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RETHINKING HIGH-LEVEL WASTE DISPOSAL: SEPARATE DISPOSAL OF HIGH-HEAT RADIONUCLIDES (^{90}Sr AND ^{137}Cs)

RADIOACTIVE WASTE MANAGEMENT AND DISPOSAL

KEYWORDS: *cesium, strontium, high-level waste, repository*

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An alternative approach for disposal of high-level waste (HLW) is proposed. HLW would be separated into two fractions: (a) the high-heat radionuclides (HHRs), e.g., ^{90}Sr and ^{137}Cs, and (b) the low-heat radionuclides (LHRs), which are all the remaining radionuclides. These two categories of waste would be disposed of separately in different sections of the repository or different facilities.

The LHRs in the HLW contain the long-lived radionuclides that control the repository performance requirements that in turn necessitate (a) expensive waste packages (WPs) and (b) limiting the repository temperatures to avoid repository performance degradation. To limit repository temperature, the amount of HLW per WP is limited and the WPs are spread over a large area. If the decay-heat–generating HHRs are removed from HLW, the repository design is not controlled by decay heat. The resultant LHR repository size (area, number of WPs, total tunnel length) may be reduced to <20% of the size of a conventional repository. With a waste partitioning and transmutation process that includes removal of the minor actinides (americium and curium) from the LHR wastes, significant further reductions in repository size are possible. The minor actinides are the next largest heat generators in LHR wastes.

Separate management of HHRs does require (a) separation of the HHRs from the HLW and (b) a separate HHR disposal facility. The HHRs are disposed of in a separate lower-cost facility made possible by the limited lifetimes ($T_{1/2} \sim 30$ yr) of the HHRs. There are potentially significant gains in economics and repository performance for separate management of HHRs and LHRs in some types of fuel cycles.

I. INTRODUCTION

There is a renewed interest in investigating alternative nuclear fuel cycles and waste management systems. There are several reasons for this interest. There have been major institutional difficulties in siting and building repositories. As a consequence, Europe, Japan, and the United States are investigating alternative waste management concepts such as waste partioning and transmutation (P-T) for the destruction of long-lived radionuclides.[1–3] Repository costs have increased because of changing repository design philosophies. In early designs, the geology was the primary barrier against release of radionuclides to the environment. Today, most repositories propose a multibarrier system that includes high-cost waste packages (WPs) (Ref. 4). Furthermore, there is the added requirement that there be very high assurances that the repository will perform as expected. Last, if nuclear power is to be used on a very large scale, such as to replace fossil fuels to minimize climatic change, there are strong economic, institutional, and environmental incentives to find better ways to manage radioactive wastes. While it is unclear whether there will be changes in fuel cycle policies by different countries, a starting point for discussions on alternative fuel cycle policies is an understanding of the options.

Historically, repositories were designed to accept whatever waste was generated with the exception that some requirements were imposed on the chemical characteristics of the waste form, i.e., to convert liquid high-level waste (HLW) into HLW glass. More recently, there has been renewed investigation of various P-T systems,

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where selected long-lived radionuclides are removed from the waste and destroyed in a reactor or accelerator. This reduces the long-term radiotoxicity of the waste. A third approach is proposed here. The waste characteristics that control the cost and performance of the repository are identified. The repository is then designed to separately dispose of several categories of waste, where the waste categories are chosen to maximize repository performance and minimize repository costs. The viability of this approach depends upon the gains in the repository versus the costs in added waste processing.

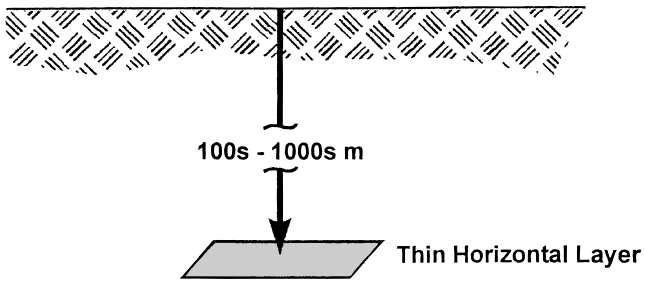

Fig. 1. Idealized HHR repository.

I.A. Alternative Design

The high cost and complexity of a repository are partly the consequence of disposing of HLW that contains two classes of radionuclides, each with different characteristics.

1. *High-heat radionuclides (HHRs)*: The HHRs (^{90}Sr and ^{137}Cs) generate almost all the heat produced by HLW. These radionuclides have relatively short half-lives (~30 yr) and do not need to be isolated from man for thousands to millions of years. HHRs decay to nonradioactive isotopes within a few hundred years.

2. *Low-heat radionuclides (LHRs)*: The LHRs include all the remaining radionuclides in the HLW. The LHRs include the radionuclides (^{99}Tc, ^{129}I, ^{237}Np, etc.) that remain hazardous for long periods of time and that necessitate long-term performance requirements on the repository.

The long-lived radionuclides in LHR wastes necessitate low repository temperatures. The U.S. Nuclear Waste Technical Review Board,[5] the Congressionally mandated review board for the Yucca Mountain (YM) repository project, recently summarized temperature constraints. High temperatures alter rock properties with reductions in the capability of the geology to retard radionuclide releases. It is extremely difficult to predict how rock properties change with exposure to high temperatures. The performance of WPs becomes uncertain. Waste forms degrade. If separate repositories were designed for HHRs and LHRs, respectively, they would greatly differ because of the different requirements to ensure radionuclide isolation from the environment.

I.A.1. HHR Repository

The controlling HHR-repository design requirement is to cool the waste. The optimum design to aid cooling is a repository in which the HHRs are spread out underground as a thin, horizontal layer (Fig. 1). This configuration allows decay heat to be conducted away from the HHRs to the surface of the earth. The package size is limited to a few centimetres in diameter because of the high heat-generation rates (see Sec. II.D). Lower-cost packages and a simplified repository performance assessment are possible because of the limited lifetime of these radionuclides.

I.A.2. LHR Repository

The controlling design requirement is to isolate the LHRs for very long times from the environment. The optimum design (assuming very low heat-generation rates) is a single, large sphere buried deep underground (Fig. 2). The primary radionuclide release mechanism is (a) dissolution of radionuclides in groundwater and (b) the transport of that groundwater to the open environment. The large spherical form minimizes radionuclide release by two mechanisms.

1. *Mass transfer*: Various barriers can be placed around the waste to slow the flow of water through the waste. With a smaller surface-to-volume ratio, fewer economic constraints are placed on the design of such barriers; consequently, higher performance barriers may be used.

2. *Solubility limits*: The release of radionuclides from a WP after its failure is proportional to the groundwater flow through the waste and the solubility limits of the radionuclides in groundwater. Let us consider two

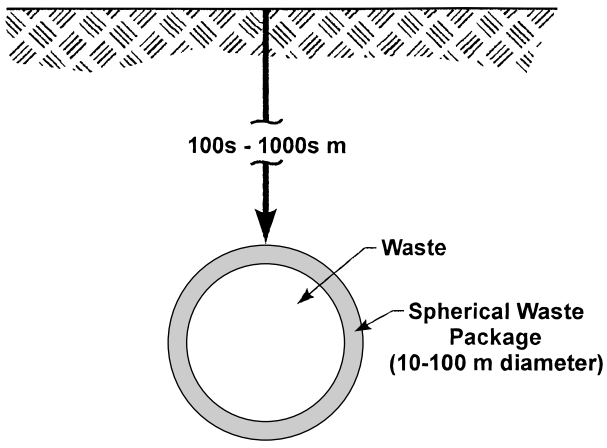

Fig. 2. Idealized LHR repository.

examples (Fig. 3), where radionuclide A will dissolve in groundwater up to its solubility limit. In the first example, a fixed quantity of radionuclide *A* is spread over 25 m^2 (WPs). In the second example, the same quantity of radionuclide *A* is spread over 1 m^2 (WP). Assuming uniform groundwater flow, the groundwater flow through this 1-m^2 area is one twenty-fifth of that flowing through the 25-m^2 area. Because the flow of water through the waste is reduced by a factor of 25, the release rate is reduced by a factor of 25. Larger WPs with the smaller cross-section areas per unit of waste lower the total repository radionuclide release rates.

While an ideal LHR repository would be a single, spherical WP, there is some LHR decay heat. This decay heat implies that a practical repository design would use a few, large cylindrical WPs.

