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Information Basis for Developing Comprehensive Waste Management System

- US-Japan Joint Nuclear Energy Action Plan Waste Management Working Group Phase I Report -
(Joint Research)

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Dirk GOMBERT, Kevin McMAHON, Scott JAMES and Florie CAPORUSCIO

Geological Isolation Research and Development Directorate

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Japan Atomic Energy Agency
日本原子力研究開発機構

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Mikazu YUI, Hirohisa ISHIKAWA, Atsuo WATANABE[*1], Kyoji YOSHINO[*1], Hiroyuki UMEKI, Kazumasa HIOKI, Morimasa NAITO, Toshihiro SEO[+1], Hitoshi MAKINO, Chie ODA, Yukio TACHI, Seiichiro MITSUI, Yoichi MIYAMOTO[+2], Masahiro SHIBATA, Mark NUTT[*2], Mark PETERS[*2], James BRESEE[*3], Sue LESICA[*3], Patrick SCHWAB[*3], Steve GOMBERG[*3], Jay JONES[*3], Bill HALSEY[*4], John MARRA[*5], John VIENNA[*6], Dirk GOMBERT[*7], Kevin McMAHON[*8], Scott JAMES[*8] and Florie CAPORUSCIO[*9]

Geological Isolation Research and Development Directorate
Japan Atomic Energy Agency
Tokai-mura, Naka-gun, Ibaraki-ken, Japan

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The activity of Phase I of the Waste Management Working Group under the United States – Japan Joint Nuclear Energy Action Plan started in 2007. The US-Japan JNEAP is a bilateral collaborative framework to support the global implementation of safe, secure, and sustainable, nuclear fuel cycles (referred to in this document as fuel cycles). The Waste Management Working Group was established by strong interest of both parties, which arise from the recognition that development and optimization of waste management and disposal system(s) are central issues of the present and future nuclear fuel cycles.

This report summarizes the activity of the Waste Management Working Group that focused on consolidation of the existing technical basis between the U.S. and Japan and the joint development of a plan for future collaborative activities.

+1Policy Planning and Administration Department
+2Horonobe Underground Research Center
*1 Agency for Natural Resources and Energy, Ministry of Economy, Trade and Industry
*2Argonne National Laboratory
*3 U. S. Department of Energy
*4 Lawrence Livermore National Laboratory
*5 Savannah River National Laboratory
*6 Pacific Northwest National Laboratory
*7 Idaho National Laboratory
*8 Sandia National Laboratories
*9 Los Alamos National Laboratory

Firstly, the political/regulatory frameworks related to nuclear fuel cycles in both countries were reviewed. The various advanced fuel cycle scenarios that have been considered in both countries were then surveyed and summarized. The working group established the working reference scenario for the future cooperative activity that corresponds to a fuel cycle scenario being considered both in Japan and the U.S. This working scenario involves transitioning from a once-through fuel cycle utilizing light water reactors to a one-pass uranium-plutonium fuel recycle in light water reactors to a combination of light water reactors and fast reactors with plutonium, uranium, and minor actinide recycle, ultimately concluding with multiple recycle passes primarily using fast reactors. Considering the scenario, current and future expected waste streams, treatment and inventory were discussed, and the relevant information was summarized.

Second, the waste management/disposal system optimization was discussed. Repository system concepts were reviewed, repository design concepts for the various classifications of nuclear waste were summarized, and the factors to consider in repository design and optimization were then discussed. Japan is considering various alternatives and options for the geologic disposal facility and the framework for future analysis of repository concepts was discussed. Regarding the advanced waste and storage form development, waste form technologies developed in both countries were surveyed and compared. Potential collaboration areas and activities were next identified. Disposal system optimization processes and techniques were reviewed, and factors to consider in future repository design optimization activities were also discussed. Then the potential collaboration areas and activities related to the optimization problem were extracted.

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Keywords: US-Japan Collaboration, Waste Management, Fuel Cycle, System Optimization, Repository Design Concept, Advanced Waste Form

包括的廃棄物管理システム開発のための知識基盤
-日米原子力エネルギー共同行動計画廃棄物管理ワーキンググループフェーズ I 報告書-
（共同研究）

日本原子力研究開発機構　地層処分研究開発部門

油井　三和，石川　博久，渡邊　厚夫[*1]，吉野　恭司[*1]，
梅木　博之，日置　一雅，内藤　守正，瀬尾　俊弘[+1]，牧野　仁史，
小田　治恵，舘　幸男，三ツ井　誠一郎，宮本　陽一[+2]，柴田　雅博，
Mark NUTT[*2], Mark PETERS[*2], James BRESEE[*3], Sue LESICA[*3],
Patrick SCHWAB[*3], Steve GOMBERG[*3], Jay JONES[*3], Bill HALSEY[*4],
John MARRA[*5], John VIENNA[*6], Dirk GOMBERT[*7], Kevin McMAHON[*8],
Scott JAMES[*8], Florie CAPORUSCIO[*9]

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日米原子力エネルギー共同行動計画の下での廃棄物管理ワーキンググループの活動は平成 19 年に開始された。日米原子力エネルギー共同行動計画は，安全で持続可能な核燃料サイクルの国際的な実現を支援するための二国間の協力の枠組みである。廃棄物管理と処分システムの開発とその最適化は，今日および将来の核燃料サイクルの中心的課題であるとの認識に基づいて，両国の強い関心によって廃棄物管理ワーキンググループは設立された。

本報告書は，両国における既存の技術基盤の集約と将来の協力計画の作成に焦点をあてて活動をしてきた同ワーキンググループの成果を取りまとめたものである。

はじめに，両国の核燃料サイクルに関連する政策と規制の枠組みについて概観するとともに，両国で検討されている様々な先進的核燃料サイクルシナリオを調査し，それらを取りまとめた。ワーキンググループでは，将来の協力活動のために，日米両国で検討されている燃料サイクルシナリオに対応する「作業用参照シナリオ」を構築した。この「作業用参照シナリオ」は，軽水炉による燃料のワンススルーでの利用から始まって，軽水炉による MOX 燃料のリサイクル，

核燃料サイクル工学研究所（駐在）：〒319-1194 茨城県那珂郡東海村村松 4-33
+1 経営企画部
+2 幌延深地層研究センター
*1 経済産業省資源エネルギー庁
*2 アルゴンヌ国立研究所
*3 米国エネルギー省
*4 ローレンスリバモア国立研究所
*5 サバンナリバー国立研究所
*6 パシフィックノースウエスト国立研究所
*7 アイダホ国立研究所
*8 サンディア国立研究所
*9 ロスアラモス国立研究所

高速炉によるウラン，プルトニウム，マイナーアクチニドのリサイクルと軽水炉の組み合わせ，そして究極的には主として高速炉を用いる複数のリサイクル経路に至るといった，燃料サイクルの変遷を取り扱う内容となっている。この「作業用参照シナリオ」を前提として，現在と将来に予測される廃棄物の発生量や処理方法といった廃棄物の流れが議論され，関連する情報がまとめられた。

次に，廃棄物管理と処分システムの最適化が議論された。処分場システム概念を概観するとともに，様々な分類の放射性廃棄物に対する処分場設計概念を調査し，それらを取りとまとめた。また，処分場の設計と最適化で検討すべき要因などが議論された。先進的な廃棄体についても，両国で開発されている技術について調査と比較を行い，協力可能と思われる分野を特定した。処分システム最適化プロセスと最適化技術についても概観し，将来の処分場設計の最適化で考慮されるべき要因についても議論され，最適化問題についても，協力可能と思われる分野の抽出を行った。

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ACRONYMS

ADS:	Accelerator Driven System
AFCI:	Advanced Fuel Cycle Initiative (USA)
BWR:	Boiling Water Reactor
CCIM:	Cold Crucible Induction Melter
COL:	Combined License
DOE:	U.S. Department of Energy
CRIEPI:	Central Research Institute of Electric Power Industry (Japan)
DUPIC:	Direct Use of Spent PWR Fuel in CANDU
EBS:	Engineered Barrier System
Echem:	Electro-Chemical Reprocessign (sometimes referred to as pyroprocessing)
EIA:	U.S. Department of Energy, Energy Information Agency
FCRD:	Fuel Cycle Research and Development Program (USA)
FR:	Fast Reactor
FS:	Japanese Feasibility Study of fast reactor commercialization
GIF:	Generation IV International Forum
IWMS:	Integrated Waste Management Strategy
JAEA:	Japan Atomic Energy Agency
JNEAP:	U.S. - Japan Joint Nuclear Energy Action Plan
JNFL:	Japan Nuclear Fuels Limited
LFCM:	Liquid Fed Joule-heated Ceramic Melter
LLRWPA:	Low Level Radioactive Waste Policy Act (USA)
LLW:	Low Level Waste
LWR:	Light Water Reactor
METI:	Japan Ministry of Economy, Trade, and Industry
MOX:	Mixed Oxide ($[Pu,U]O_x$) Fuel
NRC:	U.S. Nuclear Regulatory Commission
NUMO:	Nuclear Waste Management Organization of Japan
NWPA:	Nuclear Waste Policy Act (USA)
OREOX:	Oxidation and Reduction of Oxide Fuel
PEM:	Prefabricated EBS Modules
PUREX:	Plutonium and Uranium Extraction aqueous reprocessing
PWR:	Pressurized Water Reactor
R&D:	Research and Development
RWMC:	Radioactive Waste Management Funding and Research Center (Japan)
SNF:	spent nuclear fuel
TRU:	transuranic
UREX:	Uranium solvent extraction
WMWG:	Waste Management Working Group, U.S. - Japan Joint Nuclear Energy Action Plan

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INFORMATION BASIS FOR DEVELOPING COMPREHENSIVE WASTE MANAGEMENT SYSTEM – US-JAPAN JOINT NUCLEAR ENERGY ACTION PLAN WASTE MANAGEMENT WORKING GROUP PHASE I REPORT

1. INTRODUCTION

The United States – Japan Joint Nuclear Energy Action Plan (JNEAP) was signed by representatives from the United States and Japan in April, 2007. The JNEAP is based on shared U.S. and Japanese objectives for global implementation of safe, secure, and sustainable, nuclear fuel cycles (referred to in this document as fuel cycle(s)). Six research and development working groups have been established to execute the JNEAP. These working groups are:

- Fast Reactor Technology
- Fuel Cycle Technology
- Simulation and Modeling
- Small and Medium Reactors
- Safeguards & Physical Protection
- Waste Management

The JNEAP released a Joint U.S. – Japan Cooperative Research and Development (R&D) Plan in January, 2008. This plan described the activities that would be conducted under Phase I of the JNEAP, through June, 2008.

This report presents the results of the Phase I activities completed by the JNEAP Waste Management Working Group (WMWG).

2. Waste Management Working Group

The mission of the Waste Management Working Group (WMWG) is the development of advanced waste forms and generic repository concepts for utilization in the development and consideration of advanced nuclear fuel cycles. The overall objective of the WMWG is the development of general approaches and methodologies for the optimization of waste management systems and waste form and process technologies. Phase I activities, conducted in 2007 and 2008, focused on consolidation of the existing technical basis between the U.S. and Japan and the joint development of a plan for future activities to be completed jointly under the working group.

The specific Phase I goals of the WMWG were:

- Develop a mutual understanding of the important issues pertaining to waste management under advanced nuclear fuel cycles
- Develop a mutual understanding of the current waste management research and development (R&D) program and future plans for both countries

- Develop a joint plan for collaborative waste management R&D including:
  - Future fuel cycle scenarios and waste management requirements
  - Waste inventory definition
  - Waste forms and processes for closed fuel cycle options
  - Advanced geologic disposal concepts
- Begin to provide an integrated view for waste management in the context of the safe, secure, and sustainable nuclear fuel cycles,
  - Develop a basis for optimizing and selecting waste management systems
  - Inform advanced nuclear fuel cycle system decisions in terms of waste management impacts

The activities of the WMWG link to the implementation of safe, secure, and sustainable nuclear power to help meet growing energy demands. The WMWG will develop a general approach and methodology for optimization of waste management systems with applications to a range of future advanced fuel cycles and broad geologic settings and geochemical environments through two interrelated tasks:

- Task 1: Analysis of credible fuel cycle scenarios and resulting wastes;
- Task 2: Waste management system approaches and benefits.

In Phase I, the WMWG considered six sub-tasks as part of Phase I completion and Phase II planning:

- Task 1: Analysis of credible fuel cycle scenarios and resulting waste inventories
  - 1-1: Future scenarios for nuclear energy use
  - 1-2: Waste inventory development
- Task 2: Waste management systems
  - 2-1: Development of repository system concepts
  - 2-2: Advanced waste forms development
  - 2-3: Definition of waste management system evaluations
  - 2-4: Systems analysis

During Phase I, the WMWG exchanged information on relevant programmatic activities in the U.S. and Japan and reviewed available information world-wide. The WMWG also held several information exchange meetings:

- 1st WMWG Meeting, June 20, 2007, Washington, DC, USA
- 2nd WMWG Meeting, September 11 – 12, 2007, Boise, ID, USA
- 3rd WMWG Meeting, December 10 – 14, 2007, Tokyo, Japan
- Waste Form S&T and Modeling & Simulation Workshop, January 29 – 31, 2008, Ann Harbor, MI, USA
- Preparatory meeting for WGs Plenary Meeting, March 13, 2008, Tokyo, Japan
- 4th WMWG Meeting, March 17 – 19, 2008, Argonne, IL, USA

3. Phase I Results

This section discusses the results of the Phase I activities. Section 3.1 discusses the results of Task 1, the analysis of credible fuel cycle scenarios and resulting waste inventories. Section 3.2 discusses the results of Task 2, waste management system studies

3.1 Analysis of Credible Nuclear Fuel Cycles and Resulting Waste Inventories

This section presents the analysis of credible nuclear fuel cycles for consideration within the WMWG. Future scenarios and nuclear energy use in the United States and Japan are discussed first in Section 3.1.1. Section 3.1.2 follows with a discussion of potential waste inventories.

3.1.1 Future Scenarios for Nuclear Energy Use (Task 1-1)

This section first discusses current electricity energy demand, the contribution of nuclear generated electricity to this demand, and future projections in both the United States and Japan. The political and regulatory framework related to nuclear energy and specifically to waste management in the United States and Japan is discussed next. Advanced fuel cycle alternatives, including transition scenarios, considered in the United States and Japan are then discussed. This section concludes with the reference scenario established by the WMWG.

3.1.1.1 Energy Demand and Nuclear Contribution

It is clear that the demand for electricity will increase worldwide through the next century both within developed countries, such as Japan and the United States, and in developing countries. Nuclear power is seen as a potential source of electricity for meeting increased demand. The deployment of nuclear power plants and their types (i.e., light water reactor [LWR], fast reactor [FR], fast breeder reactor [FBR]) and the associated fuel cycles depends on several factors, including technological maturity, cost, regulatory framework, national policy, and national investment. The contribution of nuclear power to total energy generation varies from country to country, driven primarily by domestic factors within those countries.

While nuclear power is a carbon-emission free source of electricity, it is not without waste management challenges. The waste streams can differ significantly between fuel cycles, both in quantities and types of waste. However, there is a commonality in that all wastes must be effectively and safely disposed of so as to be protective of public health and the environment. Thus, any country having nuclear power must also address, either directly or indirectly, the management and disposal of nuclear waste.

Policy of Energy Use in the United States

In 2006, the U.S. Department of Energy’s (DOE)Energy Information Administration (EIA) reported that over 4,000 terawatt-hours of electricity were generated in the United States [Ref. 1]. Figure 1 shows the various sources of electricity generation used in the United States in 2006 [Ref. 1]. The United States depends primarily on fossil fuels with nuclear power contributing roughly20% of the total electricity generated, or 787 terawatt-hours.

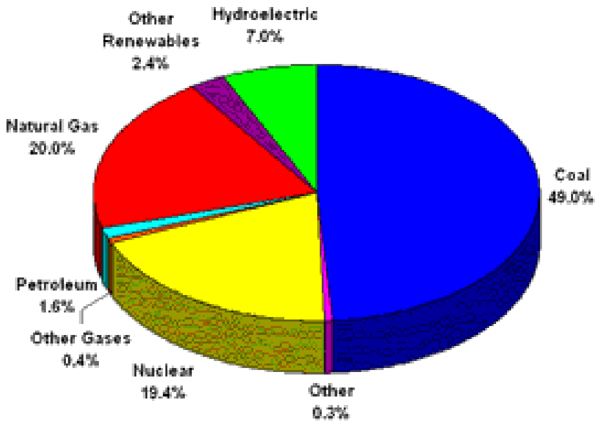


Figure 1. U.S. Electric Power Generation Capacity, 2006

There are 104 U.S. commercial nuclear generating units that are licensed by the U.S. Nuclear Regulatory Commission (NRC). These power plants operated at an estimated annual net capacity factor of roughly 92 percent during 2008 [Ref. 1]. Nuclear power plants in the United States were initially licensed by NRC to operate for a period of 40 years. The NRC has established a license renewal process with clear requirements to verify safe plant operation for up to an additional 20 years of plant life. As of 2009, 54 of the 104 licensed plants have been granted license extensions, applications for license renewal for 18 reactors have been submitted to NRC and are under review, and applications for 15 reactors are planned to be submitted [Ref. 2].

A combined operating license (COL), when issued, is authorization from NRC to construct and, with conditions, operate a nuclear power plant at a specific site and in accordance with laws and regulations. Before issuing a COL, the NRC staff will complete safety and environmental reviews of the combined license applications in accordance with the Atomic Energy Act, NRC regulations, and the National Environmental Policy Act. As of 2009, seventeen applications for a COL have been submitted to NRC for new nuclear power plants [Ref. 2].

