema, the commercial fuel cycle arm of the CEA, is a government corporation that owns and operates reprocessing and HLW immobilization facilities at the Marcoule and La Hague Centers.

The costs of managing, transporting, reprocessing, vitrifying, and finally disposing of radioactive wastes are borne by the producers of the waste.

Because all civilian HLW results from the nuclear power program of the single national utility (EdF), HLW management is financed directly out of the government's national budget and no fee is assessed on individual waste generators. For a program of 60 gigawatts of nuclear generating capacity and reactor lifetimes of 30 years, the cost of the French waste program (including R&D, construction, and operation of repositories) is projected at $7.5 to $8.3 billion (U.S. dollars), or $0.008 (U.S. dollars) per kWh (IEAL 1987).

4.1.6 Repository Licensing Process, Regulation Status, Major Regulations, Safety Bases (OECD/NEA 1988; Schneider et al. 1988; IEAL 1985)

Authorization for construction of the repository is subject to approval of a safety report by the Central Service for the Safety of Nuclear Installations (SCSIN) and to a public inquiry.

In the public inquiry process, all necessary information concerning the project is made available to the public, and registers are made available for the public to make written comments. An Inquiry Commission is designated by the local administrative court and can request supplementary information from the license applicant. The Commission report, which considers the public comments and contains recommendations, is sent to the local administrative court, the applicant, the regulatory authorities, and each of the local town councils where the inquiry took place. The report is also available to the public for one year.

After the public inquiry and technical studies by specialized branches of the CEA and other Ministries are favorably concluded, a decree authorizing installation must be signed by the Minister of Industry and the Prime Minister. The Parliament has preemption rights in siting decisions, and a land use veto by local governments can be overruled.

Operational licensing is subject to approval of the final safety report by the SCSIN and a preliminary radioactive effluent release impact study.

No regulations have been issued for the disposal of long-lived radioactive waste. The report by the Ad Hoc Commission on Technical Criteria for Underground Disposal, which was made public in November 1987, is being reviewed by safety authorities.

4.1.7 Roles of the Public, Local Organizations, Multinational Organizations

A formal public inquiry process allows the local community to examine a nuclear project and to express opinions. Nearby communities, within a five-mile radius of the proposed project, may also vote on the nuclear project. If the vote is favorable, the applicant needs only to receive federal government authorization. If the vote is unfavorable, then the federal Parliament is called upon to make a final decision deemed to be in the national interest (IEAL 1985).

France is a member of the IAEA, the OECD/NEA, and the CEC and is actively involved in their overall waste management activities and in many specific cooperative R&D projects. Also, bilateral R&D cooperative agreements on radioactive waste management have been signed with Sweden, Switzerland, Spain, and the U.S., and cooperative projects are actively pursued (OECD/NEA 1988).

4.2 OVERALL WASTE MANAGEMENT SYSTEM

The overall HLW management strategy is to 1) store spent fuel first at the reactor site and then in an AFR facility at the reprocessing plant until it is reprocessed, 2) vitrify the HLW, and 3) store the canisters in a vault (a storage period of 30 years or more is envisioned) until they can be placed in a geologic repository. The French policy is that no spent fuel will be disposed of in the repository.

The spent fuel resulting from nuclear power generation should total about 20,000 MT by the year 2000, and reprocessing of French spent fuel is expected to produce about 3000 cubic meters of vitrified HLW for disposal by the year 2000 (IEAL 1987). All radioactive wastes resulting from French reprocessing of foreign spent fuel are to be returned to the originating countries (OECD/NEA 1988).

4.2.1 Reprocessing (OECD/NEA 1988; Cogema 1987)

Reprocessing is a key step in the nuclear fuel cycle in France. Reprocessing of spent fuel is done at the La Hague (LWR fuel) and the Marcoule [gas-cooled reactor (GCR) fuel] installations. Fast breeder reactor fuel has been reprocessed both at La Hague and in the pilot plant at Marcoule. The La Hague plant has been operational since 1966, and the Marcoule plant since 1956. Capacity at La Hague will be increased from 400 to 1600 tons per year in the early 1990s in one new and one expanded facility.

The French reprocess fuel for both domestic and foreign customers. Over the past ten years, Cogema has reprocessed over 80% of all civilian LWR fuel reprocessed in western countries.

4.2.2 Interim Storage Before and After Reprocessing

Spent fuel is stored at reactor sites in pools until transferred to spent fuel basins at the reprocessing facilities within about one year of discharge. Reracking in reactor bays and dry storage are not planned for LWR fuel. The La Hague spent fuel storage facility has 10,000 MTU wet storage capacity and can unload up to 800 MTU/yr under either wet or dry unloading conditions (Schneider et al. 1988).

A vault facility for dry storage of gas-cooled, heavy water moderated reactor (GCHWR) and other non-standard fuels is under construction at Cadarache with a capacity of about 200 MT. For the Superphenix fast breeder reactor (FBR) fuels, an onsite storage facility is to be built using dry storage in gas-filled Castor shipping casks in an interim storage hall (Schneider et al. 1988).

Following reprocessing, the liquid HLW will be vitrified by the Atelier de Vitrification de Marcoule (AVM) process in use at the Marcoule Center (OECD/NEA 1988). The AVM process involves calcination of the wastes in a rotating, inclined tube followed by conversion to borosilicate glass in a metallic pot using induction heating. Vitrification has been conducted at Marcoule since 1978 and will be conducted at La Hague in two plants, each equipped with three vitrification lines (Schneider et al. 1988). The HLW glass containers are stacked and stored in air-cooled concrete dry wells that are covered by a containment building (IEAL 1987).

4.2.3 Geologic Repository

A deep underground final disposal site for long-lived wastes, both TRU and HLW wastes, is planned. A candidate site for the repository will be selected, and an In Situ Site Validation Laboratory (ISVL) will then be built for detailed site characterization. Once the repository site is selected and qualified, a TRU waste disposal facility will be built in parallel with construction of a nearby pilot repository for tests with HLW canisters (OECD/NEA 1988; Schneider et al. 1988).

Following site selection, the construction of the ISVL is scheduled to start in 1990. Assuming favorable results from the ISVL, the startup of the TRU disposal facility is scheduled for the early 2000s and the HLW glass disposal facility for 2010 (OECD/NEA 1988).

4.2.4 Other System Considerations

Immobilized TRU wastes, as well as vitrified HLW wastes, will be disposed of in the geologic repository. Two recommendations made by the Castaing Commission that are not being adopted presently are 1) stripping of alpha-emitters from wastes followed by "incineration by neutron bombardment or launching into outer space," and 2) the use of monitored retrievable storage, rather than irretrievable disposal, for the remaining alpha wastes (IEAL 1985).

No information was found on France's interest in an international repository. The French are generally favorable on the concept of seabed disposal.

4.3 WASTE STORAGE AND TRANSPORTATION

Spent fuel is stored at the reactor site and is then transported in specially designed casks to the reprocessing plant, where it is stored until it is reprocessed. The resultant HLW is vitrified. Onsite storage of the HLW glass canisters is provided until the repository is ready for receipt of wastes.

4.3.1 Spent Fuel Storage

French policy is for spent fuel to be stored at the reactor site for approximately one year and then at the reprocessing plant site for two to three years. Pool storage is used for LWR spent fuel, and dry storage is used for FBR spent fuel. The La Hague spent fuel storage facility has 10,000 MTU wet storage capacity and can unload up to 800 MTU/yr under either wet or dry conditions. The longest storage period to date for some spent fuels has been about 10 years (OECD/NEA 1988).

4.3.2 High-Level Waste Storage

Liquid HLW are stored in high-integrity storage tanks at La Hague (from LWR fuel) and Marcoule (from GCR fuel). The tanks are constructed of stainless steel and contain means for cooling and agitation. The HLW liquids are stored for about one year to allow for some radioactive decay prior to vitrification (Cogema 1987). The vitrified wastes are stored in air-cooled vaults at La Hague and Marcoule. The capacity of the first module at La Hague, commissioned in 1988, is 4500 canisters (5625 MT equivalent of spent fuel) (OECD/NEA 1988). Air-cooled concrete caissons are used at La Hague, with canisters stacked nine high. The canisters will be stored at the La Hague site for about 20 years. The HLW will be stored for approximately 30 years prior to disposal. Interim dry storage for HLW of 20 to 30 years will be available at the disposal site (Schneider et al. 1989).

4.3.3 Transportation

The national policy in France for transporting spent fuel or radioactive waste is for the nuclear industry to transport it safely and effectively according to French transport regulations. The industry may make shipments, as needed, provided that all regulations and requirements are met. The regulations regarding transport of spent fuel and radioactive waste are nearly all modeled on IAEA transportation standards (OECD/NEA 1988).

Transport packagings, or casks, are used to transport high-level radioactive material, e.g., spent fuel assemblies. The containers are designed to withstand a severe transportation accident (drop + crushing + fire + water immersion), verified by testing. Each transport packaging model must be certified by the French Transport Ministry. If transport also involves travel in other countries, this certification must also be validated by the authorities of the countries through which the package travels. The Institute for Nuclear Protection and Safety (IPSN) performs technical assessments of all waste packagings for the Transport Ministry. A Transportation Safety Commission oversees compliance with regulations. Transport licenses are issued by the Ministry of Industry, with the IPSN performing the required inspections (French Nuclear News 1989).

For spent fuel transport to the French reprocessing plants, Cogema has a set of standard cask designs that includes dry transport casks, a steel cask body, double containment, large capacity, and standard sizes and equipment (OECD/NEA 1988). Cogema relies on three firms for international transport operations: 1) Transnucleaire, which has developed transport systems used throughout the world; 2) Nuclear Transport Limited, a specialist in spent fuel transportation in Europe; and 3) Pacific Nuclear Transport Limited, which handles the transportation of spent fuel between Europe and Japan. The equipment used to serve the Cogema plant at La Hague includes 90 large rail casks, about 30 specially-designed railway cars, and five double-hull ships with redundant equipment for safety (French Nuclear News 1989).

Transport is made preferably by rail from power plants in continental Europe and by ship from Japanese plants. Road transport, using heavy-haul trucks, are generally limited to short distances between a nuclear site and a rail siding equipped for truck to rail car transfer. About 340 spent fuel cask shipments are made every year in France (French Nuclear News 1989). Prior to 1987, 14,800 MT of uranium in spent fuel had been transported from power plants around the world to the La Hague and Marcoule reprocessing plants in 4100 different cask movements (OECD/NEA 1988).

4.4 GEOLOGIC WASTE REPOSITORY

A candidate site for the In-situ Site Validation Laboratory (ISVL) is planned to be nominated by the end of 1990. The intention is to investigate this site for its suitability to become an actual geologic waste repository. The goal of the ISVL is to validate the site selection by demonstrating the technical feasibility and the economics of the repository. Construction will take about two years, and performing the tests will take between two and three years. Site evaluation should be complete before the end of 1995.

A preliminary safety impact report will be prepared at the end of the site validation period to show that the future repository's environmental impacts are acceptable. Once the repository site is selected and qualified, a pilot repository will be built for tests with HLW canisters, in parallel with construction of a TRU waste disposal facility (a separate part of the same repository site). The startup of the disposal facility for TRU waste is scheduled for the early 2000s, and the startup of the HLW glass disposal facility is scheduled for about 2010 (OECD/NEA 1988).

4.4.1 Safety Requirements and Approach

The Central Service for the Safety of Nuclear Installations (SCSIN) is responsible for the development of technical and safety criteria for the waste repository (for both siting and disposal), based on recommendations from the CEA, IPSN, ANDRA, and special committees (IEAL 1987). The general objective is that the repository provide enough protection that no further human action is required after closure. No radiation protection limits have been set for the boundary of the HLW repository. No performance requirements exist yet for the repository; but multiple barriers are required, and each barrier should contribute significantly to the safety of the repository. A 1000-year package is desired (Schneider et al. 1989).

In France, the geologic setting should have short- and long-term stability, including stability from effects of the man-made changes. Existing or potential future resources should be absent to minimize the potential for human intrusion. The safety of the repository must be demonstrated by modeling for at least 10,000 years (Schneider et al. 1989).

4.4.2 Siting

Sites will be evaluated in four types of geologic formations: granite, salt, clay, and schist. A candidate site for the repository will be selected, and an ISVL will then be built for detailed site characterization. The intention is to investigate this site for its suitability to become an actual repository for long-lived radioactive wastes (OECD/NEA 1988).

The site selection process has involved choosing eight sites (two of each geologic formation type) based on a geological survey and a literature search. A national inventory of possible sites was compiled, based on criteria which included long-term stability, favorable hydrogeology with very low permeability, and good geochemical properties (such as nuclide retention). The national inventory was completed at the end of 1983. The number of sites was reduced from eight to four (one of each formation type) in 1987. Most investigations have centered on a granite formation. Field investigations of the sites will include geophysical measurements from the surface and several deep core drillings. The proposal of one site for the ISVL is scheduled for the end of 1990 (Schneider et al. 1988; OECD/NEA 1988).

The geological disposal site investigation is the responsibility of ANDRA, financed by the waste producers according to their future delivery forecasts. The site investigation program is estimated to cost a total of 1 billion French francs (FF), and construction and operation of the ISVL is expected to cost 1.5 billion FF ($237 million U.S. dollars) (OECD/NEA 1988).

4.4.3 Design Concept(s)

The repository design capacity is 240,000 cubic meters of alpha wastes and 15,000 cubic meters of vitrified HLW (125,000 MT equivalent of spent fuel) (Schneider et al. 1988). ANDRA has not yet decided on a disposal facility layout. Various concepts are presently being examined for the reception, handling, and emplacement of the waste packages and for the design of drifts or silos (OECD/NEA 1988). The minimum depth of the disposal horizon is 150 to 200 meters, and the maximum depth is about 1000 meters (Schneider et al. 1989).

Three scenarios under consideration for HLW disposal in granite at a depth of 500 to 1000 meters are 1) disposal in a spread-out repository following surface interim storage for 30 years, which appears to be the reference case; 2) disposal in a compact repository following a one- to two-century period of engineered surface storage, cooled by natural air convection; and 3) early disposal in a compact repository with in situ cooling provided for one to two centuries (Schneider et al. 1988).

Two families of disposal concepts are under study. One concept is disposal in boreholes in the floors of emplacement drifts with or without interim natural convection air cooling. (Such air cooling is being considered for interim storage to allow for a smaller repository for older waste.) The holes with interim cooling would be large caissons with about 18 stacked canisters in each; the holes without interim cooling would have one canister in a hole. Another concept under consideration is direct disposal in drifts with or without interim natural convection air cooling (Schneider et al. 1989).

Concepts under study for waste package handling are 1) assembling the waste canister and buffer/packing material in a metal basket and lowering the assembly into the disposal hole, and 2) pre-emplacing a clay-type packing into the disposal hole, followed by emplacement of the waste canister (Schneider et al. 1989).

4.4.4 Retrievability and Monitoring (Schneider 1988, 1989)

Retrievability requirements have not yet been determined in France. A test and evaluation facility is to be built and used for demonstration of the retrievable emplacement of HLW and TRU wastes.

It is planned that monitoring of the repository site will not be required for longer than about 300 years. The general objective is that the repository should provide enough protection that no further human action is required after closure. Maintenance of administrative control over the repository area is desirable for several centuries.

4.4.5 Waste Package System

The reference wastes for disposal in the geologic repository are HLW borosilicate glass, cooled several decades after discharge, and immobilized TRU wastes. These wastes would be stored in separate parts of the repository. The waste package concept is a cylindrical stainless steel canister containing about 400 kg of monolithic HLW borosilicate glass. The glass from La Hague is to contain about 12% fission products and actinide oxides. High-level waste from about 1.25 MTU is in one canister. The canister volume is 170 liters, and the canister will be filled 88% full of glass (150 liters). The vitrified HLW quality is based on control of vitrification process parameters (Schneider et al. 1988, 1989).

A 1000-year package is desired, which may be achieved with an overpack or a new canister material. The wall thickness of the canister is 5 mm, and there is no corrosion allowance. The canister is made of special stainless steel designated as Z 15 CN 24-13. Overpacks of metals and ceramics are being considered (Schneider et al. 1988, 1989).

Use of packing or buffer materials remains to be determined, but clays are being investigated actively. Requirements are not yet specified, but studies are aimed at providing heat transfer and a seal between the canister and the excavation in the host rock. The seal would prevent water from contacting the container and retard migration of radionuclides from the container. In addition, the buffer would help keep the container in place. Pressing of the buffer into cylindrical blocks that would surround the waste canister is under study. Filling of the final gap between the container and the buffer with a grout is being considered (Schneider et al. 1989).

4.4.6 Research and Development

A sizable R&D program has been conducted (in situ and in laboratories) to improve understanding of basic hydrological and geochemical mechanisms and to develop instrumentation and methodology for the demonstration of long-term safety (OECD/NEA 1988). Field studies have been conducted at a granite formation near Auriat, with boreholes sunk to depths of 500 and 1000 meters. An underground research laboratory in granite, in a gallery of a uranium mine near Limoges, has been used to conduct permeability studies, tracer migration tests, and other hydraulic studies in fractured granite (Schneider et al. 1988).

The ISVL studies will include in-depth exploration of the whole volume of rock involved in the repository construction. In situ experiments will be conducted to confirm thermal and mechanical behavior 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 (OECD/NEA 1988).

Research and development being conducted on the waste package includes studies of HLW glass properties and reactions with materials in the repository environment; improved canister, overpack, and barrier materials; and the potential for sealing fractures with silica (Schneider et al. 1988).

4.4.7 Approach to Proving the Safety of the Repository

The ISVL will be used by the French to conduct an intensive program of site-specific experimental and performance assessment modeling work for three years. A safety assessment for a repository at the ISVL location will be the result of this work.

The total system performance model under continuing refinement in the French program is the deterministic MELODIE code. MELODIE is used for best estimate calculations, for evaluating scenario effects, and for determining sensitivities and uncertainties. Its component codes are CONDIMENT, a source-term code based on the advection-dispersion equation with diffusion and reprecipitation capabilities; METIS, a two-dimensional finite-element code assuming an equivalent porous medium and accounting for numerous processes important to migration; and ABRICOT, which calculates pathways and doses. Other codes are also under development. The French program is diverse in terms of media being investigated and the participating organizations conducting numerous laboratory, field, and analogue studies. The performance assessment function is recognized as an integral part of all of these activities.

French performance assessment modeling is being coordinated to allow French workers in a number of locations to use the same software and data. The system being developed would allow the model developers and data collectors to inject new information into the model and data bases in a structured yet flexible way.

The French program contributed performance assessments to the CEC's PAGIS exercise for a repository in bedded salt and two for a repository in granite. The analysis based on a repository in bedded salt focused on the role of overburden properties. The expected case had no releases. Doses on the order of 0.1 mrem/yr were calculated after 300,000 years assuming a water intrusion scenario.

The reference analysis for a repository in granite was based on the Auriat site, for which there is significant in situ data. Sensitivity studies showed bentonite backfill to be less important in the long term than the first few meters of host rock. Vertical fractures were shown to be most important when connected to horizontal fractures intersecting the entire repository. Dose results were sensitive to the amount of dilution provided by surficial aquifers and surface waters. Doses for the expected case at Auriat were 0.02 mrem/yr after 250,000 years, peaking at 0.6 mrem/yr after three million years. A human intrusion case wherein a cavity is mined close to the waste emplacement area resulted in a calculated dose to the miners of 1.6 mrem/yr and in regional individual doses of 10 mrem/yr for worst-case data assumptions.

The variant granite case was based on data for the Barfleur site, a coastal location with a nearby low-flow stream and other rivers somewhat further away entering the ocean. Doses were most significant from the low-flow stream and insignificant from the rivers or the ocean. Doses were computed at 300,000 years, with a maximum of 0.1 mrem/yr at three million years.

4.4.8 Peer Review Activities

As mentioned earlier, the French Working Group on Spent Fuel Management, an advisory group composed of independent senior scientists, periodically reviews aspects of the French radioactive waste management activities. No information was found concerning foreign peer review activities of the French radioactive waste management system.

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5.0 FEDERAL REPUBLIC OF GERMANY

5.1 LEGISLATIVE/INSTITUTIONAL BASES

The Federal Republic of Germany (FRG) has an active nuclear power program moving in the direction of complete independence in the nuclear fuel cycle, though recent events stemming from economic and political pressures have slowed progress considerably. About 35% of the FRG electricity demand is supplied by nuclear power stations, primarily PWRs. Reprocessing of spent fuel with recycling of plutonium in nuclear power plants is an integral part of the official FRG nuclear policy. The original plan to conduct the reprocessing in the FRG was modified recently after the FRG utilities, for economic reasons, elected to have the reprocessing done in French and United Kingdom (U.K.) reprocessing facilities. Plans for development of advanced reactors in the FRG have been slowed, also, with the recent decision to shut down the high-temperature reactor, THTR-300, and the FRG utilities' withdrawal of support for the SNR-300 fast breeder reactor at Kalkar. All radioactive wastes, including spent fuel, are being collected and stored at central storage facilities pending reprocessing of the spent fuel and eventual disposal of the radioactive wastes in deep geologic repositories. Two deep geologic repositories are under development, one for LLW and ILW and one for spent fuel and HLW. Another recent policy change has been the establishment of the new FRG Ministry of the Environment, Protection of Nature, and Reactor Safety (BMU) with responsibility for nuclear safety and radiation protection. This change is resulting in realignments of responsibilities among the numerous organizations involved in radioactive waste management in the FRG.

5.1.1 Nuclear Power Policy

The 21 commercial power reactors, primarily LWRs, in the FRG supply about 35% of the nation's electric power. Key directions in the FRG's nuclear plans include an increase in electricity production by nuclear power, use of reprocessing of spent fuel with recycling of the recovered plutonium for power production, interim storage of radioactive waste pending disposal, and use of deep geologic repositories for disposal of all radioactive wastes (Schneider et al. 1988). Growth of nuclear power generation has been slowed, however, with no plans for new stations, abandonment of the 300 MW THTR, and a decision by FRG utilities to discontinue support for the fast breeder project at Kalkar (Leigh 1989).

The FRG's nuclear plans include interim storage for spent fuel at reactor (AR) and for spent fuel and HLW storage at away-from-reactor (AFR) sites. Though delayed by public opposition, central storage sites are under construction or operating at Ahaus, Gorleben, and Wackersdorf. The recent redirection from reprocessing within the FRG to sending FRG fuel to France and the U.K. for reprocessing has simultaneously increased German interest in direct disposal of spent fuel. An indicator of that interest is the recent announcement of funding for the PKA spent fuel conditioning facility at Gorleben (Nuclear Fuel 1988; Nuclear News 1989b).

Disposal of all the FRG's radioactive waste will be in deep geologic repositories. Vitrified HLW and spent fuel not to be reprocessed will be emplaced in a salt repository at Gorleben, near the border with the German Democratic Republic. Startup of that repository is expected around 2008 (Schneider et al. 1988).

5.1.2 Major Legislation

The major legislation covering spent fuel and HLW management is contained in the federal constitution and in the Atomic Energy Act of 1959 and their subsequent amendments. The federal constitution defines the roles of the federal government (Bund) and the individual states (Lander). It generally declares that the federal authority takes precedence over state authority and that the states may legislate in the nuclear field only as far as the federal authority has not acted. The Atomic Energy Act of 1959 empowers the federal government to issue ordinances concerning licensing of nuclear materials and operations. These ordinances are largely implemented by the individual states as agents of the federal government (Schneider et al. 1988).

An important provision of the Atomic Energy Act of 1959 gives priority to recovery of fissile materials from spent fuel over disposal of spent fuel; direct disposal of spent fuel is permissible only for fuel for which reprocessing is neither technically feasible nor economically justifiable (Schneider et al. 1988). Also, the utilities are required to demonstrate provisions for spent fuel management in order to obtain a license to operate a reactor. Before the first partial operating license is granted, the utility must demonstrate that the safe location of spent fuel is assured for six years after plant commissioning. The original program for closing the fuel cycle was issued in 1974 (IEAL 1987). A resolution passed in September 1979 by the heads of the state and federal governments confirms the integral waste management concept; it is based in general on onsite and offsite interim storage, followed by reprocessing of spent fuel, recovery of radioactive materials for reuse, and conditioning of the radioactive waste for disposal (Peehs 1988).