I.B. Conflicting HLW Design Requirements

A repository for HLW or spent nuclear fuel (SNF) is designed to dispose of a wasteform that is a combination of HHRs and LHRs. The cost and technical difficulties are directly affected by the different requirements for HHR and LHR disposal:

1. *Performance*: To maximize long-term performance by minimizing release of LHR radionuclides from the WP, large WPs should be used. However, the presence of HHRs prohibits this. With a large WP, the HHRs generate sufficient decay heat that temperature limits are exceeded with subsequent degradation of the wasteform, WP, and geology.

2. *Repository economics*: Closely spaced, large WPs minimize costs. The HHRs limit WP size and force placement of WPs in long drifts to minimize repository temperature. For example, as a consequence of temperature constraints, it is estimated that the proposed YM repository will have ~10 000 SNF WPs distributed over ~100 km of disposal drifts at a cost of several tens-of-billions of dollars.

The conflicting design requirements imposed on HLW and SNF disposal by the existence of both LHRs and HHRs in the same waste form have been recognized in Sweden, Great Britain, France, and several other countries. These countries have selected a policy of storing SNF and HLW for many decades before disposal to allow HHRs to decay with reduced heat generation rates and thus (a) improve repository performance and (b) reduce costs. What is proposed here is to extend the concept to the logical end point—separation of wastes into HHR and LHR waste streams with separate disposal of each waste.

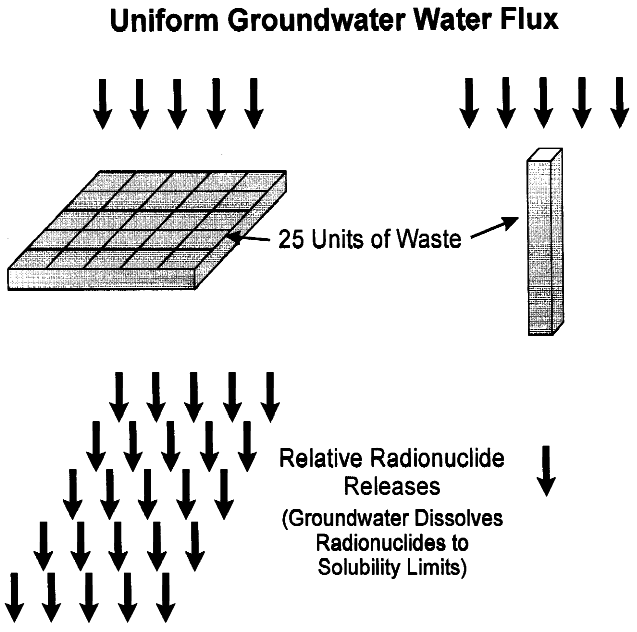

Fig. 3. The geometry of the waste (surface-to-volume ratio) strongly impacts radionuclide releases.

II. WASTE CHARACTERISTICS

II.A. Categories

II.A.1. HHR Waste

HHRs include cesium, strontium, and their decay products. The defining characteristics are (a) high heat-generation rates and (b) limited lifetimes. The characteristic of a limited lifetime is necessary to allow inexpensive disposal of HHRs; consequently, this waste stream cannot have significant quantities of long-lived radionuclides–except ^{135}Cs. It is not practicable to separate long-lived ^{135}Cs from other cesium isotopes. The issues associated with ^{135}Cs are discussed in Sec. VIII.B.

In the United States, there is considerable experience in the separation and storage in capsules of large quantities of HHR wastes. HLW from defense operations is stored in tanks. HHRs have been separated from this HLW on an industrial scale for several purposes:

1. *Minimize HLW storage costs*: HLW at Hanford, Washington, is stored in million-gallon underground tanks that do not have internal cooling coils to remove decay heat. Tank capacity was limited by the heat-generation rate of the HLW. Removal of the HHRs allowed better volume utilization of the tanks.

2. *Beneficial use of HHRs*: The HHRs were separated into two streams to produce ^{90}Sr and ^{137}Cs capsules for research. There was an interest in ^{90}Sr for heat sources and an interest in ^{137}Cs for irradiation sources.

The 6.67-cm-diam HHR ^{90}Sr and ^{137}Cs capsules are currently stored at Hanford, Washington.[6] The total volume of the capsules is only ~3.5 m^3, but the capsules contain 1.4×10^8 Ci of radioactivity with a heat generation rate of ~400 kW.

In the future, large quantities of HHRs will be separated from existing HLW (Ref. 7). It is currently planned to separate ^{137}Cs from HLW at the Savannah River Site (SRS) in Aiken, South Carolina, as part of the program to convert HLW salts and sludges into HLW glass. Thus, the HLW sludges are converted into HLW glass. The salt cake is primarily a mixture of soluble salts containing ^{137}Cs. To reduce costs, (a) the salt will be dissolved, (b) the cesium with ^{137}Cs and other hazardous radionuclides will be removed from the liquid, and (c) the liquid will be treated and disposed of as low-level waste (LLW). The cost of LLW disposal is much lower than that for HLW disposal. The separated cesium may be sent to the vitrification plant with the sludges and converted into HLW glass. The option exists to separately dispose of this HHR waste stream.

II.A.2. LHR Waste

LHRs include all other radionuclides in HLW. Small amounts of cesium and strontium are allowed in this waste stream.

II.A.3. Very Low Heat Radionuclide Waste

In some fuel cycles, it is possible to create very low heat radionuclide (VLHR) wastes from HLW. Strontium and cesium are the dominate heat generators. After they have been removed, the heat generation rate is controlled by the minor actinides (MAs) americium and curium. The added removal of these radionuclides from HLW creates a VLHR waste stream. There are additional options for management of such wastes (see Sec. III.C).

II.B. Waste Sources

The characteristics of this alternative HHR–LHR waste management system depend partly upon the fuel cycle. Different fuel cycles define the potential range of characteristics:

1. *Conventional reprocessing of SNF*: France, Great Britain, India, Russia, and Japan reprocess commercial SNF to recycle and use plutonium and uranium in their power reactors. These processes produce acidic HLW solutions that are converted to HLW glass. The HLW can be separated into HHR and LHR wastes. The LHR wastes would contain (a) the fission products (excluding strontium and cesium) and (b) the MAs that are not recovered for recycle as fuel.

2. *Waste partitioning and transmutation*: Europe,[1] the United States,[2] and Japan[1] are investigating the selective removal of certain long-lived actinides and fission products from waste streams and the subsequent destruction of these radionuclides using accelerators or reactors. This reduces the long-term waste hazard; however, in all of these systems, some long-lived radionuclides remain that require repository disposal. Cesium and strontium are the dominant heat sources in HLW; however, the next largest heat-generation sources[3] are the MAs—americium and curium. In a P-T system, these are removed and destroyed. Consequently, in a P-T system, the HLW could be separated into an HHR waste and a VLHR waste.

3. *Defense*: In the United States, large quantities of HLW have been generated from defense operations. Defense wastes were generated primarily as a byproduct of producing weapons-grade plutonium (WGP), which is ^{239}Pu with low concentrations of higher plutonium isotopes. To make WGP, uranium is irradiated to a low burnup in a production reactor. These reactor conditions produce targets that contain low concentrations of MAs. As a consequence, much of this HLW can be partitioned into an HHR waste and a VLHR waste.

II.C. Characteristics

To provide a basis for analysis, the wastes from processing 40 000 MWd/t pressurized water reactor SNF were used to define representative HHR, LHR, and VLHR wastes. Six different streams were identified.

1. *Plutonium and uranium*: This stream includes plutonium, uranium, and their decay products that build up after separation from the SNF.

2. *Minor actinides*: The MAs include all actinides (except plutonium and uranium) and their decay products that build up after separation from the SNF.

3. *Volatiles*: This stream includes all volatiles and their subsequent decay products that would be expected to be released during processing operations and separately managed. These include inert gases (He, Ne, Ar, Kr, Xe, and Rn), halogens (F, Cl, Br, I), hydrogen, nitrogen, and carbon.

4. *Structure*: This stream includes the metallic components of a fuel assembly including the clad.

5. *HHRs*: The HHRs are defined as all cesium and strontium isotopes. Initially, these are the only elements in the HHRs. Over time, the decay products of these radioactive isotopes build up in this waste stream.

6. *VLHRs*: The VLHRs are the fission products minus the volatile fission products, cesium, and strontium. HLW from P-T can be separated into a VLHR waste stream and an HHR waste stream. Combining VLHRs with the MAs creates an LHR waste stream that is typical of the LHR waste stream that would be obtained from commercial processing of SNF.