Figure 2 shows the EIA projections for all sources of electricity generation based on current conditions [Ref. 3]. The EIA forecasts that electricity demand will grow at an average rate of 1.3 percent per year through 2030. The majority of this increase in demand is forecast to be met through fossil fuel generation. Nuclear generating capacity is forecast to increase from 100.2 gigawatts in 2006 to 118.8 gigawatts in 2030. An additional 20 gigawatts of newly built nuclear capacity is forecast, partially offset by 4.5 gigawatts of capacity retirement.

The United States has recognized that nuclear power is a dependable source of carbon-emission free electric power. As such, the United States has initiated programs to help increase the deployment of nuclear generating capacity over the near- and long-terms, both domestically and internationally [Ref. 4].

<u>Nuclear Power 2010</u>: The Nuclear Power 2010 program was initiated in 2002 and is a joint government/ industry cost-shared effort to identify sites for new nuclear power plants, develop and bring to market advanced nuclear plant technologies, evaluate the business case for building new nuclear power plants, and demonstrate untested regulatory processes. Accomplishing these program objectives will pave the way for industry decisions to build new advanced LWR nuclear plants in the United States that would begin operation early in the next decade.

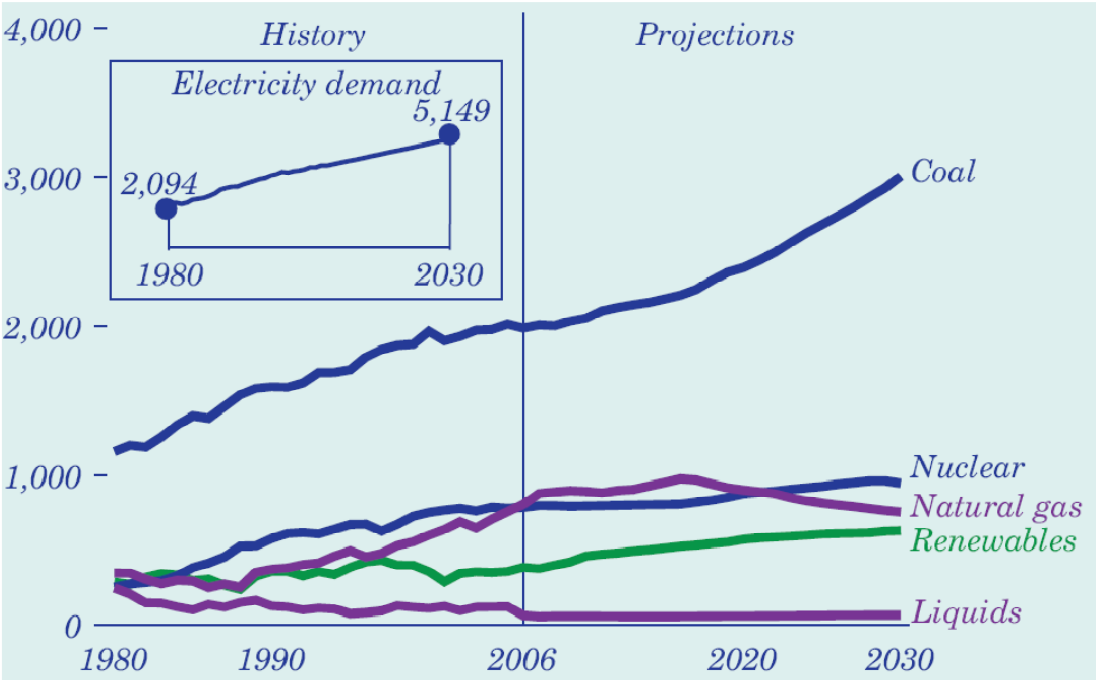


Figure 2. Electricity Generation by Fuel, 1980–2030 (billion kilowatt-hours)

Generation IV Nuclear Energy Systems: Ten countries have joined together to form the Generation IV International Forum (GIF) to develop future-generation nuclear energy systems that can be licensed, constructed, and operated in a manner that will provide competitively priced and reliable energy products while satisfactorily addressing nuclear safety, waste, proliferation, and public-perception concerns. The objective for Generation IV nuclear energy systems is to have new nuclear power plants available for international deployment before the year 2030, when many of the world's currently operating nuclear power plants will be at or near the end of their operating licenses.

Fuel Cycle Research and Development: The mission of the Fuel Cycle Research and Development program (FCR&D) is to develop options to the current commercial fuel cycle management strategy to enable the safe, secure, economic, and sustainable expansion of nuclear energy while reducing proliferation risks by conducting research and development focused on nuclear fuel recycling and waste management to meet U.S. needs. The FCR&D program aims to recycle used fuel so it can be used as fuel again, thereby "closing the fuel cycle." The Fuel Cycle R&D program is charged with the investigation and preparation of tomorrow's U.S. nuclear fuel cycle.

The success of these programs would support increased nuclear generation of electricity in the U.S. This resurgence could result in nuclear power contributing a larger fraction of total electricity generation than is projected in Figure 1.

A key aspect of any fuel cycle is the management of wastes. As discussed in Section 3.1.2.1.1, the U.S. policy had been the direct disposal of spent nuclear fuel (SNF); however the U.S. will be evaluating alternative approaches for waste management. A decision by the U.S. to choose to recycle spent nuclear fuel in an advanced nuclear fuel cycle would present the opportunity to change the current approach for managing and disposing nuclear waste.

Policy of Energy Use in Japan

Japan's energy policy has been driven by considerations of energy security and the need to minimize dependence on current imports. The main elements regarding nuclear power are:

- Continue to have nuclear power as a major element of electricity production.
- Recycle uranium and plutonium from SNF, initially in LWRs, and have reprocessing domestically from 2005.
- Steadily develop fast breeder reactors in order to improve uranium utilization dramatically.
- Promote nuclear energy to the public, emphasizing safety, security and safeguards (non-proliferation).

In March 2002 the Japanese government announced that it would rely heavily on nuclear energy to achieve greenhouse gas emission reduction goals set by the Kyoto Protocol. A 10-year energy plan, submitted in July 2001 to the Minister of Economy Trade & Industry (METI), was endorsed by the cabinet. It called for an increase in nuclear power generation by about 30 percent (13,000 MWe), with the expectation that utilities would have 9 to 12 new nuclear plants operating by 2011.

As of January 2009, Japan has 53 reactors totaling 47,935 MWe on-line, with 3 (3,668 MWe) under construction and 10reactors (13,562 MWe) planned.

In October 2005 the Atomic Energy Commission reaffirmed policy directions for nuclear power in Japan, while confirming that the immediate focus would be on LWRs [Ref. 5]. The main elements are that a "30-40% share or more" shall be the target for nuclear power in total generation after 2030, including replacement of current plants with advanced light water reactors. Fast breeder reactors will be introduced commercially, but not until about 2050. Used fuel will be reprocessed domestically to recover fissile material for use in MOX fuel.

As discussed above, the basic policy of Japan for the nuclear fuel cycle is to reprocess SNF and make effective use of the recovered uranium, plutonium, and other usable elements. SNF generated in nuclear power reactors is sent for reprocessing after a period of onsite storage. Until now, part of SNF has been reprocessed at the Tokai Reprocessing Plant of the Japan Atomic Energy Agency (JAEA) and in overseas facilities. In the meantime, Japan Nuclear Fuel Ltd. (JNFL) is constructing the SNF reprocessing planting Rokkasho-mura in Aomori Prefecture, which is currently completing start-up testing leading to full-scale operation. JNFL received and began placing SNF in the storage facility of the reprocessing plant in 1998 as a test, and full-scale SNF storage officially started in 1999. The interim storage of SNF before reprocessing allows flexibility in the nuclear fuel cycle. The Reactor Regulation Law was amended in 1999 to incorporate provisions on interim SNF storage, and a new company named Recycle Fuel Storage Company [RFSC] was established. In 2007, RFSC filed an application for license for the commercial operation of central interim storage facility to be constructed in Mutsu city of Aomori Prefecture in 2007. The facility is planned to start operation in2012.

3.1.1.2 Political/Regulatory Framework

This section summarizes the political and regulatory framework as related to nuclear energy, and more specifically to waste management, within both the United States and Japan. Pertinent laws and regulations are presented and summarized. A key component of this discussion is the classification of nuclear wastes, roles and responsibilities between government and commercial entities, and how the various classes of waste are to be managed and disposed. In this context, a classification scheme being proposed by the International Atomic Energy Agency (IAEA) is also referred to in this section.

Political/Regulatory Framework in the United States

In the United States, energy is generated, for the most part, by private companies. This includes nuclear power. There must be a financial incentive for private companies to invest in nuclear power plants. The investment in nuclear power plants is significant, and these companies must have high confidence that new power plants can be licensed efficiently, risks can be effectively managed and mitigated, and the power plants can be operated reliably. In addition, they must have confidence that the spent nuclear fuel can be managed and that ultimately there will be a path for disposal of nuclear wastes.

Laws pertaining to nuclear power are codified in various acts, including the U.S. Atomic Energy Act, the Nuclear Waste Policy Act (NWPA) [Ref. 6], and the Low Level Radioactive Waste Policy Act (LLRWPA) [Ref. 7]. Responsibility for portions of the nuclear fuel cycle is provided under these acts.

The U.S. DOE has the responsibility to dispose of SNF and HLW under the NWPA. The U.S. Environmental Protection Agency (EPA) has authority to establish radiation protection standards and the NRC regulates the disposal of spent nuclear fuel and HLW under the NWPA. The LLRWPA assigns the responsibility for disposing of Class A, B, and C LLWs to those that generated the waste. Regulatory authority over disposing of Class A, B, and C LLW resides with either the NRC or with individual states that have enacted regulations that reflect the requirements in NRC regulations (called agreement states). The LLRWPA assigns the responsibility for disposing of waste with concentrations of specific radionuclides is in excess of the limits of Class C LLW to DOE, with NRC regulating disposal. LLW classification is discussed later in this section.

National policy for managing and disposing of SNF and HLW is established in the NWPA. In 1982, the NWPA established a process for the nomination of at least five sites suitable for site characterization and the recommendation of three of those sites to be characterized for a first repository and a second site. This ultimately led to the selection of the Deaf Smith County (bedded salt), Hanford (basalt), and Yucca Mountain (tuff) sites for characterization for the first repository site. In 1987, the NWPA was amended to terminate site characterization activities at all candidate sites other than the Yucca Mountain Site (Subtitle E). In addition, activities related to a second repository site were also terminated.

The definition of HLW was developed in 1982 based on SNF reprocessing technologies present at that time, in particular the PUREX process. Only plutonium and uranium are recovered in the PUREX process with all other transuranics and the vast majority of the fission product elements remain as waste. The resultant waste form remained extremely hazardous for a very long period of time, presenting a large risk to the public. As such, the NWPA defines the term high-level radioactive waste as [Ref. 6]:

1. The highly radioactive material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and
2. Other highly radioactive material that the Commission, consistent with existing law, determines by rule requires permanent isolation.

The NWPA "prohibit[s] the emplacement in the first repository of a quantity of SNF containing in excess of 70,000 metric tons of heavy metal or a quantity of solidified HLW resulting from the reprocessing of such a quantity of SNF until such time as a second repository is in operation." The NWPA states that site-specific activities with respect to a second repository may not be conducted unless Congress has specifically authorized and appropriated funds for such activities. Further, the NWPA directs the Secretary of Energy to report to the President and to Congress between January 2007 and January 2010 on the need for a second repository; this report was submitted in 2008.

The LLRWPA establishes responsibilities for the disposal of LLW for both the States and the Federal Government. Each State, either by itself or in cooperation with other States, is responsible for the disposal of:

- Class A, B, or C radioactive wastes generated within the state
- LLW that is generated by the Federal Government, except for waste that is owned or generated by the U.S. DOE
- Class A, B, or C radioactive waste generated outside the state and accepted for disposal.

The Federal Government is responsible for the disposal of:

- LLW owned or generated by the U.S. DOE
- Any other LLW with concentrations of radionuclides that exceed the limits for class C radioactive waste.

The LLRWPA also established the policy of the Federal Government that the responsibilities of the States for the disposal of LLW can be most safely and effectively managed on a regional basis and that States may enter into compacts to provide for the establishment and operation of regional disposal facilities for LLW.

The LLRWPA establishes the authority of States to regulate the disposal of LLW as under an agreement with NRC (as "agreement states").

The classification of radioactive waste involves two considerations. First, consideration must be given to the concentration of long-lived radionuclides (and their shorter-lived precursors) whose potential hazard will persist long after such precautions as institutional controls, improved waste form, and deeper disposal have ceased to be effective. These precautions delay the time when long-lived radionuclides could cause exposures. In addition, the magnitude of the potential dose is limited by the concentration and availability of the radionuclide at the time of exposure. Second, consideration must be given to the concentration of shorter-lived radionuclides for which requirements on institutional controls, waste form, and disposal methods are effective.

- Class A low-level radioactive waste contains the lowest concentration of radioactive materials, and most of those materials have half-lives of less than five years.
- Class B contains the next lowest concentration of radioactive materials, and it contains a higher proportion of materials with longer half-lives. Class B waste must meet rigorous requirements on the waste form to ensure its stability in the disposal system.
- Class C low-level waste has the highest concentration of radioactive material allowed to be buried in a low-level waste disposal facility. The waste form must meet rigorous requirements to ensure stability and the disposal system must include additional measures to protect against inadvertent human intrusion.
- The concentration of radioactive materials in Greater Than Class C (GTCC) exceeds the limits for Class C waste. GTCC waste s not generally acceptable for near-surface disposal and the waste form and disposal methods must be different than, and in general more stringent, than those specified for Class C waste. In the absence of specific requirements, such waste must be disposed of in a NRC licensed geologic repository.

National regulations relating to energy, including nuclear power, are codified in Title 10 of the U.S. Code of Federal Regulations. Environmental protection regulations are codified in, Chapter I:

National regulations for protection of the environment are codified in Title 40 of the U.S. Code of Federal Regulations. Specific regulations include:

- 10 CFR Part 50: Domestic Licensing of Production and Utilization Facilities
- 10 CFR Part 51: Environmental Protection Regulation for Domestic Licensing and Related Regulatory Functions
- 10 CFR Part 52: Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants
- 10 CFR Part 60: Disposal of High-Level Radioactive Wastes in Geologic Repositories
- 10 CFR Part 61: Disposal Requirements for Land Disposal of Radioactive Waste
- 10 CFR Part 63: Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada
- 10 CFR Part 71: Packaging and Transportation of Radioactive Material
- 10 CFR Part 72: Licensing Requirements for Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste.
- 40 CFR 190: Environmental Radiation Protection Standards for Nuclear Power Operations
- 40 CFR 191: Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes
- 40 CFR 197: Public Health and Environmental Radiation Protection Standards for Yucca Mountain, Nevada

Political/Regulatory Framework in Japan

The overall system of legislations and regulations for utilization of nuclear energy is based on the Atomic Energy Basic Law that was established in 1955. The objectives of the law are quoted as "to secure future energy resources, achieve progress in science and technology, and promote industry, by encouraging research, development, and utilization of nuclear energy, and thereby contribute to the improvement of the welfare of human society and the national living standard." The basic policy is prescribed as follows: "The research, development and utilization of nuclear energy shall be limited to peaceful purposes, on a basis of ensuring priority to safety, and performed on a self-disciplined basis under democratic administration, and the results thereof shall be made public and actively contribute to international cooperation."

In order to attain these objectives and achieve the basic policy, the law prescribes the following:

- Establishment of the Atomic Energy Commission and the Nuclear Safety Commission, and their duties, organization, administration, and authorities
- Regulations on the nuclear fuel materials
- Regulations on the construction, etc. of reactor facilities.
- Prevention of radiation hazards.

The Atomic Energy Basic Law also prescribes the assignment of these matters to the respective laws.

Major laws established for the purpose of providing safety regulations on the utilization of nuclear energy and related laws include:

- Law for the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors (the Reactor Regulation Law),
- Electricity Utilities Industry Law,
- Law Concerning Prevention from Radiation Hazards due to Radioisotopes, etc. (the Radiation Hazards Prevention Law) and
- Medical Care Law.

Other laws that pertain to nuclear energy include:

- Basic Law for General Emergency Preparedness,
- Special Law of Nuclear Emergency Preparedness (the Special Law for Nuclear Emergency),
- Law for Technical Standards of Radiation Hazards Prevention, and
- Specified Radioactive Waste Final Disposal Act

The Reactor Regulation Law was established in 1957. This law and the relevant regulation were amended in October 2003 and the clearance system was put into operation.

In Japan, HLW is specified as solidified waste generated from reprocessing of SNF. All non-HLW is classified as LLW. SNF from power reactor facilities is being held in storage at SNF management facilities within power reactor facilities, at the SNF management facilities within the Tokai Reprocessing Facility of JAEA and Rokkasho Reprocessing Plant of JNFL, as mentioned in 3.1.1.1.2. SNF stored in the JNFL storage facility totals approximately 3,000 tons as of the end of June, 2009. As of the end of June, 2009, 1,310 vitrified waste packages are stored at JNFL. It is anticipated that a SNF equivalent to 40,000 vitrified waste packages will be generated by 2020. These packages are to be disposed of in a geological disposal facility deeper than 300 meters as specified in the Specified Radioactive Waste Final Disposal Act.

LLW with relatively high radioactivity, such as control rods and core internals, is to be disposed of in an intermediate-depth (e.g. 50 to 100 meters) disposal facility. LLW with relatively low radioactivity is to be disposed of in a near surface (approx. 10 meters) disposal facility (concrete pit). LLW with very low radioactivity is to be disposed of in a near surface disposal facility (trench). Transuranic (TRU) waste and uranium waste are to be disposed of in either geological, intermediate, shallow-depth, or near-surface disposal facilities, depending on their radioactivity (also see Section 3.1.2.2.1).