5.1.3 Linkage Between Nuclear Power and Waste Management

A stipulation-type law links licensing of nuclear power plants to an adequate demonstration of spent fuel management, e.g., reprocessing and interim storage. Safe location of spent fuel for six years after plant commissioning must be assured before the first partial operating license is granted (IEAL 1987).

5.1.4 Policy on Spent Fuel and Waste Management

The current spent fuel and waste management policy is based on the Atomic Energy Act of 1959 and a waste management resolution passed in September 1979. The policy emphasizes reprocessing but permits direct disposal of spent fuel for which reprocessing is not technically or economically justifiable. The integral waste management concept is based on onsite and offsite interim storage, followed by reprocessing of spent fuel, recovery of radioactive materials for reuse, and conditioning of the radioactive waste for disposal.

There are several key elements in the policy. An increase in spent fuel burnup is sought to improve economics, to conserve resources, and to minimize waste volumes. Spent fuel is to be stored in reactor pools for 5 to 10 years in compact racks. AFR dry storage facilities (using metal casks) are to be provided for spent fuel, when needed, until reprocessing or disposal is accomplished. Spent fuel is to be reprocessed 7 to 10 years after reactor discharge; recovered plutonium and uranium are to be recycled to light-water and breeder reactors. Plans to build a reprocessing plant at Wackersdorf were cancelled in 1989. Instead, reprocessing contracts have been or are being negotiated with French and British sources. HLW is to be vitrified and then stored (typically at least 10 years) until the repository is available, in about the year 2008. Deep geologic disposal of all radioactive wastes is to be practiced. Reprocessing wastes are to be placed in salt domes, which can accommodate all types of waste. Non-heat-producing (LLW/ILW and decommissioning) wastes are expected to be emplaced in the Konrad iron mine facility and in the Gorleben repository. Tritium-containing water is to be pressure injected into deep geologic strata or resolidified in cement-bentonite for disposal in the Konrad facility. Development of technology is to continue for direct disposal of spent fuel as an alternative and for disposal of certain spent fuels (20% of all spent fuel) that are not planned to be reprocessed (Schneider et al. 1988; Nuclear News 1989b).

5.1.5 Organization/Responsibilities for Waste Management

There is no single central authority in the FRG in which all executive responsibilities related to nuclear energy are vested. The constitution of the FRG contains detailed provisions on the legislative and administrative authorities of the federal government and the states (Schneider et al. 1988).

The responsibilities for spent fuel management and waste disposal are divided among the federal government, the states, and the utilities. The federal government is to coordinate the FRG nuclear program, sponsor R&D, build and operate radioactive waste disposal facilities, and set licensing rules. The states are to license nuclear installations, while the utilities are to provide for spent fuel and reactor waste storage, reprocessing, and waste treatment, as well as to pay for waste disposal. The FRG waste disposal system is financed under the concept that the waste producer pays the costs (IEAL 1987).

There are several key organizations in the FRG for management of radioactive wastes. The BMFT (Federal Ministry for Research and Technology) is the cognizant federal ministry for R&D on radioactive waste management. The BMU (Federal Ministry for Environmental Protection and Reactor Safety) is responsible for storage, transportation, and disposal of radioactive wastes. It is also responsible for nuclear safety and radiation protection and for supervision of state licensing procedures (ERCE 1989; Leigh 1989; Schneider et al. 1988).

In 1989, an Office for Radiation Protection (BfS) was established under the BMU with responsibility for licensing transport and storage of waste, for constructing and operating waste repositories, and for conducting nationwide radiological monitoring. It has assumed the radioactive waste-management functions formerly assigned to the PTB (Federal Science/Engineering Institute) under the Ministry for Economics (BMWI) (ERCE 1989; Nuclear News 1989a).

The RSK (Federal Reactor Safety Commission) and the SSK (Radiation Protection Commission) issue licensing requirements on behalf of the BMU (Schneider et al. 1988).

The German Fuel Reprocessing Company (DWK), established and funded by the nuclear power utilities, was previously responsible for spent fuel management, including reprocessing, and for radioactive waste storage and treatment. (Due to the recent demise of reprocessing in the FRG, DWK's responsibilities are being transferred to others.) Its responsibilities formerly included the spent fuel AFR storage facilities at Gorleben and Ahaus, HLW vitrification R&D, and operation of the German reprocessing pilot plant at Karlsruhe (to be decommissioned) (Schneider et al. 1988).

The Company for Construction and Operation of Waste Disposal Facilities (DBE) is a consortium of mining companies reporting to the BfS and is responsible for repository construction and operation. The Company for Radiation and Environmental Research/Institute for Underground Storage (GSF/IfT), under the BMFT, is to manage the waste disposal R&D program and operate the Asse mine facility (Schneider et al. 1988).

While the utilities remain legally responsible for waste management, the current federal Environment Minister has imposed a plan for reorganizing the FRG industrial participation in the back-end of the fuel cycle. Specifically, competition for back-end services has been eliminated. A new subsidiary of the federal railway, Nuclear Cargo and Service, is now a monopoly transporter of spent fuel and radioactive wastes in the FRG. The firm GNS (Company for Nuclear Service) now holds a monopoly on waste treatment. The DWKs' share of the spent fuel AFR facilities at Gorleben and Ahaus will be taken over by GNS (ERCE 1989).

5.1.6 Repository Licensing Process, Regulation Status, Major Regulations, Safety Bases (ERCE 1989; IEAL 1987)

The initial license application for a HLW repository is not necessarily site specific. The applicant, either DWK (pre-repository, including interim storage and reprocessing) or BfS (repository), proposes sites for the waste facility within a particular region. They submit an application and a technical evaluation to the local town or district council, which then agrees or disagrees with the location of such a facility within its boundaries. This local approval, a requirement for the general legal procedure, is also useful in determining the types of acceptance problems the applicant may encounter later on. The local authority must have a valid reason for rejecting a project, so the applicant has an early opportunity to alter the plans accordingly or to abandon the project if the opposition appears insurmountable. If there is agreement at the local level, the site is selected by the applicant, the licensing application documents are completed, and the formal authorization procedure begins.

The site-specific application is initially submitted to the responsible state authority, the "Supreme Land Authority," as designated by each respective state. This authority immediately forwards a copy of the application to the federal Minister of Environment so that federal review proceedings occur simultaneously. A copy of opinions from all involved local entities, such as planning authorities, building authorities, nature conservancy authorities, etc., is also sent to the federal Minister of Environment. The federal review is primarily a technical/safety review, while the state review is much broader, covering the more specific aspects of the locality and their relationship to the proposed facility.

On the state level, the review process begins when the state licensing authority commissions independent experts to evaluate the acceptability of the licensing documents. These experts are generally appointed from one of the eleven regional Technical Inspectorates and/or from the Company for Reactor Safety, which are both private consulting associations under government supervision. They are officially recognized as competent authorities for supervising technical safety matters on behalf of the states.

If the documents satisfy the expert panel, i.e., they do not require modification or additional information, the state authority informs the public with respect to the project applied for, provides for public inspection of the licensing documents, convenes a public hearing, and then incorporates any objections or observations raised at the hearing into the licensing documents. On the basis of the information available, particularly that obtained from the public hearing, the experts make a recommendation on the proposed project. This recommendation is also attached to the licensing documents.

A similar review is conducted simultaneously at the federal level. The license application is reviewed primarily by the Reactor Safety Commission (RSK) and the Radiation Protection Commission (SSK) on behalf of the Minister of Environment. The Minister of Environment may also request the advice of other federal authorities. After having examined these opinions, and the state licensing authority's files, if desired, the Minister of Environment may issue a directive to the state licensing authority to license the facility. This directive is legally binding in terms of the final decision; however, such a directive overriding a state government decision has so far been avoided.

In practice, the final decision is made by the state licensing authority, based on the conclusions drawn from all proceedings. All citizens or groups directly affected by that decision may challenge it in the local administrative court. Thus, both the applicants and the objectors may appeal unfavorable decisions to a local court which, in this event, inherits decision-making authority. Furthermore, local court decisions can be challenged in the State Appeals Court and finally with the Federal Administration Court.

Although involving both federal and state authorities, the formal licensing procedure for an interim spent fuel or HLW storage facility (AFR) is simpler than the above procedure, because the spent fuel is not altered in any way at the temporary location. In this case, DWK applies to the local authorities of the state for a construction permit. At the same time, DWK also applies to the state for an operating license. The legislation establishing the BfS also requires a public hearing during the spent fuel storage licensing process. Public hearings are also required for significant design changes in a facility. Pending the advice of the state authority, the community associations, and the BfS, the state Prime Minister issues or does not issue the license. This final decision may also be challenged in the local administrative court by any or all parties affected by the decision, with the possibility of appeal.

In the FRG, partial licenses (first, second, and third stages) are granted for construction of a nuclear installation. At each stage, the competent authorities also examine the facility's overall capabilities and vulnerabilities, e.g., its siting, financial considerations, public acceptance, and whether it is in the national and regional interest. Depending on the construction or operating phase, the repositories must be licensed by mining and nuclear authorities. For example, Gorleben excavation is being conducted under the Mining Law, but if there is a decision to use Gorleben as a repository, it will have to be licensed under the Atomic Energy Law. However, the Federal Administrative Court ruled in 1988 that construction of all parts of a nuclear facility have to be licensed in the course of nuclear licensing procedures and cannot be built under construction permits issued by the local authorities.

5.1.7 Roles of the Public, Local Organizations, Multinational Organizations

Public involvement is facilitated by conducting an extensive public information campaign, with visitors' centers maintained at major waste management sites, and by holding public hearings on license applications. Local council and state authorities have veto rights, which can be overridden by the national Minister of the Environment. Strong public dissent has been a factor in several delays and changes in deployment of FRG waste management technology.

The FRG is an active member in the IAEA, the OECD/NEA, and the CEC, and participates in their waste management activities (Schneider et al. 1988).

5.2 OVERALL WASTE MANAGEMENT SYSTEM

The FRG spent fuel and HLW waste management concept involves five major elements. These are 1) interim storage of spent fuel at nuclear power plants and in offsite interim storage facilities, 2) reprocessing of spent fuel and reuse of the nuclear fuel thus recovered in nuclear power plants, 3) development of direct disposal for spent fuel for which reprocessing is not technically feasible nor economically viable, 4) conditioning and intermediate storage of HLW in interim storage facilities, and 5) disposal of spent fuel and HLW in a deep geologic repository (Peehs 1988).

5.2.1 Reprocessing

The FRG policy is to close the nuclear fuel cycle by reprocessing. The proposed recycling strategy uses MOX fuel assemblies for recycling plutonium, preferably with reprocessed uranium as the carrier material. The remainder of the reprocessed uranium is planned to be enriched separately and used for enriched uranium fuel assemblies. In 1980, the German utilities decided to cooperate so that all plutonium that cannot be used immediately in fast breeder reactors is pooled and used for thermal recycling. The large-scale technical and economic feasibility of plutonium use has therefore been demonstrated primarily for PWRs. The program is scheduled to be extended to include BWRs, as well.

There are several principal aspects of the German approach to reprocessing. The FRG had initially contracted with France and the U.K. for reprocessing until they had developed domestic capability (Malting et al. 1985). The Karlsruhe experimental reprocessing plant (WAK) has been in operation since 1971 at the nuclear research center in Karlsruhe (Peehs 1988). Along with the extensive R&D on reprocessing, the WA-350 commercial reprocessing plant at

Wackersdorf was designed and partially completed. The design rate was 350 MT/yr or 2 MT/d. Construction was initiated in 1986 but was abandoned in 1989 because of a cost advantage offered by an agreement with France for a joint reprocessing venture (Nuclear News 1989a). The FRG is also presently negotiating for reprocessing services with the U.K. (British Nuclear Fuels) (Nuclear News 1989b). Concurrently, the incentives for reprocessing of FRG fuel are being reviewed.

5.2.2 Interim Storage Before and After Reprocessing (Schneider et al. 1988)

Wet storage of spent LWR fuel is provided at reactors. Most reactors have storage capacity of 3 to 10 years, but some with less capacity use dry storage at reactors in dual-purpose metal casks. Dry storage of spent LWR and other fuel types in dual-purpose nodular cast iron casks at AFRs at Gorleben or Ahaus is planned but not presently implemented due to litigation.

The transfer of spent fuel to foreign reprocessing facilities will normally occur within about one to five years from discharge. Interim storage of acidic high-level liquid waste (HLLW) to be vitrified is conducted in metal tanks at the WAK reprocessing pilot plant at Karlsruhe. Dry storage of vitrified HLW in metal casks at AFRs is planned but has not yet been implemented.

5.2.3 Geologic Repository (Peehs 1988; Schneider et al. 1989)

The candidate HLW repository site is located at Gorleben in the state of Lower Saxony. The disposal concept involves waste emplacement in a salt dome at a depth of about 800 meters. Above-ground exploration of the salt deposit has been carried out, and below-ground exploration was started in 1986 with the sinking of shaft No. 1. If the Gorleben site is ruled acceptable for a repository, the facility is expected to be operational in about 2008. The current concept for the repository estimates a final inventory of 300×10^9 curies of beta/gamma and 3×10^9 curies of alpha radioactivity. Canisters containing HLW will be stacked vertically in bore holes; spent fuel in Pollux casks will be emplaced horizontally in drifts.

Other FRG geologic repositories include the Konrad iron mine, planned to be used for LLW/ILW disposal; and the Asse salt mine, to be used for a demonstration of disposal and storage of a wide range of LLW/ILW and other R&D activities, including HLW demonstrations.

5.2.4 Other System Considerations

The FRG participated in a Nuclear Energy Agency study of an international repository concept but has taken no formal position on a multinational repository. The FRG also participated in an NEA assessment of seabed disposal, but again, the FRG has not established a formal position. There are no current FRG activities in the areas of waste transmutation and partitioning. The FRG disposal plans for LLW and ILW include some combination of underground repository facilities at Asse, Konrad, and Gorleben.

5.3 WASTE STORAGE AND TRANSPORTATION

FRG spent fuel management plans begin with wet storage in reactor pools. A wet storage AFR also operates at the WAK reprocessing pilot plant at Karlsruhe. In addition, the FRG has taken a leadership role in developing, demonstrating, and licensing dual-purpose nodular cast iron or forged steel shipping and storage casks. The dry storage and shipping concept has been demonstrated at reactor sites, as well as by loading LWR fuel (including assemblies with failed rods) into casks, followed by shipment to and storage at R&D sites. The plan to implement dry storage in metal casks at AFRs is on hold pending outcome of litigation.

5.3.1 Spent Fuel Storage

Spent fuel in the FRG is predominantly from PWRs and BWRs, with a small amount from special reactor types, i.e., thorium high-temperature reactor (THTR); gas-cooled, graphite-moderated reactor (GCR); liquid metal fast breeder reactor (LMFBR); and mixed (plutonium-uranium) oxide light water reactor (MOX-LWR). The German spent fuel arisings by the year 2000 are estimated to be 11,000 MT (Schneider et al. 1988). Approximately 20% of the spent fuel is expected to be the special types identified above.

Onsite storage in pools is used at nuclear power plants for the interim storage of spent fuel generated in those plants. Interim dry storage is planned at several central facilities, using dual-purpose shipping and storage casks.

The total interim storage capacity is composed of both onsite and offsite storage. Onsite storage capacity at nuclear power plants is approximately 5610 MT; of this, approximately 5250 MT have been licensed. License applications have been submitted for a further capacity of approximately 360 MT. A full-core reserve is maintained at every nuclear power plant. In principle, spent fuel trans-shipment between reactors is not allowed.

Offsite dry interim storage facilities having a capacity of 3000 MT, including two 1500 MT facilities at Gorleben and Ahaus, are completed, but usage is delayed by the courts. The Ahaus facility was intended principally for storage of non-LWR fuel; however, a license for LWR fuel has been granted, and a license for storage of THTR fuel is pending (late 1989). One of the principal spent fuel sources, the THTR-300, is being shut down. The Gorleben and Ahaus facilities are essentially identical, comprising large concrete halls, designed for convective air cooling, and sized to accommodate 420 metal storage casks.

German nodular cast iron casks (three designs) provide for combined transport and storage of spent fuel and HLW. The GNS Castor cask has a maximum capacity of 4.8 MTU fuel (9 PWR or 25 BWR assemblies), a weight of 60-100 MT, and a design thermal capacity of 30 to 50 kW. It is licensed in the FRG for both storage and transportation. The Transnuklear TN-1300 cask has a maximum capacity of 12 PWR or 33 BWR assemblies, a weight of 116.5 MT, and a design thermal capacity of 50 kW. The Pollux cask system for combined storage, transport, and disposal of spent fuel is under development by DWK; the loaded weight is 65 MT (Schneider et al. 1988).

5.3.2 High-Level Waste Storage

Liquid HLW at the Karlsruhe experimental reprocessing pilot plant is stored in stainless steel tanks. In 1987, a new storage facility (LAVA) replaced a prior installation used for 15 years. It has two 60-cubic meter storage tanks for concentrated wastes having up to 1100 Ci/L (Schneider et al. 1988).

The current plans for interim storage of vitrified HLW are to encapsulate the vitrified waste in metal canisters, followed by shipment and storage in dual-purpose casks at the Gorleben AFR. The first HLW from French reprocessing of FRG fuel is expected to be returned to the FRG in about 1992.

5.3.3 Transportation

The FRG has combined the requirements of transport and dry storage for irradiated fuel assemblies into its cask designs. Germany's concept for direct disposal of spent fuel involves a "Pollux" triple-purpose cask container for interim storage, transport and disposal (see later section on Waste Package System) (Schneider et al. 1988).

The two suppliers of FRG dual-purpose shipping and storage casks are GNS and Transnuklear (TN). The GNS casks are manufactured from nodular cast iron. The TN cask material is forged steel. The cask weights are 80 to 120 MT. The casks have a range of capacities, up to 26 PWR or 52 BWR assemblies. The Pollux cask has a steel body, with a steel and polyethylene overpack. The composite loaded weight is 65 MT.

Shipping and storage cask technology has been applied to spent LWR fuel and has also been demonstrated for spent research reactor fuel, spent fast breeder fuel and spent high-temperature gas-cooled reactor (HTGR) fuel. The designs of casks of the Castor and TN-1300 series are based on Type B licensing criteria as established by the IAEA. For storage purposes, additional design features for protection against external and internal events (e.g., aircraft crashes) include the two barrier cover concept with interspace pressure monitoring; sufficient passive decay heat dissipation capability; the fuel integrity concept, in which storage conditions prevent fuel failure; and the activity retention capability concept, which assumes 100% fuel rod failure (Peehs 1988).

Plans are underway for shipping up to about 100 cubic meters of liquid HLW from the German WAK reprocessing pilot plant at Karlsruhe to the Pamela vitrification facility in Mol, Belgium. Shipment would occur in Castor-type casks by barge and rail, but no shipments are expected to occur before about 1992. The shipments would be funded 50% by DWK and 50% by the FRG federal government (Schneider et al. 1988).

5.4 GEOLOGIC WASTE REPOSITORY

The FRG disposal plans include three underground facilities: the Asse salt mine for LLW and ILW disposal and for HLW R&D, the Gorleben salt dome for characterization as the HLW repository, and the Konrad iron mine for LLW and ILW disposal. The FRG is presently focusing on the Gorleben HLW repository; the facility is scheduled for characterization by the mid-1990s, and if judged to be acceptable, for commissioning as a repository by about 2008. If Gorleben is not accepted, another site will be characterized (Schneider et al. 1988).

5.4.1 Safety Requirements and Approach

Repository safety is based on the concept of multiple barriers: waste form (the waste canister for HLW is not considered a barrier), repository, and overall geologic environment. Each barrier should provide "appropriate" isolation. The waste form and canister should provide the necessary level of containment during transport and handling.

The integrity of the container around the waste form canister may be required until near-field convergence of salt rock has encapsulated the waste package (from a few years for the borehole disposal concept to about 200 years for the disposal drift concept).

The maximum allowable dose to the most exposed member of the public for unavoidable occurrences is 30 mrem/yr; this applies to both the operational phase and to the post-closure period. The maximum allowable dose to workers is the same as for reactor operations, or 5 rem/yr whole body dose (Schneider et al. 1989).

5.4.2 Siting

The German preference for salt dome geology for the HLW repository was strongly influenced by early disposal testing in the Asse Salt Mine (1967 to 1978). In the 1970s, the Gorleben site in Lower Saxony was selected for development of an integrated nuclear technology center, including reprocessing and waste disposal, largely because of its proximity to a large salt dome. The site was offered by the Lower Saxony government for the center, and site investigation began in 1979. Opposition to the integrated site concept resulted in scaling back the scope by eliminating the reprocessing plant, but a dry interim storage AFR facility for spent fuel and HLW, a LLW interim storage facility, and the leading candidate for an FRG HLW repository survived. Between 1979 and 1984, 590 boreholes were drilled for characterization of the Gorleben site, including examination of hydrogeology, exploration of the salt dome, and construction of shaft pilot boreholes (Grübler and Pitz 1985).

The major ongoing activity is excavation of two shafts, each 7.5 meters in diameter, the first to a depth of 800 meters and the second to a depth of 940 meters. Drilling of the first shaft began in 1986 but stopped in 1987, due to a fatal accident caused by dislodging of a shaft support ring at the 239-meter depth. Drilling resumed in 1989 and will continue to a depth of about 800 meters (Hibbs 1989). Excavation of the shafts is expected to be completed in 1992.

Site characterization is continuing. The geometry and composition of the dome and protective clay overburden have been approximated. A major focus of site characterization is to determine the existence and extension of the main anhydrite formation to determine the location of the disposal area. Four deep boreholes (2000 meters) were drilled into flanks of the dome; 300 boreholes were drilled to investigate groundwater, and 50 holes were drilled into the top of the dome (Schneider et al. 1988).

The exploratory level at 840 meters (30 meters above disposal level) will have a total of 25 kilometers of drifts and 50 to 60 kilometers of boreholes.

Starting from shafts, an 18-square-kilometer area is to be investigated by constructing tunnels, galleries, and other drillings.

Meanwhile, other FRG field tests are underway at the Asse Salt Mine, the Hope Potash Salt Mine in northern Germany, and the Grimsel (Switzerland) rock laboratory. The FRG program includes keeping abreast of nonsalt geologic studies in other countries that have potential relevance to FRG geologies.

If site characterization (scheduled for completion by about 1997) at Gorleben reveals that the site may be unsuitable, another site will be selected and characterized (Schneider et al. 1988).

5.4.3 Design Concept(s)

Reference wastes will consist of three principal types: 1) vitrified HLW with about a 15% waste oxide content; 2) off-standard spent fuel elements, including HTGR fuels, second cycle MOX fuels, LWR fuels with high burnup (higher than licensing limit of 50,000 MWd/MT), or LWR fuels beyond fuel reprocessing capacity; and 3) other heat-producing wastes.

In the reference design concept, the underground structure in the salt dome will be at a depth of 870 meters; the repository is to occupy an area 5 km x 1 km in size. The repository will have two wings, one for final disposal of heat-producing wastes and the other for nonheat-producing wastes. HLW canisters are to be stacked vertically in boreholes (300 meters deep and 50 to 60 meters apart) in the gallery floor. Canistered fuel rods and other heat-producing wastes will also be emplaced in boreholes. The cavities are to be backfilled with salt, and each chamber will be sealed off from other chambers with clay and concrete dams. Maximum temperatures will be 150 to 200°C at the salt/canister interface (130°C if other minerals are present).

The FRG program includes development of technology for spent fuel disposal, both for off-standard spent fuel and as an alternative to vitrified HLW. The spent fuel disposal concept has the following features. The underground structure in the salt dome will be similar to that for the reference concept. Packaged, intact THTR spent fuel elements or consolidated LWR spent fuel rods in transport-storage-disposal casks ("Pollux" casks) will be emplaced horizontally in small rooms and galleries mined out of salt. Emplacement will be carried out in a retreating mode. The concept involves early filling and closing of emplacement spaces.

The Gorleben repository design concept does not specify use of a buffer material. The backfill will be compacted salt from the shaft and drift excavations.