Table I characterizes these six streams and how they vary over time. The SNF is assumed to be processed 5 yr after discharge from the reactor. For the purposes herein, idealized processes were assumed with perfect separations. The times shown in Table I are years out from when the reactor discharged the SNF. Five properties are shown per tonne of initial heavy metal:

1. *Mass*: The HHR mass is very small. If 1 t of 40 000 MWd-light water reactor SNF is processed, there are only 4.1 kg of cesium, strontium, and their respective decay products (barium and yttrium). In contrast, the VLHR mass is 30.3 kg.

2. *Decay heat*: At 10 yr after the SNF was discharged from the reactor, the HHRs, the VLHRs, and the MAs are, respectively, 71, 4.4, and 7.8% of the decay heat from SNF. The LHRs from conventional reprocessing include the MAs and VLHRs, and thus this LHR waste decay heat is 12.2% of the SNF. The HHRs are the primary source of decay heat for several hundred years after which time the MAs are the dominant decay-heat source.

3. *Radiation*: At 10 yr after the SNF was discharged from the reactor, the HHRs, the VLHRs, and the MAs are, respectively, 89, 6.7, and 0.5% of the gamma rays from SNF. Most of the high-energy gamma rays are from the HHRs. This implies that less shielding is required for the LHRs and VLHRs than for the HHR wastes, HLW, or SNF.

4. *Ingestion hazard*: The ingestion hazard measures the hazard from drinking radioactively contaminated water. At 10 yr after the SNF was discharged from the reactor, the HHRs, the VLHRs, and the MAs are, respectively, 99, 0.17, and 0.24% of the ingestion hazard from SNF. The HHR ^{90}Sr is the dominant hazard. Transmutation studies[8] indicate that it is not feasible to transmute ^{90}Sr.

5. *Inhalation hazard*: The inhalation hazard measures the hazard from inhalation of radioactively contaminated air. At 10 yr after the SNF was discharged from the reactor, the HHRs, the VLHRs, and the MAs are, respectively, 2, 0.05, and 11% of the inhalation hazard from SNF. The plutonium is the dominant hazard.

TABLE I

Streams from Processing 1 Tonne Initial Heavy Metal of 40 000 MWd/t of Pressurized Water Reactor SNF

				LHRs			
	SNF	U/Pu	HHRs	VLHRs	MAs	Volatiles	Structure
Mass[a] (g)	1.427×10^{6}	9.576×10^{5}	4.132×10^{3}	3.030×10^{4}	1.192×10^{3}	7.206×10^{3}	2.918×10^{5}
Decay heat[b] (W)							
At 10 yr	1.443×10^{3}	1.851×10^{2}	1.024×10^{3}	6.359×10^{1}	1.132×10^{2}	8.900×10^{0}	4.799×10^{1}
At 20 yr	1.096×10^{3}	2.113×10^{2}	7.554×10^{2}	2.239×10^{1}	8.989×10^{1}	4.663×10^{0}	1.284×10^{1}
At 50 yr	6.578×10^{2}	2.276×10^{2}	3.726×10^{2}	1.972×10^{0}	5.457×10^{1}	6.708×10^{-1}	3.051×10^{-1}
At 100 yr	3.555×10^{2}	2.014×10^{2}	1.154×10^{2}	8.611×10^{-2}	3.858×10^{1}	2.668×10^{-2}	4.551×10^{-2}
At 1000 yr	6.308×10^{1}	5.395×10^{1}	1.818×10^{-4}	2.353×10^{-2}	9.097×10^{0}	1.739×10^{-4}	1.531×10^{-2}
Radiation[b] (γ/s)							
At 10 yr	9.474×10^{15}	4.601×10^{13}	8.444×10^{15}	6.392×10^{14}	4.356×10^{13}	5.517×10^{13}	2.460×10^{14}
At 20 yr	6.630×10^{15}	6.929×10^{13}	6.214×10^{15}	2.133×10^{14}	3.924×10^{13}	2.890×10^{13}	6.444×10^{13}
At 50 yr	3.217×10^{15}	9.345×10^{13}	3.068×10^{15}	1.750×10^{13}	3.212×10^{13}	4.156×10^{2}	1.433×10^{12}
At 100 yr	1.071×10^{15}	9.090×10^{13}	9.520×10^{14}	6.138×10^{11}	2.754×10^{13}	1.658×10^{11}	1.784×10^{11}
At 1000 yr	3.008×10^{13}	2.189×10^{13}	7.706×10^{8}	2.337×10^{11}	7.844×10^{12}	1.905×10^{9}	1.095×10^{11}
Ingestion hazard[b] (m^3)							
At 10 yr	2.488×10^{11}	1.597×10^{9}	2.460×10^{11}	4.240×10^{8}	6.073×10^{8}	7.965×10^{5}	1.245×10^{8}
At 20 yr	1.957×10^{11}	1.647×10^{9}	1.934×10^{11}	1.360×10^{8}	5.097×10^{8}	7.209×10^{5}	4.603×10^{7}
At 50 yr	9.670×10^{10}	1.605×10^{9}	9.470×10^{10}	1.417×10^{7}	3.594×10^{8}	6.395×10^{5}	1.524×10^{7}
At 100 yr	3.052×10^{10}	1.397×10^{9}	2.883×10^{10}	1.643×10^{6}	2.827×10^{8}	6.222×10^{5}	1.023×10^{7}
At 1000 yr	4.461×10^{8}	3.761×10^{8}	5.460×10^{3}	6.379×10^{5}	6.829×10^{7}	6.210×10^{5}	5.200×10^{5}
Inhalation hazard[b] (m^3)							
At 10 yr	1.197×10^{17}	1.039×10^{17}	2.609×10^{15}	6.417×10^{13}	1.313×10^{16}	2.419×10^{10}	1.074×10^{13}
At 20 yr	1.070×10^{17}	9.397×10^{16}	2.045×10^{15}	2.279×10^{13}	1.091×10^{16}	1.368×10^{10}	3.056×10^{12}
At 50 yr	8.420×10^{16}	7.569×10^{16}	1.002×10^{15}	2.373×10^{12}	7.512×10^{15}	3.638×10^{9}	2.858×10^{11}
At 100 yr	6.556×10^{16}	5.941×10^{16}	3.058×10^{14}	1.913×10^{11}	5.842×10^{15}	1.955×10^{9}	1.643×10^{11}
At 1000 yr	2.022×10^{16}	1.876×10^{16}	1.817×10^{8}	3.074×10^{10}	1.454×10^{15}	1.879×10^{9}	1.601×10^{10}

[a]SNF includes 1.334×10^{5} g oxygen. Component streams exclude oxygen.
[b]Time measured from reactor discharge. Separations assumed to occur 5 yr after reactor discharge.

II.D. Wasteforms

II.D.1. LHRs

The LHRs and VLHRs are chemically similar to HLW. The same wasteforms as used for HLW are acceptable. Borosilicate glass is currently the preferred HLW form and thus would likely be the preferred LHR and VLHR wasteform.

II.D.2. HHRs

For the purposes herein, the HHRs are assumed to be packaged in capsules with an outside diameter of 6.67 cm—the same diameter as that of existing Hanford ^{137}Cs and ^{90}Sr capsules.[9] The same cooling requirements apply to future HHRs; thus, future HHR capsules will have similar dimensions. The chemical form of future HHRs may be different (see the following). The existing Hanford ^{90}Sr and ^{137}Cs capsules have somewhat different characteristics.

1. *Strontium capsules*: Each capsule was designed to receive 1.5×10^5 Ci of strontium in the form of SrF_2, which was added as a powder to an inner capsule with an inside diameter of 5 cm and which was made of Hastelloy C-276. The powder was compacted to 68% of its theoretical density, and the inner capsule was welded shut. The inner capsule was then overpacked with an outer capsule made of Type 316L stainless steel. The external package has a length of 51.05 cm. In air, a fully loaded capsule with a heat output of ~900 W has a centerline temperature of 860°C and a surface temperature of 430°C. Dosimeter measurements of capsules with 7×10^4 Ci had radiation levels of 4.9×10^3 rad/h at 20 cm.