The Specified Radioactive Waste Final Disposal Act, established in May 2000, provided the framework for the planned and steady final disposal of “Specified Radioactive Waste,” which is HLW resulting from reprocessing of spent nuclear fuel. The major points of the law are, (1) the government establishes basic policy and plan on final disposal of specific wastes (Final Disposal Plan), (2) establishment of an implementing organization, (3) measures to secure financial resources for disposal, and (4) site selection procedure. The Act also requires that safety regulations should be developed separately by the safety authorities.

The Minister of the Ministry of Economy, Trade, and Industry (METI) establishes the Basic Policy on Specified Radioactive Waste Final Disposal and the Specified Radioactive Waste Final Disposal Plan pursuant to the Act. The Basic Policy and Final Disposal Plan are focused on a ten year period, being reviewed and revised every five years. The Nuclear Waste Management Organization of Japan (NUMO), which was established as an implementing organization based on the Act, carries out the activities of final disposal according to the Basic Policy and Final Disposal Plan as well as the Act. Utilities shall contribute financial resources to the fund reserved for disposal, which is managed by an organization (Radioactive Waste Management Funding and Research Center, RWMC) designated by the Minister of METI. The NUMO promotes site selection by a three step procedure specified in the Act, that is, selection of the preliminary investigation area(s), detailed investigation area(s) and selection of a site for a disposal facility, and obtains approval of the Minister of METI at each step of procedure. At each step of selection, the Minister of METI needs to consult with governors and local governments, and revises the Final Disposal Plan as appropriate considering the opinions. A specific feature of NUMO siting process is to ask volunteers for accepting a geological disposal facility.

A brief summary of the evolution of policies and plans pertaining to waste management after the establishment of the Act follows:

- The Basic Policy spells out the basic directions for the disposal of specified radioactive waste. The Specified Radioactive Waste Final Disposal Plan describes the amounts and estimates of specified radioactive waste to be disposed of, the time frame for the selection of a disposal site, and so forth. Both were approved by the Cabinet in September 2000 pursuant to the Act.
- The Basic Policy and the Final Disposal Plan were reviewed and revised without major changes and, approved by the Cabinet in October 2005.
- The Act was amended to include the geological disposal of certain types of TRU waste in April 2007, which was followed by the approval of the Cabinet on the modifications of the Basic Policy and Final Disposal Plan in March 14, 2008. These amendments came into force on the 1st of April 2008.
- In May 2007, the NSC (Nuclear Safety Commission Japan) issued a report on the upper limit of radioactivity concentrations of low level solid radioactive waste that can be disposed of respectively in near surface disposal facility and intermediate depth disposal facility.

The revisions of the Basic Policy and the Plan include the addition of certain types of TRU waste with a long half-life to the list of wastes subject to final disposal by NUMO and the revision of the time frame for the selection of a disposal site in consideration of the latest situation that no volunteers had not appeared by that time.

The amended "Specified Radioactive Waste Final Disposal Act (Final Disposal Act)" adds the fund arrangement to cover the costs and the implementing organization for geological disposal of TRU waste generated from nuclear fuel cycle, such as reprocessing of spent nuclear fuel and MOX fuel fabrication. The term "specified radioactive waste" therefore means vitrified HLW and TRU waste and this enables NUMO to be implementer for geological disposal of TRU waste in addition to vitrified HLW.

The associated amendment was made on "Law for the Regulations of Nuclear Sources Material, Nuclear Fuel Material and Reactors" to cover TRU waste and HLW by providing:

- The classification of waste disposal facilities with radioactivity concentration limit between geological disposal and near surface/intermediate-level depth disposal;
- Inspection scheme and physical protection before repository closure as a part of the legal framework of safety regulations for the geological disposal.

Concerning the disposal of low-level radioactive waste generated from the research, medical and industrial facilities for using radioisotopes and radiation generators, Ministry of Education, Culture, Sports, Science and Technology, MEXT amended the Law for the Independent Administrative Agency, Japan Atomic Energy Agency, in May 2008, and the JAEA was designated as the implementation entity of the disposal of the waste. In Feb 2009, JAEA established "Low-level radioactive waste disposal project center" for this purpose.

IAEA Proposed Waste Classification System

The International Atomic Energy Agency (IAEA) is proposing a revision to the waste classification system. This waste classification system could potentially be used both in the U.S. and Japan when considering the management of wastes from advanced fuel cycles.

A draft of IAEA draft Safety Guide, 'Classification of radioactive waste' (DS390) [Ref. 8] was approved by the Waste Safety Standards Committee (WASSC) in April 2008 and is planned to be published in 2009. The following six classes are proposed in this draft:

(1) Exempt waste (EW): Waste that meets the criteria for clearance, exemption or exclusion from regulatory control for radiation protection purposes.

(2) Very short lived waste (VSLW): Waste that can be stored for decay over a limited period of up to a few years and subsequently cleared according to arrangements approved by the regulatory authority, for uncontrolled disposal, use or discharge. This would include radioactive waste containing primarily radionuclides with short halflives often used for research and medical purposes.

(3) Very low level waste (VLLW): Waste which does not necessarily meet the criteria as EW, but which does not need a high level of containment and isolation and therefore is suitable for disposal in near surface landfill-type facilities, with limited regulatory control. Such landfill-type facilities may also contain other hazardous waste. Typical waste in this class would include soil and rubble with low activity concentration. Concentrations of longer lived radionuclides would generally be very limited.

(4) Low level waste (LLW): Waste that contains material with radionuclide content above clearance levels, but with limited amounts of long lived radioactivity. It requires robust isolation and containment for periods of up to a few hundred years and is suitable for disposal in engineered near surface facilities. It covers a very broad range of materials that can include short lived radionuclides at higher activity levels and long lived radionuclides, but only at relatively low activity levels.

(5) Intermediate level waste (ILW): Waste which, because of its content, particularly of long lived radionuclides, requires a higher level of containment and isolation than is provided by near surface disposal. However, it needs no or only limited provision for heat dissipation during its storage and disposal. It may includes long lived radionuclides, in particular alpha emitting radionuclides, that will not decay to an activity level acceptable for near surface disposal during the time which institutional controls can be relied upon and therefore requires disposal at greater depths in the order of tens up to a few hundred meters.

(6) High level waste (HLW): Waste with radioactivity levels high enough to generate significant quantities of heat by the radioactive decay process or with large amounts of long lived activity which need to be considered in the design of a disposal facility for these waste. Disposal in deep (usually several hundred meters or more below the surface), stable geological formations is the generally recognized option for its long term management.

A conceptual illustration of the classification scheme is presented in Figure 3 [Ref. 8], the vertical axis representing the activity content of the waste and the horizontal axis representing the half lives of the radionuclides contained in the waste. Considering Figure 3 vertically, radioactivity levels range from negligible to very high concentrations of radionuclides. As the level rises, there is an increased need to contain the waste and to isolate it from the biosphere. At the lower range of the vertical axis, below the clearance levels, the management of the waste can be carried out without consideration of its radiological properties.

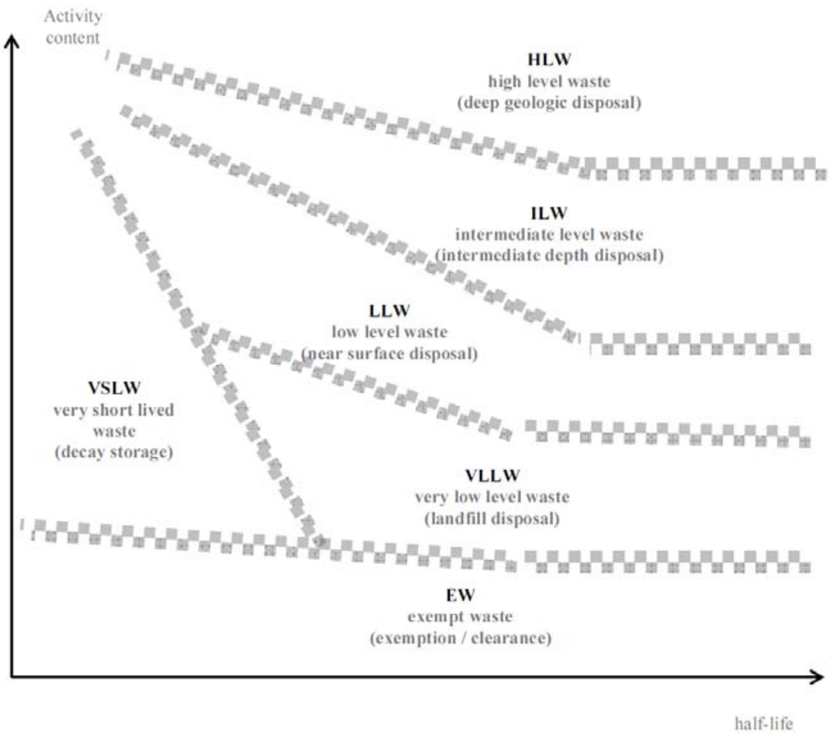


Figure 3. Conceptual Illustration of the Proposed IAEA Waste Classification System

3.1.1.3 Advanced Fuel Cycle Alternatives

This section summarizes the various fuel cycles and transition scenarios that have been considered both within the United States and Japan.

Survey and Review of Fuel Cycle Alternatives and Transition Scenario(s)

Forecasting the type of fuel cycle that would ultimately be deployed many years into the future is not possible due to the multitude of factors involved both nationally and internationally. However, the development and analysis of future fuel-cycle scenarios is necessary to inform decisions regarding the types of fuel cycles that should be pursued in research and development activities.

As an example, the Nuclear Energy Agency (NEA) considered a variety of fuel cycle scenarios to evaluate their effect on waste management policies. This evaluation is documented in a report entitled *Advanced Nuclear Fuel Cycles and Radioactive Waste Management* [Ref. 9]. Several fuel cycle schemes were evaluated:

- LWR—Once-Through
- LWR—Conventional Reprocessing
- LWR—CANDU DUPIC

- LWR—Plutonium Burning
- LWR—Plutonium and Americium Burning
- PWR—FR Heterogeneous Americium Recycling
- PWR—FR TRU Burning
- PWR-FR—Double Strata (variant including ADS)
- FR—Recycle

These fuel cycles considered a variety of reactor types (LWR, CANDU, FR), accelerator-driven systems, and fuel processing technologies (OREOX, PUREX, UREX, Electro-Chemical).

U.S. Perspective

The United States continues to explore a number of fuel-cycle scenarios in the Fuel Cycle Research and Development (FCR&D) program. The fuel-cycle strategies under consideration inlclude once-through, limited recycle where TRU elements are recycled once, transitional recycle where TRU elements are recycled repeatedly until destroyed, and sustained recycle. A variety of reactor types (LWR, VHTR (Very High Temperature Reactor), FR) and fuel processing techniques were considered (PUREX, UREX, Electro-Chemical).

An advanced fuel-cycle scenario envisioned by the U.S. DOE that could ultimately be implemented is shown in Figure 4 below. In this scenario, advanced LWRs would be deployed both domestically and possibly internationally in fuel-user countries. SNF from these reactors would be processed in a recycling center with the recovered TRU elements being fabricated into fuel for advanced burner reactors. The SNF from the advanced-burner reactors would also be processed to recover TRU elements for subsequent recycle. The U.S. DOE is also considering a transition scenario utilizing LWRs and one-pass U-Pu recycle in addition to the current "once-through" fuel cycle.

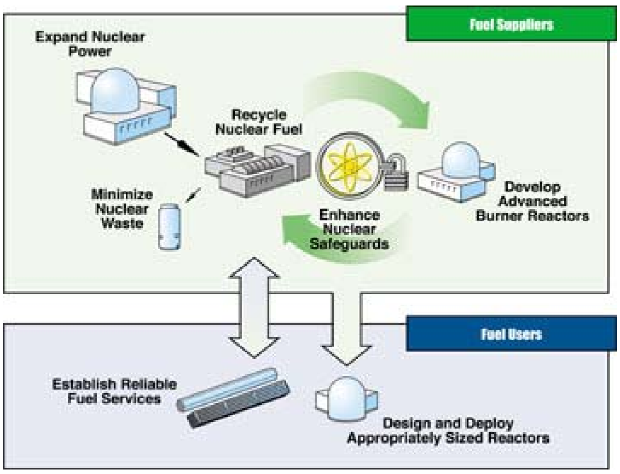


Figure 4. Advanced Fuel-Cycle Scenario Under Consideration in the U.S.

The U.S. DOE has developed scenarios for the deployment of nuclear reactors and SNF recycling facilities in the United States. These scenarios are not forecasts, but rather a set of scenarios that can be used to inform decision makers about the effects of implementing an advanced nuclear-fuel cycle. Both one-tier and two-tier recycling scenarios are considered. Specific assumptions used in these scenarios are:

- Nuclear energy resumes growth in 2015 at an annual rate of 1.75%, resulting in 200 GWe-year of electricity generated in 2060 and 400 GWe-year in 2100. Note that current generation is about 90 GWe-year.
- Separation of LWR used fuel begins in 2020, initially with a small plant (800 MTHM/year capacity) with additional plants added as needed to eliminate excess stores of used fuel by 2100.
- A small fast reactor starts up in 2022 to demonstrate the reactor and transmutation fuel technologies.
- Follow-on commercial fast reactors use a TRU conversion ratio of 0.5, metal fuel, and on-site recycling.
  - For the one-tier scenario, commercial fast reactors follow 10 years later (2032).
  - For the two-tier scenario, the MOX cycle takes 15 years (5 years in the reactor, 10 years cooling) before the used fuel is available for recycle into fast reactor fuel, so commercial fast reactors are delayed 15 years.

These scenarios were evaluated using the **V**er**i**fiable Fuel Cycle **Si**mulati**on** (VISION) system analysis tool [Ref. 10], which is discussed later in this report. Sensitivity analyses were conducted where key assumptions and parameters were varied to evaluate the effects on the fuel cycle. Metrics pertaining to the entire fuel cycle can be obtained and evaluated. Figure 5 shows one example, the electricity generated by fast reactors in a one-tier recycling scenario for various growth rates of nuclear generated electricity. Such metrics, also discussed later in this report, are important in determining the overall effects of fuel-cycle scenarios with regard to waste management and disposal.

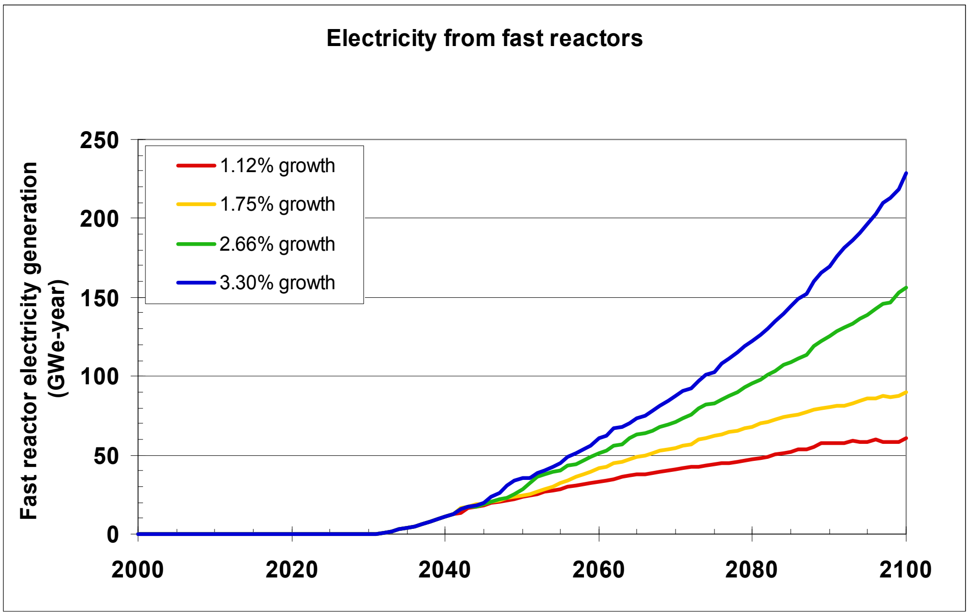


Figure 5. Fast Reactor Electricity Generation from a One-Tier Fuel Cycle Scenario

Japan Perspective

Taking into consideration the recommendations on December, 1997, by the "Round-Table Conference on Fast Breeder Reactor (FBR)" of the Atomic Energy Commission of Japan (AEC) and other discussion results, the Japan Atomic Energy Agency (JAEA; JNC (Japan Nuclear Cycle Development Institute) at that time) and electric utilities initiated the Feasibility Study (FS) in July, 1999, in collaboration with the Central Research Institute of Electric Power Industry (CRIEPI) and nuclear industries [Ref. 11]. The objective of this study is "to present both an appropriate picture of commercialization of the Fast Reactor (FR) cycle and the research and development (R&D) programs leading up to the commercialization in approximately 2015" as shown in Figure 6 [Ref. 11]. Knowledge accumulated from the experience in construction/operation of an experimental FR, JOYO, and a prototype FBR (Fast Breeder Reactor), MONJU, as well as from the design study of the demonstration FBR was effectively used in the study. A wide range of technical options have been evaluated to select several promising concepts as a candidate for the commercialization in the Phase I study from July 1999 to the end of Japanese fiscal year (JFY) 2000 (April 2000 - March 2001). The Phase II study (FS-II) was initiated in JFY 2001, aiming "to identify the most promising candidate concept for the commercialization of the FR cycle, as well as to draw up the future R&D program", and completed at the end of JFY 2005. Based on the outcome of the FS-II, it was required in the "Framework for Nuclear Energy Policy" [Ref. 5] issued by the AEC in 2005 to present "a principle for prioritizing R&D as well as R&D programs until approximately 2015, and the potential future issues."