5.4.4 Retrievability and Monitoring

Postclosure retrievability is not considered part of a viable waste disposal concept in the FRG. Retrievability ends once an emplacement gallery has been backfilled and sealed. Long-term retrievability is believed to compromise achieving safe isolation of radioactive waste from the biosphere. In particular, intentional retrievability in a salt repository is believed to be counter-productive because, due to pronounced creep and high temperatures, canisters in boreholes will be completely encapsulated by surrounding salt within a matter of months (IEAL 1987).

During the operation of a repository, monitoring will be required to assess the exposures resulting from the radioactively contaminated air and water that are discharged to the environment. This monitoring will be nuclide-specific both within the repository system and at the land surface. For the post-closure period, specific monitoring requirements were not found.

Human intrusion is not considered to be a matter of major concern. It is acknowledged that inadvertent human intrusion can occur due to loss of documentation on the repository. However, the FRG authorities conclude that human beings who mine to depths in excess of 1000 meters in the far future can be assumed to be capable of detecting radiation (IEAL 1987).

5.4.5 Waste Package System (Schneider et al. 1989)

The reference HLW waste package contains 150 liters of borosilicate glass (about 500,000 curies) in a 180-liter stainless steel canister (0.43 meters in diameter, 1.35 meters tall, and with a 2.5 kW heat output). The HLW canister will be the same as the French and U.K. HLW canisters. Vitrified waste canisters will contain waste originating from about 1.2 MT of spent fuel. The

French HLW canister weighs 70 kilograms, and the glass contents weigh 360 to 400 kilograms. It has a wall thickness of 5 millimeters, with no corrosion allowance. The French HLW canister is made of a special stainless steel designated Z 15 CN 24-13. No information was found on the life expectancy for either the French or the U.K. canisters for HLW.

For direct disposal of spent fuel, the waste package will include consolidated spent fuel rods, intact spent fuel assemblies, or chopped spent fuel pins; the canister/cask; and the shielding overpack for gamma and neutrons (Schneider et al. 1989).

In the reference waste package concept for direct spent fuel disposal, consolidated rods from eight PWR assemblies are encapsulated into a thin-walled canister. Canisters are loaded into the perimeter of the Pollux disposal cask, and compacted fuel hardware (also contained in a canister) is loaded into the center of the cask. The disposal cask is sealed by welding; then it is inserted into a shielding overpack for transport. The package can handle spent fuel out-of-reactor for three years with a total heat generation rate of 20 kW.

The sealed Pollux cask and the overpack of polyethylene-lined nodular cast iron (not leak tight) will be an integral unit and will also serve as the disposal container. The Pollux cask and overpack has an expected life of about 500 years in salt. It must withstand the lithostatic head of about 300 atmospheres. The design basis temperatures are 300°C maximum for emplaced spent fuel and 150 to 200°C maximum surface temperature for the disposal package. The design surface dose rate is less than 20 mrem/h. The empty waste package (cask and overpack) will weigh 57.8 MT, and the loaded cask will weigh 65 MT. The thick-walled cylindrical waste package will be 0.154 meters outside diameter by 5.46 meters long.

The reference Pollux cask will be made of 15 Mn-6.3 Ni steel (15 centimeters thick) with an external coating of Hastelloy C4 (8 millimeters thick). The overpack will be made of ductile cast iron (GGG 40) lined with polyethylene (100 millimeters thick) for neutron shielding.

An alternative concept for direct spent fuel disposal is disassembled fuel pins from half of a PWR fuel assembly cut into pieces about 1 meter long and placed in the same canisters as vitrified HLW (0.43 meter in diameter and 1.33 meters long). Emplacement would be in boreholes, as with HLW (Schneider et al. 1988). The alternative concept canister would weigh about 1.18 MT when loaded and would have a surface dose rate of about 1000 mrem/h. The alternative concept would have canister walls that are 50 millimeters thick, with a coating of corrosion-resistant material.

Low-level waste and ILW disposed in the HLW repository will be contained in waste packages that are cylindrical concrete or cast iron containers; sheet metal, concrete or cast iron box-type containers; and standard 200- and 400-liter drums in sheet metal containers. Heat-producing ILW will be emplaced in 0.9-meter diameter boreholes at a depth of 300 to 600 meters. The temperature at a borehole is to be less than 100°C. Low-level waste containers are to be emplaced in a salt cavern by top loading, tumble down, or remote stacking (Schneider et al. 1988).

5.4.6 <u>Research and Development</u> (Schneider et al. 1988)

Waste management R&D is conducted at the Karlsruhe and Jülich Nuclear Research Centers, Hahn-Meitner Institute, Nukem, Alkem, and supporting institutions. Dry storage cask test activities have included mechanical, thermal, and shock tests on casks; behavior of intact and defective spent fuel rods in inert atmospheres; full-scale demonstrations of spent fuel storage in dry casks; and computer modeling of cask performance. Information will be continually gathered on disposal R&D in nonsalt rocks in other countries. Waste package R&D (a limited emphasis will be placed on the container as a barrier) includes investigations of properties of HLW borosilicate glasses, corrosion of metallic and ceramic materials in brines, and leaching of spent fuel. Laboratory and theoretical geosciences R&D studies cover mechanical and thermal properties of rock salt, deformation of cavern and borehole walls, release of liquids and gases from salt, radionuclide migration as a result of repository flooding, retention of fission gases by backfill materials (for spent fuel disposal), and development of rock mechanical computer codes. The FRG participates in the CEC Natural Analog Working Group and conducts relevant studies. Performance assessment work currently underway is summarized in the following section.

Field tests are conducted at the Asse Salt Mine (underground test program), the Hope Potash Salt Mine (flooded mine test), the Grimsel (Switzerland) facility (FRG test program--discontinuities, stresses, water flow in rock mass, tiltmeter measurements, and ventilation), and at Gorleben (underground exploration of proposed repository location).

In 1986, DWK submitted an application for a license to construct and operate a pilot conditioning plant (PKA) at Gorleben to develop conditioning techniques for direct disposal of spent fuel. The facility will have the following capabilities. The hot cell will have equipment for rod consolidation, compaction of fuel assembly skeletons, and loading of canisters into the Pollux cask system. The maximum throughput will be 35 MTU/yr. It will also have the capability for unloading of vitrified HLW from transport to storage casks and for performing maintenance for transport and storage casks. The facility will also demonstrate handling and transport, and possibly the simulation of emplacement of canisters at a repository. Its startup is planned for 1994 (Schneider et al. 1988). DWK expects a license in late 1989, but licensing may be postponed until after the 1990 state elections (Nuclear News 1989b).

5.4.7 Approach To Proving the Safety of the Repository (Schneider et al. 1988)

The radiation protection objectives serve as de-facto performance objectives for a specific repository. The site-specific safety analysis defines the limiting criteria on radionuclide inventory and waste package leakage, consistent with meeting the overall radiation protection objectives. A five-year study (1980 to 1984) compared safety aspects for direct disposal of spent fuel versus reprocessing and disposal of HLW; the conclusions indicated that both approaches can be implemented safely. A seven-year project is underway to develop and test safety assessment methodology for the back end of the fuel cycle.

Early performance assessments have been carried out for each of the three FRG disposal sites: 1) the Asse salt dome, used for R&D and LLW and ILW disposal; 2) the salt dome near Gorleben, being investigated for the HLW repository; and 3) the Konrad abandoned iron mine in sedimentary rocks, planned for LLW and ILW disposal.

Site-specific studies addressing the salt domes have focused on identifying possible pathways for water migration in and/or out. All of these pathways are highly unlikely because the expected case is that these salts are and will remain dry. For Gorleben, a postulated pathway is anhydrite passageways from the salt dome surface to the repository horizon, allowing relatively rapid brine migration. For the Asse mine, another highly unlikely scenario being investigated is the intrusion of water from overlying strata. Even for these unlikely scenarios, the calculated doses for the 10,000-year regulatory period are either zero or well below the regulatory limit of 3×10^{-4} Sv/yr. The Konrad site is expected to become resaturated from surrounding rock, and the calculated 300,000-year water travel time to the nearest discharge zone is the major line of performance assessment defense for the undisturbed case.

In 1985, the Project Sicherheitsstudien Entsorgung presented a preliminary deterministic safety assessment for a salt-dome repository at Gorleben. The salt dome regional model, the mine and its workings, were modeled. The planned inventory was 3% by volume HLW, about 12% ILW, and the rest LLW. Waste totals were on the order of 1000 MT.

Current performance assessment-related activities center on obtaining data during the shaft sinking at Gorleben and continuing in situ experimental work at the former salt mine at Asse. Asse experiments have produced much of the data and conceptual approaches that will be used to model the Gorleben site. After shafts are sunk at Gorleben and underground exploration and development are complete in the mid-1990s, a safety assessment will be prepared for the license application. The regulatory period for which safety must be demonstrated is 10,000 years. Conceptual safety assessments for LLW and ILW sites have recently been completed.

To date, the FRG assessments appear to rely more on deterministic rather than probabalistic approaches. The approaches can be characterized as conservative and seem to be bounding. A strong approach to validation is not yet evident but may emerge as the status of acceptance of Gorleben as a repository site is better defined. There has been some FRG participation in the CEC Natural Analog Working Group (Côme and Chapman 1989).

5.4.8 Peer Review Activities

Through bilateral working agreements and joint research projects with a number of other countries and through international organizations, the FRG cooperates in the development of methods and processes for safe geological disposal of spent fuel and HLW. Currently, the FRG approach does not include formal international peer review of its HLW management methods.

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6.0 JAPAN

6.1 LEGISLATIVE/INSTITUTIONAL BASES

The following sections discuss Japanese policy, regulations, and organizations that pertain to spent fuel and high-level waste (HLW) management.

6.1.1 Nuclear Power Policy (OECD/NEA 1988; Schneider et al. 1988; Leigh 1989)

Japan has a strong nuclear power program to lessen its dependence upon foreign energy sources. About 32% of Japan's electricity is generated by nuclear power. The nuclear power capacity of Japan is about 26 GWe and is anticipated to grow to 53 GWe by the year 2000. Japan currently operates 19 BWRs, 16 PWRs, one FBR, one GCR, and one HWR; 12 additional nuclear power reactors are scheduled to be on line by 1995.

Development of a complete fuel cycle capability, including uranium enrichment, fuel fabrication, spent fuel reprocessing, waste treatment and storage, and waste disposal, is well underway. However, Japan is presently largely dependent upon foreign services for enrichment and reprocessing.

6.1.2 Major Legislation

Nuclear activities are governed by a series of laws, each dealing with a specific sector, such as nuclear installations and radioactive material, radiation protection, and compensation for nuclear damage. Most of these laws were enacted under the Atomic Energy Basic Law of 1955. This law provides the basic legal authority for the Prime Minister of Japan and his cabinet agencies to perform and manage the various nuclear regulatory activities (IEAL 1987).

In May 1986, the Japanese Diet approved an amendment to the Nuclear Regulations Law ("The Law for the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors," enacted in June 1957). With this new amendment, regulatory procedures for organizations involved in radioactive waste management (including interim storage of HLW) have been set up (OECD/NEA 1988; IEAL 1987).

6.1.3 Linkage Between Nuclear Power and Waste Management

No stipulation law exists that links reactor licensing to final disposition of HLW, but reactor owners must identify the means for reprocessing of the spent fuel, either in Japan or abroad (Schneider et al. 1988).

6.1.4 Policy on Spent Fuel and Waste Management

The policy in Japan is to reprocess spent fuel and recycle the recovered plutonium and uranium. High-level waste from reprocessing will be vitrified, stored for 30 to 50 years for cooling, and then disposed of in a domestic deep geologic repository (OECD/NEA 1988).

6.1.5 Organization/Responsibilities for Waste Management (Schneider et al. 1988; IEAL 1987; Leigh 1989; OECD/NEA 1988)

The organization of Japanese nuclear activities is complicated, involving numerous government ministries, government agencies, institutes, and industrial organizations. The Japanese federal government funds nuclear R&D and is responsible for waste management regulation and HLW disposal. Industry handles the commercial fuel cycle, is responsible for TRU waste and LLW disposal, and pays for HLW disposal.

The ultimate authority for regulating nuclear energy in Japan is the Prime Minister and his Office. The Science and Technology Agency (STA) was established as an extra-ministerial agency of the Prime Minister's Office, to provide technical support for the development of regulations. In addition to its regulatory activities, the STA is responsible for nuclear R&D activities. The STA is organized into a Nuclear Safety Bureau (NSB) and an Atomic Energy Bureau (AEB), which provide technical support to the Atomic Energy Commission (AEC) and the Nuclear Safety Commission (NSC).

The AEC, the NSC and the Radiation Council report directly to the Prime Minister. Long-term planning for the overall nuclear program is overseen by the AEC, while safety is overseen by the NSC. The Radiation Council establishes technical standards for radiation protection.

Nuclear power reactors are licensed by the Ministry of International Trade and Industry (MITI). Research reactors and fuel cycle facilities are licensed by the STA. It has not yet been decided whether the STA or MITI will have the responsibility to license a HLW repository.

The Ministry of Transport, together with the STA, has jurisdiction over the regulation of transport of radioactive materials and nuclear fuel.

Development of fuel cycle technology is carried out by the Power Reactor and Nuclear Fuel Development Corporation (PNC), which is a semi-governmental organization reporting to the AEB of the STA. The PNC organization carries out R&D pertaining to the vitrification, storage, and disposal of reprocessing wastes. It operates a pilot reprocessing facility and is constructing a vitrification facility and storage facility for the vitrified wastes at Tokai. It also does the field surveys necessary to select candidate disposal sites and performs R&D on geological disposal techniques and performance assessment.

Safety research, including evaluation of geological disposal safety, is performed by the Japan Atomic Energy Research Institute (JAERI), a semi-governmental organization under the STA. The institute also performs research on advanced waste management technologies, such as subseabed disposal and partitioning of HLW.

Private organizations involved in radioactive waste management are the Japan Nuclear Fuel Industries Company, Inc. (JNFI), established in 1985 and responsible for shallow-land burial of LLW; and the Japan Nuclear Fuel Service Company, Ltd. (JNFS), established in 1980 and responsible for treatment and storage of HLW, LLW, and TRU waste resulting from reprocessing. These two companies were established by the Japanese utility companies. The Central Research Institute of the Electric Power Industry (CRIEPI), the research arm of the utility industry, has performed R&D on spent fuel storage technologies, including dry cask technology (ERCE 1989), as well as studies on waste transportation and storage/disposal of HLW.

The Radioactive Waste Management Center was originally established for investigating sea dumping of LLW. It now performs research on LLW land disposal and tests related to the returned reprocessing wastes of Japanese spent fuel from European reprocessing facilities. Its owners are Japanese industry, MITI, and STA.

6.1.6 Repository Licensing Process, Regulation Status, Major Regulations, Safety Bases (ERCE 1989; IEAL 1987; OECD/NEA 1988)

The licensing procedures for waste repository construction and operation are prescribed in the 1986 amendment to Japan's regulation law ("The Law for the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors"), in which the regulation of waste management activites was included. The licensing procedures are similar to the procedures for siting of nuclear power plants, described in this section. However, initial steps relating to siting examination, including primary public hearings, are not included in the procedures for waste management facility licensing. It has not yet been decided whether the STA or MITI will have the responsibility to license a HLW repository. The advice of the NSC would be required in either case.

Potential repository sites are selected on an informal basis after consultation with local authorities. Generally, a resolution is passed by the local council to "invite" the utility to investigate the site, including drilling and seismic analysis. After the site is investigated and selected, environmental surveys and impact reports are submitted for review by MITI and its Advisory Committee on Environmental Matters. With the consent of the local government, MITI then holds a public hearing. This hearing is a forum for MITI to explain the results of the environmental investigation and to solicit public opinion. On the basis of the hearing, the Electric Power Resources Development Coordination Council reviews the construction plans and reports the result of its review to the Prime Minister's Office. When approved, an application for a construction permit is initiated.

MITI then reviews the application for a construction permit with the assistance of its Advisory Committee on Nuclear Power Technology, emphasizing safety aspects of the plant. MITI then requests inquiries by both the AEC and NSC. The AEC reviews development matters, such as planned performance, economics, and other operational aspects of the plant. The NSC conducts a second safety examination supported by its Committee on Examination of Reactor Safety, and another public hearing is held. This hearing is similar to that conducted previously by MITI; the purpose is to explain the results of the safety studies to the public and to solicit opinions. The findings of the AEC and NSC are reported to the Prime Minister's Office, which in turn directs MITI to grant a construction permit.

An operating license is also required by law for a HLW repository. MITI reviews the application, including the safety report, plant design, construction performance, operating plans, etc., and after final inspection of the completed plant and preoperational tests, it may grant an operating license.

The Japanese facility siting system involves a consensus approach by local entities and the federal government. Outright vetoes by localities are rare. Under Japanese law, the federal government may override a local veto on siting. In practice, however, the Japanese prefer concurrence to the exercising of preemptive powers. Although land ownership is well protected, there are provisions for eminent domain of the federal authority. This law is rarely invoked, however, because the Japanese have a tradition of amicable resolution of such difficulties, which often involves financial settlements.

6.1.7 Roles of the Public, Local Organizations, Multinational Organizations

Since January 1979, public hearings have become a part of the licensing requirements for nuclear plant siting and are expected to be required for repository siting and licensing. In the present system, public hearings are divided into two types: a primary hearing relating to siting and a secondary hearing relating to environmental and safety matters of the nuclear facility. Primary hearings are held by MITI, and secondary hearings are held by the NSC. Secondary hearings are not limited to nuclear power stations, and they may be held on the nuclear fuel cycle facilities planned to be constructed in Rokkasho-mura (ERCE 1989).

Japan is a member of the IAEA and OECD/NEA and actively participates in many of the international joint R&D projects on waste management, such as the Stripa Project and the Alligator Rivers Analogue Project (OECD/NEA 1988).

6.2 OVERALL WASTE MANAGEMENT SYSTEM

In Japan, management of radioactive waste is carried out according to the "Long-Term Program for Development and Utilization of Nuclear Energy," set up by the AEC (most recently published in June 1987) and the policies related to safety regulation authorized by the NSC (OECD/NEA 1988; ERCE 1989; IEAL 1987).

In general, the overall Japanese HLW management program involves the following:

- at-reactor storage of spent fuel for a cooling period of two to three years
- shipment offsite for reprocessing of all spent fuel
- vitrification of the HLW from reprocessing into borosilicate glass
- storage of the glass canisters for 30 to 50 years for cooling
- deep geological disposal of the vitirfied HLW (several hundred meters from the surface) (IEAL 1987).

6.2.1 Reprocessing

The spent fuels generated in Japanese LWRs have been reprocessed in the PNC Tokai pilot-scale plant (0.7 MTUHM/d) and in United Kingdom and French reprocessing plants. Foreign contracts are in place with British Nuclear Fuels Ltd. (BNFL) in the U.K. and with Cogema in France for reprocessing of 4800 MTU of LWR spent fuel and 1100 MTU of gas-cooled reactor (GCR) spent fuel. The HLW from foreign reprocessing will be returned to Japan as vitrified waste for disposal (Schneider et al. 1988).

The first private industrial-scale Japanese reprocessing plant is planned to be constructed at Rokkasho-mura by JNFS and will start operation in the mid-1990s, with a capacity of 800 MT/yr. The amount of spent fuel to be generated is estimated to be at least 1100 MT/yr in the year 2000 and greater than 2000 MT/yr in 2030. Operation of a second domestic private reprocessing plant is planned for about 2010 (OECD/NEA 1989).

6.2.2 Interim Storage Before and After Reprocessing

Spent fuel is stored in pools at the reactor sites for a two- to three-year cooling period and is then stored in pools at the reprocessing plants until it is reprocessed. Japan Nuclear Fuel Service Company Ltd. (JNFS) is planning to construct a storage pool for spent fuel (about 3000 MTU) at the site of the reprocessing plant at Rokkasho-mura. The radioactive wastes generated from reprocessing will be stored for about 30 to 50 years until final disposal. High-level radioactive liquid waste is presently stored at the Tokai pilot reprocessing plant, awaiting vitrification starting in about 1990 (OECD/NEA 1989; Schneider et al. 1988; Leigh 1989).

6.2.3 Geologic Repository

The reference concept for disposal of HLW in Japan is emplacement in a geologic repository. The repository will be in operation after about the year 2030 (Schneider et al. 1989).

6.2.4 Other System Considerations

The Japanese are carrying out basic research on subseabed disposal technology for HLW (ERCE 1989). Japan also participates in the OECD/NEA cooperative program on seabed disposal of HLW (OECD/NEA 1989).

The Japanese are actively pursuing development of partitioning and transmutation for HLW. In 1988, Japan initiated a long-term R&D program on partitioning of long-lived and potentially useful radionuclides from HLW and transmutation of the long-lived radionuclides to short-lived radionuclides. In October 1988, the Japanese AEC submitted a proposal to the OECD/NEA for a cooperative program to study 1) physical and chemical properties of elements generated in the fuel cycle, 2) advanced technologies for separation of elements generated in the fuel cycle, and 3) nuclear physics and engineering of transmutation (Atoms in Japan 1988).

Disposal methods for wastes containing transuranic (TRU) elements will be established, depending on their classification; the definition of TRU waste has not yet been clearly defined (OECD/NEA 1988; ERCE 1989).

6.3 WASTE STORAGE ANO TRANSPORTATION

In Japan, conditioned HLW will arise from 1) vitrified HLW generated at the PNC Tokai reprocessing facility; 2) the returned vitrified waste from Cogema in France and BNFL in the U.K., based on reprocessing contracts; and 3) vitrified HLW to be generated at the JNFS Rokkasho-mura reprocessing facility (ERCE 1989).

6.3.1 Spent Fuel Storage

Spent fuel is stored for cooling in pools in compact racks at the nuclear power reactors for two to three years for cooling until it is transported to foreign and domestic reprocessing facilities. The planned commercial reprocessing facility at Rokkasho-mura will provide pool storage for about 3000 MTU. Research and development is being performed on dry storage technologies by JAERI and CRIEPI (Schneider et al. 1988).

6.3.2 High-Level Waste Storage

High-level waste glass canisters are to be stored in dry, forced-air-cooled systems for 30 to 50 years for cooling before emplacement in the geologic repository. JNFS is planning to construct (starting in 1990) a HLW vitrified waste storage facility at its Rokkasho-mura reprocessing plant site. PNC wastes are to be stored in a proposed PNC Storage Engineering Center at Horonobe-sho.

Vitrified HLW returning from foreign reprocessors (starting in about 1990) are to be stored with domestic vitrified wastes at both the JNFS and PNC reprocessing plant sites (Schneider et al. 19B8; Leigh 1989; ERCE 1989).

6.3.3 Transportation (ERCE 1989; Schneider et al. 1988)

The Japanese legislative technical criteria pertaining to transportation of radioactive materials are consistent with the technical criteria of the IAEA (OECD/NEA 1988).

Spent fuel is transported to foreign reprocessing facilities in France and the U.K. using specially designed ships operated by Pacific Nuclear Transport Ltd. Japan is currently moving approximately 500 MTU annually to

Europe by ship. These ships weigh about 3000 tons and will also be used for transport of vitrified waste returning to Japan from Europe. The casks in use are the British EXCELLOX for shipments to the U.K. and the French TN for shipments to France.

A ship (Hinoura-maru) is also used to transport spent fuel from Japanese nuclear power plants to the Tokai reprocessing facility, using Japanese HZ casks. Spent fuel is also transported overland for short distances in Japan using heavy-haul trucks. A nuclear fuel transport company was formed in mid-1986 to provide transport associated with fuel-cycle activities at Rokkashomura.

The plutonium being returned from French and British reprocessing plants may be transported to Japan by air (Atoms in Japan 1984).

6.4 GEOLOGIC WASTE REPOSITORY (OECD/NEA 1988; Schneider et al. 1988; ERCE 1989)

The geological disposal of HLW will be carried out in four stages: 1) selection of effective host geologic formations; 2) selection of candidate disposal sites; 3) demonstration of the disposal technology at the candidate disposal sites; and 4) construction, operation, and closure of disposal facilities. Operation of the HLW repository is not planned until after 2030.