2. *Cesium capsules*: Each capsule was designed to receive 7.0×10^4 Ci of cesium in the form of CsCl, which was added as a liquid to an inner capsule with an inside diameter of 5.24 cm, and which was made of Type 316L stainless steel. The typical loading was 65% of the inner capsule volume. The inner capsule was then overpacked with an outer capsule made of Type 316L stainless steel. The external package has a length of 52.77 cm. In air, a fully loaded capsule with a heat output of ~300 W has a centerline temperature of 450°C and a surface temperature of 200°C. Dosimeter measurements of capsules with 7×10^4 Ci had radiation levels of 3.0×10^5 rad/h at 20 cm.

There are two important HHR characteristics:

1. *Linear heat generation rates*: The heat-generation rates of capsules, measured in kilowatts per metre, are about the same as those of typical heat-limited WPs.

2. *Diameter*: The diameter of any HHR capsule is small. The capsules contain relatively pure SrF_2 or CsCl. Modern wasteforms typically contain ~25 wt% wastes. If one of these modern wasteforms were used as a strontium or cesium wasteform, the diameter would increase to only ~12 cm.

III. MANAGEMENT OF LHRs

The LHR repository requirements are identical to those of a conventional HLW or SNF repository. The design is different because the removal of HHRs allows the use of larger WPs and closer spacing of WPs without exceeding temperature design limits (Fig. 4). For the purposes herein, the simplifying assumptions are made that (a) wastes are disposed of 10 yr after the SNF is discharged from the reactor and (b) the thermal characteristics of a wasteform can be characterized by its decay heat at 10 yr after the SNF is discharged from the reactor. The design of the proposed YM repository in the United States is used as a basis of comparison between current repository designs and the proposed alternative repository design. Similar results would be expected when comparing any other conventional repository design with the alternative design described herein.

III.A. Repository Size

The repository size is determined by the temperature limits for the wasteform, WP, and geology. The temperature is controlled by limiting the waste decay-heat load per unit area. For LHRs (VLHRs + MAs) from HLW, the heat generation rate at 10 yr after the SNF was discharged from the reactor is 12.2% of that of SNF. Using simplifying assumptions, the LHR repository area and the required length of tunnels will be 12.2% of a comparable SNF or HLW repository. For VLHRs, the heat generation rate after 10 yr is 4.4% of that of SNF. Using the same assumptions, the VLHR repository area will be 4.4% of the comparable SNF or HLW repository.

III.B. WP Limits and Numbers

There are two limits on the quantity of waste in a WP: decay heat and volume. Removal of HHRs increases the allowable quantity of LHR waste per WP. Using idealized assumptions (perfect separations, full-WP volume utilization, repository waste placement 10 yr after SNF reactor discharge, WP heat limit exactly matches 10-yr-old SNF), the capacity of YM WPs for LHRs was determined. The largest proposed YM SNF WP contains ~10 t of commercial SNF. The WP loading is limited by the radioactive decay heat from the HHRs. With removal of the HHRs, larger quantities of LHRs can be placed in a WP without exceeding the WP thermal limits:

1. *LHR wastes*: The decay heat is 12.2% of SNF; thus, a WP that accepts 10 t of SNF could accept the LHRs from 82 t of SNF before exceeding WP thermal limits.

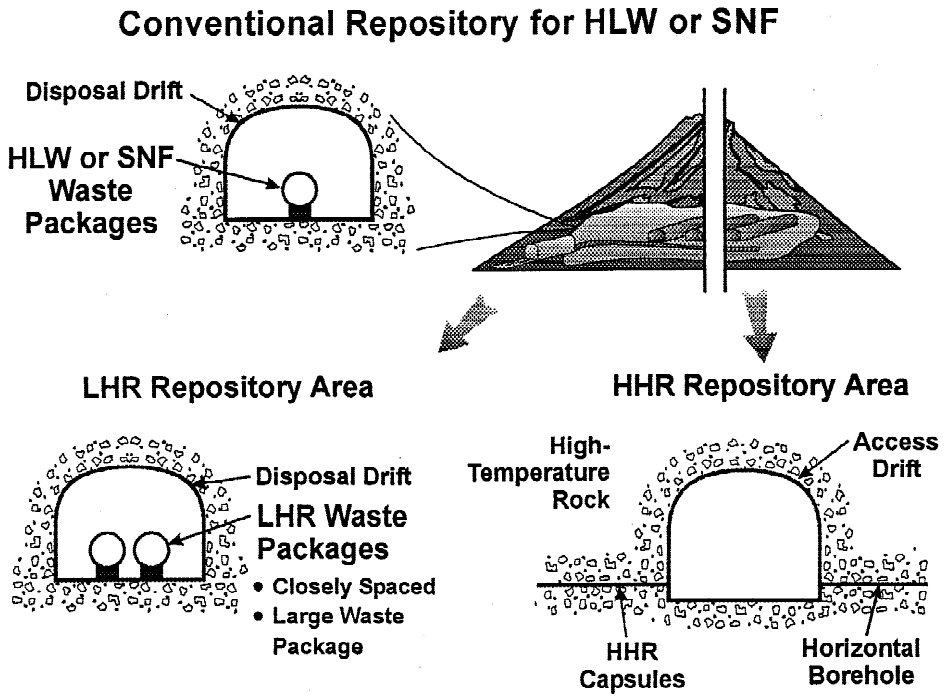

Fig. 4. Proposed YM repository with an LHR and HHR repository.

2. *VLHR wastes*: The decay heat is 4.4% of SNF; thus, a WP could accept LHRs from ~227 t of SNF before exceeding WP thermal limits.

There are volume limits. The largest YM WPs have internal volumes of between 9 and 10 m^3—similar to large SNF rail transport casks[10] and accept ~10 t of SNF. The historical assumption is that SNF processing plants[11] produce about 0.16 m^3 HLW glass/t of SNF. Lower volumes [0.083 m^3/t (Ref. 12) and 0.10 m^3/t (Ref. 13)] have been reported for newer facilities. Assuming identical waste volumes for either HLW or LHR wastes (0.16 m^3/t of SNF), a YM WP with a cavity of 10 m^3 could accept the LHR waste from 62 t of SNF.

Considering both the heat-generation and volume limits of WPs (Table II), the following conclusions are drawn:

1. *LHR wastes*: The WP capacity is controlled by volume limits. LHRs from 62 t of SNF can be put into a single WP. The WP thermal limit would not be exceeded until the LHRs from 82 t of SNF were placed in the package. The number of WPs required compared to an SNF repository is reduced by a factor of 6. This reduces cost and improves performance.

2. *VLHR wastes*: The WP capacity is controlled by volume limits. VLHR wastes from 62 t of SNF can be put into a single WP. The number of WPs required compared to an SNF repository is reduced by a factor of 6. This is far below the thermal limit of LHRs from 227 t of SNF.

Larger WPs with internal volumes up to ~30 m^3 may be feasible for repositories with underground truck or rail access. Both the proposed U.S. and Swedish repositories will have such access. This upper limit is based on transportation constraints between the WP fabrication shop and the repository. In the United States, most railroads limit cargo diameters to 3.05 m (128 in.). Assuming

TABLE II

Thermal and Volume Limits of WPs for LHR and VLHR Wastes*

Time from Reactor Discharge (Yr)	WP Limit Thermal or Volume (Type)	Waste Package Size			
		10 m^3		30 m^3	
		LHR (MTIHM)	VLHR (MTIHM)	LHR (MTIHM)	VLHR (MTIHM)
10	Thermal	82	227	82	227
	Volume	62	62	187	187
50	Thermal	255	732	255	732
	Volume	62	62	187	187

*Example: Consider the case of a YM WP that is accepting LHR wastes from SNF that was discharged from the reactor 10 yr ago. The LHR wastes from 82 tonnes of initial heavy metal (MTIHM) of SNF could be placed in the WP before the thermal limit of the WP was exceeded. The LHR wastes from 62 MTIHM of SNF could be placed in the WP before the volume limit (10 m^3) of the WP was exceeded. In this case, the WP capacity is limited by volume, not decay heat.

(a) an outside WP diameter of 3 m, (b) 0.2 m for the wall thickness and lifting lugs, and (c) a WP external height-to-diameter ratio is two, the WP internal volume would be ~30 m^3. This is three times the volume of the WP discussed earlier. Assuming the same maximum decay heat limit per package but larger volume WPs, the following conclusions are drawn:

1. *LHR wastes*: The WP capacity is controlled by decay heat with larger-volume WPs accepting LHR wastes from 82 t of SNF.