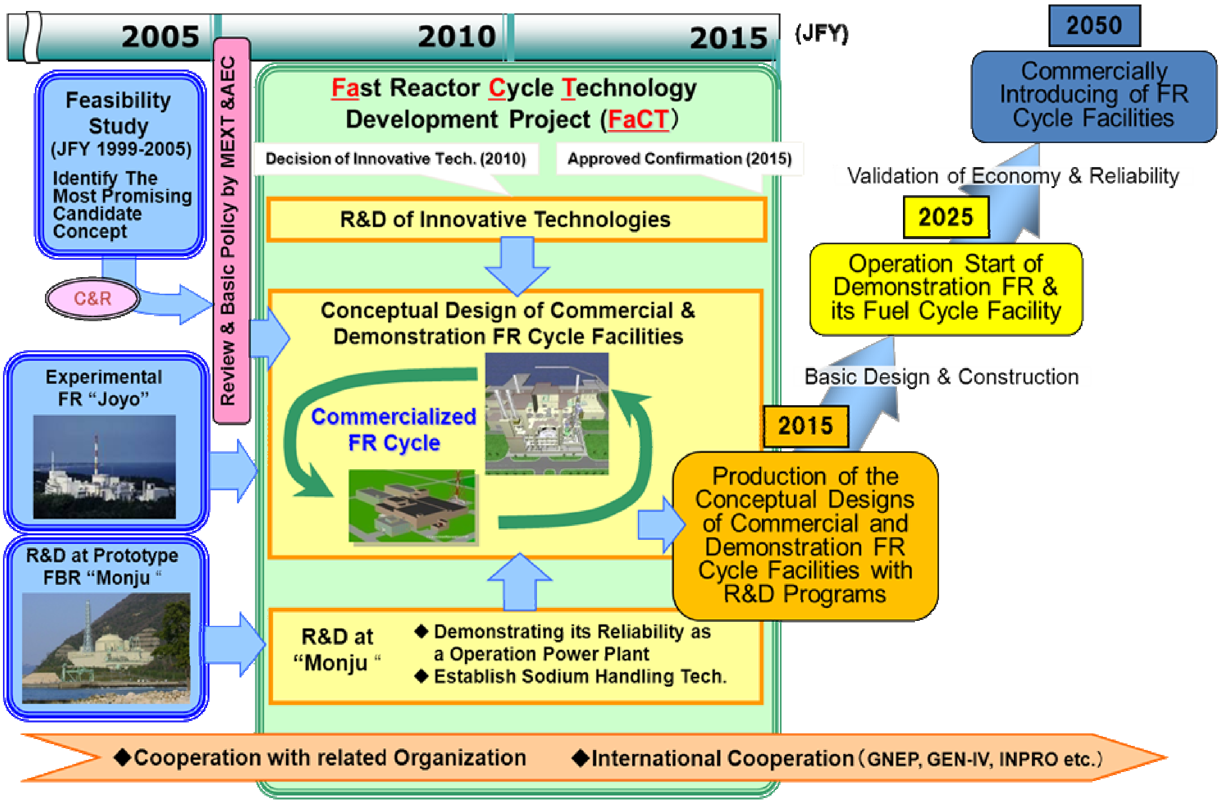


Figure 6. Plan for Fast Reactor Development in Japan

<u>FR system</u>

Promising systems have been identified based on the FS-II technical analysis of candidate concepts for the FR system. The selected concepts are shown in Figure 7 [Ref. 11]. The sodium-cooled reactor has been identified as the most advantageous among the concepts from the perspective of both potential conformity to the design requirements and technical feasibility. Furthermore, it has the potential to be adopted as an international standard concept, which may help to enhance technical feasibility.

The helium gas-cooled reactor has the potential to meet all the design requirements, and also has the potential to accommodate the various needs as a high-temperature heat source, which makes the helium gas-cooled reactor different from the other reactor types. Although it is a major challenge to evaluate conceptual applicability of the helium gas-cooled reactor, several countries, including the United States and France, has been considering its development, and therefore this seems to be an area for international cooperation.

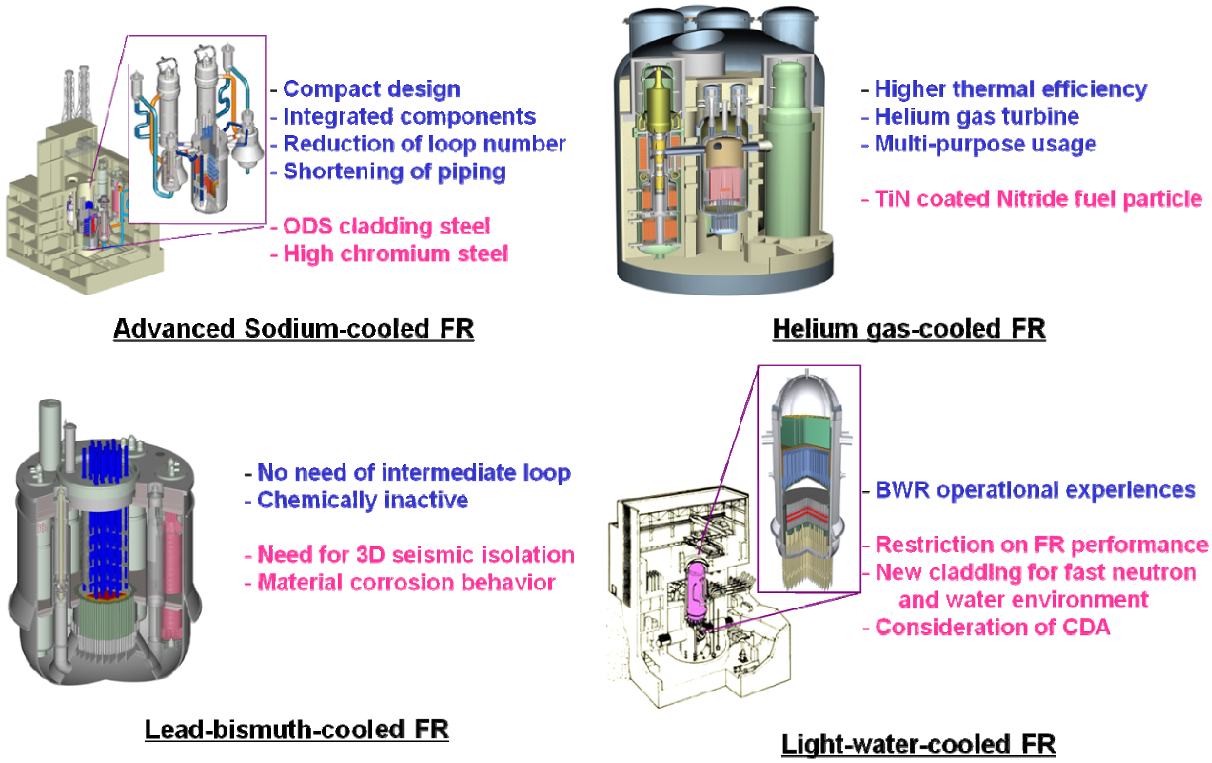


Figure 7. Fast Reactor Concepts Considered in Japan

Fuel-Cycle System

Promising fuel-cycle systems have also been identified through the FS-II technical analysis of candidate concepts. These promising concepts are shown in Figure 8 [Ref. 11].

The combined system of the advanced aqueous reprocessing and the simplified pelletizing fuel fabrication, which can be applied for either MOX or nitride fuel, has been identified as the most promising. This system has a potential to conform to the design requirements, as well as a high level of technical feasibility, because it can be developed with an extension of the existing technologies. In addition, it is expected to be developed through international cooperation.

The combination of the metal electro-refining reprocessing and the injection casting fuel fabrication applied to metallic fuel that can improve core performance has also a potential to meet design requirements and is likely to be more appropriate in particular for the small-scale cycle facility economy than the other candidate concepts. Although it would require a long-term R&D program to demonstrate technical feasibility, this concept could be a promising option as international cooperation with the United States and other countries could be expected.

Transition into the FR cycle

To facilitate smooth transition from LWRs to FRs around the year 2050 and beyond, when the introduction of FRs on a commercial basis is anticipated, a long-term mass-flow analysis concerning the mass balance of U, Pu, etc., has been performed to calculate the required reprocessing amount and the SNF stockpile from the perspective of fuel supply.

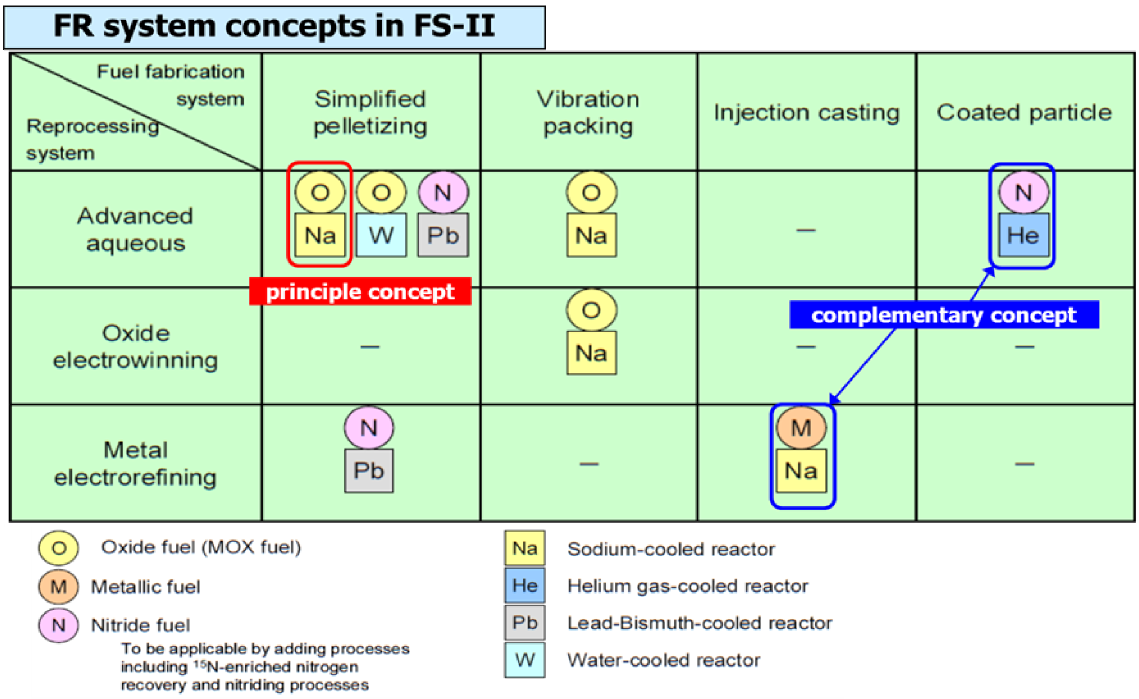


Figure 8. Fuel Cycle Systems Considered in Japan

The mass-flow analysis was performed by assuming the fixed nuclear power capacity (58 GWe) beyond 2030, and the initial deployment of an FR fleet in 2050 and the complete transition from LWRs to FRs within approximately 60 years [Ref. 12]. The achievement of the transition will require the reprocessing of not only FR fuel but also LWR fuel to supply Pu (TRU fuel) for FRs. In such an LWR fuel reprocessing, it will be effective to employ reprocessing technology for FR fuel (the advanced aqueous reprocessing system) that can streamline the LWR fuel reprocessing system. For discussion on a strategy for the transition in a reasonable manner, the applicability of FR fuel reprocessing technology to LWR fuel was investigated to identify issues for further R&D program.

As it has been indicated that it is generally possible to cope with the LWR fuel reprocessing by employing the reprocessing technologies for FR fuel, it is not necessary for the moment to reexamine the R&D program. However, since a discussion on a new reprocessing plant to follow the Rokkasho reprocessing plant is planned to begin in approximately 2010, it is considered effective further to investigate the applicability more in detail by that time.

Reference Scenario in Japan

Starting from the current LWR system with recycling of U and Pu, a transition would be made to use MOX fuel in LWRs with once-cycle U-Pu recycle. Then another transition would be made to the TRU recycling, first in LWRs and FBRs and then primarily in FBRs only. This reference scenario is shown in Figure 9 [Ref. 11]. It culminates with the scenario shown in Figure 10 [Ref. 11].

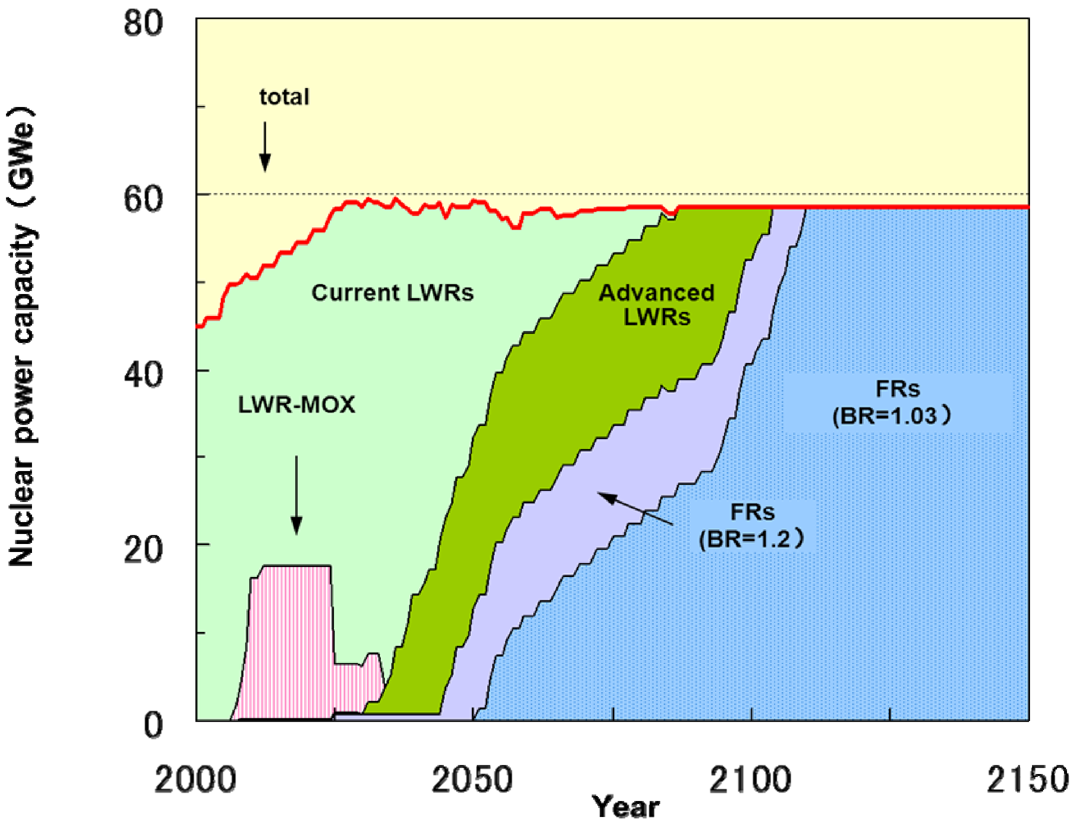


Figure 9. Reference Scenario for Japan

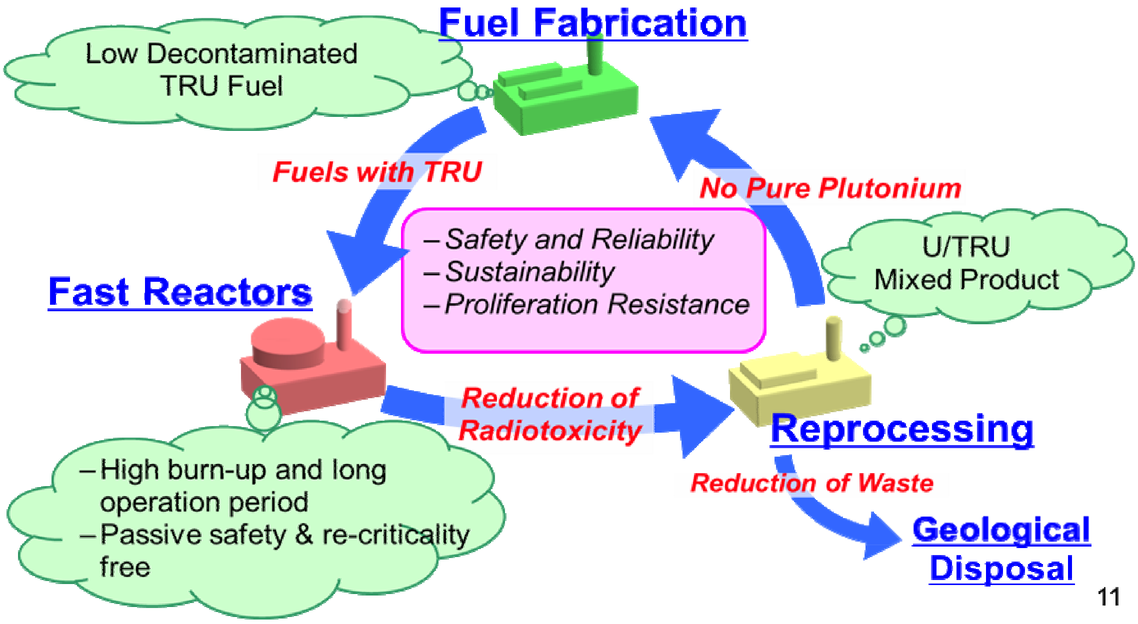


Figure 10. Ultimate Fuel Cycle Scenario for Japan

<u>Input to Waste Management Studies</u>

Vitrified waste generated from reprocessing typically contains little U and Pu (only process losses). Therefore, radio-toxicity of the vitrified package is smaller than that of SNF that is generated from the same amount of electricity production. If minor actinides are also recycled, the radio-toxicity becomes even smaller as shown in Figure 11 [Ref. 11]. Minor actinide recycling can also reduce the number of vitrified packages generated in the FR cycle because more fission products (FPs) can be contained in a vitrified package. The ultimate fuel cycle scenario will include following major processes:

- Fast Reactor —High burn-up and long operational period
- Reprocessing —Advanced aqueous process, U/TRU mixed
- Fuel Fabrication—Low-decontaminated TRU fuel

As mentioned above, JAEA completed the "Feasibility Study on Commercialized Fast Reactor Cycle Systems" [Ref. 11] and its successor project is carrying out development of advanced technologies in aqueous processes, such as solvent extraction or centrifugal separation, and also in non-aqueous processes, such as molten-salt processes. Fuel-fabrication tests using advanced packaging processes are underway.