The first stage was completed in 1985, and Japan is presently performing activities related to the second stage. To provide support for selection of the candidate disposal site(s), activities being conducted in stage 2 include the R&D on geological disposal technology and surveys to evaluate the suitablity of the geological environment.

6.4.1 Safety Requirements and Approach

Regulations for disposal of HLW have not been established to date (OECD/NEA 1988). Although no individual dose limit has been formally established for long-term releases from the repository, consideration is being given to a limit of 0.05 mSv/yr (5 mrem/yr). The time period over which the regulatory criteria for the overall HLW repository should be applied has not yet been established (IEAL 1987).

Performance criteria for a HLW repository have not yet been established. However, the performance assessment methodology is under development, with emphasis being placed on engineered barriers (IEAL 1987).

6.4.2 Siting

The Japanese repository program completed its first stage in 1985 with a determination that there were several suitable geologic formations in Japan for deep geologic disposal. A second stage is now in progress, which is to result in selection of candidate sites by 1995. This phase includes limited site characterization and performance modeling for candidate sites. Development of exploratory, testing, and perfomance assessment techniques is a large part of current second-stage activities. These activities are being carried out in a number of mines and quarries in gabbro, granite, diabase, and tuff. This effort has provided much experience with a variety of rock types. Sites may be selected because they have more than one rock type contributing to waste isolation. The third stage will complete site characterization and will demonstrate the safety of disposal at the candidate site. The fourth stage, starting after the year 2000, will verify and finalize the selected technical approach. Construction, operation, and final closure of the repository will complete this final stage.

6.4.3 Design Concept(s)

The reference HLW disposal concept uses a mined repository in crystalline or sedimentary rock. A preliminary analysis of repository performance in granite or sedimentary rock assumed a depth of about 1000 meters in saturated rock. The depth of the repository should be greater than several hundred meters. Use of bentonite or cement as backfill material is under consideration, and boreholes and rock fractures will be sealed with grout (Schneider et al. 1988). More detailed design concepts are presently being developed.

6.4.4 Retrievability and Monitoring

The concept of retrievability of emplaced wastes has not been addressed in Japan, and no regulatory requirements have been established. Monitoring requirements for the Japanese HLW repository have not been established (IEAL 1987).

6.4.5 Waste Package System

The conceptual waste package consists of borosilicate glass containing 25% waste oxides. The waste canister is constructed of 304L stainless steel. About 300 kg of glass will be contained in each canister. The use of a metallic or ceramic overpack for emplacement is under consideration (Schneider et al. 1988).

6.4.6 Research and Development (ERCE 1989)

Major centers for R&D are operated by PNC and JAERI at Tokai and Oarai. The AEB developed a "5-Year Program for R&D on Geological Disposal of High-Level Radioactive Waste" in November 1986. This program gave R&D topics and their general schedule to establish basic principles on long-term reliability of geological disposal and to identify fundamental specifications for the geological disposal system, taking into account unfavorable siting conditions in Japan, such as earthquakes, relatively small rock formations, etc. Research and development will be performed to 1) establish the multi-barrier system appropriate for the Japanese geological environment; 2) demonstrate the feasibility of design, construction and closure of the repository; and 3) establish site characterization technology. The Storage Engineering Center project, planned for construction at Horonobe, will have an important role in establishing geological disposal technology.

No underground research laboratory (URL) has yet been established, and no major drilling investigations for site selection have been carried out. Research will be conducted for two rock types, crystalline and sedimentary. Therefore, a URL for each rock type is considered to be necessary. A URL is planned for construction in sedimentary rock at Horonobe, and a URL will be developed at the final candidate repository site (OECD/NEA 1988; Schneider et al. 1988).

6.4.7 Approach to Proving the Safety of the Repository

Current Japanese performance assessment efforts are focused on developing suitable approaches and tools. Computer codes, some obtained through the NEA data bank, are being tested, and selected codes will be modified to allow their use in a total waste management systems code. Performance assessment modeling complements the Japanese experimental program. A probabalistic system assessment code named MGRAT03/CIRCLE is being developed by JAERI (Sasahara 1989).

Because of limited land area and a high rate of tectonic and seismic events, the Japanese repository program is emphasizing the performance of man-made engineered barriers in addition to natural barriers. These engineered barriers include the borosilicate glass waste form, metallic canister, a possible overpack, and tailored clay or cement backfill/buffer material. Performance assessment codes for this near-field of engineered barriers is a key area for development in the current second stage of the Japanese repository program.

The Japanese approach to radionuclide release modeling is currently based on the use of batch water/crushed-host-rock interaction data for selected radionuclides. To determine the usefulness of this laboratory data in modeling long term in situ radionuclide behavior, a uranium-series disequilibrium investigation is in progress at the sedimentary rock Tono Uranium Deposit to determine distributions between in situ host rock and groundwater. The role of groundwater geochemical conditions on the behavior of radionuclides is also being investigated at the Tono site.

6.4.8 Peer Review Activities

No information was found on peer review activities for the Japanese repository program.

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7.0 SWEDEN

7.1 LEGISLATIVE/INSTITUTIONAL BASES

7.1.1 Nuclear Power Policy

Sweden's 12 nuclear reactors provided about half of its electricity in 1988. However, despite Sweden's economic and safety successes with nuclear power reactors, the 1980 parliamentary decision to install no more nuclear power and shut down all nuclear power plants by 2010 is still the nuclear power planning basis in Sweden. A recent poll indicates eight out of ten Swedes do not believe the phase-out will ever happen, but legislation has been introduced to shut down one reactor in 1995 and one in 1996 (Leigh 1989; ERCE 1989; IEAL 1987).

7.1.2 Major Legislation

Sweden was one of the first countries to legislate, in 1977, preconditions regarding waste management. Permission to initially load fuel in new reactors would be contingent on satisfying these conditions. The 1977 Stipulation Law required the utilities to demonstrate to government satisfaction either that a reprocessing contract and a plan for final disposal of HLW with "absolute safety" or a plan for direct disposal of spent fuel, again with "absolute safety," was in hand.

There are five national nuclear-specific laws that delineate the legal framework for nuclear waste management (ERCE 1989).

- The Act on Nuclear Activities of 1984 establishes the overall framework for licensing, regulatory criteria and requirements, and supervisory responsibilities to assure the safety of nuclear activities. The 1984 Act replaces the 1956 Atomic Energy Act and the 1977 Stipulation Act. The latter change is especially important, because the Stipulation Act had set in motion the forces that led to the elaboration of the Swedish waste management program.
- The 1981 Act on Financing of Future Expenses for Spent Nuclear Fuel (as amended in 1984) makes the nuclear utilities responsible for ensuring safe disposal of spent fuel and other nuclear waste from

power plants and for decommissioning those reactors, as well as for financing such activities. This obligation extends to associated research and development (R&D).

- The 1958 Radiation Protection Act (as amended in 1984) regulates the handling of radioactive substances.
- The Nuclear Liability Act.
- The Emergency Planning Act.

A report prepared by the Nuclear Fuel Safety Project (KBS), KBS-1 (KBS 1977), was accepted by the parliament as satisfying the Stipulation Law for disposal of vitrified HLW from nuclear power. The KBS-2 report (KBS 1978a) studied the alternative of direct disposal of spent nuclear fuel. When Sweden changed its waste management policy from reprocessing to direct disposal of spent fuel, a second plan was required for a repository to take spent fuel instead of HLW. The KBS-3 report satisfied this requirement, and permission was granted for fuel loading of Sweden's last two reactors. The Swedish government has now accepted that a safe geological disposal of both vitrified HLW and spent fuel in Sweden's crystalline bedrock is feasible.

Once a repository site is designated, four laws must be satisfied: Nuclear Activities Law, Radioactive Protective Law, Environmental Protection Law, and the Building (land use) Law. The Environmental Protection Law will require formal public hearings. The Building Law gives the local government absolute veto rights on land use. There is currently no mechanism for parliament to override the local veto. However, a general change in the Building Law to provide a mechanism for overriding a land-use veto is currently being discussed in Sweden.

7.1.3 Linkage Between Nuclear Power and Waste Management

The 1977 Stipulation Law, discussed above, required parliament's approval that the demonstration of waste management feasibility and safety was satisfactory before subsequent nuclear power reactors could be loaded with nuclear fuel and operated (IEAL 1987).

7.1.4 Policy on Spent Fuel and Waste Management

The current policy is to implement direct disposal of spent fuel following an appropriate period of interim storage. Sweden will dispose of its spent fuel in a single deep geologic repository. No reprocessing or fuel recycling is planned.

Costs of waste management and future decommissioning of nuclear power plants are paid by fees collected from the nuclear utilities (Leigh 1989).

7.1.5 Organization/Responsibilities for Waste Management

Swedish national law makes the Swedish nuclear power utilities responsible for planning and implementing the waste management program. The four utility groups that own and operate the nuclear power plants formed a jointly owned nuclear fuel company, the Swedish Nuclear Fuel Supply Company, in 1973. The company's responsibilities were later broadened to include waste management and the name changed to the Swedish Nuclear Fuel and Waste Management Company (SKB). SKB is responsible for R&D, commercial arrangements, and the development of facilities for nuclear waste disposal on behalf of the nuclear utilities. This delegation of responsibility to SKB was confirmed by Parliament in 1984 (IEAL 1987).

The work of SKB in waste management is supervised by a special governmental body, the National Board for Spent Nuclear Fuel (SKN), which was organized in July 1981. One of SKN's special functions is to administer the waste management program funds accruing from fees paid by the nuclear power producers in proportion to the electric power they produce. This function is authorized by the Act on the Financing of Future Expenses for Spent Nuclear Fuel. This act requires the licensee (SKB) to defray the costs for the following:

- handling and final disposing of spent fuel, as well as the radioactive waste derived from it
- reactor decommissioning and dismantling
- R&D necessary to achieve these activities.

Since 1986, three agencies involved in regulating nuclear activities report to the Ministry for Environment and Energy (Leigh 1989):

- the SKI (Swedish Nuclear Power Inspectorate), which is responsible for licensing of construction and operation of nuclear facilities, for possession of spent nuclear fuel and radioactive waste, and for waste storage and disposal facilities
- the SSI (Swedish National Institute of Radiation Protection), which is responsible for enforcing radiation protection regulations
- the SKN (National Board for Spent Nuclear Fuel) (described previously).

An Advisory Committee on Nuclear Waste, with personnel from all three of these agencies and from universities, also reviews Swedish waste management efforts to ensure coordination and completeness.

The National Swedish Franchise Board for Environmental Protection issues permits for nuclear facility siting and construction (IEAL 1987).

7.1.6 Repository Licensing Process, Regulation Status, Major Regulations, Safety Bases

The formal licensing procedure requires the applicant, SKB, to obtain a series of permits according to several laws, not all of which are nuclear specific. First, a site permit must be obtained from the municipal administration under the Building Act and the Law on the Conservation of Natural Resources (applies to all commercial structures). The municipal council is in a position to veto the entire project, and the law currently provides for no national government override of this veto for a nuclear repository.

Second, SKB must apply to the SKI for a permit under the Nuclear Activities Act. The SKI evaluates the safety of the design of the proposed facility, ensuring adherence to the standards promulgated by such authorities as the National Institute of Radiation Protection. Both SKI and SSI must reach a favorable determination before a license is issued by the Ministry of Environment and Energy.

Third, SKB must also apply for a permit from the National Franchise Board for Environmental Protection. In the end, all nuclear permits are reviewed by the SKI and, if approved, are forwarded with its evaluation to the central government, which then issues the final decision. If approved, a detailed timetable for commencement of construction and a manpower forecast is submitted to the Country Employment Board, which then issues a building permit or requires certain modifications so that the building permit may be issued (ERCE 1989).

Primarily because of the planned phase-out of nuclear power, formal design criteria for the safety of nuclear facilities, analogous to the United States NRC's 10 CFR 50 Appendix A, have not been developed in Sweden. The SKI has a staff of approximately 85, who are divided between inspection and regulation/research. The inspectorate's main task is to set safety goals, review licensees' technical approaches to achieving these goals, and audit the quality of the licensee's technical and administrative safety performance. It does not as a rule develop detailed criteria in advance of a license application, but rather reviews the applicant's detailed documentation to ensure that the overall objectives can be achieved with high confidence. This practice is exemplified in HLW management in that the Swedish safety authorities have not established specific quantitative health and safety goals for a HLW repository, nor have they issued specific quantitative objectives or criteria (IEAL 1987).

The SKB's policy and planning for Sweden's waste management programs are structured to gain broad public support. The Swedish approach places high priority on public acceptance strategies, including institutional credibility and openness, major public relations efforts, and siting sensitivities. The following basic premises and procedures are followed:

- a high level of conservatism in concept, supported in great depth technically, well documented, and well communicated to the public
- review and endorsement of Sweden's waste disposal plans (in the KBS-1 and KBS-3 reports) by prestigious scientific bodies around the world
- credibility of an extensive in situ research program (Stripa), enhanced by participation of top scientists of many other countries

- one central deep geological repository (SFL) for spent fuel (IEAL 1987).

7.2.1 Reprocessing

Sweden originally planned to use foreign reprocessing. However, in 1980, the Swedish government decided that Swedish utilities will not sign any additional reprocessing contracts. The contractual rights for reprocessing have been traded or are expected to be traded to other countries. If any vitrified HLW is disposed of, the quantity will be small (IEAL 1987).

7.2.2 Interim Storage Before and After Reprocessing

With future expansion, the central facility for storage of spent fuel will be able to hold all spent fuel and/or vitrified waste (if any) to be produced by Sweden's 12 reactors during their anticipated operational life (IEAL 1985).

7.2.3 Geologic Repository

A national deep underground final disposal facility (SFL) for all long-lived and HLW is planned to be in operation about 2020. It is expected that a siting decision will be made about 2000 and that construction will begin about 2010 (IEAL 1987).

7.2.4 Other System Considerations

Because Sweden is not reprocessing its spent fuel, it will not have HLW to separate into fractions. Fuel channels from BWR assemblies and poison rods from PWR assemblies will be removed from spent fuel before encapsulation and will be packaged and disposed of in a separate geologic repository for long-lived wastes at the same site as the repository for spent fuel (OECD/NEA 1988).

Sweden has not expressed serious interest in other disposal concepts (e.g., sealed or transmutation) or in having an international repository.

7.3 WASTE STORAGE AND TRANSPORTATION

7.3.1 Spent Fuel Storage

Spent fuel in Sweden is from PWRs and BWRs. For repository design purposes, the reference fuel is taken to be PWR fuel with burnup at 38,000 MWd/MTU and maximum burnup of 40,000 MWd/MTU (KBS 1983, Vol. I). Sweden plans to phase out all nuclear power by 2010, and its total quantity of spent fuel at that time will be about 7800 MTU.

The Swedish reactors have spent fuel storage capacity in their reactor pools equivalent to four to five years' reactor operation. In some cases, dense racks have been installed to expand this capacity. Spent fuel is stored in water pools at the power plants for one to five years, then transferred to the central intermediate storage facility (CLAB) for 30 to 40 years before disposal (OECD/NEA 1988).

Located near the Oskarshamn nuclear station, the CLAB is a manmade cavern mined out of granite bedrock. It lies beneath 30 meters of rock, is designed for wet storage of up to 3000 MTU of spent fuel for as long as 40 years, and is sited to allow expansion to 9000 MTU if needed. It also is designed to store canisters of HLW glass, if necessary, from foreign reprocessing of Swedish spent fuel until the final repository is ready. The CLAB facility was commissioned in 1985 and is designed for a life of 60 years, when its contents will have been transferred to a repository. Construction cost was about $250 million (U.S. dollars), and fuel storage costs are estimated at $50 U.S. dollars per kilogram (in 1980 U.S. dollars) (SKB 1986).

7.3.2 High-Level Waste Storage

Sweden does not plan to reprocess spent fuel and thus should have no HLW to store.

7.3.3 Transportation

All Swedish nuclear power stations, as well as the CLAB, are built on the coast, and spent fuel and radioactive wastes are transported by sea (KBS 1983, Vol. I; OECD/NEA 1988). A ship (M/S SIGYN), built for this purpose, has transported spent fuel from the Swedish reactor plants to France for reprocessing. In 1985, the vessel began transporting spent fuel from the reactors to the CLAB and from France back to Sweden. In 1988 it also started transporting LLW and ILW to the SFR repository. The M/S SIGYN has a double hull, double bottom, and several watertight bulkheads, ensuring high floatability. The single hold, 57 meters by 10 meters by 5.6 meters deep, is designed to accommodate ten TN17 Mk 2 transport-only casks holding a total of 70 PWR or 170 BWR fuel assemblies (about 32 MTU). For short distances of transport on land, the spent fuel casks are carried by heavy-haul trucks.

7.4 GEOLOGIC WASTE REPOSITORY

7.4.1 Safety Requirements and Approach

Performance assessments of each of the final candidate repository sites will be carried out in connection with the R&D program beginning in 1992 (IEAL 1987). This will provide a basis for a preliminary evaluation of site suitability and guidelines for the detailed investigation of the final candidate sites.

Nuclear criticality in the repository is considered to be beyond the realm of reasonable possibility. If criticality should occur, the nuclear reaction would stop when the water is boiled away, and any effects would be localized.

There are no specific regulatory requirements for quality assurance, but stringent QA will be applied to preclude any deviations from the desired quality that could significantly impair system safety (OECD/NEA 1988).

7.4.2 Siting

Site investigations for the deep geological repository for spent fuel and HLW (the SFL) have been completed at eight sites and started at seven more.

Site investigations will continue up to the year 2000, when a choice of the final site is foreseen. A screening procedure is planned in two steps, the first around 1990 and the last in the year 2000. In the first step, two or three out of 10 or more possible sites will be chosen for further, more detailed studies. In the second step, one of those sites will be chosen as the desired final site.

A hard-rock underground research laboratory (HRL) will be constructed in the early 1990s for detailed, realistic R&D studies. Detailed site characterization studies will be carried out at the selected site between about 1998 and 2005.

The host rock will be pre-Cambrian crystalline rock in granite or gneiss or gabbro at about 500 meters deep and will likely be in a saturated water environment. Selection of candidate host rocks was based on availability of suitable rock formations in Sweden (IEAL 1987).

7.4.3 Design Concept(s)

The geologic repository concept was developed by 1983 as a reference to satisfy the requirement of showing that safe spent-fuel disposal is feasible. Alternative concepts are being studied, but the reference concept has not changed. The concept depends on multiple barriers. It uses a long-lived package (up to perhaps one million years) and places most of the reliance of post-closure safety on the waste package. The concept is intentionally conservative.

It should be recognized, however, that the reference concept for the spent fuel repository proposed in the KBS-3 study is not necessarily what SKB will eventually build. Rather, it is a concept that SKB regulators believe can be built and would be acceptable. What will actually be done will reflect the results of additional investigations and experience in Sweden and abroad.

The KBS-3 concept envisions emplacing encapsulated spent fuel in a deep geologic repository. The repository will consist of a series of parallel drifts at the reference depth of 500 meters with disposal boreholes in the floors of the drifts. One package will be emplaced in each borehole. Disposal drifts will be inverted U-shaped, with a flat floor, and will be 3.3 meters wide and 4.5 meters high. The disposal drifts may be at one or two levels, with an elevation difference of about 100 meters and a minimum distance to significant fractures of 100 meters (KBS 1983, Vol. I).

7.4.4 Retrievability and Monitoring

The Swedish regulatory authorities have no requirements on the retrievability of waste from a repository, and no special provisions for retrievability are included in the reference concept. The type of repository envisioned in the KBS-3 study would, however, allow for spent fuel retrievability during the operational period and for a long time afterwards, even if no specific measures are taken. Retrievability is considered to be a matter of cost and keeping records (IEAL 1987; OECD/NEA 1988).

Monitoring is not expected to be needed for long-term safety. However, a monitoring program may be valuable for scientific reasons or reasons of public acceptance. Also, requirements for markers have been discussed but not specified (IEAL 1987; OECD/NEA 1988).

7.4.5 Waste Package System

In the reference waste package concept, components of the packages will be spent fuel, a matrix material, a canister, and a buffer/packing material. Unconsolidated spent fuel assemblies are to be embedded in a lead or pressed copper powder matrix. The canister will be thick-walled copper, and compacted bentonite "donuts" will be placed around the canister. The bentonite "donuts" will minimize intrusion of water to the canisters (OECD/NEA 1988).

The canister is to perform the functions of containing the spent fuel and preventing dispersal of radioactive substances with the groundwater, providing shielding of personnel during handling, and providing shielding of the host rock and groundwater after emplacement. The canister is expected to last about one million years (KBS 1983, Vol. III).

Copper is the present choice for the canister wall because it is the noblest of the common metals and is highly corrosion resistant. However, other materials are continuing to be studied (see the following section) (OECD/NEA 1988).

The canister is cylindrical, with a 0.8-meter outside diameter, 4.5 meters long, and 10 centimeters thick. The total weight of the reference canister with a lead matrix is 22 MT; it contains 1.4 MTU of fuel assemblies. The total weight of the alternate canister with a copper matrix is 18.5 MT; it contains 1.6 MTU in fuel assemblies (KBS 1983, Vol. I; OECD/NEA 1988).

7.4.6 Research and Development

The current SKB six-year research plan[a] lays out planned R&D activities for 1986 to 1992 (IEAL 1987). The planned activities fall into six main areas:

- research on alternative engineered barriers; waste form (spent fuel) studies; canister materials (principally copper, but also passive materials such as stainless steel and titanium); corroding materials (lead and carbon steel) and non-metallic materials (ceramics and cement); and backfill materials; selection of final engineered barrier materials by mid-1990s
- geoscience studies, including groundwater movement in bedrock, and stability of bedrock; study-site investigations; instrument development; and detailed studies at the hard rock laboratory (HRL), scheduled for construction near the CLAB, starting 1990
- biosphere studies to quantify uncertainties resulting from changes in the biosphere, to improve data/models, and to validate biosphere dispersal models
- chemistry studies to describe release and transport of radionuclides
- safety research, especially development of safety assessment models in time for selection of candidate sites in 1992
- International cooperation to ensure availability of foreign R&D results.

Underground research into the disposal of radioactive waste has been carried out at the former Stripa iron-ore mine in central Sweden since 1976. The Stripa project is managed by SKB under the auspices of the OECD/NEA and under the guidance of a Joint Technical Committee composed of representatives from each participating country. The principal goals of the first two phases

(a) The R&D plan must be updated every three years, as required by the Nuclear Activities Act.

of this project were to develop techniques to assess the geology, hydrology, and geochemistry of potential sites for the disposal of radioactive waste, as well as to perform tests to examine groundwater flow within fractured rock and to assess properties of backfilling and sealing materials. Each of these main objectives was met, leading to considerable advancement of both investigative techniques and practical knowledge for repository siting and design.

Phase 3 work commenced in 1986 and is due to be completed in 1991. This third (and final) phase includes participation from seven countries: Canada, Finland, Japan, Sweden, Switzerland, the United Kingdom (U.K.), and the United States (U.S.). Research is focussed in three main areas: 1) site characterization and model validation, 2) development of advanced methods and instruments for site investigation, and 3) techniques and materials for sealing groundwater flow paths in fractured rock. Progress was made toward investigating a previously "undisturbed" granite rock volume and developing and validating conceptual and mathematical models for groundwater flow through this rock volume. Development work advanced on several technologies, including seismic, radar, and hydraulic instruments. Finally, the properties of different materials for injection grouting were studied and plans were laid for a large-scale grouting test (OECD/NEA 1989).

7.4.7 Approach to Proving the Safety of the Repository

The Swedish regulatory approach emphasizes the performance of the overall repository system with a limit on the ultimate radiation dose to man. No quantitative criteria will be used that would specify, for example, performance of the immobilizing matrix or groundwater travel time.

The Swedish regulator's purpose of a safety analysis for a repository is to obtain quantitative measures of the safety of the proposed disposal method by comparing calculated individual doses resulting from releases with individual dose limits. SKI believes that 1) the evaluation of long-term safety should be conducted using mathematical models that describe the total repository system; 2) detailed analysis of the performance of individual subsystems or components are necessary, because they provide a basis for determining which components should be included in overall performance analyses; 3) the performance assessment should serve as a driving force for SKB's R&D program, and SKB should conduct overall performance assessments at suitable time intervals based on the knowledge available at the time; and 4) both deterministic and probabilistic models should be developed and are complementary in the safety assessment.