2. *VLHR wastes*: The WP capacity is controlled by the new volume limits and can accept VLHR wastes from 187 t of SNF. This is significantly below the WP thermal limit for VLHR wastes from 227 t of SNF.

The foregoing analysis assumes waste disposal 10 yr after SNF discharge from the reactor. Both the LHR and VLHR wastes contain several short-lived radionuclides that generate significant amounts of heat. If one is willing to store the wastes for 50 yr, the decay heat generation rates drop dramatically. In these cases, WP volume limits the quantity of waste placed in a YM or jumbo WP.

In addition to WP thermal limits, the repository has a temperature limit that is typically expressed in decay heat per unit area. If WPs are volume limited, not decay-heat limited, the WPs can be closely packed in disposal tunnels to minimize tunnel construction.

III.C. Construct-in-Place WPs

For aged VLHR wastes, the decay heat generation rates are so low that decay heat does not control the design of any WP that could practically be transported into a repository. In such cases, an alternative WP design can be used—build the WP in the repository and haul the wastes to the WP. Furthermore, this approach can also be used for secondary SNF processing wastes that contain low concentrations of radionuclides (and low heat-generation rates) but still require geological disposal. For modern large-scale SNF processing facilities, the reported volumes [0.4 m^3/t (Ref. 13) and 1.14 m^3/t (Ref. 12)] for all other wastes requiring geological disposal are small. Wastes in this category include items such as SNF hulls and hardware. There are economic incentives to build a very small number of very large WPs for all low-heat-generating wastes requiring geological disposal.

An example of a WP design that may be suitable for this application (with modifications) is the intermediate-level-waste high-activity silo in the Swedish Final Repository for Radioactive Operational Waste (SFR). The SFR silo[14] was excavated in granite at a depth of more than 50 m under the Baltic Sea (Fig. 5). Access is by tunnel. The silo is 50 m high and 25 m in diameter. A thick bentonite clay barrier surrounds the silo and fills the space between the rock cavern and the silo. The clay barrier is (a) a barrier to water migration and (b) a mechanism to retard radionuclide migration. The wastes are placed in the silo and cemented in place using a special cement grout.

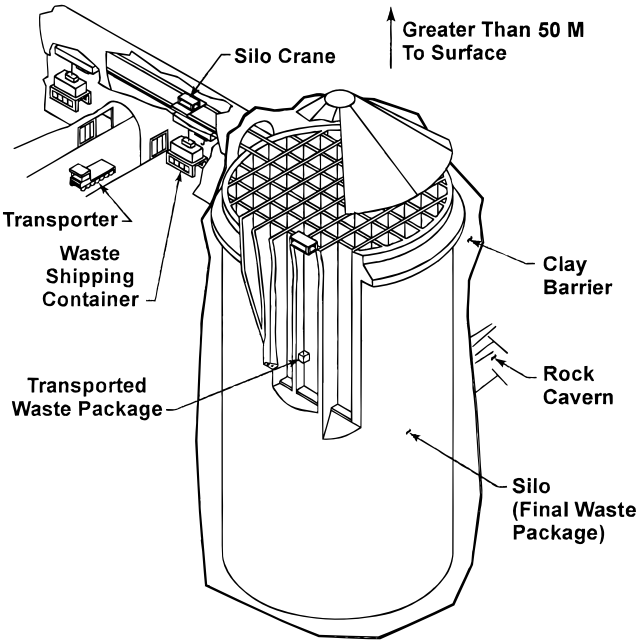

Fig. 5. Swedish SFR silo for intermediate-level radioactive wastes.

For repository applications, the silo depth of burial would be greater, there are additional constraints on silo materials of construction, and there are constraints on wasteform. This type of structure begins to approach the theoretical ideal LHR repository—a sphere. It is a potential low-cost, high-performance option.

IV. MANAGEMENT OF HHRs

IV.A. Design

The HHR capsules may be disposed of in a separate section of the proposed YM repository. One design option is shown in Fig 4. (The Appendix describes several other potentially lower-cost options, but these options would require different geological environments.) A tunnel is mined through the middle of the HHR disposal zone. Horizontal boreholes (10 to 15 cm diam) many hundreds of metres long are drilled into the rock from this tunnel. The horizontal boreholes are then filled with small-diameter HHR capsules. This design minimizes excessively high temperatures in the rock and significant radiation fields in the working spaces, and avoids construction of large disposal drifts. The repository areal heat load is determined by the spacing between boreholes. The option also exists to use vertical boreholes.

It is a low-cost option because large disposal drifts (5.5 m diam) for SNF WPs are replaced by small boreholes (~15 cm-diam) for HHR capsules. The linear heat generation rate of a capsule is essentially identical to a WP; thus, the linear length of boreholes equals the linear length of disposal drifts per unit of decay heat.

The HHR section of the repository would be designed as an "extended-dry" repository in unsaturated rock. The proposed YM repository is in unsaturated rock above the water table in a desert region with low rainfall. In an extended-dry repository, the boreholes with capsules are placed closely together to raise the local rock temperature above the boiling point of water for thousands of years. The expected failure mode of a repository is capsule failure, groundwater dissolution of radionuclides or formation of transportable colloids, transport of radionuclides to the open environment by groundwater, and inhalation or ingestion of the radionuclides by man. If the rock temperature is above the boiling point of water, there can be no groundwater flow and, therefore, no transport of radionuclides to the open environment. The HHRs decay before the repository cools below the boiling point of water.

IV.B. Performance Assessment

The YM project has examined extended-dry repository designs[15,16] for SNF disposal because such designs have much lower costs than more conventional repository designs. In extended-dry repository designs >150 t of SNF can be emplaced per acre, and the heat loads can exceed 0.027 kW/m^2 (110 kW/acre). Analysis indicates that the decay heat will ultimately heat above the boiling point of water a zone extending from 200+ m above the WPs to 150+ m below the WPs. The water is boiled out of the rock.

The center of the repository remains above the boiling point of water for ~10 000 yr, while the edge of the repository remains above the boiling point of water for ~5000 yr. Resumption of groundwater flow is delayed because large quantities of water are (a) required to cool the large masses of hot rock below the boiling point of water and (b) adsorbed by the rock as it cools.

HHR extended-dry repository design (decay heat per unit area) and performance would be similar. No HHR repository analysis has been done; however, since most of the decay heat in SNF is from the HHRs, similar results are expected. There are some differences. The maximum SNF temperature in the proposed YM repository is limited to 350°C to minimize SNF clad degradation. HHR capsules can operate at higher temperatures (see earlier). The SNF has longer-lived radionuclides that produce some heat for an extended period of time and thus slow repository cooling. However, since HHRs decay in ~300 yr, the HHRs repository performance is excellent because the HHRs decay before water reaches the capsules.

Because there are significant uncertainties about the performance of extended-dry repository designs for SNF, the current YM repository design temperatures are significantly below those of an extended-dry repository with only limited quantities of rock above the boiling point of water. The U.S. Nuclear Waste Technical Review Board,[5] the Congressionally mandated review panel for the YM project, recently recommended that repository design temperatures be lowered further and that "serious consideration be given to alternatives that keep waste package surface temperatures below the boiling point of water." The uncertainties include

1. *Radionuclide retention*: Exposure of the rock to high temperatures can reduce the long-term capability of the rock to retard the migration of specific radionuclides.

2. *Water flow*: The thermal pulse causes changes in the rock properties such that water-flow behavior is unpredictable.

3. *Climate*: The proposed YM repository is above the groundwater in unsaturated rock in a dry climate. The rate of repository cooling is dependent upon estimated rainfall.

The performance uncertainties associated with extended-dry repository designs exist only if the radionuclides exist after the repository cooldown. Unlike SNF or HLW, the HHRs decay away before the repository cools below the boiling point of water.

IV.C. Design Philosophy

Different repositories have different approaches to isolate the wastes from man. Each approach implies different wasteform requirements. Several examples are described herein to contrast the differences between an HHR extended-dry repository and other repository design approaches:

1. *YM repository*: For waste isolation, this repository depends upon the WP and wasteform to limit radionuclide releases. The WP is made of a high-nickel alloy, and the wasteforms must be leach resistant to groundwater.

2. *Waste Isolation Pilot Plant (WIPP)*: WIPP is an operating geological repository for defense transuranic wastes near Carlsbad, New Mexico. The geology (salt) is used to isolate the radionuclides from the environment. Consequently, the WIPP WP is a conventional 55-gal drum, and there are no wasteform performance requirements.