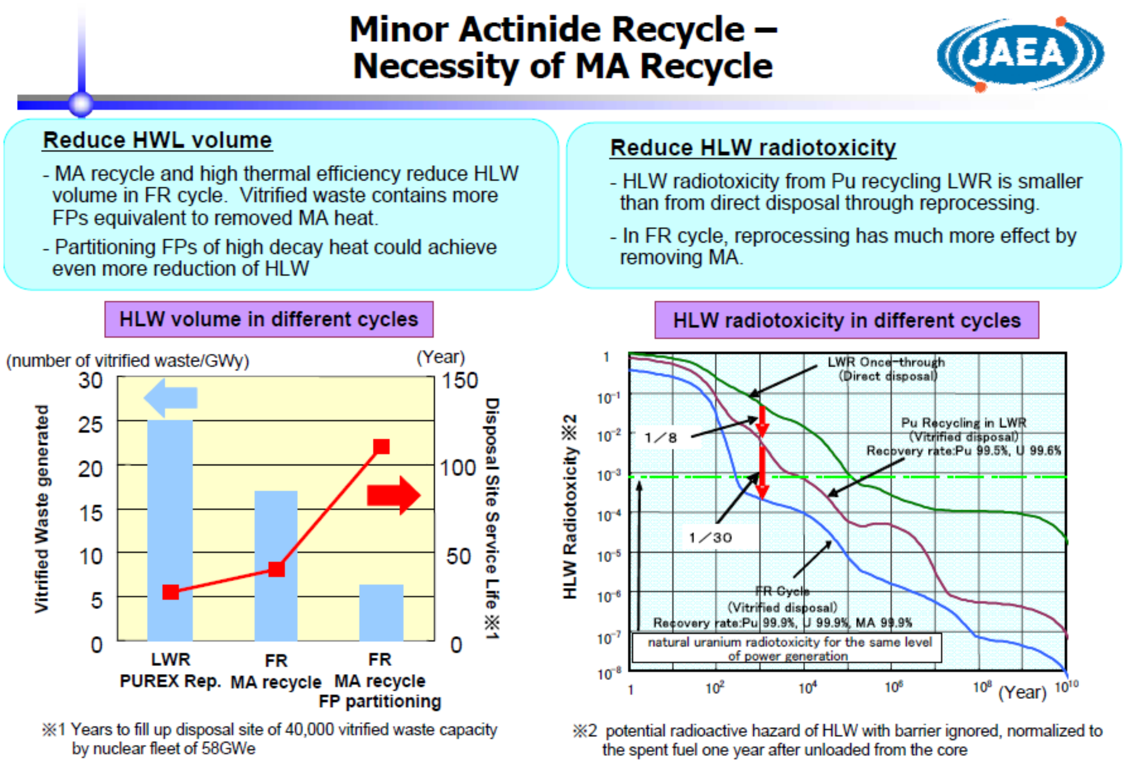
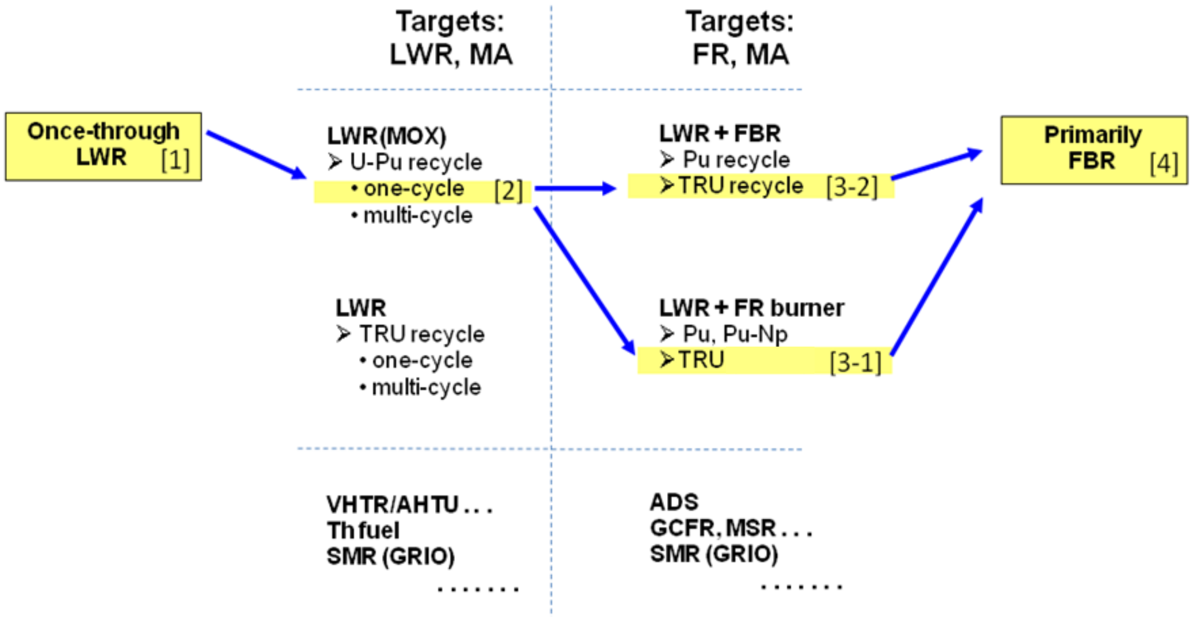


Figure 11. Benefits of Minor Actinide Recycle

3.1.1.4 Working Group Reference Scenario

The concept using a combination of aqueous, such as the Uranium Extraction (UREX) family of aqueous processes, and electro-chemical (Echem) separations and advanced fuel cycle system focused in Japan's FS make it possible to tailor waste forms to specific wastes and could extend the life of a single repository for generations. A credible strategy for managing radioactive wastes from any future nuclear fuel-cycle must provide acceptable disposition paths for all wastes regardless of reactor technology, fuel reprocessing scheme(s), and/or the degree of fuel-cycle closure. Fuel processing (separations) is to be closely coupled to material recycle/disposition and act together in closing the fuel cycle. Integrating the strategy into the fuel cycle depends on continued analyses to optimize fuel design, separations, and reuse-recycle options with consideration of treatment, storage, and disposal systems for all wastes. Thus, an integrated waste management strategy (IWMS) is critical to the success of the entire fuel cycle. The IWMS should provide guidance to optimize separations and waste treatment processes to provide a safe, secure, and cost-effective practicable system to support advanced nuclear fuel fabrication and waste/storage form(s).

The working scenario established by the WMWG, shown in Figure 12 below, is identical to advanced fuel cycle scenarios being considered both in Japan and the U.S. The working scenario involves transitioning from a once-through fuel cycle utilizing LWRs to a one-pass uranium-plutonium recycle fuel cycle also utilizing LWRs to a combination of LWRs and FRs with full minor actinide recycle, ultimately concluding with a fuel cycle primarily using fast breeder reactors with full minor actinide recycle.



NB: A reference scenario [(1) → (2) → (3-1)/(3-2) → 4] is defined in Task 1-1 taking into account a possible US strategy [(1) → (3-1) → (4)] and Japan strategy [(1) → (2) → (3-2) → (4)].

Figure 12. Waste Management Group Working Fuel Cycle Scenario

3.1.2 Waste Streams, Treatment, and Inventory (Task 1-2)

This section discusses waste streams (current and future possibilities) in Japan and the United States. Waste treatment alternatives and resultant waste forms are also discussed.

3.1.2.1 U.S. Perspective

The current waste streams generated from operation of commercial nuclear reactors in the U.S. and the potential waste streams that could arise under and advanced nuclear fuel cycle are discussed in this section.

Current Waste Streams in the U.S.

The U.S. currently utilizes a once-through nuclear fuel cycle. All used nuclear fuel is currently stored at reactor sites, except for a very small amount that was either reprocessed in the 1970s or was transferred to the U.S. Department of Energy for research, project demonstration, or other purposes. Through 2008, the U.S. had accumulated over 60,000 MTHM of spent nuclear fuel from commercial nuclear power plants [Ref. 13]. As shown in Figure 13, projected discharge rates indicate that this amount could more than double by 2035 if the once through nuclear fuel cycle is continued and all currently operating plants continue operation beyond their initial 40-year license period (20 additional years).

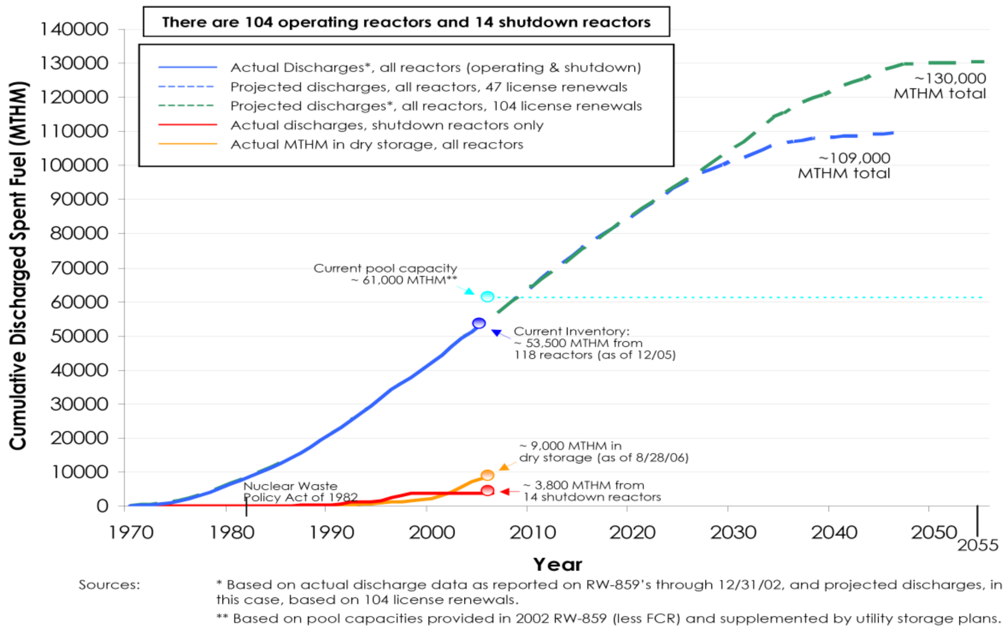


Figure 13. SNF Discharge Projections from the Current Generation of U.S. Reactors

By 2035, the U.S. will have accumulated about 2,400 MTHM of spent nuclear fuel from reactors that produced materials for the nation's nuclear weapons program, from research reactors, from reactors on nuclear-powered naval vessels, and from reactor prototypes [Ref. 14].

Large volumes of high-level radioactive waste were created in the past when spent nuclear fuel was reprocessed to separate uranium or plutonium isotopes that could be reused from the other elements in the fuel. The high-level radioactive waste left over from this process exists in both liquid and solid form; liquid wastes are stored in underground tanks at DOE sites near Hanford, Washington; Savannah River, South Carolina; and Idaho Falls, Idaho. It is estimated that 11,000 MTHM worth of HLW will have to be disposed (assuming 0.5 MTHM equivalent per glass canister) [Ref. 14].

Spent nuclear fuel and high-level radioactive waste are presently stored in 39 states, as shown in Figure 14 [Ref. 15]. These storage sites are located in a mixture of urban, suburban, and rural environments.

Through 2010, the U.S. was developing a deep geologic repository at the Yucca Mountain, Nevada site in accordance with the Nuclear Waste Policy Act (NWPA, discussed in Section 3.1.1.2). The U.S. DOE submitted a license application for construction of the repository to the U.S. NRC in June of 2008 [Ref. 16]. In accordance with the NWPA, this license application considers the disposal of 63,000 MTHM of commercial spent nuclear fuel and 7,000 MTHM of U.S. DOE owned spent nuclear fuel and high level nuclear waste. In addition, the U.S. DOE submitted to the U.S. Congress, a report on the need for a second repository in December of 2008 [Ref. 17].

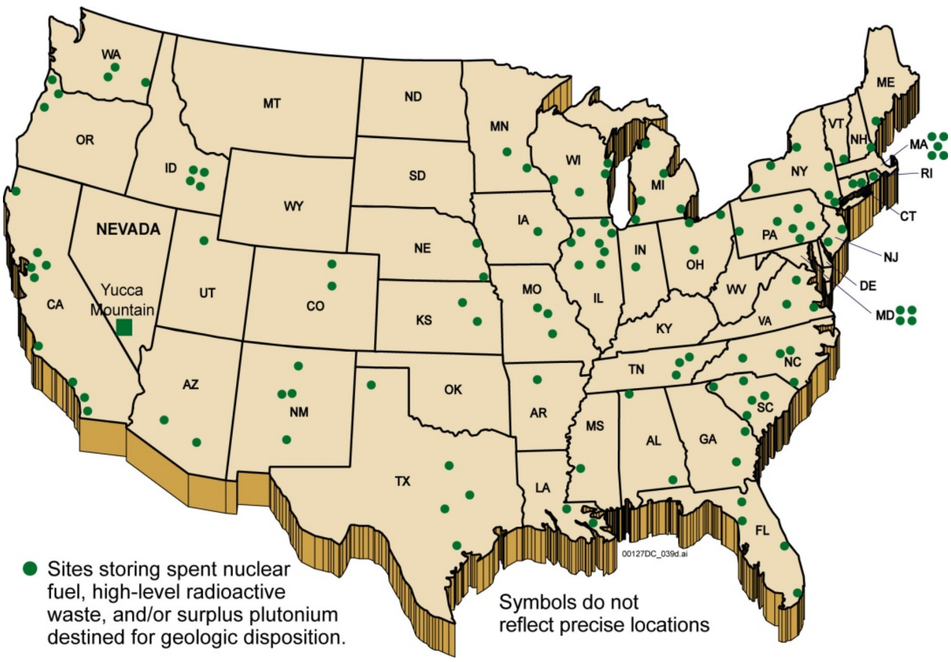


Figure 14. Locations of Spent Nuclear Fuel and High Level Nuclear Waste in the U.S.

The U.S. President has made clear his intent that Yucca Mountain is not an option for waste storage. On March 3, 2010, the Department of Energy filed a motion with the Nuclear Regulatory Commission to withdraw the license application for a high-level nuclear waste repository at Yucca Mountain with prejudice. The President's fiscal year 2011 budget request eliminates funding for the Office of Civilian Radioactive Waste Management. The DOE has formed the Blue Ribbon Commission on America's Nuclear Future that will conduct a comprehensive review of policies for managing the back end of the nuclear fuel cycle, and will provide recommendations for developing a safe, long-term solution to managing the Nation's used nuclear fuel and nuclear waste.

Since 1998, the U.S. has disposed transuranic wastes in the Waste Isolation Pilot Plant (WIPP) located in New Mexico. The waste disposed at the WIPP is limited by law to defense-generated transuranic wastes. Disposal of commercial, low level, and high level wastes at the WIPP is prohibited by the WIPP Land Withdrawal act. The transuranic wastes consist of clothing, tools, rags, residues, debris, and other items contaminated with transuranic radionuclides, primarily plutonium.

Potential Reprocessing Wastes Under an Advanced Nuclear Fuel Cycle

Using 50 GWd fuel aged 20 years as a basis and one potential aqueous separations processing scheme (know as UREX+1a), a summary of the expected amounts of each type of waste per MTHM is shown in Table 1.

Table 2 lists similar waste statistics for processing metal FR 107 GWd fuel aged 5 years using an electrochemical separations scheme. Both tables list the amount of wastes expected from processing one MTHM, but note that the fast reactor fuel has generated 107/51~2× the energy. Thus, when comparing the waste production of the two processes, the volumes from Echem should be halved to compare on an equal-energy basis. The mass of the elements in the waste divided by the percent waste loading gives the mass of waste expected. The mass of waste divided by density gives the unpackaged waste volume. Waste loading and density ranges were provided by leading authorities in the DOE complex with decades of experience developing waste forms. Minimum waste loadings are supported by data, and maximum waste loadings are projections of what is believed possible based on waste-form chemistry. Much more detail on the technical basis for these waste forms is provided in the following discussion.

Note that two entries are shown in Table 1 for the Ln and FP streams. Taken together, as has been done historically with PUREX, these streams are vitrified in HLW glass. However, when separated, the Ln stream can be loaded to a greater percentage of the glass, and the FP stream can be added to the Tc metallic waste form. In this manner, the glass volume is reduced, and essentially no volume is added to the metal waste form because the FP stream contains iron needed to make the Tc waste form anyway. This concept requires two waste forms and two processing systems, but the Tc is more stable as a metal and is difficult to keep in glass at melting temperatures. The overall economic value of this concept must still be evaluated. Note also in Table 1 that each waste stream has two possible disposal pathways: HLW as dictated by the current policy, or GTCC or decay (temporary storage) pathways based on risk, which are described in the waste-disposition schematic below. Table 2 also shows two options. If the lanthanides can be separated effectively with Echem, they could be stabilized in a high-waste-loading glass similar to the concept in Table 1. If not, the Ln and Cs/Sr stream are combined in a bonded sodalite in a much lower waste-loading glass.