The work to date on performance assessment has been based on a generic site with assumed characteristics, using the reference repository concept in the KBS-3 report (KBS 1983). The approach appears to be continued for the KBS-4 study, expected about 1991. It is probably too early to determine the approach that will be used for the final repository licensing assessments. The usefulness of probabilistic approaches is being evaluated. The current approach basically uses deterministic safety assessments for a series of scenarios of expected naturally occurring events. Potentially disruptive natural events of low probability are not considered. Human intrusion scenarios are also not considered. A variety of pathways to man are being modeled.

The current approach is considered by the SKB staff to be conservative because of numerous conservative assumptions used, and as such, the approach is believed to represent bounding conditions and results. The conservatism of some of the assumptions, such as matrix diffusion of radionuclides through the host rock and the uniform corrosive fracture of the canisters over the long time period of 100,000-1,000,000 years, is being challenged by some. However, the Swedish regulatory review in 1984 checked the approach with some more conservative assumptions and obtained very similar results. R&D is being carried out to confirm the appropriateness of the key assumptions. KBS-4 will include probabilistic modeling.

7.4.8 Peer Review Activities

Extensive international peer reviews of Swedish plans for management of HLW and spent fuel have been requested by and carried out through the Swedish Nuclear Power Inspectorate and the Swedish Ministry of Industry. The "KBS Report on Handling of Spent Nuclear Fuel and Final Storage of Vitrified High-Level Reprocessing Waste," called the KBS-I report (KBS 1977), was reviewed by

Canada, Norway, the OECD/NEA, the U.S. (Pacific Northwest Laboratory), Denmark, the Federal Republic of Germany (FRG), and Belgium (KBS 1978b). Reviewers of the "KBS-2 Plan for Handling and Final Storage of Unreprocessed Spent Nuclear Fuel" (KBS 1978a) included Canada, France, the USSR, IAEA, the U.K., OECD/NEA, and the U.S. (Environmental Protection Agency, National Academy of Science, U.S. Geologic Survey, and Lawrence Livermore National Laboratory) (KBS 1980). Reviewers of the "KBS-3 Plan" included the IAEA, OECD/NEA, Canada, France, the U.K., and the U.S. (NAS) (KBS 1984). International reviews of the Swedish research and development program for management and disposal of spent nuclear fuel have also been carried out for the Swedish National Board for Spent Nuclear Fuel (SKN 1987).

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8.0 SWITZERLAND

8.1 LEGISLATIVE/INSTITUTIONAL BASES

The following sections discuss the Swiss policy, regulations, and organizations that pertain to spent fuel and high-level waste management.

8.1.1 Nuclear Power Policy

The Swiss nuclear power program includes five reactors (2.9 GWe) that presently supply approximately 40% of the country's electricity. Nuclear power is a key energy source, along with hydroelectric power, which provides the balance of electricity supply for Switzerland. Although the federal government favors the development of nuclear energy, local opposition has delayed the expansion of nuclear power. Swiss electricity demand is increasing rapidly, and imports of largely nuclear-generated electricity from France are being planned in the 1990s. A referendum will be held in 1991 to consider the options of phasing out nuclear power in short order or of imposing a ten-year moratorium on building new nuclear plants while various alternatives for meeting electricity demand are explored. Waste management policy has played a key role in the ongoing debate on the use of nuclear energy in Switzerland (IEAL 1987; Schneider et al. 1988; ERCE 1989).

8.1.2 Major Legislation

The federal Parliament amended the Swiss constitution in June 1957 so that nuclear legislation would fall within the sole jurisdiction of the Confederation, rather than at the Cantonal level. This was approved by a national referendum and all the Cantons in November 1957 (IEAL 1987).

In 1978, the Swiss Parliament revised its Atomic Energy Act of 1959 to require that producers of radioactive waste be strictly responsible for its disposal. The revisions also stipulated that in order to license a new nuclear power plant in Switzerland, proof and guarantee of final waste disposal must be demonstrated. The federal Department of Transport, Communications, and Energy (EVED) ruled in 1979 that all current operating licenses would expire at the end of 1985 if no such guarantee project was available (IEAL 1987; ERCE 1989).

The National Cooperative for the Storage of Radioactive Waste (NAGRA) submitted its Project Gewahr report in 1985 to satisfy the requirement for proof and guarantee of final disposal (NAGRA 1985). The Federal Council reached an affirmative decision in June 1988 that Project Gewahr had satisfied the stipulation. The Federal Council concluded that safe disposal of HLW and TRU waste had been demonstrated to be feasible but that further proof of the availability in Switzerland of an acceptable and sufficiently large site was needed (ERCE 1989).

8.1.3 Linkage Between Nuclear Power and Waste Management

Proof of safe disposal of radioactive waste must be shown in applying for new reactor licenses. Utilities were required to guarantee safety and feasibility of final disposal of nuclear waste as a prerequisite to extension of nuclear plant operating licenses beyond 1985 (Schneider et al. 1988). Project Gewahr was completed by NAGRA to satisfy this requirement (NAGRA 1985).

8.1.4 Policy on Spent Fuel and Waste Management

The 1985 Project Gewahr report set out a multi-element approach to nuclear waste management, with the total interim storage time from discharge to final disposal of HLW being 40 years (NAGRA 1985). Spent fuel is currently reprocessed in France and the United Kingdom (U.K.), but the option for direct disposal of spent fuel is open. Centralized interim storage is planned for HLW, spent fuel, and some LLW/ILW. All radioactive waste is to undergo final disposal in geological formations. Two repositories are planned: one primarily for HLW (a Swiss Type C repository) and another primarily for LLW/ILW (a Swiss Type B repository). TRU waste from reprocessing is now intended to be disposed of in the Type C HLW repository. Construction of the Type B repository for LLW/ILW appears certain, and development of the Type C repository is also planned at least up to the point of site selection. The Swiss waste program prefers the option of HLW and TRU waste disposal outside Switzerland, in a foreign national repository or an international repository (ERCE 1989; Schneider et al. 1988; OECD/NEA 1988).

8.1.5 Organization/Responsibilities for Waste Management

In 1972, the electric utilities of Switzerland, together with the Swiss Confederation (being responsible for the waste from medicine, industry, and research), founded the National Cooperative for the Storage of Radioactive Waste (NAGRA), a private organization whose function is to provide facilities for the disposal of radioactive waste. The utilities are responsible for spent fuel reprocessing, transport, and interim storage. Because HLW produced from the reprocessing of Swiss spent fuel at foreign facilities was not then expected to be returned to Switzerland, it was not considered part of NAGRA's responsibilities. Since the late 1970s, when the terms of new reprocessing contracts required the return of HLW, NAGRA's responsibilities were expanded to include the disposal of HLW. NAGRA was also charged with formulating the demonstration project to guarantee the capability to safely and permanently dispose of radioactive waste. Costs of NAGRA's waste disposal program are paid by the utilities through a tariff in electricity rates (OECD/NEA 1988; IEAL 1987).

The Federal Council of the federal government's ministers represents Switzerland's executive branch. The Council formulates the administration's policy decisions in the nuclear energy field and decides on applications for general, construction, and operating licenses for nuclear installations. With respect to waste repositories, the Federal Council will grant all licenses for preparatory measures for the construction of repositories, as well as all licenses for their construction and operation. In the case of a general license, the Federal Council's decisions must be approved by the Federal Assembly (the Swiss Parliament) (IEAL 1987).

The Swiss regulatory authority falls under the Federal Council's Department of Transport, Communications and Energy (EVED), which is responsible for preparing legislation on nuclear energy. The department includes the Federal Energy Office, which is responsible for reviewing applications for nuclear installations and making recommendations to the Federal Council. It also grants licenses in the areas of transport, import, and export of nuclear materials and equipment (IEAL 1987).

Two principal organizations within the EVED work in close cooperation and share the main regulatory functions: the Nuclear Safety Inspectorate (HSK) and the Swiss Federal Commission for the Safety of Nuclear Installations (KSA). The HSK provides expert opinion on the technical safety reports relating to the various licenses required in Switzerland for nuclear installations, including nuclear waste repositories. The KSA advises EVED and the Federal Council regarding applications for licenses and comments on HSK's reports. The opinions provided by HSK and KSA are the basis for licensing (IEAL 1987).

The Federal Department of Interior (EDI) develops rules to implement the radiation protection regulations adopted by the Federal Council. It collects and stores radioactive wastes originating at facilities other than nuclear power plants and nuclear fuel reprocessing plants. Within the EDI, the Federal Office of Public Health is the competent authority for licensing the production and use of radioactive substances, except for activities in nuclear installations. The Federal Commission for Protection Against Radiation advises the EDI on protection of the population from radiation. The Federal Commission for the Monitoring of Radioactivity, under the EDI, maintains information on environmental radioactivity (IEAL 1987).

The Paul Scherrer Institute (PSI) was recently formed (1987) through a merger of the Federal Institute for Reactor Research (EIR) and the Swiss Institute for Nuclear Research (SIN). It conducts research in nuclear energy, including waste management. The PSI is owned by the Swiss government, under the Department of Interior (Leigh 1989).

The Interagency Working Group on Nuclear Waste Management (AGNEB) is responsible for preparing the necessary technical materials in support of decisions by the Federal Council and EVED for licensing nuclear waste facilities (IEAL 1987).

8.1.6 Repository Licensing Process, Regulation Status, Major Regulations, Safety Bases

Nuclear installations in Switzerland, which include nuclear waste repositories, are licensed by the Federal Council. The federal Parliament must ratify a decision by the Federal Council to issue a general license, and the

Federal Council can subsequently issue specific licenses without further need of parliamentary approval. Responsibility for the regulation of the facilities lies with the federal authorities, and the cantonal (i.e., state) authorities are in charge of concerns involving non-nuclear permits, fire protection and water. There is no explicit federal authority to preempt actions by cantonal officials involving non-nuclear permits. However, the cantons cannot deliberately change a cantonal law specifically to block a nuclear facility (IEAL 1987; ERCE 1989).

The granting of licenses for nuclear waste repositories is very similar to that of licensing nuclear power plants. However, one additional step is necessary for repository approval. Any preparations for construction of radioactive waste repositories, such as geologic investigations that penetrate the surface (test drillings) to determine if a site is suitable for a repository, must first be approved by the Federal Council. In order to receive a license, the applicant must provide maps and plans and a report on the effects of the proposed preparatory measures on the environment. The granting of the license does not presuppose the right to construct the repository (IEAL 1987). The Department of Transport, Communication, and Energy publishes the request for preparatory measures in the "Federal Notebook" and also in designated public places in the appropriate canton(s). The Federal Council makes a decision after examining the request, the advice of experts, and the response of the cantons, including any objections presented (ERCE 1989).

8.1.7 Roles of the Public, Local Organizations, Multinational Organizations

Before the revision of the Atomic Energy Act in 1978, test drilling for a nuclear waste repository was considered totally subject to local mining laws and, therefore, under the sole jurisdiction of the canton. Localities were thereby able to veto drilling applications. The Ordinance on Preparatory Measures, issued in 1979, considered the test borings to be preparatory actions for a nuclear installation, and, therefore, under federal jurisdiction (ERCE 1989).

The Federal Order of 1978 reduced the veto rights of the cantons in siting matters to those of consultation. The cantons, in turn, deal with communities and voice their objections to the federal government. The federal government has preemptive rights if the consultative system does not culminate in approval of the action, but the intent of the Swiss siting system is to employ expropriation only as a last resort. A few cantons have laws that require them to oppose, "by all possible political and legal means," nuclear plants or waste disposal site plans. The Federal Tribunal is empowered to resolve conflicts between cantonal and federal laws (ERCE 1989).

NAGRA is an active participant in international research programs, including the Stripa project in Sweden and the Poços de Caldas analogue project in Brazil. Switzerland has cooperative agreements with several foreign countries, as well as membership in the OECD/NEA and the IAEA. Several foreign countries (the FRG, the U.S., France, and Sweden) have also participated in research work conducted at the Swiss underground research laboratory (Grimsel Test Site) (OECD/NEA 1988).

8.2 OVERALL WASTE MANAGEMENT SYSTEM

The Swiss transport spent fuel for reprocessing in France and the U.K., although the option of direct disposal of spent fuel is open. Central interim storage of HLW, spent fuel, and some LLW and ILW is planned. All Swiss radioactive wastes are to be disposed of in geologic repositories. A Type C repository is planned for HLW (possibly spent fuel) and TRU waste disposal, but the Swiss would prefer to use a foreign or international repository. A Type B repository in Switzerland is planned for LLW and ILW disposal. The option for a Type A repository for LLW only remains open (OECD/NEA 1988; IEAL 1987; ERCE 1989; Schneider et al. 1988).

8.2.1 Reprocessing

The Swiss have selected a fuel cycle involving reprocessing and plutonium recycling. However, the option of direct disposal of spent fuel is also open. All Swiss nuclear fuel to be discharged through 1993 is contracted to be reprocessed (Schneider et al. 1988). Swiss nuclear power plants generate about 90 MT of spent fuel annually. Contracts are in place for a total of 599 MT of spent fuel to be reprocessed by COGEMA in France and 165 MT by BNFL in the U.K. (IEAL 1987).

8.2.2 Interim Storage Before and After Reprocessing

Until spent fuel is transported offsite for reprocessing, it is stored at each nuclear power plant site in water-filled pools to allow cooling (ERCE 1989). In the future, spent fuel is also to be stored in dry casks at a central interim storage facility (OECD/NEA 1988).

High-level waste returned from foreign reprocessors is to be stored in a central interim storage facility for approximately 40 years after unloading from the reactor, until the heat load is reduced and a low repository temperature can be maintained (OECD/NEA 1988; IEAL 1987).

8.2.3 Geologic Repository

All radioactive wastes are to undergo final disposal in repositories situated in suitable geologic formations (OECD/NEA 1988). Project Gewahr assumed two types of repositories: one (Type C) for HLW and certain alpha-containing ILW, and a second (Type B) for LLW and ILW. However, the option of a Type A repository for LLW only remains open (IEAL 1987).

NAGRA presently operates under the following target schedule for the HLW geologic repository (Type C). After 1989, one alternative sedimentary host rock will be selected, and application for field investigations will be made. After 1992, the crystalline and sedimentary options will be evaluated, and one site will be selected for further investigations. In 2020 at the earliest, operations will be started in the Swiss HLW repository, if foreign or international options do not materialize (ERCE 1989).

8.2.4 Other System Considerations

The Swiss prefer to dispose of HLW in a foreign national or international repository (OECD/NEA 1988; ERCE 1989).

Intermediate-level waste and alpha-bearing waste from reprocessing will be disposed of in either the Type B or Type C repository, according to the maximum allowable radionuclide concentrations for the Type B repository, as derived from safety analyses based on actual site data. These wastes would be disposed of in a separate part of the Type C repository using a concept different from that for HLW disposal (OECD/NEA 1988; Schneider et al. 1989).

8.3 WASTE STORAGE AND TRANSPORTATION

Spent fuel is stored in pools at reactors and will be stored in dry casks at a planned central interim storage facility. HLW returned from foreign reprocessors will also be stored in dry casks in the central facility. Spent fuel and HLW is transported by rail and road, following Swiss and international regulations (OECD/NEA 1988; Schneider et al. 1988).

8.3.1 Spent Fuel Storage

Spent fuel is stored at each nuclear power plant site for about 10 years in water-filled pools to allow cooling. Spent fuel is also planned to be stored in transport containers (CASTOR) in a central interim storage facility. Plans are to start operation of the facility for dry cask storage of spent fuel in 1991 (OECD/NEA 1988; Schneider et al. 1988, 1989).

8.3.2 High-Level Waste Storage

Return of vitrified HLW from foreign reprocessors will not begin until about 1993 (ERCE 1989). A central interim storage facility is planned to provide dry storage of HLW or spent fuel in transport containers (CASTOR) in surface halls. Low-level waste and ILW would be stored in separate surface halls. The storage capacity would be sufficient for the present nuclear power plants. Construction of the storage facility would be conducted in two stages, with the first stage having the capacity to meet the requirements of the next 15 to 20 years (OECD/NEA 1988). Interim storage of HLW is planned for about 40 years (Schneider et al. 1988).

8.3.3 Transportation

For four of five nuclear reactors, spent fuel is shipped by rail to foreign reprocessing plants (ERCE 1989; Schneider et al. 1988). Transport of other spent fuel and wastes is conducted using standard transport containers on road vehicles. Spent fuel has been transported to La Hague and Sellafield since the early 1970s, using different casks weighing up to 120 MT. Both wet and dry cask systems have been used for transport (Schneider et al. 1988). International (IAEA) and national regulations for the transport of radioactive materials are observed (OECD/NEA 1988).

8.4 GEOLOGIC WASTE REPOSITORY

The schedule for the Swiss geologic repository for HLW is as follows: selection of one site (crystalline or sedimentary) for further investigations in 1993, application for an underground research laboratory (URL) at the repository site in 1998, conclusion of the final site characterization in 2010, and engineering and construction of the repository (or participation in an international repository project) until about 2025 (OECD/NEA 1988). The Swiss repository for HLW will begin operation in about 2020, provided that the results of geological investigations are favorable and that the licensing process is not delayed (ERCE 1989).

The construction and operational phases of the repository will occur simultaneously. Appropriate arrangement of the repository installations will ensure that mechanical excavation with tunnelling machines is spatially separated from the emplacement operations (OECD/NEA 1988).

8.4.1 Safety Requirements and Approach

The safety conditions which the final repositories must satisfy are defined in the Guideline R-21, established in October 1980 by 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 that escape into the biosphere must not at any time lead to individual doses exceeding 10 mrem (0.1 mSv) per year, and 2) a repository must be designed in such a way that it can be sealed at any time within a few years. After it has been sealed, it must be possible to dispense with safety and surveillance measures (KSA/HSK 1980; OECD/NEA 1988).

No specific performance criteria have been established to supplement the protection goals in Guideline R-21. This strategy is based on the view that a multiple barrier approach should be used to achieve safe disposal, and the overall safety of the disposal system should be evaluated (IEAL 1987).

NAGRA has also developed its own criteria governing the evaluation of suitable geologic environments for HLW disposal. The four primary factors are 1) suitability of the host rock; 2) permissible ambient rock temperature of the repository and the resulting maximum depth of the repository; 3) sufficient distance from large, tectonically-disturbed zones; and 4) suitable hydrodynamics, i.e., sufficiently long flow paths and low volumes of groundwater flow (IEAL 1987).

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 (OECD/NEA 1988).

8.4.2 Siting

To date, no specific site selection program has been performed. Rather, the investigations have concerned a region for potential sites, rather than specific sites (OECD/NEA 1988). Impermeable clay, anhydrite, and crystalline rock formations have been considered by NAGRA, but since 1978, the greatest emphasis has been placed on the crystalline rock medium (IEAL 1987). Six deep boreholes have been drilled in crystalline bedrock, and work on a seventh is under way (ERCE 1989). The Federal Council concluded in its review of Project Gewahr that alternative media to the crystalline rock option should be studied. This was because the candidate crystalline rock areas have proven to be smaller in extent, due to the discovery of a permo-carboniferous trough, and more broken up by fault zones than earlier anticipated. It was questioned whether a sufficiently large crystalline formation between major fault zones could be found. NAGRA is presently focusing on molasse and clay deposits. However, this work to date has been primarily paper studies to identify technical criteria and issues (ERCE 1989).

By late 1984, several geophysical measuring projects and deep boreholes were completed and a URL at Grimsel was in operation. Between 1989 and 1995, licensed research is scheduled to be conducted at one or more of the potential repository sites. After provisional choice of a site in the mid-1990s, detailed investigations, including testing in a URL, will be conducted to allow the definitive site selection after the year 2000 (ERCE 1989).

8.4.3 <u>Design Concept(s)</u> (OECD/NEA 1988; IEAL 1987; Schneider et al. 1988)

Options for the repository design concept include 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. The repository and waste package concepts are presently based on disposal in granite. Repository concepts for sedimentary rock are just being developed.

In Project Gewahr, the reference disposal concept for the Type C repository was a system of mined tunnels and silos at a depth of about 1200 meters in the crystalline basement of northern Switzerland. The repository was designed such that HLW would be disposed of in horizontally-mined tunnels (3.7 meters in diameter with the HLW canisters axially placed 5 meters apart), while ILW was disposed of in 55-meter deep vertical silos (10 meters in diameter). Each ILW silo would consist of a concrete structure standing free of the rock in which the waste was emplaced and immobilized with a special concrete. The space between the rock and the silo wall would be backfilled with bentonite. During emplacement of HLW, the waste canisters would be surrounded by prefabricated, compacted bentonite blocks, emplaced by a special handling machine. Two vertical access shafts were to be used for the repository: one for construction operations and a fresh air inlet, and the other for conveying the radioactive waste and the backfill material and also for an air outlet. The silo and tunnel areas were separated to avoid chemical or other adverse interactions.

The tunnels and silos would be positioned to take into account the geometry of disturbed zones in the host rock at the repository depth without compromising long-term safety. Large disturbed zones of host rock in the repository area would be avoided by observing a sufficient safety clearance.

Zones of lesser disturbance intersecting the tunnel system would be 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, shafts and cavities underground would be isolated and infilled. Selected key zones would be sealed using bentonite blocks or a bentonite/quartz sand mixture, the remaining space being infilled by other materials.

However, the Project Gewahr concept was formulated to demonstrate feasibility and is not fixed; it does not affect later decisions with regard to the host rock, the repository site, or the engineering design. The safety analyses in Project Gewahr were based on a representative geological situation, the assumption being that the repository would be located in a stable granite block between two major faults of the crystalline basement overlayed by several hundred meters of sediment.

In Project Gewahr, the total design capacity for the Type C repository was approximately 1200 cubic meters of vitrified HLW, corresponding to a spent fuel inventory of 7999 MTU and up to 53,000 cubic meters of TRU waste. This would be sufficient for 40 years of operation of twice the nuclear power capacity presently installed.

NAGRA estimates that the total cost of development and construction of the Type C repository would be about 2 billion 1986 Swiss Francs (about 1 billion U.S. dollars), all to be expended prior to the start of loading in about 2020. The project costs would be paid directly by the waste producers; no Swiss organization exists for collecting and redistributing the funds. Swiss electricity tariffs include costs of NAGRA's work and future repository construction.

8.4.4 Retrievability and Monitoring

Swiss guidelines require that a repository be designed such that it is possible at any time to seal it within a few years. HSK has stated that retrievability should never be considered as a last safety resort, and if there is not confidence in the safety of a waste disposal system, then that system is not yet sufficiently developed and another proven method should be pursued. Since retrievability may be required for socio-political reasons, Swiss guidelines neither require nor rule out retrievability (IEAL 1987).

Swiss guidelines prohibit reliance on surveillance measures after repository closure. However, HSK states that surveillance of a sealed repository is not ruled out, but that such surveillance must be optional and not an essential condition for meeting the protection goal of 10 mrem/yr (IEAL 1987).

8.4.5 Waste Package System

The current reference waste package system for HLW includes a leach-resistant glass matrix, a stainless steel canister, a thick steel overpack (GS 40 cast steel), and a layer of highly-compacted bentonite. The waste form is borosilicate glass with about 13 wt.% waste oxides. The configuration of HLW in a canister is the same as in France and the U.K., or 360 to 400 kilograms (150 liters) of monolithic glass in a cylindrical canister (0.43 meter in diameter, 1.335 meters high, with a 5-millimeter wall thickness and a 170-liter volume). High-level waste from about 1.3 MTU will be contained in one canister. The cast steel overpack has a maximum total length of about 2.0 meters, a maximum outer diameter of 0.94 meter, and a wall thickness of about 25 centimeters. The total weight of the filled and sealed disposal container is 8.5 MT. The buffer material will be prefabricated blocks of bentonite in the shape of annular circular segments for stacking within the circular disposal tunnels and around the horizontal-lying disposal containers. The total thickness of bentonite around each container would to be about 1.4 meters (OECD/NEA 1988; Schneider et al. 1988, 1989).

Direct disposal of some spent fuel may be used in the future. Considerations such as consolidation and matrixing are yet to be decided (Schneider et al. 1989).