3. *HHR extended-dry repository*: The concept uses heat to create a steam bubble and isolate the radionuclides from the environment. This is dependent upon (a) decay heat and (b) limited heat transfer rates that ensure that the temperature at the HHR capsule surface remains above the boiling point of water until the HHRs decay away.

The only wasteform requirement for the system to work is that the HHRs not be in a volatile form that would allow gas transport.

If desired or required by regulation, additional barriers to HHR release can be added to an HHR extended-dry repository: a WP or use of an insoluble HHR wasteform. A practical WP would be a modified borehole liner. For operational purposes, the disposal boreholes will be lined with a metal drill pipe to prevent hole collapse during operations. The drill pipe can be designed to also serve as a waste package—a secondary barrier to radionuclide releases.

There are better HHR wasteforms than the salts that were used in the Hanford HHR capsules. It is likely that better HHR wasteforms will be used. The original Hanford HHR capsules were produced to meet two objectives: (a) reduce the costs of HLW management in tanks and (b) provide gamma and heat sources. To meet the second requirement, pure cesium and strontium salts were chosen to maximize the concentration of HHRs and the effectiveness of the capsules as radiation sources or heat sources. In the context of waste management, there are two disadvantages of these soluble salts: (a) they are more expensive to produce than many other candidate HHR wasteforms and (b) the high solubility is a disadvantage in transport, surface storage, and repository disposal. There are many alternative HHR waste forms.

An example of a potential advanced HHR wasteform is thermally converted silicotitanate.[17] Crystalline silicotitanate ion exchangers are highly selective for separating cesium from aqueous, sodium-rich waste streams. They are currently used to separate cesium from relatively low-activity waste streams and are being considered for separation of cesium from certain HLW streams. Recent research indicates that heating this ion-exchange material to 900°C for several hours converts it into a wasteform superior to borosilicate glass. The combined capability to separate cesium from wastes and the simple process to convert the inorganic ion exchange into a high-quality HHR wasteform offers potential economic and safety advantages.

In a similar way, strontium can be absorbed onto titanium oxides and converted to $SrTiO_3$. This is a highly insoluble form and has been used in ^{90}Sr radioisotopic thermoelectric generators.

IV.D. Comparative Size

The area of a YM, HHR, extended-dry repository would be ~28% of that required for a conventional SNF repository. The proposed areal decay-heat generation rate for the current YM repository design is ~40% of that of an extended-dry SNF repository.[16] The HHR decay heat is 71% of that of SNF. These two factors combined drastically reduce the HHR repository area.

A conventional and extended-dry repository for SNF have the same length of disposal tunnels per unit of decay heat. The difference is that in an extended-dry repository, the tunnels are spaced much closer together. This reduces the areal size of the repository and also reduces the length of access and ventilation tunnels. In an HHR extended-dry repository, the 5.5-m disposal drifts are replaced with 15-cm horizontal boreholes. The total length of boreholes is ~71% of that of SNF drifts because the HHR decay heat is 71% of that of SNF.

The linear heat generation rates of HHR capsules and SNF WPs are similar. The linear metres of containers (adjusted for decay heat) are the same.

V. SOCIETAL LONG-TERM RISK FACTORS

The potential for large-scale release of HHRs is much lower in a geological repository than in surface storage facilities in the event of war or a major natural catastrophe. Consequently, rapid geological disposal of wastes is a desirable social objective. The HHRs are the dominant hazard associated with nuclear activities for the first several hundred years (Sec. II.C). Separate management of HHRs allows rapid geological disposal of HHRs. Without HHR removal, rapid geological disposal of HLW is expensive and difficult because large numbers of WPs and large spacing between WPs is required to control temperatures in a conventional repository.

VI. SEPARATIONS AND SOLIDIFICATION

The preferred HHR separation and solidification processes depend upon the specific fuel cycle. Consequently, no detailed discussions of separation technologies are included herein. In some cases, HHR separations and solidification may be expensive. In other cases, the HHR separations have low costs because (a) a separated HHR is an inadvertent byproduct of some other separation process for some other radionuclide or (b) only a simple separation is required. The same is true for solidification processes.

VI.A. Separation Processes

There are many HHR separation processes. The preferred process will depend upon the specific fuel cycle. The operations[18] at Hanford have separated and encapsulated both cesium and strontium from HLW on an industrial scale. In the four decades since the Hanford cesium and strontium separations, major advances in separation technologies have occurred. Recent reviews and reports.[7,19–24] have described multiple separation processes for cesium and strontium from acid, neutral, and basic aqueous streams. Other options have identified HHR separation systems for a variety of nonaqueous separation systems.[2]

These technologies are being developed for several purposes. Liquid waste streams require cesium or strontium removal to reduce costs by converting an HLW stream to an LLW stream or an LLW stream to a nonradioactive waste stream. There is a limited demand for recovery of cesium or strontium for research and other purposes. Last, cesium and strontium removal is part of several proposed P-T flowsheets[1,21,25] to simplify other separation and processing operations.

There is renewed interest in new processes to recover uranium, plutonium, and other elements from SNF. This reflects interest in (a) more-economic recovery of uranium and plutonium from SNF, (b) P-T, and (c) recovery of other elements such as noble metals. In many of these new processing systems, low-cost separation of HHRs is an unintended consequence of other separations.

In Japan, new radiation-resistant ion-exchange processes are being investigated to recover plutonium, uranium, MAs, noble metals, and other elements from SNF.[26,27] The SNF is dissolved in nitric acid, and the feed solution is sent to an anion-exchange column where various elements separate out as bands on the ion-exchange column and sequentially leave the column. In this type of system, a relatively concentrated HHR stream can be removed by diverting the HHR-containing streams as they exit the ion-exchange column to separate receiving tanks.

In the United States, investigations are underway on the separation of actinides and long-lived radionuclides from SNF with their subsequent destruction using an accelerator.[2] The separation processes include a nonaqueous separation step where one of the waste products is a cesium-rich stream.

VI.B. LHR Wasteforms and Production Methods

Given the similar chemical characteristics of LHRs and HLW, the same wasteform would be used—borosilicate glass. However, the production of LHR borosilicate glass may be significantly easier and less expensive than the production of HLW glass:

1. *Decay heat*: HLW glass centerline temperatures are limited in HLW canisters and WPs to avoid long-term degradation of the glass properties. This limits the size of HLW glass logs. With the removal of the HHRs, lower-cost, larger glass logs become viable.

2. *Cesium*: Cesium is the primary, volatile radionuclide under the conditions found in an HLW glass melter and controls both the design and risks. To minimize volatilization of the cesium from the molten glass in an HLW melter, special glass compositions with low softening points and unique melter designs are required. Scrubbers in the off-gas system recover escaping cesium and return it to the melter. If cesium is removed from the feed stream (LHR wastes), more conventional, higher-throughput, industrial vitrification systems can be used. For a given size of system, this may double the throughput.

VII. ECONOMICS

Three costs determine the viability of separate management of HHRs: repository costs, separation costs, and development costs. If only small quantities of HHRs exist, the development costs would make this waste management option not viable. If there is a long-term worldwide use of nuclear power, this cost is not significant.

VII.A. Physical Comparison of Repository Concepts

The primary economic benefit of separate management of HHRs and LHRs is the reduction in repository costs. The relative reduction in the repository size is shown in Table III for key design parameters using the assumptions and system described earlier. The HHRs are assumed to be disposed of 10 yr after SNF discharge from the reactor. Two possible LHR repositories are shown. In the first case, the waste is disposed of 10 yr after SNF reactor discharge. In the second case, the waste is disposed of 50 yr after SNF reactor discharge. In each case, the relative sizes of the HHR–LHR repositories are compared to a conventional SNF repository. The physical cost of a combined HHR–LHR repository would be expected to be significantly less than half of a conventional repository:

1. *Area*: Separate management of HHRs and LHRs drastically reduces the area and thus cost of the repository. In addition, there are important indirect benefits. The small LHR repository size allows more local flexibility in siting for disposal of the long-lived radionuclides. The best local rock can be used for these wastes.

2. *Underground works*: There are two changes underground. The length of tunnels for LHRs is drastically reduced. More importantly, in the HHR repository, the 5.5-m-diam disposal tunnels are replaced with 15-cm boreholes.

3. *Waste packages and capsules*: The HHR–LHR systems would drastically reduce the number of high-performance, high-cost WPs and replace most of them with HHR capsules.