The waste forms and volumes reported in these tables serve only as an example and the U.S. is currently investigating the use of alternative waste forms with improved performance and/or cost. The results of these studies will be considered and used during Phase II of the WMWG.

As can be seen in Table 1 and Table 2, the fuel cycle concept is based on advanced separations that make it possible to partition SNF into several fractions rather than composite HLW, thereby partitioning the radionuclides into groups of common chemistry and to a greater extent, risk. Advanced separations allow greater flexibility in managing the individual waste streams based on duration, type, and magnitude of risk and to develop specialized waste forms to sequester radionuclides effectively per the IWMS. Recovering actinides is a central goal of an advanced fuel cycle to reduce the potential for proliferation, benefit from the fuel value of plutonium, and reduce the long-term radiotoxicity and heat in a geologic repository. Next, partitioning readily oxidized alkali and alkaline earth elements that are relatively short-lived but generate substantial heat (Cs/Sr) allows a waste form to be created that can be managed to dissipate heat, thereby mitigating heat limits on repository capacity. Similarly, very short-lived gaseous radionuclides, including tritium and krypton, can be captured and allowed to decay in storage before eventual disposition. Segregating the lanthanides allows a high-waste loading lanthanide-based glass to be produced, thereby mitigating volume limits on a repository.

Table 1. Estimated Volumes and Disposition of Recommended Waste Forms for UREX+ Process Wastes

	Fuel	Waste Form	Density MT/m^3		Waste Loading wt%		Waste Form Mass MT		Waste Volume m^3		Disposal
Stream	**g/MTIHM**	**Type**	**Lo**	**Hi**	**Lo**	**Hi**	**Lo**	**Hi**	**Lo**	**Hi**	**Options**
Hydrogen	5.90E+00	Grouted HTO	2.1	2.5	25	30	1.8E-04	2.1E-04	7.1E-05	1.0E-04	Decay/LLW
Iodine	3.92E+02	Grouted Zeolite	2.1	2.5	2	6.9	5.7E-03	2.0E-02	2.3E-03	9.3E-03	HLW/GTCC
Krypton	5.31E+02	Gas 1–50 atm	0.004	0.185	9	100	5.3E-04	5.7E-03	2.9E-03	1.5E+00	Decay/LLW
Carbon	1.31E+03	Grouted $CaCO_3$	2.1	2.5	6	10	4.8E-02	8.0E-02	1.9E-02	3.8E-02	HLW/GTCC
If Cs/Sr are separated, and the balance of transition metal FP can be reduced to metals, the following three waste forms are made possible. The FP stream includes iron added during separations, and this iron replaces the iron needed to make the Tc metal alloy; thus there is essentially no increase in volume. The lanthanides can be stabilized in high-waste-loading glass.											
Cs/Sr	7.99E+03	Glass/Ceramic	1.5	4.0	20	50	1.8E-02	4.5E-02	4.5E-03	3.0E-02	Decay/HLW
Tc/UDS/FP*	4.33E+04	Metal Alloy	7.6	8.2	40	85	5.1E-02	1.1E-01	6.2E-03	1.4E-02	HLW/GTCC
Ln	1.59E+04	LaBSG	3.0	4.0	30	60	3.1E-02	6.2E-02	7.7E-03	2.1E-02	HLW/GTCC
If the metallic FPs are not reduced to metal, they will be immobilized with the lanthanides in glass, thereby increasing the glass volume 6-8×. The Tc metal alloy will require the addition of iron, possibly from the activated hardware stream. The hardware stream would be reduced accordingly.											
Ln/FP*	6.20E+04	Glass	2.5	3.2	20	30	2.1E-01	3.1E-01	6.4E-02	1.2E-01	HLW/GTCC
Hulls	2.51E+05	Metal	4.6	6.6	93	100	2.5E-01	2.7E-01	3.8E-02	5.9E-02	HLW/GTCC
Hardware	5.65E+04	Metal	4.6	6.6	93	100	5.7E-02	6.1E-02	8.6E-03	1.3E-02	GTCC

Notes:

Calculations are based on 51 GWd burnup of UO_X LWR fuel, aged 20 years out of reactor and reprocessed using the UREX+1a flowsheet (UREX+1a is one of the UREX separation processes).

Hydrogen (HTO) stream includes estimate of humidity in dried voloxidation sweep gas, tritium disposed as cemented tritiated water without sorbent.
Iodine stream also captures bromine, halogens captured on silver zeolite and grouted on sorbent.
Krypton waste loading reflects low end contaminated with Xe, high-end captured pure, stored for decay as compressed gas.
Carbon stream also captures natural CO_2 from dissolver aeration, disposed of as grouted carbonate.
Cs/Sr stream includes Ba and Rb stabilized as ceramic or glass for decay storage.

Technetium reduced to metal and alloyed with UDS/transition metal FP/portion of Zr cladding/portion of SS hardware.
Lanthanides vitrified as borosilicate glass.
*Ln+FP stream includes iron added as TRUEX reagent. Assume sulfur can be volatilized during vitrification and captured in offgas.

UDS combined in Tc alloy waste form.
Hulls and hardware based on PWR fuel, compacted as low density or melted to yield high density, 93% reflects metals added to lower melting point if metals are melted.
Tc/UDS/FP waste loading could approach 85% because this waste is dominated by Fe added in separations.

Table 2. Estimated Volumes and Disposition of Recommended Waste Forms for Echem Process Wastes

All mass in MT/MTHM	Mass from Fuel	Mass from Separations	Waste Form	Density MT/m^3	Waste Loading mass%	Waste Form Mass MT	Waste Volume m^3	Disposal
Stream								
Metals[1]	8.79E-01	1.10E-01	Metal Alloy	7.75	100%	9.88E-01	1.27E-01	HLW/GTCC
Cs/Sr[2]	1.88E-02	4.54E-01	Glass-bonded Sodalite	2.4	33%	1.43E+00	5.97E-01	HLW/GTCC
Ln	2.47E-02	0.00E+00	Glass	4	50%	4.95E-02	1.24E-02	HLW/GTCC
Cs/Sr and Ln forms can be considered separately as above or as one form as shown below, but not both.								
Ln/Cs/Sr[3]	4.36E-02	4.72E-01	Glass-bonded Sodalite	2.4	33%	1.56E+00	6.51E-01	HLW/GTCC
SS Hardware[4]	1.75E+00	0.00E+00	Compacted Metal	5.5	100%	1.75E+00	3.18E-01	GTCC

Notes:

Waste/MTHM cannot be compared directly to aqueous wastes, FR fuel used was for 107 GWD/MTHM or roughly 2× the electrical generation as the 51 GWD/MTHM fuel used in Table 1.

Waste loading calculated as mass% of radionuclides and separation process chemicals in final waste form.

[1]Metal waste form also incorporates SS Echem processing basket.

[2]Cs/Sr waste form contains iodine and is likely >100nCi/g TRU, thus not potentially LLW after decay.

[3]Lanthanides combined with Cs/Sr in glass bonded sodalite if they can not be separated and put into glass.

[4]The SS hardware stream is activated metal from the fuel element from above/below the reactor core.

Currently, the baseline waste form for HLW worldwide is borosilicate glass containing waste elements as oxides. Unfortunately, several fission-product elements have limited solubility in glass (Ru, Rh, Pd, I, Tc, etc), which results in low waste loading and production of more glass. With the other elements removed, the remaining metallic fission products that have historically limited HLW loading in glass can potentially be alloyed with cladding and undissolved solids to minimize volume.

Matching the waste form to the individual waste-stream chemistry allows the disposal system to achieve more optimum waste loading with less heat and radiotoxicity, achieving comparable or improved performance. Not only can the waste form be matched to the waste, but the disposal environment can be matched. Some elements are more stable in a low-oxygen reducing environment, while others are more stable as oxides. Thus, a more-efficient waste-management system that uses the most effective waste form and disposal design for each waste is made possible by this proposed change in technology. The following sections discuss the options evaluated for the primary radionuclides separated in advanced separations.

Reducible Element Waste Streams

For the aqueous process, the reducible element waste streams are the undissolved solids (UDS) from the fuel dissolution step, dissolved Tc recovered from the UREX product, and transition metal FP. Depending on the conditions that are used, it is expected that at least 75% of the Tc in the fuel will be dissolved in the initial dissolution, and it is assumed that all of the dissolved Tc will be recovered from the UREX product solution. The balance of the Tc will be in the UDS. The dissolved fractions of the other transition-metal fission products (FP), including Zr, Mo, Ru, Rh, and Pd, will be recovered in the TRUEX raffinate. For waste-management efficiency, this FP waste stream could potentially be blended into the Tc/UDS-bearing waste form because these transition metals are generally the same elements that make up the UDS.

For the Echem process, the reducible metals (anything more noble than U in chloride salts), including Tc and the transition metals described above, are captured together as the metallic wastes that are retained in the anode basket of the electrorefiner.

In addition to metals from the fuel itself, some waste streams will include metals used in processing. One method being studied for recovering pertechnetate from the UREX product solution is to evaporate the solution and reduce pertechnetate to the metal by steam reforming. The metallic Tc would then be alloyed with Zr or other metals. Another method being studied is the reductive deposition onto iron or other metal substrate (including electro-deposition). This would eliminate the need to evaporate the solution and conduct steam reforming, but would add Fe or another metal to the waste form. Ferrous sulfamate may be added to reduce Np(V) to Np(IV) in the solution before TRUEX separation. If so, the impacts of iron and sulfur on waste processing must be considered. This would make iron the dominant metal in the FP-bearing waste solution and in the blended Tc, UDS, and FP streams. Using reductants other than ferrous sulfamate to eliminate Fe and S from the waste stream is being studied.

In the Echem process, the anode basket used in the electrorefiner is expected to be included with the waste and would dominate the waste-stream composition. The basket is constructed primarily of Type 316 SS and will contribute a significant amount of Fe to the waste stream as well as Cr and Ni. It is anticipated that the cladding hulls from the oxide fuels that are treated with either the UREX+ or other processes will be cleaned and disposed of separately from the Tc-bearing wastes. The cladding hulls may form metallic fuels that are treated with the Echem process and will be retained with the Tc-bearing metallic wastes in the anode basket and alloyed with them in a metallic waste form.

Immobilizing only the aqueous waste streams in a multi-phase metal-alloy waste form or a borosilicate glass was evaluated. Echem UDS wastes include massive amounts of metal, and vitrification is not practical. Vitrification was considered because these same elements are being immobilized in glass internationally, albeit at very low concentrations. However, vitrification of the reducible metal wastes is probably not practical because of:

1. Restrictive limitations on the amounts of metal that can be tolerated in a melter
2. Additional process steps required to oxidize this stream
3. Volatilization of Tc at vitrification temperatures
4. Low solubility of noble metals in glass.

Consolidation of the UDS, recovered soluble Tc, and noble metal fission products within a single alloy waste form is expected to approach the maximum achievable waste loading and minimize the volume of waste, although this remains to be demonstrated. Applying the materials and methodology developed for metallic wastes from Echem treatment of spent sodium-bonded fuel was considered for the Tc-bearing waste streams from both the UREX+ and Echem processes. Insights drawn from the development of those materials and from binary phase diagrams were used to estimate the capacities of various mixtures to accommodate the waste streams. Producing separate waste forms for individual streams could increase the individual waste loadings, but would increase the total volume of waste.

Cesium/Strontium Waste

There are three broad classes of waste forms that were considered for stabilizing this waste stream: ceramics (including synthetic minerals), glasses, and cements. These three classes were selected because the process used to make the waste forms included in each class is essentially the same and potentially applicable to remote processing of this highly radioactive waste stream. The first process for all three classes is feed evaporation. For ceramics, the processes involve 1) mixing raw materials: clay or individual element compounds, 2) firing or heat-treating these materials together with the waste to affect the desired phase assemblage, 3) optional consolidation with a press before or after heat treatment, and (4) packaging. For glass the process involves 1) mixing the feed with additives, 2) feeding the slurry to a melter where it is converted to a molten glass, and 3) casting the molten glass into the disposal package where it solidifies into a glass. For cement, the evaporated feed is mixed with cement-forming materials and solidified into a final waste form. Within each class, there is some variation on the overall process for specific materials, but, for the most part, these overall process steps are applicable. For example, in the fluidized bed steam reformer (FBSR) process, Steps 1 and 2 are combined in a steam environment. Further consolidation of the sintered material may be needed.

Metal matrix waste forms are also discussed. However, this option is better considered as a canister and storage strategy than a waste-form option, per se, because the waste itself is still fixed first in a glass, a ceramic, or even a simple oxide. These stabilized materials could then be dispersed in a metal matrix to enhance the overall conduction of heat and provide some radiation shielding. This option allows the waste loading in the primary waste form to be much higher than would be possible for the material alone while reducing the centerline temperature of the waste package or allowing a much–larger-diameter waste form to be stored.

For all of these materials, transmutation and decay heat are the major technical material challenges to resolve. For example, ^{137}Cs decays to ^{137}Ba and ^{90}Sr decays to ^{90}Y, which then decays to ^{90}Zr. With these decays come changes in charge on the valence and ionic radius. The effects of these changes on all the materials considered are largely unknown, although glass is expected to be less sensitive to the change in ionic radius than crystalline materials. It was assumed that the change in charge can be compensated for by including the crystalline or vitreous matrix elements that have one or more than one available oxidation state (e.g., Fe(III) can go to Fe(II)) with the consumption or release of oxygen. The capacity for charge transfer is found in naturally occurring minerals such as garnet, where Fe(III) and Fe(II) on different crystal sites can exchange charge. It is also assumed that changes in ionic radius can be accommodated, but this has not been demonstrated. From a processing point of view, the presence of high concentrations of ^{137}Cs and ^{90}Sr in the waste stream means that careful processing may be needed to handle the high β-γ dose and associated decay heat.

A composite ceramic waste form (CWF) was specifically developed to immobilize radionuclides in chloride salt wastes from the Echem treatment of spent sodium-bonded fuel. During processing, the salt reacts with zeolite 4A to sequester chloride in a sodalite phase, and the sodalite is encapsulated in a borosilicate glass to produce a multiphase glass-bonded sodalite waste form. Most of the radionuclides either dissolve in the glass or form phases that become included in the glass. The principle role of the sodalite phase is to contain the chloride. Thus, the waste loading that can be achieved depends on the concentration in the salt and will probably be limited by the amount of zeolite that is needed to sequester the chloride and the amount of binder glass needed to encapsulate the sodalite. Due to the chloride intrinsic to this waste stream, no other options were initially considered. However, ongoing research is considering alternative high halide waste forms.

Lanthanides and Balance of Fission Products

In both aqueous and electrochemical processing, separation of the lanthanides (Ln) is being developed. In UREX+1a, the non-Ln FP, including the transition metal isotopes of Zr, Nb, Mo, Ru, Rh, and Pd, are segregated, but could be combined with the lanthanides as one waste stream.

The Echem process flowsheet may leave the rare-earth element chlorides in waste salt. The CWF would then be used for the salt waste stream as described above for Cs/Sr, and no options were evaluated due to the chloride content intrinsic to this waste. However, if the lanthanides are recovered electrochemically, the salts can be distilled away, and the lanthanides could be stabilized using any of the waste forms described in this section. In Echem transition metal, FPs are incorporated into a metal waste form as described above; no options were considered here because of the high waste loading and density of this demonstrated waste form.

The candidate waste forms for the Ln/FP streams can be grouped into four primary classes: glasses (e.g., borosilicate and phosphate), mineralized forms (e.g., aluminosilicate or phosphate minerals), ceramics (e.g., Synroc-type), and composites (e.g., glass bonded zeolite). Metallic waste forms are also included as a potential means to treat fractions of the fission product streams.

Vitrifying the Ln and FP streams would have several advantages, including relatively high loadings, a proven technology, and similarity in form to waste forms currently accepted for repository disposal. Waste loadings of 30 to 60 wt% for lanthanides have been demonstrated in the laboratory, and 20 to 30 wt% combined Ln/FP, limited by noble metal solubility, is estimated. Glasses have been developed and tested, and fabrication processes have been demonstrated for a wide variety of applications with similar compositions and disposal requirements to the Ln/FP wastes. A second glass option considered was an iron-phosphate glass (IPG). The IPG is an attractive option due to the relatively high solubility of salt components, such as molybdate and sulfate in the phosphate liquid. While most phosphate-based glasses are relatively low in chemical durability and are not suitable for nuclear waste forms, two phosphate-based glass families have shown superior durability: the alkali-alumino-phosphate family and the alkali-iron-phosphate families. In the vitrification process, the feeds will be prepared for vitrification by mixing the feeds with glass frit or appropriate glass-forming chemicals, melted in a high temperature melter, cast into a container, and stored.

The primary melter technologies considered include the Joule Heated Melter (JHM) and Cold Crucible Induction Melter (CCIM). Joule-heated ceramic melters are currently used in radioactive operations to treat HLW in the United States. CCIMs are currently receiving increased interest because of their ability to process at higher temperatures, minimize melter corrosion by use of a skull layer to contain the melt, and allow processing of crystalline inclusions in the melt.

In a mineralized waste form, varying solid mineral phases can be produced depending on the type of co-reactant fed with the waste (e.g., a high sodium waste with an aluminosilicate clay co-reactant produces a sodium aluminosilicate [NAS] mineral waste form). For the Ln/FP wastes, a number of waste-specific mineralized forms were considered based on the major constituents of the wastes. An FBSR was considered as the primary process to produce mineralized-type waste forms, though this is only one of several potential reactors (i.e., a rotary calciner). Waste fed to a steam reformer can be either liquid or slurry. The resulting mineralized product is collected from the reaction vessel. The particulate waste form would probably have to be consolidated in some form before disposal.