For ILW, the waste package system is a dissolution-resistant solidification matrix (cement or bitumen), concrete backfill material, concrete silo walls, and bentonite backfill (OECD/NEA 1988; IEAL 1987).

The repository containers for HLW are designed to withstand the repository conditions for a minimum lifetime of 1000 years. The compacted bentonite (greater than 1 meter thick) is expected to retain radionuclides for about 100,000 years (Schneider et al. 1988). The heat from radioactive decay will result in a maximum temperature of about 150°C at the outer wall of the disposal container. The swelling pressure of the backfill material and the hydrostatic pressure will not exceed a value of 30 MPa (OECD/NEA 1988).

8.4.6 Research and Development

A URL was established at the Grimsel Pass in the Swiss Alps. The Grimsel URL is situated in granite about one kilometer inside the mountain. The rock overburden is about 450 meters. The research program includes projects within the scope of international cooperative agreements. The granite at Grimsel Pass is particularly suitable for rock mechanical, geophysical, and hydrogeological investigations, because within a restricted area, dry and impermeable rock areas, damp zones, and water-bearing fissures can be found. An extensive research program has been carried out at the Grimsel Test Site since 1984, including methods for non-destructive rock examination, rock movement measurements by tiltmeters, various tests regarding rock mechanics, and an extensive hydrogeological experimental program (OECD/NEA 1988).

Research has included corrosion tests on container and overpack materials, which has resulted in the selection of cast steel as the reference concept. Laboratory investigations have been conducted on the mechanical and chemical behavior of bentonite materials. The Grimsel area is not under consideration as a final repository site, and no tests using radioactive wastes are planned (Schneider et al. 1988).

8.4.7 Approach to Proving the Safety of the Repository

The Swiss program has completed its concept feasibility and safety studies and is now beginning the site-selection phase. The concept safety assessment work has provided a foundation upon which the site selection and data development work is being planned. Analyses of the relative importance of selected information to system performance helps guide priorities in site testing, and the relative importance of modeling assumptions is helping guide priorities in validation-oriented studies. Performance assessment is being used to coordinate and guide a comprehensive laboratory, field and natural analogue investigations program.

In the Project Gewahr report, a reference model site was depicted based on data obtained mainly from one particular borehole. The safety analysis showed that the assumed site characteristics allowed the performance objectives in Guideline R-21 to be met, provided that a sufficiently large area of host rock with the modeled properties existed. The latter premise was not demonstrated, and the Federal Council ordered NAGRA to undertake additional investigations (ERCE 1989).

NAGRA completed a deterministic safety assessment for a hypothetical repository in crystalline rock in Switzerland in 1985 (Project Gewahr). This generic calculation assumed a flowpath of 5000 meters ending in fluvial gravels of the Rhine River basin. The annual total flow through the repository was conservatively estimated at 4.3 cubic meters per year. The canister is assumed to disappear at 1000 years, although the canister corrosion products ensure reducing chemical conditions around the waste matrix over the entire release period. Resulting doses for the base case of realistically conservative assumptions were completely insignificant.

A parameter variation study was performed to see what hypothetical doses would have resulted from taking less conservative values for hydrological and geochemical parameters into the calculations. Some of these analyses yielded more significant dose commitments, but the conservatism used to obtain these results requires probabilistic evaluation of the likelihood of encountering such pessimistic conditions. No significance was attached to the actual analytical results, but the results are being used in guiding and challenging site studies to define, as realistically as possible, the range of site parameters. Alternative future scenarios were discussed but not included in the analysis.

The Swiss regulatory authority has accepted the results of this and similar preliminary analyses for LLW and ILW disposal concepts as sufficient to suggest that waste can be safely disposed. Siting is now the main activity of the program in Switzerland, with site characterization and performance assessment being the focus. A deep borehole program investigating the properties of the buried granite of northern Switzerland continues, with associated hydrologic modeling, and migration experiments are being conducted and modelled at the Grimsel underground laboratory. Model development includes a probabilistic capability. Although the safety of the HLW disposal concept was demonstrated for a granite site, other geologic media are also to be investigated.

8.4.8 Peer Review Activities

No information was found on peer reviews of the Swiss radioactive waste management program. However, international cooperation at the Grimsel site allows for international review of Swiss repository R&D projects.

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9.0 UNITED KINGDOM

9.1 LEGISLATIVE/INSTITUTIONAL BASES

The United Kingdom (U.K.) involvement in nuclear technology is both comprehensive and complex. Four nuclear reactor types currently contribute to the national grid, and a fifth type is under development. Reprocessing is the dominant spent fuel management approach but requires differing strategies for the various spent fuel types. Development of geological waste disposal is presently being deferred, and long-term storage of high-level waste (HLW) is being pursued for the interim.

9.1.1 Nuclear Power Policy

The U.K. nuclear power base includes 26 gas-cooled reactors (GCRs) (Magnox type), 14 advanced gas-cooled reactors (AGRs), 1 liquid metal fast breeder reactor (LMFBR), and 1 steam-generating heavy water reactor (SGHWR). One 1.1 GWe pressurized water reactor (PWR) is on order, with further PWR additions planned. The GCRs are expected to be phased out by 2005, to be replaced by the addition of PWRs. About 20% of electricity in the U.K. is generated by nuclear power stations (OECD/NEA 1988). A central policy is to reprocess Magnox fuel and recycle uranium to AGR systems (Leigh 1989). Reprocessing of oxide fuel (AGR and foreign LWR types) is scheduled to begin in 1992. Liquid HLW will be vitrified and stored until a geologic repository is developed.

9.1.2 Major Legislation (Schneider et al. 1988)

In the U.K., the basic regulatory framework was initially established in 1946, with passage of the first atomic energy act governing the industrial use of nuclear energy. Subsequent acts have amended or modified language of the earlier acts or have shifted authority from one organization to another. In general, nuclear-related legislation in the U.K. includes acts that are primarily directed at the occupational aspects of radiation control and protection of the public, such as the Radioactive Substances Act of 1960, the Nuclear Installations Acts of 1965 and 1969, the Radiological Protection Act of 1970, the Nuclear Installations Regulations of 1971, and the Health and

Safety at Work Act of 1974. The Nuclear Installations Act provides for a system of licensing to control the operation of commercial power stations and associated fuel manufacturing and reprocessing facilities. The Control of Pollution Act of 1974 is not specific to radiation but does contain provisions relating to the control of water pollution by radioactive waste and also to the control of radioactive waste that is toxic.

In addition to basic legislation, there are orders and regulations known as "Statutory Instruments" which are made within the framework of the basic legislation, e.g., the Ionizing Radiation Regulations of 1985. Generally, the regulatory philosophy is that the organizations that produce radioactive wastes must carry the burden of responsibility for the safe and effective management of such wastes and for meeting the full cost of waste management. It is the responsibility of the licensee to prevent harm to the general public from operations at the site. Licensees are required to establish standards, procedures, and provisions for self-regulation. Enforcement through detailed regulations is not regarded in the U.K. as the most appropriate way of exercising regulatory control, because the delay inherent in changing statutory regulations and in developing detailed guides or codes could inhibit the introduction of new safety practices. The licensees are monitored by the relevant government organizations to see that they comply with the overall regulatory guidelines.

9.1.3 Linkage Between Nuclear Power and Waste Management

No stipulation law exists that links nuclear reactor construction to waste management plans.

9.1.4 Policy on Spent Fuel and Waste Management (IEAL 1987)

Until now, it has been the policy of the U.K. that spent fuel from its nuclear power stations (including Magnox and AGR spent fuel) be sent to the Sellafield site for reprocessing. No decision has been made regarding reprocessing of LWR fuel from U.K. reactors; however, LWR fuel from foreign reactors is scheduled for reprocessing in the Thermal Oxide Reprocessing Plant (THORP) under construction at Sellafield (Sills 1989).

The position in the U.K. is that a HLW repository will not be needed before 2040, and there are advantages to letting the waste age. The policy now established by the U.K. is to store HLW for 50 years after its solidification prior to disposal. The U.K. does not view storage as a substitute for disposal, and it is the policy to ultimately dispose of HLW in an environmentally safe manner in a deep geologic repository. Safe storage and transportation, however, warrants early stabilization of the liquid HLW. Furthermore, the U.K. does not want to foreclose on disposal options until a disposal facility is available. Its strategy is under continual development (Schneider et al. 1989).

Although the Central Electricity Generating Board (CEGB) has a major commitment with British Nuclear Fuels Ltd. (BNFL) to reprocess its spent fuel, consideration is being given to the possibility of long-term storage of future accumulations of discharged fuel and to deferment of reprocessing. One reason is the availability of dry storage that was not developed 25 years ago when the nuclear power program began. Although dry storage of spent fuel might be contemplated, until there is more certainty about the feasibility, cost, etc., the U.K. position is unchanged regarding ultimate reprocessing of the fuel and disposing of the vitrified waste in a geologic repository.

The organizations that produce radioactive wastes must carry the burden of responsibility for safe and effective management of the wastes; therefore, spent fuel and HLW management are the responsibility of the utilities and reprocessors.

9.1.5 Organization/Responsibilities for Waste Management (Schneider et al. 1988; Leigh 1989)

The overall organizational structure for the management of radioactive wastes in the U.K. involves three components: the government, the nuclear industry and generating boards, and the private sector.

Responsibility for radioactive waste management policy is vested in the Secretary of State for the Environment, together with the Secretaries of State for Scotland and Wales. Each consults with other Ministers as appropriate and is responsible for authorizing and administering discharges and disposal of radioactive wastes.

Her Majesty's Inspectorate of Pollution (e.g., the Radiochemical Inspectorate) of the Department of the Environment and inspectors of the Ministry of Agriculture, Food, and Fisheries are responsible for monitoring all radioactive discharges to ensure that they are within authorized limits. Regulatory activities in Scotland, Wales, and Northern Ireland are different but cover the same functions.

The Nuclear Installations Inspectorate (NII), which is part of the Health and Safety Executive, according to the Health and Safety at Work Act of 1974, is responsible for ensuring that high standards of waste management are maintained at licensed nuclear sites and that potential hazards are reduced to as low as reasonably achievable (ALARA).

An independent expert committee, the Radioactive Waste Management Advisory Committee (RWMAC), advises the government through the Secretary of State for the Environment in matters concerning national policy for radioactive waste management.

Responsibility for implementing strategy, however, clearly lies with the nuclear industry. BNFL carries out reprocessing of the spent fuel from the reactors operated by the Central Electricity Generating Board (CEGB) and the South of Scotland Electricity Board (SSEB). For now, BNFL has responsibility for the storage of the HLW arising from those operations. The decision to reprocess spent fuel immediately or to store it for some time lies with the CEGB in England and Wales and with the SSEB in Scotland.

National Power, PowerGen, and National Grid Company, successor companies to the CEGB, are to take CEGB's non-nuclear business into the private sector on March 31, 1990. CEGB's nuclear plants would be kept under government control as the company Nuclear Electric (Nucleonics Week 1990).

In 1982, the government established the Nuclear Industry Radioactive Waste Executive (NIREX) to develop and operate radioactive waste management facilities for low-level waste (LLW) and intermediate level waste (ILW). NIREX was originally established as a partnership of the U.K. Atomic Energy Agency (UKAEA), BNFL, CEGB, and SSEB but is now a separate legal entity as U.K. Nirex Limited. UKAEA performs research and development (R&D) on nuclear technologies, including reprocessing and waste management. It is a government-owned nuclear research agency, which has been operating on a fully commercial basis since 1986.

9.1.6 Repository Licensing Process, Regulation Status, Major Regulations, Safety Bases

The Department of the Environment believes that it is not yet necessary to identify and investigate candidate disposal sites, because HLW is expected to be stored for at least 50 years before disposal. Therefore, detailed regulatory requirements for the disposal of HLW in the U.K. have not been developed. With the policy in the U.K. being extended storage of HLW prior to disposal, the federal government [the Inspectorate of Pollution (HMIP) or the Nuclear Installations Inspectorate (NII)] has no plans in the near future to develop such regulatory requirements.

As required by Section 6 of the Radioactive Substances Act of 1960, radioactive waste disposal requires prior authorization by the authorizing departments, i.e., the Department of the Environment; the Scottish Office; the Welsh Office; and the Ministry of Agriculture, Fisheries and Foods. The responsibilities of these authorizing departments are 1) to approve radioactive waste disposal routes and disposal sites, 2) to assess the radiological implications of disposal, 3) to set other necessary conditions in authorizations and to ensure compliance with the limits and conditions, and 4) to carry out necessary monitoring of the environment for radioactivity.

9.1.7 Roles of the Public, Local Organizations, Multinational Organizations

Public involvement is accomplished by several means. The U.K. Nirex, Ltd. has an extensive public relations program, including local information offices, a mobile exhibit, and speaker services. A public inquiry system helps inform the public. Past problems with discharges at the Windscale site have aroused public concern with all nuclear activities, and the BNFL has responded to the concern with an extensive television ad campaign and major plant upgrades.

The U.K. participates in numerous international activities. It is an active member of the CEC, IAEA, and the OECD/NEA and is involved with most of their waste management initiatives. It has agreements and partnerships with various countries. In particular, the U.K. is a partner with France and the Federal Republic of Germany (FRG) in Nuclear Transport, Ltd., which provides for spent fuel transport, and in United Processors GmbH, which deals with reprocessing. Analogue studies in Brazil and in Australia also involve the U.K. (Schneider et al. 1988).

9.2 OVERALL WASTE MANAGEMENT SYSTEM

The U.K. approach to HLW management currently emphasizes reprocessing and storage and defers major repository development. The U.K. waste management system must deal with a range of spent fuel types, including AGR, LWR, HWR, GCR (Magnox) and LMFBR fuels. The system must also accommodate a wide range of fuel designs and materials (e.g., corrosion characteristics). It deals with both domestic and foreign spent fuel. It must allow for potential delays in reprocessing, and therefore, must provide buffer storage. While deferring major repository development, the management system needs to anticipate some aspects of repository development.

9.2.1 Reprocessing (Sills 1989; IAEA 1988)

The elements in the U.K. approach to reprocessing are 1) to reprocess Magnox fuel and recycle the uranium to AGRs, 2) to reprocess AGR fuel and foreign LWR fuel in THORP after its expected completion in 1992, 3) to reprocess a relatively small inventory (about 10 MT) of LMF8R fuel, and 4) to defer a decision on reprocessing PWR fuel from U.K. reactors by providing 18 years of storage capacity at each PWR.

The commissioning of THORP in 1992 will mark the start of commercial-scale oxide fuel (LWR and AGR) reprocessing in the U.K. The design rate of THORP is 1200 MT/yr, but it is expected to process about 7000 MT in its first

10 years of operation (Schmeider 1989). High-level waste conditioning at THORP is based on the French vitrification process (Schneider 1989). Light-water reactor fuel is destined for reprocessing in THORP as a result of contracts with overseas customers. Contracts for reprocessing of 6000 MTU of spent fuel have been secured (Sills 1989).

A major capital investment program will be completed in the early 1990s to ensure that Magnox reprocessing operations continue uninterrupted until the Magnox program ends around the year 2005. As of 1989, 30,000 MTU of Magnox fuel have been reprocessed at Sellafield, and over 15,000 MTU have been recycled to the AGR program; plutonium is stored for use in the fast breeder program and for mixed-oxide fuel for other systems. The chemical reactivity of Magnox fuel precludes direct disposal of the fuel, in contrast with well-proven reprocessing methods.

9.2.2 Interim Storage Before and After Reprocessing

Interim storage in the U.K. must accommodate a variety of fuel types, including GCR (Magnox and AGR), SGHWR, foreign and U.K. LWR, and LMFBR spent fuel. Storage facilities include at-reactor (AR) wet storage, AR dry storage (at one Magnox plant), AFR wet storage at Sellafield, and LMFBR storage at Dounreay. An above-ground engineered facility is planned for vitrified HLW storage in stainless steel canisters over a period of at least 50 years.

9.2.3 Geologic Repository (Schneider et al. 1988)

Some site appraisal work was conducted in the U.K. in the late 1970s, but no specific site is under active investigation. Tentative repository criteria have been proposed, including a minimum depth of 300 meters; entrances to shafts at least 60 meters above sea level; and deep mining, dam construction, etc., absent within a 15-kilometer radius of the facility. The present strategy is interim storage of vitrified HLW until the repository is available in about 2040.

9.2.4 Other System Considerations

The U.K. has supported the international assessment of sub-seabed disposal of HLW. For ILW, the possibility exists for an underground or an undersea storage facility to be built, possibly at Sellafield, with construction beginning in the mid-1990s (Schneider et al. 1988).

9.3 WASTE STORAGE AND TRANSPORTATION

The U.K. has a storage and transportation system that accommodates several fuel types, including U.K. and foreign fuel. Transportation is by truck, rail, and ship. Interim storage of HLW will become an important element in the system, to manage vitrified waste over a period of at least 50 years until the repository is available.

9.3.1 Spent Fuel Storage

The various spent fuel types in the U.K. impose a variety of storage requirements. Fuel with Magnox (magnesium-base alloy) cladding is susceptible to aqueous corrosion. Wet storage is practiced for short periods at reactor sites and at the Sellafield site, primarily to allow for decay of fission product gases, until the fuel is reprocessed. Dry storage of Magnox fuel in vaults has been practiced at the Wylfa power station since 1971.

AGR fuel has stainless steel cladding that sensitizes and becomes susceptible to aqueous corrosion, but at lower rates than the rates for Magnox fuel. The AGR fuel is stored in water pools AR and at the Sellafield AFR facility. Before the fuel is reprocessed, it is dismantled and consolidated. The graphite component is removed, and the fuel rods are then placed into canisters in close-packed arrays, achieving a 3:1 consolidation ratio. The canisters are stored in water until the AGR fuel is reprocessed in THORP, currently scheduled to be commissioned in 1992.

A dry storage option for AGR fuel is also being developed. Feasibility studies have been undertaken on dry storage systems for AGR fuel, carried out for the CEGB by the National Nuclear Corporation (NNC). The type of dry storage facility being developed by NNC is equally suitable for fuel from any type of thermal reactor. The NNC dry storage facility can also accommodate vitrified waste. A vault-type facility has been designed and a site has been identified. Dry storage would provide a buffer against reprocessing plant outages, provide the option to reprocess after longer storage, and provide the option for direct disposal of spent fuel, if economic and technical considerations are favorable. The same basic dry storage facility design could be adapted to other fuel types (Sills 1989).

Storage of LWR fuel in water over periods of several decades without significant degradation has been demonstrated. Foreign spent LWR fuel has been stored in water at the Sellafield site between one and two decades. The excellent wet storage characteristics of LWR fuel prompted a decision to install 18 years of wet storage capacity at each of the four or five PWRs that may be built in the U.K. over the next decade (Sills 1989).

9.3.2 High-Level Waste Storage

Liquid HLW is stored in acidic condition in cooled and agitated, double-walled, stainless steel tanks at the Sellafield site (Schneider 1988). It is estimated that 2860 cubic meters of liquid HLW will have accumulated by the year 2000. When converted to vitrified form, the waste volume will be reduced by about two-thirds, to 985 cubic meters. The vitrified HLW storage facility at the Sellafield site will store up to 8000 vitrified HLW containers, equivalent to 3300 cubic meters of waste in liquid form. The vitrified HLW will be stored in an above-ground engineered facility. It will be stored 50 years or longer in stainless steel thimbles in a vault cooled by natural convection air (Schneider 1988).

9.3.3 Transportation

BNFL, Pacific Nuclear Transport, Ltd. (PNTL), and Nuclear Transport, Ltd. (NTL) transport fuel within the U.K. and from other countries. Several types of casks are used, principally the Magnox cask, the Excellox cask, and the TN cask.

The wall construction of most Magnox casks is steel, but an early Chapelcross Magnox cask was constructed of cast iron. Magnox casks weigh 43 to 49 MT when loaded. Both the Excellox and TN casks have steel wall construction, with the laden weight being 76 to 102 MT for the Excellox casks and 85 to 111 MT for the TN casks. The Excellox casks can handle 5 to 7 PWR or 14 BWR assemblies; the TN casks, 6 to 12 PWR or 17 to 32 BWR assemblies, depending on the design. The casks containing Magnox fuel are shipped filled with water, the Excellox cask with air or water, and the TN cask with nitrogen.

The Chapelcross Magnox cask is transported only by road, but the others are transported by road, rail, and ship. Magnox fuel has been transported by ship from Italy and Japan beginning in the late 1960s. Oxide fuel has also been transported by ship from Japan and from European countries (IAEA 1988).

9.4 GEOLOGIC WASTE REPOSITORY (Schneider 1988)

The U.K. is planning for eventual geologic waste disposal but is presently pursuing long-term interim storage. Vitrified HLW will be stored for about 50 years before final disposal; by then the heat generation and radioactivity will be greatly reduced, and both transport and disposal will be simplified. Moreover, the 50-year period will give more time to identify and prove the suitability of a geologic disposal site. The U.K. strategy is to postpone further efforts to identify a repository site until a definite need is near. The U.K. will instead concentrate research on confirming applicability to the U.K. of findings from work in other countries. In deferring disposal, the U.K. does not view storage as a substitute, and its policy is to ultimately dispose of HLW in an environmentally safe manner in a deep geologic repository.

9.4.1 Safety Requirements and Approach

Detailed regulatory requirements for the disposal of HLW have not been developed in the U.K. because disposal is presently being deferred. Also, the U.K. does not plan to develop detailed prescriptive regulations; instead it will develop only general performance requirements (Bellington et al. 1990). The U.K. Department of the Environment will seek to assure that any exposure from a solid waste disposal site will not exceed an annual dose of 0.1 mSv (10 mrem) and that exposures to radiation will be as low as reasonably achievable (ALARA).

Tentative specifications and criteria for geologic repositories are that the minimum depth be 300 meters to avoid inadvertent intrusion; that entrances to shafts be at least 60 meters above sea level (projected shoreline should be higher than sea level to allow for a potential polar ice cap melt); and that deep mining, dam construction, etc., are absent within a 15-kilometer radius of the facility.

9.4.2 Siting

In the late 1970s, the U.K. was investigating 12 regions where geologic field research could be conducted for a permanent HLW disposal facility. An exploratory drilling program to investigate candidate repository sites was initiated but met with stiff local resistance, and only one application for drilling was accepted. The exploratory drilling program was suspended because the Department of the Environment took the view that the U.K. already has enough geologic data from domestic drilling operations, as well as from those of other nations. In addition, construction of pilot HLW disposal facilities, planned for the late 1990s, was no longer considered necessary. An internal review conducted by the government on whether it was essential to seek a site for an HLW repository concluded that waste could instead be stored safely on the surface for about 50 years. The government believes that it is not yet necessary to identify and investigate candidate disposal sites.

The inventory of potential sites had led to selection of several crystalline rock, argillaceous rock, and evaporite areas for further characterization. Exploratory drilling into granite was completed at one site in northern Scotland. In addition, a borehole was drilled into an argillaceous formation under the Harwell site for hydrologic studies. The present intention is to closely monitor developments involving underground facilities in Sweden and Canada for granite, in Belgium for clay, and in the FRG for salt.

9.4.3 Design Concept(s)

The reference waste form is vitrified HLW (borosilicate glass) stored from 50 to 100 years before disposal. The waste container will be designed to keep water from contacting the HLW glass for 500 years or longer; it is to be a stainless steel canister (similar to the French design) with an overpack of thin titanium, thick cast iron, or steel. The preferred host rock for disposal is granite. The emplacement of waste in the repository will be in-vertical boreholes drilled from gallery floors or within horizontal tunnels. The backfill will be bentonite or cement. The construction of the entire repository is to be completed before starting waste emplacement. Closure of the repository is to be delayed as long as is feasible to allow extended preclosure monitoring.

9.4.4 Retrievability and Monitoring

No requirements have been developed to date on retrievability of spent fuel or HLW from a repository, although strong public support has been indicated for retrievability (Marshall 1989). Also, requirements have not been developed for monitoring of the repository (IEAL 1987).