If defense or P-T HLW is divided into HHR and VLHR streams, the repository size is further decreased. In these cases, a few silos for VLHR wastes replace the WPs (Sec. III.C). Using the example of YM, 10 silos would replace ~10 000 WPs and ~100 km of tunnel.

VII.B. Separations Versus Disposal Costs

The economic benefits of separate management of HHRs and LHRs is a trade-off between separation costs and disposal costs. Comparisons using relative facility sizes suggest that repository savings would exceed the incremental costs for HHR–LHR separation and solidification in many fuel cycles. This assumes that (a) a

TABLE III

Relative Size of a Conventional SNF Repository and HHR–LHR Repositories

Parameter	Conventional Repository[a]	HHR, LHR, and Combined Repositories					
		10-yr-old LHR Waste			50-yr-old LHR Waste		
		HHR[a]	LHR	Total	HHR[a]	LHR	Total
Area (relative size)	100	28	12	40	28	4	32
Disposal sites (relative length)							
Tunnels (5.5-m-diam)	100	1	12	13	1	4	5
Boreholes (15-cm-diam)	0	71	0	71	71	0	71
Waste packages (relative length)							
Waste package	100	0	17(13[b])	17(13[b])	0	17(6[b])	17(6[b])
Capsules	0	71	0	71	71	0	71

[a]Disposal of SNF and HHRs occurs 10 yr after SNF discharge from the reactor.
[b]Number of oversized WPs. With 10-yr-old waste, YM WP is volume (10 m^3) limited. With a slightly larger WP, the WP becomes heat limited, and only 13% as many LHR WPs as SNF WPs are required. With 50-yr-old waste, all WPs are volume limited—not heat limited. Jumbo (30 m^3) WPs may be used to reduce the number of packages by a factor of 3.

decision has been made to process SNF and (b) a large-scale nuclear system exists where economics of scale are achieved. The economics of the larger decision on once-through versus recycle fuel cycles is primarily controlled by the relative cost of natural uranium versus reprocessing and plutonium fuel fabrication.

The Nuclear Energy Agency[11] has published the estimated disposal costs in 11 different countries. For direct SNF disposal, the costs varied between \$100 and \$425/kg. For disposal of HLW, the costs vary between \$60 and \$400/kg of SNF that is processed. The potential repository savings could be significantly more than half (~\$200/kg) of these repository costs (Table III).

The estimates for processing of SNF are typically ~\$600 to 1000/kg (Ref. 28). The separations processes within a reprocessing plant are typically less than 20% of the total costs (\$120 to 200/kg). The cost of separating HHRs (cesium and strontium) would be expected to be smaller than the costs for separating and purifying uranium and plutonium. The mass of HHRs is <0.5 wt% of that of uranium and plutonium. Cesium and strontium have very different chemical properties when compared to most other elements in SNF, which simplifies separations. Furthermore, many other facility subsystems, such as the glass melter, would be less expensive (Sec. VI.B) after HHR removal.

For any system where HHRs are separated from the HLW for other purposes, the economics strongly favor separate disposal of HHRs—if the quantities of HHRs are significant. There are large uncertainties. Significant work will be required to understand the cost impacts of separate management of HHRs because so many components of the fuel cycle simultaneously change.

VIII. OTHER CONSIDERATIONS

VIII.A. SNF and Other Wasteforms with HHRs

The fundamental concept here is separate management of HHRs where practicable. Adoption of a waste management strategy for HHRs does not constrain the LHR repository to contain only LHRs. It may contain SNF and HLW, where it would not be economical to separate these wastes into HHR and LHR forms.

VIII.B. Cesium-135

The HHRs contain one long-lived radionuclide, ^{135}Cs. It has a half-life of 3×10^6 yr. Performance assessments of proposed repositories[29,30] indicate that this long-lived radionuclide is not usually a significant risk to man nor a significant factor in terms of repository performance. There are several reasons for this:

1. *Geochemistry*: Radionuclides, such as ^{129}I, ^{237}Np, and ^{99}Tc, that dominate the long-term risks from a repository are those most easily transported by groundwater with little retention by the geology. There is significant retention of cesium in most types of rock and ion-exchange of radioactive cesium isotopes with nonradioactive cesium in the rock.

2. *Biological effects*: Some radionuclides accumulate in specific human organs with resultant high-radiation doses to that organ. An example is ^{129}I, which accumulates in the thyroid and often controls the long-term risk to the public from a repository. This is not an issue with ^{135}Cs. The hazard from ^{135}Cs is one-fiftieth of that of

^{129}I/Ci when measured using the U.S. Nuclear Regulatory Commission effluent standards on allowable concentrations of radionuclides in water[31] discharged from the environment.

For any HHR disposal option, a performance assessment of the risks from this radionuclide will be required. Such an assessment would be significantly simpler to make than for HLW or SNF because there is only a single radionuclide.

VIII.C. P-T Fuel Cycles

There have been P-T fuel cycle studies for ~30 yr. In these studies, the objective has been to find methods to destroy specific long-lived radionuclides to reduce the radiotoxicity of the waste. In most of these studies, it has been proposed to destroy americium and curium. Americium and curium are the third and forth largest heat generators in SNF. With their destruction, HLW can be divided into HHRs and VLHRs. With VLHRs, silos rather than thousands of waste packages may be used for VLHR disposal with major cost and repository performance advantages. This suggests that in a P-T system, the benefits of destroying long-lived heat generators (americium and curium) and altering the repository design may be as important as the benefits from the reduction in radiotoxicity by destroying these radionuclides.

IX. CONCLUSIONS

Historically repositories were designed to accept whatever waste was generated—with the exception that some requirements were imposed on the chemical characteristics of the waste; i.e., HLW liquids are to be converted to HLW glass. More recently, investigations are underway to reduce the radiotoxicity of wastes by P-T. There is a third alternative: Design the repository to separately dispose of several categories of waste, where the waste categories are chosen to maximize repository performance and minimize repository costs. The viability of this approach depends upon the gains in the repository versus the costs in added waste processing. Significant added research is required to understand the full implications of this alternative waste management strategy.

APPENDIX

ALTERNATIVE HHR DISPOSAL OPTIONS

There are alternative HHR disposal options. Some of these options are described herein. Several of them may have significant economic and performance advantages over collocating the LHR and HHR disposal facilities. The disadvantage of these stand-alone options is the necessity to license a second site.

A.I. MODIFIED CONVENTIONAL REPOSITORY DESIGNS

The limited lifetimes of HHRs allow changes in repository design. The example repository design described previously applies only to a YM-type repository that is above the groundwater table in a relatively dry climate. However, major simplifications with corresponding reductions in cost would be expected for many other types of repositories in different geologies.

A historical example is the first repository design in the United States—the Lyons, Kansas, repository. It was to be located in bedded salt. Salt was chosen as the geological media because its high thermal conductivity allowed close spacing of WPs. The design had areal heat loads significantly higher than those proposed today for a salt repository. The safety case was that groundwater had to dissolve large quantities of salt before reaching the WPs. Given the available groundwater, the radionuclides would decay before any groundwater reached the WPs.

At the time of these studies, the proposed wasteform was HLW, and it was thought that ^{137}Cs and ^{90}Sr were the dominant long-term hazards. The hazards of actinides and long-lived fission products were not fully understood. Under these conditions, almost any salt deposit with any WP could safely isolate the wastes—even if there had been nearby drilling for oil or nearby mining of salt. Such simple, inexpensive designs are viable for HHRs, but not for HLW or SNF. A salt repository for HLW and SNF requires a careful selection of the salt deposit and other design features to ensure long-term performance.

A.II. SURFACE DISPOSAL OF HHRs

HHRs can be managed in near-surface facilities until the radionuclides decay to insignificant levels. This option implies institutional control of the disposal site for several hundred years. Two options have been identified:

1. *Storage*: The HHRs would be placed in long-term dry storage in a facility similar to existing SNF dry storage facilities. The HHRs would be stored until most of the radionuclides had decayed to low levels. The waste would then be treated as LLW.

2. *Shallow-land disposal*: The HHRs would be mixed with grout, the grout would be sent to a concrete-lined shallow-land disposal facility, and the grout would be allowed to solidify into large cement monoliths. This option[32] is one of several options that are being developed

at the SRS, which is near Aiken, South Carolina, for the disposal of ^{137}Cs from defense HLW.