Ceramic-based (or more appropriately, crystalline) waste forms retain the radionuclides in the waste as part of the matrix. The primary ceramic-based materials evaluated were those from the Synroc and monazite families of compositions. In general, Synroc (i.e., synthetic rock) is an advanced synthetic crystalline ceramic composed of geochemically stable titanate-based minerals, which have immobilized uranium, thorium, and other natural radioactive isotopes in the environment, for millions of years. These minerals and their man-made analogs are capable of incorporating into their crystal structures nearly all of the transition metal FPs and the Ln/FPs. Monazite is also attractive as a waste form based on the fact that significant amounts of the naturally occurring actinides thorium and uranium are contained in natural monazites. Ceramic materials can be formed using a wide range of processes, including cold pressing and sintering, HUP, HIP, and CCIM. Of these various processing routes, HIP is the most developed for production of Synroc-type materials. Throughput limitations have hampered the utility of ceramic processes for large-scale waste-treatment operations. Using CCIM technology to produce ceramic forms may provide a solution to this limitation.

Volatile Radionuclides

For the purpose of this study, it was assumed that the primary volatile radionuclides from a large reprocessing facility could not be vented or discharged to water bodies. Once captured, ^{129}I and ^{14}C must be sequestered essentially indefinitely, but ^{3}H and ^{85}Kr can be effectively managed in decay storage because of their relatively short half-lives (12.28 years, 10.73 years, respectively). Proposed capture methods should be evaluated using parameters such as selectivity, efficiency, regeneration of sorbent, and conversion to final waste forms. Silver-zeolite (AgZ) for iodine, molecular sieve for tritium, caustic scrub for ^{14}C, and zeolite (mordenite, faujasite) for Xe/Kr are the baseline technologies selected in the conceptual design of an offgas control system.

When loaded with tritium-bearing water, the molecular sieves may be added to grout for final disposal or regenerated. A reasonable disposal path includes adding either the loaded molecular sieves or the recovered water to grout, placing the grout in a stainless steel drum, sealing the drum, and disposing of the drum by burial. The tritium will decay to safe levels before the packaging will deteriorate because of its relatively short half-life (12.26 years). Grouting is a well-developed technology for stabilizing a variety of waste forms. No problems are anticipated for this method of disposing of a purified tritium-bearing water stream. However, because the molecular sieves may also co-sequester a small amount of ^{129}I and carbon dioxide (^{14}C), the method should be evaluated for its effect on a grout waste form, and options should be identified and/or developed to cleanly separate the water from these other species.

A number of possible immobilization forms have been suggested for iodine. The dissolution of NaI or KI in standard fluoride glasses is limited to 1 mol%. Quaternary glasses permit compositions of 4 and 8% iodide, respectively, but the durability of these glasses is quite low. One possible stabilization package is based on the incorporation of the iodine-loaded AgZ into a grout matrix. Assuming that the iodine “filter” is designed as the storage package, then this would have end caps welded in place, and the sealed filter package would be placed into a secondary overpack. Grout could be added to the annular space, but this would add little to the overall containment of the ^{129}I. A second approach is to remove the loaded AgZ pellets from the filter housing and mix the loaded pellets with grout for stabilization and containment during transport. A disposition pathway for this waste form considering the 1.6×10^7 year half-life of ^{129}I in something other than a vitrified HLW form is yet to be resolved. A number of studies have also shown that iodine-loaded silver zeolites can be converted to the aluminosilicate mineral sodalite in which the iodine is more strongly bound than in the unprocessed zeolite sorbent. Iodide sodalite that does not contain silver has also been successfully synthesized, and leaching tests indicate that it may be suitably durable as a long-term waste form.

Options evaluated for Kr include storage in pressure containers and encapsulation in solid matrices. Pressurized gas containers must remain intact for ~100 years, resist corrosion because of the in-growth of elemental Rb that is chemically aggressive, and dissipate the decay heat. This method provides for easy recovery of the krypton for subsequent industrial use, but it also increases the hazard of a release. Encapsulation as a sputtered metal matrix will contain 5 to 6% Kr on an atomic level. The product is an amorphous glassy deposit. Depending on the process used, 16 to 25 liters could be loaded at standard temperature and pressure (STP) per kg metal matrix. An alternative is encapsulation in a zeolite matrix. The krypton is encapsulated in the zeolite structure by a sintering process where the pores of the zeolite are sealed. The relatively low thermal conductivity of the zeolite should be considered and may limit the maximum loading of the zeolite.

Most of the carbon immobilization studies conducted to date have considered calcium or barium carbonate that has been mixed with cement and packaged in steel drums.

Uranium and transuranics, outside of process losses, are currently considered to be recycled products. Trade studies are underway to evaluate the reuse of other materials from SNF reprocessing.

3.1.2.2 Japan Perspective

This section discusses the current classification and disposal pathways for Japanese HLW and the potential waste streams from a fast reactor scenario.

Classification and Disposal of Wastes

Radioactive waste in Japan is classified into two main categories, HLW and LLW, according to its level of activity (see also 3.1.1.2.2). Depending upon its origin, the LLW is further sub-classified into waste from power reactors, waste containing TRU radionuclides, uranium waste, and radioactive waste from medical, industrial, and research facilities, as shown in **Table 3** [Ref. 18] with disposal concepts shown in Figure 15 [Ref. 18].

Table 3. Waste Classification in Japan

Classification			Source	Disposal concept
HLW			Reprocessing plant	Geological disposal
LLW	TRU waste		Reprocessing plant, MOX fuel fabrication plant	Geological/Intermediate depth/ Near surface disposal
	Waste from NPP	Relatively higher radioactive waste	Reactor decommissioning	Intermediate depth disposal (Dispose at 50～100m from the surface)
		Relatively lower radioactive waste	Operation and decommissioning of nuclear facilities	Near surface disposal with artificial barrier
		Very low level waste	Operation and decommissioning of nuclear facilities	Near surface disposal without artificial barrier
	Uranium waste		Uranium enrichment and fuel fabrication plant	Intermediate depth/Near surface disposal (+Geological disposal)
	Waste from research facilities and radioisotope users		Research facilities, hospitals and RI licensees	Intermediate depth/Near surface disposal (+Geological disposal)

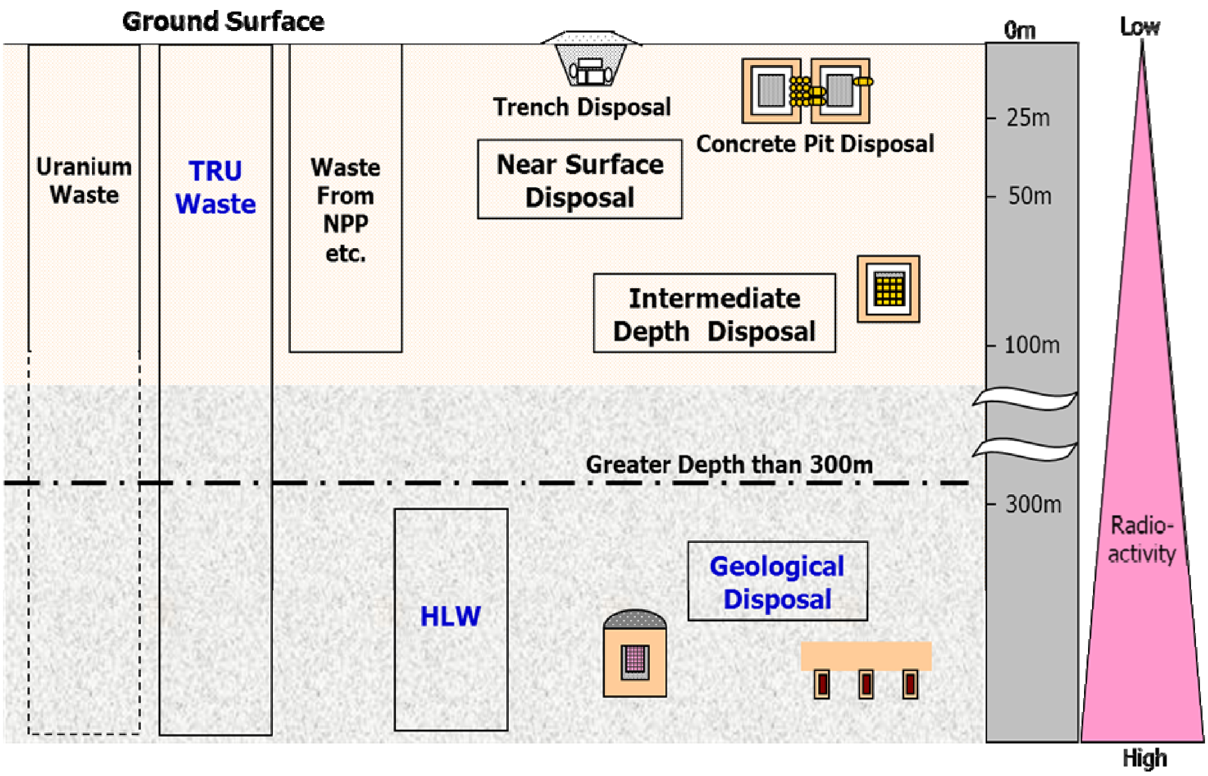


Figure 15. Disposal of Japanese Radioactive Waste

High-Level Radioactive Waste

HLW includes the highly active liquids that arise from the reprocessing of spent nuclear fuels and the solid glass waste form produced by the vitrification of these liquids. It contains substantial quantities of both fission products and actinides.

Low-Level Radioactive Waste

TRU waste, arising from the operation and decommissioning of SNF reprocessing and mixed oxide(MOX) fuel-fabrication facilities that contains TRU elements (e.g., neptunium, plutonium, americium, etc.) and other radioactive elements (e.g., iodine, carbon etc.). It should be noted that some portion of the TRU waste is disposed of in geological repository (see below).

Waste from nuclear power plant (NPP), arising from the operation and decommissioning of nuclear power plants.

Uranium production waste, arising mainly at the front end of the nuclear fuel cycle.

Miscellaneous wastes, arising from medicine, industry, and research (including decommissioning research and test reactors and nuclear laboratories).

Wastes are also classified according to their envisaged disposal route. Four types of repositories are considered as shown in Figure 15 [Ref. 18] and 16 [Ref. 19]. There are two basic concepts for land disposal, i.e., “geological disposal” and “near surface and intermediate depth disposal with institutional control.” (Figure 16) The near-surface and intermediate-depth disposal consists of near-surface disposal with artificial barrier (concrete vault; L2), near-surface disposal without artificial barrier (trench; L3), and intermediate-depth disposal (disposal at a depth sufficient to safety margin for conventional underground building; L1). HLW is disposed of solely by geological disposal, and LLW can be disposed of either by geological disposal or near-surface and intermediate-depth disposal with institutional control, depending on the property of the waste.

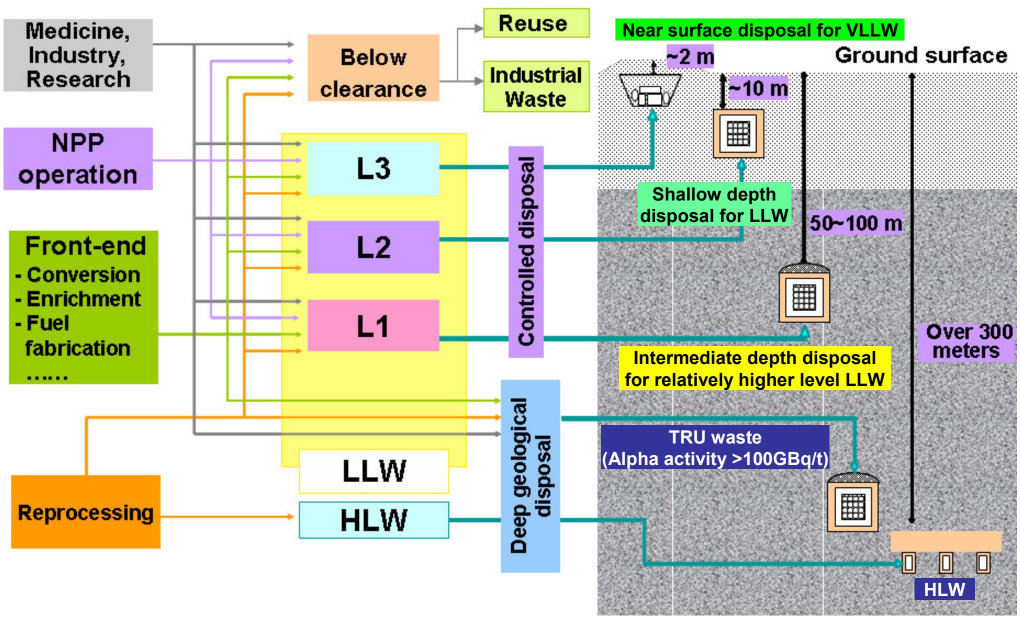


Figure 16. Disposal System Concepts for Various Classifications of Japanese Wastes

The Reactor Regulation Law provides for the setting of upper limits on the concentrations of radionuclides in waste authorized for near-surface and intermediate-depth disposals from reactor facilities. These upper limits have been formulated on the basis of a report published by the Nuclear Safety Commission (NSC) and are used in the preparation of license applications. The Reactor Regulation Law also provides for the clearance level and the procedure for its monitoring for compliance.

Mass Flow and Inventories for Fast Reactor System Scenario

In a Japanese feasibility study on a commercialized FR cycle system (FS), the combined system of sodium-cooled reactor, advanced aqueous reprocessing system, and simplified pelletizing fuel fabrication(MO_X fuel) has been evaluated as the most promising FR cycle system concept.

The process flow of advanced aqueous reprocessing [Ref. 11], which was selected as the most promising, is compared with the current plutonium-uranium extraction (PUREX) flow in Figure 17. The advanced process is composed of:

- Disassembly to remove wrapping tube and other hardware, shearing the fuel-pin bundle
- Dissolution to obtain a high concentration of dissolved solution, clarification to remove residue
- Crystallization of dissolved solution to recover the most part of the U,
- Solvent extraction of dissolved solution to recover Pu, Neptunium (Np), and the remaining U
- Extraction chromatography of raffinate of solvent extraction to recover Americium (Am) and Curium (Cm).

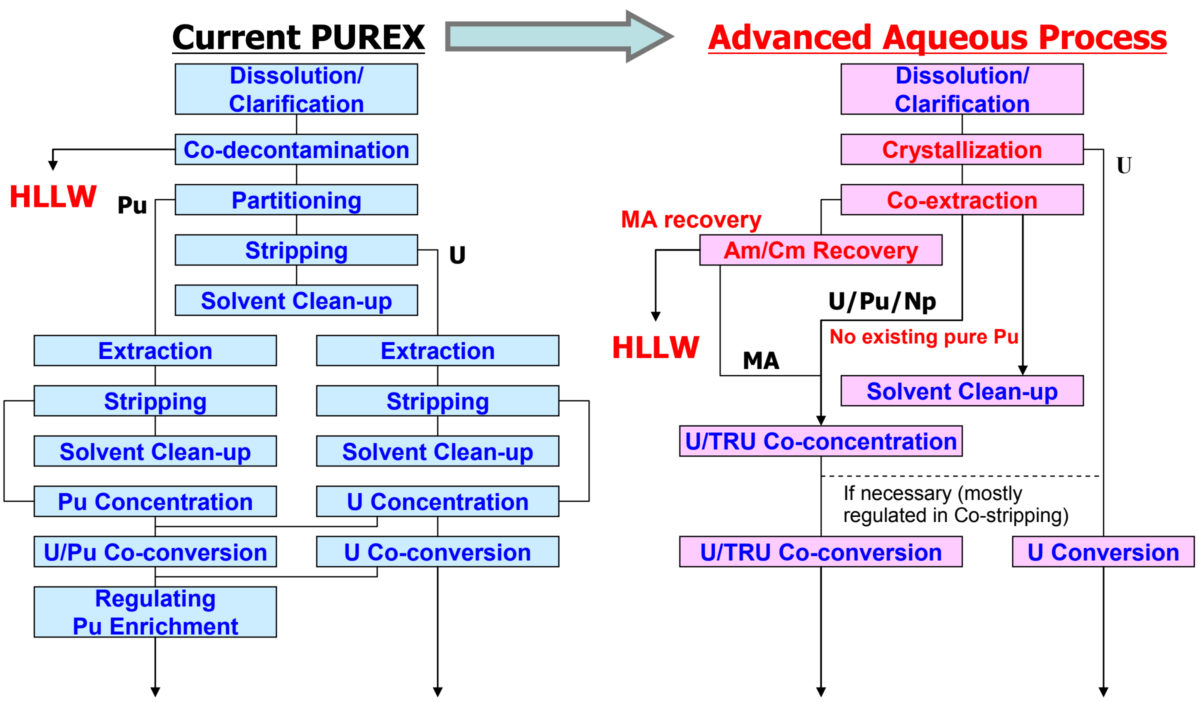


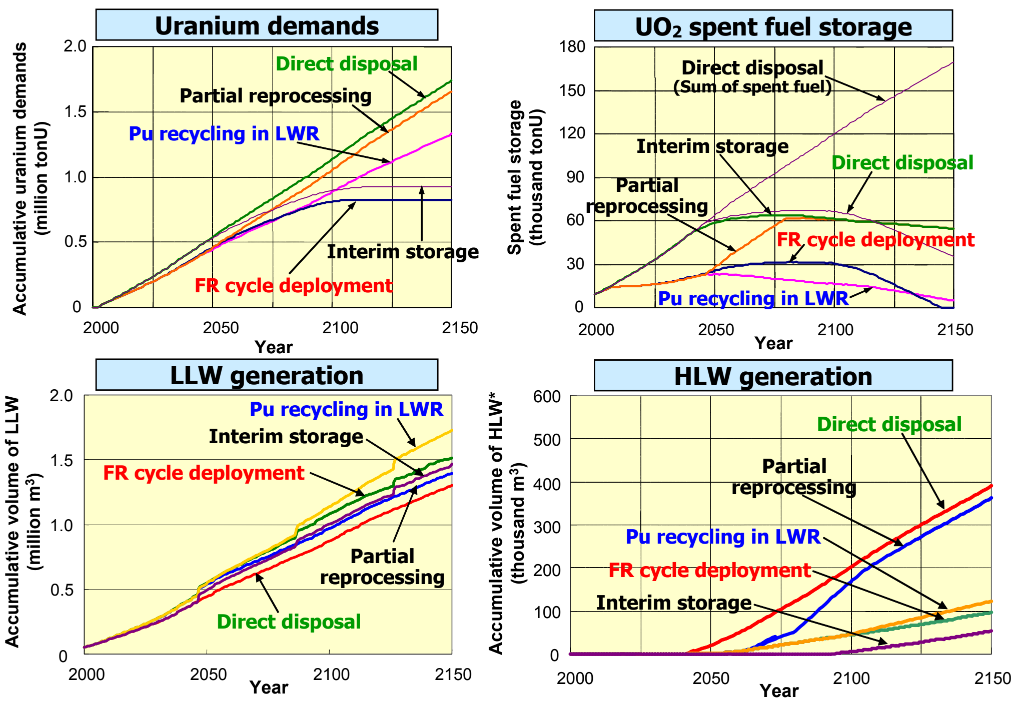
Figure 17. Comparison of PUREX with Advanced Aqueous Processing

The MA recovery in the advanced process reduces the generation of HLW. The LLW, including TRU waste, is also expected to be reduced by applying a simplified waste-treatment process with salt-free reagents.