9.4.5 Waste Package System

The HLW form will be borosilicate glass containing 25 wt% (possibly increasing to 35 wt%) of total oxides. The vitrification process is based on that developed by the French (Schneider et al. 1989). The waste canister will be the same as that used in France: made of stainless steel, 0.43 meters in diameter, 1.3 meters high, and containing 0.15 cubic meters of HLW glass (400 kilograms of glass). Each canister will hold waste from about 1.9 MTU of LWR fuel or 8.65 MTU of Magnox fuel. The 169-L canister will be 89% filled with 151 L of glass. The canister will provide containment of waste during handling, interim storage, and transportation through emplacement (Schneider et al. 1988; IEAL 1987).

9.4.6 Research and Development (Schneider et al. 1988)

The U.K. has conducted its principal research, development, and demonstration at the UKAEA's Harwell Atomic Energy Research Establishment, Dounreay Nuclear Power Development Establishment, the Springfields Nuclear Power Development Laboratories, and the Risley Nuclear Power Development Establishment. The Research Group for Nuclear Energy at AERE has worked on waste treatment, disposal, and reprocessing; the Dounreay facility has performed FBR fuel cycle development, and Springfields has worked on fuel technology and waste conditioning.

Much of the HLW vitrification R&D has been performed at Harwell. The past work on a Joule-heated ceramic melter was conducted at Harwell for possible use at Dounreay. The work at Harwell has also included preparation and characterization of active samples of Synroc, development of glasses of increased durability, and characterization and treatment of gases released during melting. The rising-level in-can vitrification process was developed at Harwell and was tested in the Fingal and Harvest (now inactive) Pilot Plants.

Waste package R&D has been conducted, including formulation and characterization of vitrified HLW from fast breeder reactor (FBR) fuel reprocessing, behavior of vitrified waste under repository conditions, corrosion studies on canister materials, and backfill technology.

Geosciences R&D has been undertaken to determine the effects of waste emplacement in a geologic repository. Work has focused on geochemistry and near-field performance. Groundwater movement in fractured granite has been evaluated. The generic properties of saturated crystalline rock were studied, including heat transfer and the thermomechanical response of granite. Studies of the alteration products that are present in rock fissures have been made. Laboratory studies have evaluated radionuclide migration through backfill materials in "mini-repositories." Other testing has included radionuclide migration in fractures, long-term migration of natural elements in saturated clays, and waste/rock interactions at various temperatures. Natural analogue studies were made to assure that important repository elements are accounted for in modelling. As part of this effort, the U.K. has been participating in natural analogue studies in Brazil and Australia.

Field tests were accomplished at two locations. Early field studies of heat transfer, fracture hydrology, and radionuclide transfer were carried out at the AERE test site in a granite quarry in Cornwall. Hydrogeological tests were made in boreholes 300 meters deep in Scotland.

9.4.7 Approach to Proving the Safety of the Repository

The U.K. program in performance assessment resulted in publishing of safety assessment principles in 1984, promulgating risk assessment procedures based on the use of the probabilistic sampling codes SYVAC A/C and D, which are modified versions of the Canadian Systems Variability Analysis Code (SYVAC) system. A stochastic treatment of climate change is part of this approach.

Early in the 1980s, a very preliminary generic assessment was made for an inland granite site. Unrealistically pessimistic assumptions were used, and doses of 500 to 6000 mrem/yr were calculated. Peak doses occurred after 10,000 years. When a seaside location was assumed, doses fell two orders of magnitude due to dilution. A second assessment was done in the mid-1980s, and a third assessment is underway.

The latest assessment by the Department of the Environment (the regulatory authority) uses a unique time-dependent simulation modeling approach. The only other country that has invested in this approach is the U.S., but it is not currently being considered for use in U.S. assessments. The approach has been criticized in international meetings as being unlikely to yield defensible results, but because VANDAL, the model under development, has not yet been released, some of these opinions may be premature. Development of this approach is ongoing and will include thorough testing and comparisons with other approaches.

Other ongoing work in developing performance assessment methods involves a radionuclide migration model (CHEMTRAN), models for the long-term evolution of deep repository hydrogeology (TIME2/4), and models for biosphere transfer and dose (ECOS, DECOS). The linkages between the larger deterministic models and the simplified models used in a probabilistic systems code are also under investigation, and performance assessment is being used to identify waste form, engineered barrier, and site data needs.

9.4.8 Peer Review Activities

The U.K. participates in several multinational activities that provide access to broadly-based viewpoints. However, with a repository target date of 2040, needs for specific, definitive reviews related to the repository have not developed.

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10.0 UNITED STATES

10.1 LEGISLATIVE/INSTITUTIONAL BASES

10.1.1 Nuclear Power Policy (Schneider et al. 1988; Leigh 1989)

The nuclear power policy in the United States is to encourage construction and operation of nuclear power stations by private and public electrical utilities. The federal government has supported nuclear power development through major federal research and development (R&D) programs. Federal support in recent years has been declining, and much of the current R&D is being done by private industry. Current federal R&D is primarily aimed at new concepts of reactors and safety-related activities.

Nuclear power and the associated nuclear fuel cycle activities are regulated strongly by federal agencies. The Nuclear Regulatory Commission (NRC), which regulates most nuclear activities, also carries out independent nuclear safety research.

The United States (U.S.) has the largest nuclear electricity production capacity in the world, with 110 operating nuclear power reactors having a capacity of 99,000 MW, which is 20% of the national capacity. Nuclear power is still growing in the U.S. but is reaching a plateau, with facilities yet to come on line being those ordered more than 10 years ago. No new orders for nuclear power facilities have been received since 1978, and numerous previously planned facilities have been cancelled. The slowdown in nuclear power growth has been brought about by the added costs of long licensing and construction times (on the order of 12 years), increasing public pressures against nuclear power, and continually more restrictive regulations.

The U.S. has complete nuclear fuel-cycle capabilities for a once-through cycle. All fuel-cycle facilities are owned and operated by private industry, except for uranium enrichment facilities. Future facilities for central storage of spent nuclear fuel and for disposal of spent nuclear fuel will be owned and operated by the federal government, with funds supplied by the nuclear electrical utilities.

10.1.2 Major Legislation

The overall legislation for nuclear power is set in the Atomic Energy Act of 1954, with subsequent amendments. This Act specifies conditions for using atomic energy and radioactive materials for civilian and national defense purposes (U.S. Congress 1954).

The primary legislation for management and disposal of civilian spent nuclear fuel and high-level waste (HLW) was the Nuclear Waste Policy Act of 1982 (NWPA). It was amended in the Nuclear Waste Policy Amendments Act of 1987. These two Acts established a federal program and responsibility for a geologic repository for permanent disposal of these wastes. They also provided authorization for transportation, gave responsibility for the waste management system to the Department of Energy (DOE), dictated the maximum capacity of the repository, and provided institutional requirements, as well as funding and a timetable. The Amendments Act named the Yucca Mountain site in Nevada for detailed site characterization to determine its suitability as a repository and authorized a Monitored Retrievable Storage (MRS) facility for interim storage of spent fuel (subject to a number of conditions before construction can start) (U.S. Congress 1983, 1987).

10.1.3 Linkage Between Nuclear Power and Waste Management

There are no national legislative acts that link the development or operation of nuclear power to assurance that radioactive wastes can be managed satisfactorily (Schneider et al. 1988).

10.1.4 Policy on Spent Fuel and Waste Management

The primary objective of radioactive waste management in the U.S. is to protect the health and safety of the public and the quality of the environment cost effectively. Management of radioactive wastes is considered to be the responsibility of the entities and immediate generations that produce the wastes and should not be left for future generations. U.S.-generated radioactive wastes will be disposed of in the U.S., and spent fuel and HLW will be disposed of in deep geological formations. Consideration of a possible second repository is deferred until the time period 2007 to 2010. Interim storage of spent fuel and HLW will be carried out at the site of the generator, although the federal waste management system has proposed a central MRS facility (U.S. Congress 1983, 1987; DOE 1988a).

10.1.5 Organization/Responsibilities for Waste Management

Management of all types of radioactive wastes up to final disposal is the responsibility of the respective waste generators. For disposal of all spent nuclear fuel and HLW, the DOE is responsible. For disposal of civilian low-level waste (LLW), the States are responsible and are joining into groups of States called "compacts" to share the responsibility for LLW disposal on a regional basis (DOE 1988a; OECD/NEA 1988a).

The DOE is responsible for the total waste management system associated with disposal of spent nuclear fuel and HLW. This includes any interim storage beyond that provided by the waste generators; transportation to any federal interim storage facility and to the final repository; and siting, development, operation, closure, and monitoring of the final repository (U.S. Congress 1983; DOE 1988a).

The DOE is also responsible for setting the fee schedule that the waste generators pay to fund the federal waste management system for spent nuclear fuel and HLW. The Nuclear Waste Fund is collected and managed by the federal Treasury Department. Authorization for use of the waste fund is done on an annual basis by the Congress, based on need. The DOE annually evaluates the adequacy of the fee schedule of the waste fund. The fee schedule has been $0.001/net kWh of electricity sold by the nuclear utilities since the inception of the fee in 1983 (DOE 1988a; Schneider et al. 1988).

The federal Environmental Protection Agency (EPA) is responsible for setting overall standards for protection of the environment and public health for all hazardous materials and activities (DOE 1988a).

The federal Nuclear Regulatory Commission (NRC) has primary responsibility for review of the nuclear safety aspects and for licensing of interim storage and repository siting, construction, operation, and decommissioning.

The NRC is also responsible for inspecting and certifying casks for use by DOE in transporting the spent fuel and HLW (DOE 1988a).

The federal Department of Transportation (DOT) regulates the safe transport of hazardous materials, including nuclear waste, under the federal civilian waste management system (DOE 1988a).

10.1.6 Repository Licensing Process, Regulation Status, Major Regulations, Safety Bases

The regulatory requirements for licensing a geologic disposal repository, an MRS facility, and the associated transportation system are relatively complete, although they are continually undergoing review and revision. The regulations are highly prescriptive in defining details of safety requirements that must be met (NRC 1983; EPA 1985).

The overall federal repository licensing process involves two major steps: obtaining a license for construction of the repository and obtaining a license for operation (i.e., emplacement of the wastes). In addition, regulatory approval is required to seal and close the repository (U.S. Congress 1983).

Submission of a license application to the NRC for construction of a repository requires DOE to make a finding that the proposed repository site is suitable as a result of detailed site characterization studies. After this finding, the DOE prepares a final environmental impact statement (EIS) and recommends the site to the President; assuming the recommendation is accepted by the President and Congress and no disapproval is received from the host State (or host Indian tribe, if applicable), the DOE may then submit the license application for repository construction. If, when the President submits the site recommendation to Congress, the host State or host Indian tribe submits a notice of disapproval of the decision, the site is disapproved unless the Congress overrides the disapproval. When the NRC finds the license application to be acceptable, authorization is given for construction. After repository construction is completed, the DOE submits an application to the NRC for operation of the repository. If the construction and new safety-related data satisfy the safety regulations, the NRC will issue a license to receive and possess radioactive materials, i.e., the operation of and emplacement of wastes in the repository (U.S. Congress 1983, 1987).

A permit is required from the host State for any major construction work to assure that air quality standards will be met during construction. This permit is required for construction of the Exploratory Shaft Facility (underground research laboratory), as well as for the repository (Nuclear Waste News 1989).

After completion of the operating and retrievability periods in the repository, the DOE must apply to the NRC for a license amendment for permanent closure. After a period of post-closure monitoring, DOE may apply to the NRC for a license amendment to terminate the repository license (NRC 1983).

The regulations for licensing of a federal MRS facility for interim storage of spent nuclear fuel and/or HLW are given in the NRC's 10 CFR 72. Licensing of an MRS facility is a one-step process, with approval of the site and design and operations occurring simultaneously. An MRS license is active for 40 years and is renewable (NRC 1988a).

The regulations for transportation of hazardous and radioactive materials are promulgated by DOT in 49 CFR 171-176 and supporting regulations. Transportation casks for spent fuel and HLW must be certified by the NRC. The U.S. transportation regulations are consistent with the international transportation standards issued by the International Atomic Energy Agency (IAEA) (DOT 1988; NRC 1988b).

The basic federal standard for environmental radiation protection for operation of uranium nuclear fuel cycle facilities is the EPA's 40 CFR 190. This regulation applies to waste storage and to filling and presealing of waste disposal repositories but not to post-sealing disposal. The standard states that the dose equivalent to any member of the public for expected performance of all operations in the nuclear fuel cycle shall not exceed 75 mrem/yr to the thyroid or 25 mrem/yr to the whole body or to any other organ. All EPA and NRC regulations on radiation protection include the requirement for keeping radiation doses from all activities as low as reasonably achievable (ALARA), over and above meeting the specified numerical requirements (EPA 1984).

The federal standard for environmental radiation protection after sealing of waste disposal repositories is the EPA's 40 CFR 191. The EPA standards limit the projected cumulative releases of radionuclides to the accessible environment for 10,000 years after disposal to specific numerical values that are based on limiting the resulting premature cancer deaths. The EPA standard also provides an annual limit of radiation doses to members of the public for 1000 years after disposal and sets numerical limitations on radionuclide concentrations in potable groundwaters for this same time period. Currently, parts of these standards have been remanded and are being revised (EPA 1984, 1985).

The basic NRC radiation protection regulations, which expand on the EPA standards and are consistent with those of the International Commission on Radiation Protection, are given in the NRC's 10 CFR 20. These regulations limit the dose to any individual member of the public to 500 mrem/yr and the dose to occupational workers to 5 rem/yr averaged over a lifetime. Other specific numerical limits are also given (NRC 1984).

The NRC regulations for disposal of spent nuclear fuel and HLW are in 10 CFR 60 and require preserving the option of waste retrieval throughout the pre-closure period of the repository. For the post-closure period, the primary technical requirements are for containment release rates and groundwater travel time, providing specific numerical limits in each case. Also contained in 10 CFR 60 are siting, facility design, and waste package criteria; criteria pertaining to land ownership and control; and requirements for establishing programs for performance confirmation, quality assurance, and personnel training and certification (NRC 1983).

In addition to the regulations by the EPA, the NRC and the DOT, the DOE has its own operating "Orders" that are compatible with the regulations of these regulatory agencies. As an example, a requirement in these "Orders" is a stipulated design objective of no more than 1 rem/yr maximum individual occupational dose for DOE facilities (DOE 1988b).

10.1.7 Roles of the Public, Local Organizations, Multinational Organizations

The public has the opportunity to play a major role in all aspects of developing the high-level radioactive waste management system. The public influences its representatives in Congress in enacting legislation. All proposed major federal actions that could affect the environment require the preparation of EISs, which require public comment and input before the decision is made to proceed with the proposed action (U.S. Congress 1969). Because the results of most regulations related to radioactive waste management will affect the environment, proposed regulations are also subjected to public review and comment before they can be promulgated. In these cases, bonafide concerns must be resolved before action can be taken.

The NWPA and its Amendments Act specify public hearings on the detailed site characterization plan for nominated potential repository sites. The NWPA also requires that, before proceeding with sinking of shafts for site characterization of a repository, the DOE submit the plan to the NRC and the potential host State for comments and hold public hearings on the proposed activities. After detailed site characterization and evaluation of a specific repository site indicates a site is suitable and the DOE plans to recommend a final site for a repository, the host State must be notified and public hearings must be held in the host State. The final recommendation of a site to the President must be accompanied by an EIS that includes earlier comments from several other federal agencies and the prospective host State, as well as DOE responses (U.S. Congress 1983, 1987).

If the President agrees with the recommendation of the proposed site for development of a repository, he submits the recommendation to Congress for approval. At this time, the prospective host State or host Indian tribe has 60 days to submit a notice of disapproval of the selection. In this case, Congress has 90 days to override the disapproval, or the disapproval becomes binding. A similar process for site recommendation and approval, State and

Indian tribe veto, and congressional override of the veto applies to an MRS facility (U.S. Congress 1983, 1987).

Potential repository host States or Indian tribes may participate in the site selection and evaluation process by reviewing the federal activities and results, carrying out independent monitoring programs, independently determining impacts on the host area, employing an onsite state representative to oversee DOE activities, and carrying out public information programs within the state. The potential host State is paid for these activities from the Nuclear Waste Fund. Until the activities cease at the site, the host repository and MRS facility States and units of local government also receive grants equivalent to taxes that would be received if the federal activities were private industrial activities. In addition, after repository or MRS facility sites have been selected, potential host States may enter into an agreement with the DOE for federal payments of benefits (fixed amounts specified) to the potential host States, who must share the benefits payments with units of local government. Any potential host State that submits a notice of disapproval of its recommended site waives its rights to these payments. Also, Nuclear Waste Funds may be paid to the host States to mitigate impacts resulting from hosting a repository or MRS facility, including training of public safety officials for transportation emergencies (U.S. Congress 1983, 1987).

A potential host State for a repository or MRS facility may participate in an oversight review panel with seven members, to be funded by the Nuclear Waste Fund. Two of the panel members, including the chairman, are to be appointed by the DOE, four are to be appointed by the potential host State and units of local government, and one is to be selected by DOE from nuclear electric utilities (U.S. Congress 1987).

The Amendments Act provides for the President to appoint a nuclear waste negotiator, who will terminate activities in 1992. The negotiator shall attempt to find a State or Indian tribe that has a suitable site and is willing to host a repository or MRS facility. If such a potentially qualified volunteer site is identified, the process for evaluating and characterizing the potential site will be much the same as for a DOE-designated site, with similar public roles (U.S. Congress 1987).

In accordance with the Amendments Act, an independent three-member MRS Commission was appointed by the President to review the need for an MRS facility and to recommend to Congress whether such a facility should be included in the national waste management system (U.S. Congress 1987). The MRS Review Commission completed its work and submitted its report to Congress late in 1989. The Commission concluded that either MRS or No-MRS options for interim storage of spent fuel are technically safe and there are no single discriminating factors that would cause the MRS alternative to be chosen in preference to the No-MRS option; an MRS facility with its implementation schedule linked to the repository implementation schedule is not justified; and some small federal interim storage facilities are in the national interest. The Commission recommended congressional authorization of a federal emergency storage facility with a capacity limit of 2,000 MTU of spent fuel; congressional authorization of a user-funded interim storage facility with a capacity limit of 5,000 MTU of spent fuel; and Congress shall reconsider the subject of interim storage by the year 2000. Congress is currently considering the MRS Review Commission's findings and recommendations (MRS Review Commission 1989).

The Amendments Act provides for establishing a Nuclear Waste Technical Review Board to evaluate the technical and scientific validity of activities undertaken by the DOE related to repository site characterization, waste packaging, and transportation. The 11-member board reports to the President, and the members have been selected from candidates identified by the independent National Academy of Sciences (NAS) (U.S. Congress 1987).

The NRC has appointed an Advisory Committee on Nuclear Waste. The three-member independent panel and its staff provide advice to the Commission on regulatory issues for all types of radioactive wastes (Nuclear News 1988).

The U.S. is a member of the IAEA and the OECD/NEA and participates actively in their waste management programs, including R&D study projects. The U.S. has participated in reviews of other country's waste management systems when requested through these agencies but has not requested these organizations to participate in or review the U.S. program.

10.2 OVERALL WASTE MANAGEMENT SYSTEM (OECD/NEA 1988a)

The overall HLW management plan for the U.S. is to store spent fuel from civilian nuclear power stations on an interim basis for 5 to 40 years and then dispose of canistered spent fuel directly in a deep geologic disposal repository. Spent fuel storage will be in the reactor pools, with some nuclear power stations implementing their own supplementary onsite storage in dry storage systems. A federal MRS facility has been proposed for interim storage of spent fuel that has been accepted within the federal waste management system. However, the MRS Review Commission and DOE positions on an MRS facility are of interest and unresolved (U.S. Congress 1983, 1987; MRS Review Commission 1989).

High-level waste from the 640 MT of civilian spent fuel that has been reprocessed at the former West Valley, New York, facility will be vitrified and disposed of in the deep geologic repository. In addition, vitrified HLW from nuclear defense programs will be disposed of in the same deep geologic repository (DOE 1989a).

The HLW repository is limited by legislation to receive no more than 70,000 MT of spent fuel (including the equivalent in HLW) until a second repository (if needed) is started up. The current reference estimate for life-time waste generation is about 86,000 MT of spent fuel and about 9,500 MT equivalent of civilian and defense HLW (DOE 1989a).

10.2.1 Reprocessing (Schneider et al. 1988; Leigh 1989)

Reprocessing of civilian spent nuclear fuel is permitted by law and is determined by the owners of the spent fuel. However, the utilities with nuclear electrical power have elected not to reprocess because of poor economics and because of uncertainties in policy regarding the nuclear fuel cycle.

The former Nuclear Fuel Service's West Valley reprocessing facility operated from 1966 to 1972 and is currently being decommissioned.

Reprocessing plants for civilian nuclear power plant spent fuel were also constructed by the General Electric Company at Morris, Illinois, and by the Allied General Nuclear Services Company at Barnwell, South Carolina. These two plants were never completed and have never operated.

10.2.2 Interim Storage Before and After Reprocessing (U.S. Congress 1983, 1987)

Interim storage of civilian spent fuels is the responsibility of the owners of the nuclear power stations. The spent fuel will continue to be stored in the reactor pools, with some nuclear power stations implementing supplementary onsite storage in dry storage systems, as needed. Storage will be at the nuclear power stations for five to about 30 years, until the spent fuel is received by the federal transportation system and transferred to the federal waste management system for disposal in the deep geologic repository. The vitrified HLW from the former civilian reprocessing facility at West Valley, New York, and vitrified defense HLW will be interim stored at the respective sites of their origin.

A federal MRS facility has been proposed by DOE and authorized by Congress. However, congressional appropriation of funding for an MRS facility is dependent on resolution of completing recommendations of DOE and the MRS Review Commission.

10.2.3 Geologic Repository

A federal repository in a deep geological formation is planned for civilian spent nuclear fuel and for both civilian and defense HLW. The repository is planned to be operational in 2010. A site at Yucca Mountain in Nevada has been selected by Congress to undergo characterization to determine its suitability for the national repository (U.S. Congress 1983, 1987; DOE 1989a).

10.2.4 Other System Considerations

There are no plans to dispose of wastes in the federal HLW repository, except canistered spent fuel from civilian nuclear power reactors; vitrified HLW from the former civilian reprocessing plant at West Valley, New York; and vitrified HLW from defense production operations. For spent fuel that might be consolidated, either at some nuclear power stations, at the repository, or at the proposed MRS facility, the non-fuel-bearing components from the consolidated spent fuel are expected to be disposed of at the same repository, although the final decision has not yet been made (DOE 1988c). Recent analyses indicate that most of the non-fuel-bearing hardware from spent fuel has sufficient radioactivity to exceed the levels acceptable as LLW, and the material will likely require disposal in a deep geological repository (Luksic et al. 1989). Furthermore, the NRC has ruled that greater-than-Class C waste is required to be disposed of in a deep geological repository unless the NRC approves disposal elsewhere (NRC 1989).

Because further reprocessing of civilian spent fuel is not planned, partitioning of civilian HLW into fractions for reuse or other disposition is not planned.

Legislation provides for carrying out research on subseabed disposal of spent fuel and HLW as an alternative technology to deep geological disposal. Activities in this area have been minimal since completion of the OECD/NEA subseabed disposal project in 1987 (U.S. Congress 1987; OECD/NEA 1988b).

The U.S. participated in the OECD/NEA's preliminary study on the feasibility of a multinational repository but has expressed no particular interest in pursuing such a project (OECD/NEA 1987).

10.3 WASTE STORAGE AND TRANSPORTATION

Storage of spent fuel and HLW is the responsibility of the generator until they are accepted by the federal waste management system for disposal. The DOE has recommended, however, the implementation of an integral MRS facility for spent fuel. The integrated MRS facility would be an in-line facility in the waste management system. The facility would receive spent fuel, provide a limited amount of storage, provide staging for transportation to the repository, and other functions (such as waste packaging) if determined to be desirable in future analyses (DOE 1987, 1989b; U.S. Congress 1983, 1987).

Transportation of the civilian spent fuel and HLW to a federal repository or MRS facility will be the responsibility of the DOE and is considered to be part of the federal waste management system (U.S. Congress 1983, 1987).