A.III. NONCONVENTIONAL HHR REPOSITORIES

Several advanced HHR disposal options require high-heat generation rates to function. These options have potentially large economic and performance advantages as compared to those of conventional repositories, but only limited information is available. In addition, some of their characteristics raise institutional questions. For example, it would be very difficult to retrieve the HHRs after disposal, and it would be difficult to inspect the disposal site after initial operation. Because of technical and institutional uncertainties, these options were not seriously considered for HLW or SNF disposal. However, these disposition options may be potential candidates for HHRs because of the limited lifetimes of HHRs.

A.III.A. Melt-Rock Repository

In the melt-rock repository, a large, spherical, underground cavity would be constructed several hundred to several thousand metres underground. Large quantities of HHRs would be placed in the cavity. During loading operations, active cooling systems control temperatures. After the cavity is loaded, the cavity would be sealed and the cooling systems would be shut off. The HHRs would melt and then melt the surrounding rock. The radionuclides would then be incorporated into the molten rock. It is large-scale vitrification of waste. Ultimately, as the decay-heat levels decrease, the molten rock would solidify into solid rock.

During the period of high-temperature operations, the high temperatures result in plastic deformation of the rock beyond the melt zone that seals all cracks. This concept is an extension of what is seen in an underground nuclear-weapons test during which the explosion creates molten rock that solidifies and traps most of the radionuclides.

The concept[33] was originally developed for the disposal of liquid HLW. A cavity 11 m in radius would be constructed several hundred to several thousand metres underground using nuclear explosives or conventional mining techniques. The cavity would accept HLW liquids for over 25 yr from a 1500 t/yr commercial reprocessing plant. The cavity would be cooled by allowing the wastes to boil. The steam would be condensed, and the resulting liquid would be returned to the cavity. At the end of this time, the HLW would boil dry, the cavity would be sealed, and the radionuclides and rock would form into a molten mass.

The decay heat would slowly melt rock, creating a molten rock zone ~96 m in radius ~90 yr after cavity closure. Beyond this time, the molten rock would slowly solidify as the decay heat decreased and thermal conduction removed heat from the molten rock. Three to four hundred years would elapse before full solidification of the molten rock. Variants of the concept[34] were later examined that used only mined cavities.

Several uncertainties have been identified with this disposal option. However, the identified uncertainties apply only to HLW, not disposal of HHRs. Further analysis would be required to determine if there are unidentified failure modes when disposing of HHRs:

1. *Short-term radionuclide releases*: With a liquid HLW feed and decades between the initial placement of liquids and the time the liquids are allowed to boil dry, liquid wastes may leak from the cavity before the melting begins. This issue does not exist for HHRs added to the cavity as solids and actively cooled until the melting begins.

2. *Long-term radionuclide releases*: There is the potential for leaching of long-lived radionuclides from the rock after cooling.[34] The cooling process will result in fractured rock and the potential for migration of groundwater through the former molten rock zone and the transport of radionuclides from the disposal site. This uncertainty does not exist for disposal of HHRs that decay before the rock solidifies.

The preliminary economic analysis[33] indicated disposal costs would be about one-fifth that of a conventional repository. These early 1970s cost estimates were arrived at before the requirements and costs of conventional repositories significantly increased. The cost advantage would be expected to be larger today.

A.III.B. Saltdiver Repository

The saltdiver repository uses the high-heat generation rates of HHR capsules to allow disposal at depths up to 10 000 m underground in salt domes. The HHRs are packaged into moderately large containers (saltdivers) that are placed in a salt dome. The high-density heat source sinks by heating the salt under the WP until the salt becomes plastic or melts. Salt melts at 800°C. The saltdiver then sinks to the bottom of the salt dome. The configuration of a salt dome is shown in Fig. A.1.

Salt domes contain relatively pure salt in the shape of a mushroom. The vertical dimension may be as large as 10 000 m. Salt domes form from deep layers of bedded salt. The density of salt is lower than other rocks, and salt is plastic. When there is a weakness in the layered rock above a deeply buried bedded salt deposit, the salt forces itself through the rock, creating salt domes. The bedded salt is squeezed into the dome. The dissolution rate of a salt dome by groundwater is slow. As salt is dissolved, there is a buildup of insoluble residues around the edges and top of the dome. These residues form a barrier to groundwater dissolution of the salt.

Scoping calculations indicate a 1-m spherical saltdiver will descend to the bottom of a large salt dome in

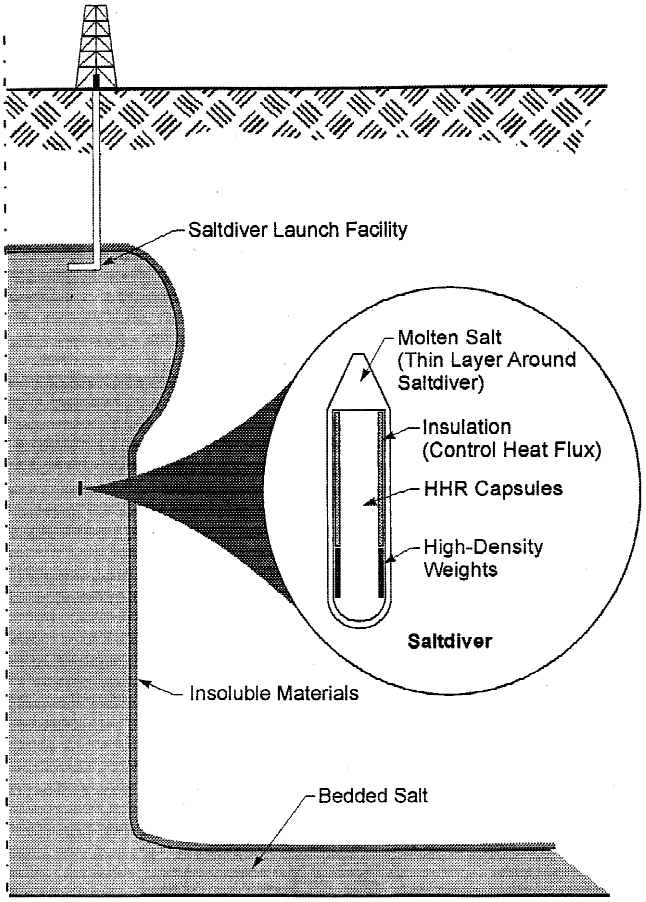

Fig. A.1. Disposal of HHRs in a salt dome using a saltdiver.

about a decade. This assumes that the salt is rigid until it melts. The saltdiver moves only through molten salt. The actual descent rate may be much faster.

1. *Salt plasticity*: Salt is plastic. Heavy objects slowly sink in salt. The plasticity increases with temperature.

2. *Water migration*: Salt contains water that migrates toward a heat source. Consequently, any heat source more dense than salt slowly sinks over time.

3. *Package design*: The WP geometry and internal distribution of heat generation were not optimized. Design of such a saltdiver involves detailed fluid-dynamics and heat transfer analysis. The optimum package would likely include higher temperatures in the nosecone of the saltdiver to assist melting of the salt or increasing its plasticity.

In conventional salt repository designs, there is a requirement to allow WP recovery for some time period. Because of the tendency of normal WPs to sink in salt, the salt temperature must be limited to prevent significant downward migration. With a saltdiver, this negative feature of a salt repository becomes a positive feature.

If successfully developed, this may be the lowest risk disposal option for HHRs. To a first-order approximation, waste isolation improves with depth. No other option exists that can dispose of wastes at such depths. There is no realistic potential for accidental human intrusion in the future. Furthermore, the safety case is simple. A typical salt dome contains cubic kilometres of salt. The time to dissolve a significant fraction of the salt far exceeds the time for the ^{137}Cs and ^{90}Sr to decay. Furthermore, the HHRs are at the bottom of the salt dome. As salt dissolves, insoluble materials will fill the void space and lower the salt dissolution rate. Radar penetrates salt domes; hence, it is possible to track saltdivers to determine if there are problems.

If successfully developed, this may be the lowest cost HHR disposal option. What would be required is a launch site in the top of the salt dome where the saltdivers are (a) filled with capsules, (b) welded shut, and (c) launched. Launching would require placing the saltdiver on the salt floor or in a hole in the salt floor of the facility. Because the same launch site can be used repeatedly, no excavation of kilometres of tunnels in the salt would be needed. Relatively inexpensive saltdivers could be viable. Materials such as carbon steel are corrosion resistant in salt and the travel time to the bottom of the salt dome is short (years).

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