10.3.1 Spent Fuel Storage

The spent fuel from nuclear power stations will continue to be stored in the respective reactor pools, with some nuclear power stations implementing supplementary onsite storage in dry storage systems, as needed, if pool storage capacity is exceeded. Storage will continue at nuclear power stations for 5 to about 40 years until the spent fuel is received by the federal transportation system and transferred to the federal waste management system for disposal in the deep geologic repository. By the year 2000, approximately 42,000 MT of spent nuclear fuel will be in storage (Strahl 1988; DOE 1988d).

Because many of the nuclear power stations were originally constructed with a small amount of in-pool storage capacity, most of them have or are initiating storage expansion. Nearly all are installing poisoned storage racks to reduce the spacing and thereby increase the storage capacity in their storage pools by a factor of about two (Strahl 1988).

Some nuclear power stations are installing dry storage to satisfy their supplemental storage needs for spent fuel. Two types of onsite dry storage facilities are currently being used. One is using metal storage casks of various types with capacities typically from 21 to 26 intact PWR assemblies or the equivalent of BWR assemblies. The casks are stored vertically on an outside concrete pad on the reactor site. The casks are loaded with spent fuel in the spent fuel pool and lifted from the pool, the contained liquid and gases are evacuated, and the casks are then back-filled with inert gas and sealed mechanically. The other concept uses a series of modular horizontal concrete chambers placed next to one another on outside concrete pads. Each module holds one canister (with integral shielding on each end) with 7 to 24 intact PWR spent fuel assemblies. The spent fuel is canisterized, evacuated, and backfilled with inert gas in the reactor storage pool, then removed from the pool in a transfer cask. The transfer cask is mated to the storage module, and the canister is pushed or pulled externally into the storage module.

Each module is cooled passively by external air convection and internal air convection chambers (Strahl 1988).

Some nuclear power utilities are considering and/or demonstrating in-pool consolidation of the rods from spent fuel assemblies into metal canisters with the same dimensions as the original intact assembly to increase their spent fuel storage capacity in existing pools. The non-fuel-bearing components remaining after consolidation of the spent fuel rods may be compacted into metal canisters and stored in the reactor pool. The rod consolidation results in a reduction in spent fuel volume by up to a factor of two, not including the volume of the separately canistered non-fuel-bearing components, which constitutes about 10% of the volume of the original spent fuel assemblies (Strahl 1988).

One small commercial away-from-reactor (AFR) storage facility is in operation at Morris, Illinois, storing about 560 MT of spent fuel. This facility uses the originally planned lag storage pool for incoming spent fuel and the interconnected interim storage pool for vitrified HLW at the commercial fuel reprocessing plant that has never operated (DOE 1988d).

A federal MRS facility has been proposed to Congress. This facility, if constructed, would provide up to 15,000 MT of AFR storage of spent nuclear fuel, and would be started up in 2003. The current concept is intact storage of spent fuel in storage canisters that are backfilled with inert gas and sealed. The spent fuel would be stored either in sealed concrete storage casks (the primary concept), in-ground sealed steel caissons, or in concrete vaults. At the end of the interim storage period, the canisters would be transferred to transportation casks for transport to the repository for further packaging before emplacement. The concept using concrete storage casks (about 12 feet in diameter and 22 feet high) would hold up to 32 intact PWR or 80 intact BWR assemblies. The casks would be stored vertically on concrete pads. If consolidation is implemented, the capacity of individual storage units would be increased, and the compacted non-fuel-bearing hardware would be placed in 55-gallon steel drums (hardware from 7 PWR or 7 BWR assemblies in each drum). Final decisions on whether to construct an MRS facility, the site, the canisters and storage concept to be used, and whether or not consolidation would be used all remain to be made (Parsons 1985; DOE 1989a, 1989b).

10.3.2 High-Level Waste Storage

The vitrified HLW from the former civilian reprocessing facility at West Valley, New York, and vitrified defense HLW will be interim stored at the respective sites of their origin. The storage facility at West Valley will utilize an existing process cell that is being modified for storage of the HLW canisters. The 300 canisters (2 feet in diameter and 10 feet long) of vitrified HLW will be stored in a vertical orientation in air (DOE 1988d).

10.3.3 Transportation

Transportation of civilian spent nuclear fuel between nuclear power plants or to licensed commercial interim storage facilities is the responsibility of the waste generator. Certification of the transport casks by the NRC is the responsibility of the cask manufacturer. These transports have been carried out for more than twenty years by truck and rail in a variety of commercial casks with capacities from 1 PWR or 2 BWR assemblies to 3 PWR or 7 BWR assemblies in truck casks and up to 21 PWR or 48 BWR assemblies in rail casks. The empty truck casks weigh 22 to 37 tons, and the rail casks weigh 65 to 89 tons. Larger casks are also being considered. No transportable storage casks have yet been licensed for transportation in the U.S., but licensing of such casks is being considered (Johnson and Notz 1988).

Transportation of civilian spent nuclear fuel between the nuclear power plants or licensed commercial interim storage facilities to the repository or MRS facility is the responsibility of the DOE and is part of the federal waste management system. Preliminary designs of the casks for transporting 10-year-old spent fuel from the civilian facilities is in progress. Two truck casks with capacities of 4 PWR or 9 BWR intact assemblies and three rail casks with capacities of 16 to 26 PWR or 40 to 52 BWR intact assemblies are under development. The empty truck casks will weigh about 28 tons, and the empty rail casks will weigh about 73 to 80 tons. Shipments from reactors by rail may be on dedicated trains with up to five casks per train. Similar rail casks are planned for transport of the vitrified HLW canisters (five canisters per cask load) from civilian and defense facilities. The rail casks could also be used for transport on barges for some shipments (DOE 1989a).

If an MRS facility is constructed as part of the federal waste management system, it has been proposed that all spent fuel be shipped from the MRS facility on dedicated trains with five large rail casks each. Development of these rail casks has not yet been initiated, but the casks are expected to weigh 112 to 125 tons empty and contain 34 PWR or 80 BWR intact assemblies, or up to 56 PWR or 140 BWR equivalent assemblies if the spent fuel is consolidated. If spent fuel is consolidated, similar casks could be used, each containing about twenty 55-gallon drums of the compacted non-fuel-bearing components of the spent fuel (DOE 1989a).

10.4 GEOLOGIC WASTE REPOSITORY

Disposal of civilian spent fuel and HLW and all other civilian wastes that are not acceptable for disposal in LLW facilities is the responsibility of the DOE. The DOE is also responsible for disposal of defense HLW, which will be disposed of in the same repository as civilian HLW. The federal government has been carrying out research and development on disposal since 1955 and began carrying out underground research in an abandoned salt mine in 1965. Development of a repository program has continued since that time and will continue until startup of the repository, scheduled for 2010 (DOE 1989c).

10.4.1 Safety Requirements and Approach

The basic federal standard for environmental radiation protection for waste disposal repositories is the EPA's 40 CFR 191. The same numerical dose limits for operation of fuel-cycle facilities are stated to cover the operational phase of waste repositories. The EPA standards limit the projected cumulative releases of radionuclides to the accessible environment for 10,000 years after disposal to specific numerical values. These values are based on limiting the resulting premature cancer deaths to no more than an average of 0.1/yr from disposal of each 100,000 MT equivalent of spent fuel and HLW. The EPA standard also limits the doses to individual members of the public to

4 mrem/yr and sets numerical limitations on radionuclide concentrations in potable groundwaters for 1000 years after disposal. Currently, parts of these standards have been remanded and are being revised (EPA 1984, 1985).

The NRC regulations for disposal of spent nuclear fuel and HLW, which must be consistent with those of the EPA, are in 10 CFR 60. They require the preservation of the option of waste retrieval for 50 years after waste emplacement is started. For the post-closure period, the primary technical requirements are for containment. These include 1) substantially complete containment within the engineered barriers for 300 to 1000 years, 2) a maximum release rate of any radionuclide from the engineered barrier system of one part in 100,000 per year of the inventory of that radionuclide calculated to be present at 1000 years after permanent repository closure, and 3) a minimum of 1000 years for groundwater to travel from the "disturbed zone" around the underground facility to the accessible environment. 10 CFR 60 also contains siting, facility design, and waste package criteria, and criteria pertaining to land ownership and control, as well as requirements for establishing programs for performance confirmation, quality assurance, and personnel training and certification (NRC 1983).

The Nuclear Waste Policy Act of 1982 requires the use of multiple barriers to assure long-term safety of the repository. The NRC regulations conform to this legislative requirement (U.S. Congress 1983; NRC 1983).

The DOE's approach to meeting the regulatory performance requirements is to have the waste package meet the 300- to 1000-year requirement, and thereafter, to use other engineered barriers and the site geohydrology to provide the primary barriers for preventing the waste radionuclides from reaching the accessible environment. The approach to assurance of meeting these requirements is by modelling of the expected performance of the repository into the distant future. The DOE, EPA, and NRC are each developing independent models to assure that the safety requirements are met (DOE 1988e).

10.4.2 Siting

The DOE and its predecessor agencies have been engaged in activities to identify potentially acceptable sites for geologic repositories since the

1960s. As required in the Nuclear Waste Policy Act of 1982, the DOE established general guidelines for recommendation of potential sites for repositories, including qualifying and disqualifying criteria. The criteria considered the environment, socioeconomics, transportation, and technical feasibility and cost of repository development (U.S. Congress 1983; 10 CFR 960).

In 1983, nine potentially acceptable repository sites were identified for the first repository, including sites in bedded and in domed salt, in basalt, and in welded tuff. From surface-based analyses and environmental assessments, five of the sites were selected as candidates for site characterization in 1986. The three final sites, which included one in each of the three types of rocks, were then nominated for site characterization in 1986. These site selections were accomplished using the DOE site-selection guidelines. In the 1987 Amendments Act, Congress selected the Yucca Mountain site (with welded tuff as the host rock) in Nevada as the sole site for characterization (U.S. Congress 1987).

Site characterization consists of surface-based investigations, studies conducted using deep and shallow boreholes, laboratory tests, and most importantly, tests conducted in the host rock at the proposed horizon of the repository. The latter is planned to be carried out in an exploratory shaft facility. All tests are to be carried out in concert with continuing performance assessment studies. A site-characterization plan for the Yucca Mountain site has been issued that prescribes the details of the site characterization tests and evaluations that will be carried out over a period of about 6 to 8 years. High priority will be given to determining potential adverse conditions and effects on meeting regulatory requirements and on identifying potentially unfavorable conditions. Currently, only surface-based tests are in progress, with construction of the exploratory shaft facility to be initiated over the next one to two years. Upon completion of site characterization and evaluation with favorable results, an EIS will be prepared, and the site will be recommended for development as the deep geologic repository (DOE 1988e).

In addition to the siting process identified above, the Amendments Act of 1987 provides for the President to appoint a nuclear waste negotiator, who will attempt to find a State or Indian tribe that has a suitable site and is willing to host a repository. If such a potentially qualified volunteer site is identified, the process for evaluating and characterizing the potential site will be the same as for a DOE-designated site (U.S. Congress 1987).

In addition to the original nine potential sites identified for the first repository, other potential sites, primarily in crystalline rocks in the eastern part of the U.S., had been identified for a possible second repository. Further work on site selection for a possible second repository was discontinued in 1986, and the Amendments Act of 1987 specified consideration for a second repository to be deferred until 2007 to 2010 (U.S. Congress 1987).

10.4.3 Design Concept(s)

The reference concept for the Yucca Mountain site is a preliminary design concept developed as the basis for carrying out site characterization. The design concept will continue to evolve until the characterization of the site is completed (MacDougall 1987).

The repository concept is designed to accept 63,000 MT of spent fuel and 7000 MT of HLW equivalent (in about 14,000 canisters) from West Valley and DOE defense sources). The steady-state receiving rate is 3000 MT/yr of spent fuel and 400 MT/yr equivalent of HLW after a capacity ramp-up over the initial five years of operation (MacDougall 1987; DOE 1989a).

The repository surface facilities will receive and handle the waste and place it into sealed disposal containers, then transfer the waste down to the underground disposal rooms. The underground portion of the repository will consist of underground structures such as shafts, rooms and drifts, and engineered barriers, backfill and seals, and other components (MacDougall 1987).

The proposed disposal horizon is unsaturated welded tuff about 13 million years old. The disposal formation is 330 to 570 feet thick at the proposed repository location and 660 to 1300 feet above the water table. The disposal horizon is 660 to more than 1000 feet below the surface of Yucca Mountain and about 3500 feet above sea level. The conceptual disposal area will be made up of 18 emplacement panels containing emplacement drifts, with each panel 1400 feet wide by 1500 to 3200 feet long (MacDougall 1987; DOE 1988e).

Packaged waste from the surface facilities will be transported down an inclined ramp to the underground disposal area on a truck transporter. At the disposal horizon, it will be transported through access tunnels to a rectangular array of disposal drifts or tunnels whose total length is about 600,000 feet. Disposal will be either in vertical boreholes in the floors or in horizontal boreholes in the walls of the disposal drifts (which are at one level). In the reference concept using disposal in vertical boreholes, one canister will be emplaced in each of the boreholes (29 inches in diameter by 25 feet deep for spent fuel or 20 feet deep for vitrified waste). The upper half of the emplacement boreholes are lined with metal, and the bottom half are unlined. Immediately after emplacement, each borehole will be fitted with a metal shield plug, backfilled with crushed tuff, and fitted with a metal cover plate. Drifts for vertical emplacement are 126 feet between centers, they are 16 feet wide and 22 feet high. The distance between spent fuel disposal boreholes is 15 feet, or 7.5 feet between alternating spent fuel and HLW boreholes, as dictated by the heat content of the packages (MacDougall 1987).

The tentative thermal load limit in the repository for reference spent fuel (10 years old and irradiated to 33,000 MWd/MT) is 57 kW/acre. The tentative maximum allowable design temperature of the borehole wall is 275°C and the temperature at 1 m from the borehole is less than 200°C. In the far-field, the tentative allowable temperature increase is 5°C in nearby aquifers and 0.5°C in the earth's surface (DOE 1988e).

10.4.4 Retrievability and Monitoring

The Nuclear Waste Policy Act of 1982 requires restrictions on the retrievability of the emplaced waste. The NRC regulations require the option to retrieve the waste throughout the emplacement period and thereafter until the completion of a performance confirmation program and NRC review to assure that the performance is confirmed. The regulations require that all waste be retrievable on a reasonable schedule starting at any time up to 50 years after waste emplacement operations are initiated. The DOE has assumed that retrieval operations, if any, should be carried out in about the same time as the emplacement operations and have allowed 34 years for this potential activity. Thus, the total period of retrievability is taken to be about 84 years after start of waste emplacement. Retrievability of the waste after this time frame has not been considered (U.S. Congress 1983; NRC 1983; MacDougall 1987).

Monitoring of the performance of the repository will be carried out during the period of retrievability, and the results will be used in the post-emplacement performance confirmation analyses. Additional monitoring after final repository closure is not expected to be required (DOE 1985).

10.4.5 Waste Package System

The waste package system in terminology used in the U.S. section of this report includes the waste form (spent fuel or vitrified HLW), the waste canister (which surrounds the vitrified HLW), the disposal container (which surrounds the spent fuel and high-level waste canisters), and buffer or packing material (if any) surrounding the outer waste container. The NRC regulations require substantially complete containment by the waste packages for a minimum of 300 to 1000 years (i.e., the final requirement within this time frame is to be determined). The DOE is designing the disposal containers for 1000-year life expectancy (NRC 1983; DOE 1989a).

The reference conceptual waste container for spent fuel has been a cylinder of stainless steel with a wall thickness of 0.375 inches, an outer diameter of approximately 28 inches, and a length of 15.6 feet. However, recent studies indicate that this reference material may be changed. Spent fuel must be at least five years old since discharge from the reactors, but the reference design basis is for 10-year-old fuel. For intact spent fuel assemblies (the current reference concept), three PWR or six BWR assemblies (or a combination of PWR and BWR assemblies) are proposed to be in one container. The assemblies will be placed between webbed separators inside the container. If rod consolidation is implemented, the fuel rods from 6 PWR or 18 BWR assemblies (or a combination of rods from PWR and BWR assemblies) would be in one container, also separated by webbed spacers within the container. The top of the container would have a dished head to which a neck with a flange/pintle for handling would be welded (MacDougall 1987; DOE 1988e).

The conceptual disposal container for vitrified HLW is similar to that for spent fuel, but with slightly different dimensions of 26 inches in outside diameter by 10.5 feet long. This container is an overpack for the stainless steel waste canister, which is 24 inches in outside diameter by 9.8 feet long, and which also has a dished head with a welded neck and a flange/pintle for handling. Disposal containers for spent fuel and for HLW are still under development, and changes from these current concepts are being considered (MacDougall 1987; DOE 1989a).

If consolidation of the spent fuel rods is implemented, the compacted non-fuel-bearing components from the spent fuel assemblies are proposed to be placed in 55-gallon steel drums in stacks of five drums each that will be placed in metal cages. These metal cages would be overpacked into thin-walled stainless steel containers for disposal. These containers are proposed to be emplaced in a configuration similar to that for spent fuel and will be in boreholes commingled with those of spent fuel and HLW (MacDougall 1987; DOE 1989).

The current U.S. conceptual waste package system does not use a buffer/packing material around the waste container in the disposal hole. The unused part of the disposal boreholes above the shield plug are proposed to be filled with crushed tuff after filling of the borehole. Similarly, the emplacement drifts, tunnels, and shafts are proposed to be filled with crushed tuff after the retrievability period, when the repository is decommissioned. Concrete grouts and clays are proposed to be used to seal off specific features in the repository (MacDougall 1987).

10.4.6 Research and Development

Research and development (R&D) of deep geologic disposal of HLW was initiated as a result of a study by the independent National Academy of Sciences (NAS) in 1955. Its report identified rock salt formations as potential host rocks for a repository (Hess 1957). From then until the 1970s, numerous paper and experimental studies were carried out on concepts for deep geological disposal, with general emphasis on salt formations. In the 1970s, investigations were broadened to include other host rock formations, and involved other paper studies, laboratory investigations, surface-based site-selection activities, and disposal concept feasibility studies (Schneider et al. 1974).

The first underground research was carried out in a former salt mine near Lyons, Kansas, from 1965 to 1968. This effort included geomechanical and irradiation effects tests with electrical heaters and irradiated fuel elements (Bradshaw 1969).

The Nevada Test Site, which is adjacent to the Yucca Mountain site, was recognized early as a possible location for an underground repository, and an investigation program was initiated in 1977. Underground research was performed in both tuff and granite. Experiments are being conducted in a tunnel to obtain data on in situ physical and mechanical properties of tuffaceous rocks. Between 1978 and 1983, canisters of spent fuel were placed in a granite test facility at the Nevada Test Site 1400 feet below the surface to help evaluate granite as a potential host repository medium. An underground test facility in basalt at Hanford was utilized to carry out geotechnical tests for disposal in basalt from 1980 to 1988 (DOE 1986; Moak 1980; Patrick 1986).

The Yucca Mountain site selected for characterization will include an underground research facility. The exploratory shaft facility for site characterization will consist of two exploratory shafts, a tunnel that connects the shafts, and other tunnels and underground rooms for testing. Underground research during site characterization will be conducted in two phases. The first phase, construction testing, will involve numerous geotechnical tests accompanying the construction of the facility. The second phase, in situ testing, will concentrate on characterizing the rock mass, including numerous other geotechnical tests, permeability tests, geochemical and migration tests, seismic tests, thermal tests, etc. Results from these tests will be integrated with evaluation of the suitability of the site relative to performance requirements and repository operations (DOE 1988e).

Major development and evaluation activities have been carried out since the passage of the Nuclear Waste Policy Act of 1982. These have included R&D

on detailed characterization of spent fuels and HLW forms, wet and dry storage of spent fuels, consolidation of spent fuel rods, design concepts and siting for a MRS facility, desk-based and surface-based site selection and preliminary characterization in a variety of host rocks, detailed development of performance assessment methodology, development of a transportation system, and numerous systems studies to optimize the total federal waste management system (DOE 1988f).

In addition to the domestic R&D activities, the U.S. has participated in a number of underground research studies with other countries and with the OECD/NEA and the IAEA (DOE 1988f).

10.4.7 Approach to Proving the Safety of the Repository

The approach to assuring the safety of the repository is based on modeling its expected performance into the distant future. The EPA is using bounding analytical calculations to set its environmental standards. The DOE is developing the models it needs to comply with the regulations. The NRC will independently use DOE models and some of its own to assess compliance.

A variety of performance measures are to be addressed by this modeling in response to the regulatory requirements of the OOE and NRC. The EPA standard requires analyses that include a probabilistic analysis of cumulative releases across a boundary around the repository. The NRC regulations require deterministic calculations of groundwater travel time, waste package containment time, and radionuclide release rates from the waste package after the containment has failed (NRC 1983; EPA 1985).

Comprehensive development of performance assessment techniques has been carried out in the U.S. since the late 1970s. Preliminary, deterministic site-specific performance assessments were carried out for the Environmental Assessments of the nine candidate repository sites in 1986. Development of site-specific performance assessment methodologies for the Yucca Mountain site is now underway and will be continued through the site characterization phase. The general approach is to develop models for major parts of the repository system (for example, the waste package and near-field, the far-field, and the environment), and integrate the appropriate results from each to obtain total system performance. Performance assessment models will be used to provide the primary long-term repository safety evaluations in the EIS and Safety Analysis Reports planned to be completed in 2001 (DOE 1988e; DOE 1989c; Alexander 1989a).

Performance assessments have been used to indicate performance of the disposal system and its parts and to provide feedback to the R&D and site characterization activities. Preliminary performance assessments will be completed in 1990 on a reference set of problems to compare and evaluate the various existing models. Subsequent assessments on the waste disposal system will be completed annually and will be reviewed by U.S. peers. The preliminary performance assessments will provide valuable input to site characterization and evaluations (Alexander 1989a).

In general, the performance assessment methodologies under development in the United States are quite detailed and comprehensive. The methodologies generally attempt to provide a realistic simulation of repository performance. Both deterministic and stochastic models, and some bounding analyses for hypothetical disruptive events, are being developed (Alexander 1989a).

Emphasis is now being placed on verification and validation of the current models to assure that they accurately represent the U.S. repository system. The U.S. approach to validation uses extensive laboratory and field data and some natural analogue studies to support the credibility and defensibility of calculations to the degree necessary for the NRC to reach a finding of compliance with its regulations. The U.S. is placing modest emphasis on studying natural analogues to support performance assessment and validation of the performance assessment models. The U.S. is actively participating in multinational comparison and validation of performance assessment models. The objective in model validation is to reduce the uncertainties and to define the role of uncertainty and sensitivity analysis in U.S. performance assessments (DOE 1988e; Alexander 1989b).

10.4.8 Peer Review Activities

Potential repository host States or Indian tribes participate in the site selection and evaluation process by reviewing the federal activities and results using their own experts and by having onsite representatives that oversee DOE activities (U.S. Congress 1987).

An oversight review panel for the waste repository has been authorized by the Nuclear Waste Policy Amendments Act of 1987. The seven-member panel includes the chairman and one other to be appointed by the DOE, four to be appointed by the potential host State and units of local government, and one to be selected by DOE from nuclear electric utilities (U.S. Congress 1987).

In accordance with the legislation, an independent three-member MRS Commission was appointed by the President to review the need for an MRS facility and to recommend to Congress as to whether such a facility should be included in the national waste management system (U.S. Congress 1987). The Commission completed its work and submitted its recommendations to Congress in late 1989 (MRS Review Commission 1989).

The legislation provides for establishing a Nuclear Waste Technical Review Board to investigate and evaluate the technical and scientific validity of activities undertaken by the DOE related to repository site characterization, waste packaging, and transportation. The 11-member board reports to the President, and the members were selected in early 1989 from candidates identified by the independent NAS (U.S. Congress 1987; DOE 1989d).

The NRC appointed an Advisory Committee on Nuclear Waste in 1988. The three-member independent panel and its staff advise the Commission on regulatory issues for all types of radioactive wastes (<u>Nuclear News</u> 1988).

All published performance assessments, including those published annually, on the waste disposal system are reviewed by numerous technical peers (DOE 1988e).

The U.S. is a member of the IAEA and the OECD/NEA and participates actively in their waste management programs, including R&D study projects. The U.S. has participated in reviews of other country's waste management systems when requested through these agencies but has not requested these organizations to participate in or review the U.S. program.

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