In the Framework for Nuclear Energy Policy [Ref. 5], four representative scenarios, shown in Table 4 and Figure 18, for a nuclear energy and fuel-cycle system in the future are evaluated from various viewpoints. The mass flows and waste inventories are also evaluated for each scenario.

Table 4. Representative Fuel Cycle Scenarios Considered in Japan

CASE	Note
I. Direct disposal scenario (CASE 1)	LWR once-through (direct disposal of all spent fuels)
II. Partial reprocessing scenario (CASE 2)	Reprocessing of a part of spent fuels and directly disposing of the remainders (Rokkasyo LWR reprocessing will terminate in 2047)
III. Reprocessing of all spent fuels	Continuation of nuclear fuel cycle policy
Pu recycling in LWR scenario (CASE 3-A)	Continuation of LWR cycle by plutonium thermal utilization in LWR
FR cycle deployment scenario (CASE 3-B)	FR cycle to be deployed after 2050 with minor-actinide (Np, Am, Cm) recycling.
IV. Interim storage scenario (CASE 4)	FR cycle will be deployed in 2050 after interim storage.



* as waste package for disposal (overpack container including vitrified HLW, canister including spent fuel)

Figure 18. Representative Fuel Cycle Scenarios in Japan

3.2 Waste Management System Optimization

This section discusses the results of Task 2, waste management system optimization. Section 3.2.1 first discusses the development of repository system concepts for consideration by the WMWG (task 2-1). Section 3.2.2 then summarizes advanced waste form development activities (task 2-2). A definition of optimization problems (task 2-3) is next discussed in Section 3.2.3 followed by a discussion of the use of system analysis by the WMWG (task 2-4).

3.2.1 Development of Repository System Concepts (Task 2-1)

This section discusses the development of repository concepts that will be considered by the WMWG(task 2-1). First a survey and review of repository design concepts for the various classifications of nuclear waste is presented in Section 3.2.1.1. The factors to consider in repository design and optimization are then discussed in Section 3.2.1.2. Japan is considering various alternatives and options for its first geological disposal facility and this work is discussed in Section 3.2.1.3. The framework for future analysis of repository concepts is then presented in Section 3.2.1.4.

3.2.1.1 Survey and Review of Repository Design Concepts

This section summarizes disposal facilities within each of the systems (deep geologic, enhanced isolation, near surface). The purpose of this section is to identify the types of disposal facilities that potentially could be used for disposing of wastes from an advanced nuclear fuel cycle. This summary is provided at a survey-level, and additional information can be found in the references cited.

Deep Geologic Disposal Facilities

Geologic disposal of HLW and SNF is internationally accepted. In 2001 the National Research Council reaffirmed their position regarding geologic disposal of HLW, stating [Ref. 20]:

> Geological disposal, the approach recommended in previous National Research Council (NRC) reports and by many other national and international scientific bodies, is the only available alternative that does not require ongoing control and resource expenditures by future generations. The science supporting this alternative has been developed by intensive work over the past 25 years. The view repeatedly expressed by a large fraction of the scientific and technical community is that geological disposal, correctly managed, is a safe approach to long-term management of HLW and that it best satisfies the ethical goal of minimizing burdens on future generations. Nevertheless, uncertainties remain, and some scientists feel that it is premature to commit fully to disposal. The biggest challenges to initiating geological disposition, however, are societal: there is a clear lack of public confidence and support in many countries for proceeding with siting and construction of geological repositories.

The Nuclear Waste Policy Act defines the term disposal as the emplacement in a repository of HLW, spent nuclear fuel, or other highly radioactive material with no foreseeable intent of recovery, whether or not such emplacement permits the recovery of such waste. However, a key aspect of deep geologic disposal becoming more important internationally is the concept of reversibility and retrievability with the context of stepwise decision making.

In 2004, the OECD Nuclear Energy Agency (NEA) published the Stepwise Approach to Decision Making for Long-term Radioactive Waste Management—Experience, Issues and Guiding Principles

[Ref. 21]. This report discusses and recommends the use of a stepwise decision-making process for managing radioactive waste. Although the report focuses primarily on ultimate disposition in geologic repositories, its findings and recommendations are relevant to the disposition of wastes generated by advanced fuel cycles under consideration in both the U.S. and Japanese programs. Key points, taken from this report are presented below.

- Radioactive waste management involves both technical and societal decision making.
- The key feature of the stepwise concept is development by steps or stages that are reversible, within the limits of practicability.
- A stepwise approach provides reassurance that decisions can be reversed if experience shows them to have adverse or unwanted effects.
- A stepwise approach to decision making has thus come to the fore as being of value in advancing long-term radioactive waste management solutions in a socially acceptable manner.

In 2008, the OECD NEA released Moving Forward with Geological Disposal of Radioactive Waste, A Collective Statement by the NEA Radioactive Waste Management Committee [Ref. 22]. The report expresses the collective view on why geological disposal remains an appropriate waste management choice for hazardous and long-lived radioactive wastes. Key points include:

- Spent nuclear fuel and high-level waste must be contained and isolated from humans and the environment for many tens of thousands of years.
- It is universally recognized that safe and acceptable disposal solutions must be pursued for existing and projected inventories of long-lived radioactive waste.
- A geological disposal system provides a unique level and duration of protection for long-lived radioactive waste.
- The overwhelming scientific consensus worldwide is that geological disposal is technically feasible.

Reversibility denotes the fact that fallback positions are incorporated in the long-term waste management policy, as well as in the actual technical program [Ref. 23]. Reversibility is meant to help a facility program respond flexibly to:

- new technical information regarding the site and design
- new technological developments relevant to radioactive waste management
- changes in economic, social, and political conditions and acceptance
- changes in regulatory guidance and its interpretation or even, possibly, in basic safety standards.

Reversibility is made possible by considering and incorporating fallback positions at any given step in the development program of a waste-management facility. This contributes both to technical confidence in the ability to manage the waste safely and, also, to confidence in wider audiences that an irreversible decision is not being made. Reversibility should not be seen as a lack of confidence in ultimate safety of a waste-management option, but rather as a desire to make optimum use of available options and design alternatives.

Retrievability denotes the possibility of reversing the action of waste emplacement [Ref. 23]. It is thus a special case of reversibility. Retrieval is the action of recovery of the waste or waste packages. Retrievability, the potential for retrieval, may need to be considered at various stages after emplacement, including after final sealing and closure. Some facility concepts for deep geologic disposal and certain geologic media are more amendable for implementing a stepwise decision-making process that includes retrievability than others. The retrievability of wastes is virtually impossible in some concepts. Retrievability aspects are discussed below for each concept within the deep geologic disposal system.

Mined Geologic Repository

Every organization actively pursuing the disposal of SNF or HLW is investigating the disposal of these wastes in mined geologic repositories. A mined geologic repository is simply a mined facility for the disposal of wastes located hundreds of meters beneath the earth's surface. They consist of both engineered and natural barriers that together serve to prevent or minimize the movement of radionuclides to a point where they can affect the population. It is recognized that a properly sited and constructed repository with passive engineered and natural barriers will provide adequate protection of public health and safety during the hazardous lifetime of the wastes without requiring additional human action.

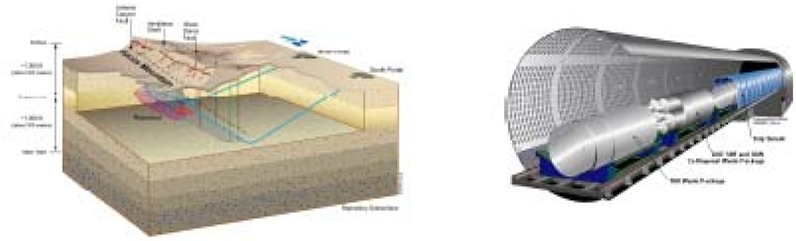
Several geologic media have been considered, including salt, unsaturated tuff, and saturated basalt, shale, granite, argillite, and clay. Repository designs differ based on the quantities and types of waste disposed of (SNF vs. HLW) and the geologic media in which the repository would be constructed. In general, they consist of access shafts or ramps and rooms, tunnels, or galleries for disposing of wastes. Design concepts for several repositories under development are shown in Figure 19 as examples [Refs. 24 – 28].

The development of a repository can generally be broken out into five phases.

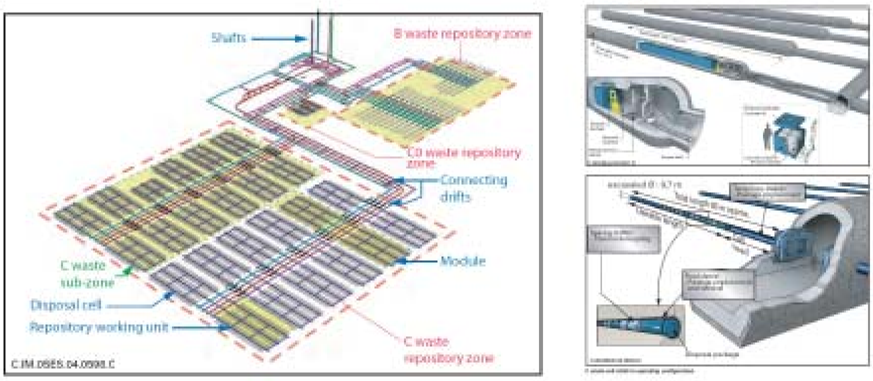
- Site Characterization, Preliminary Design, Licensing: The activities required to characterize the site, develop a preliminary design, and to develop the safety case to submit to the regulator to obtain authorization to construct the facility and to receive waste
- Detailed Design, Surface Facility Construction, Initial Subsurface Facility Construction: Detailed facility design followed by construction of the surface waste-handling facilities and the initial subsurface facilities for waste emplacement
- Subsurface Facility Construction and Waste Emplacement: Construction of additional sub-surface disposal facilities in parallel with emplacement operations
- Monitoring and Ventilation: Ventilation of the subsurface facility (active and natural) to allow for thermal decay and monitoring if needed.
- Closure: Sealing, backfilling (depending on the repository design), and repository closure.

These phases are for the most part independent, but there is some overlap. For example, in a two-step licensing approach, as is the case for the proposed Yucca Mountain repository, licensing would continue into the detailed design, surface facility construction, and initial subsurface facility construction phase. Some steps also may not be included. For example, some designs may not use backfill, or the facility may be backfilled immediately after the waste is emplaced. In addition, monitoring and ventilation may not be included in some repository concepts as immediate closure may be desired.

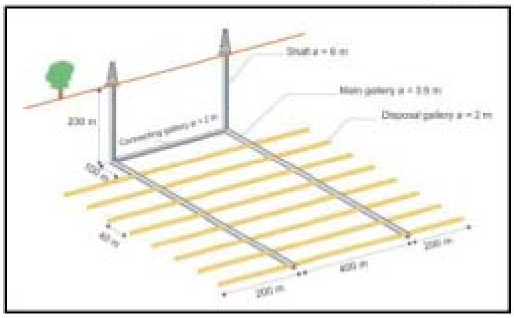
The capability to retrieve wastes depends on the geologic media and the operational phase. Although repositories are designed primarily to dispose of wastes, typically, the disposed wastes can be retrieved until backfill is placed, the repository is sealed, or both. This is not to say that repositories are designed such that the waste can be retrieved, but rather retrievability is possible. In principle, retrieving waste is possible even after the closure of a repository although technical difficulty increases. Retrieving wastes disposed in salt may be more difficult because of the propensity of salts to creep, which would be accelerated in the presence of heat-generating wastes.



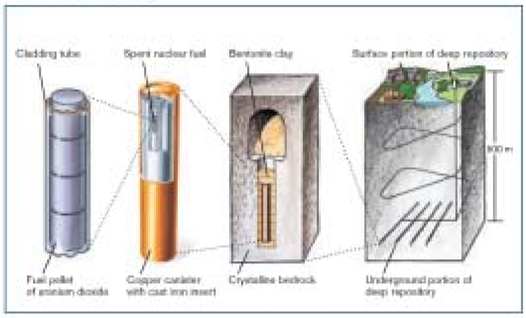
Unsaturated Tuff (U.S. DOE, Yucca Mountain)



Argillite (France, ANDRA)



Clay (Belgium, ONDRAF/NIRAS)



Crystalline Rock (Granite, SKB)

Bedded Salt (U.S. DOE, WIPP)

Figure 19. Mined Geologic Repository Concepts

The concept of HLW geological disposal in Japan is similar to that in other countries: it is based on a multibarrier system that combines the natural geological environment with engineered barriers [Refs. 29 and 30]. The approach for development of a disposal system concept in the generic phase has been to consider the wide range of geological environments throughout Japan without targeting either a specific type of rock or a specific siting area. However, particular consideration is given to the long-term stability of the geological environment, taking into account Japan's location in a tectonically active zone. Because of Japan's complex geology, an engineered barrier system (EBS) with sufficient margins in its isolation functions to accommodate a wide range of geological environments was developed. The major component of the disposal system's function to serve as an overall barrier is borne by the near field (the EBS and a limited volume of the surrounding host rock), while the remainder of the geosphere serves to reinforce and complement the performance of the EBS. The disposal concept is therefore to construct an EBS that, in a stable geological environment, provides sufficient margins in its long-term performance for isolation of the waste applicable to a range of potential geological conditions and their future evolution. A reference layout of the EBS involves either axial emplacement in a horizontal tunnel or vertical emplacement in a pit, as shown in Figure 20 [Ref. 30], which can be adopted for both hard and soft rock systems; in both cases, vitrified waste is encapsulated in a thick steel overpack surrounded by highly compacted bentonite. NUMO has also considered a range of repository concepts to increase flexibility in optimizing design to volunteered sites [Ref. 30]. Such alternatives will be discussed in section 3.2.1.3.

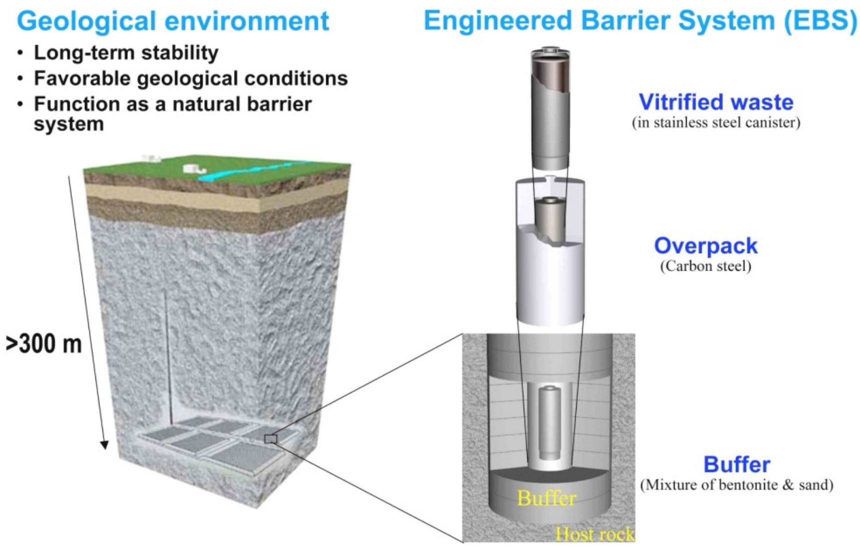


Figure 20. Japanese Disposal System Concept

The mechanical stability of tunnels is investigated based on data obtained from relevant geological environments. Rough estimates are then made of the depth range in which construction of the disposal facility is feasible. In addition, a design concept for efficient emplacement of the vitrified waste and layout of the tunnels is developed, based on thermal analyses, as shown in Figure 21. It has been determined that construction of the disposal system, emplacement of the waste forms, and backfilling of the tunnels can be realized using currently available technologies or technological advances expected in the near future.

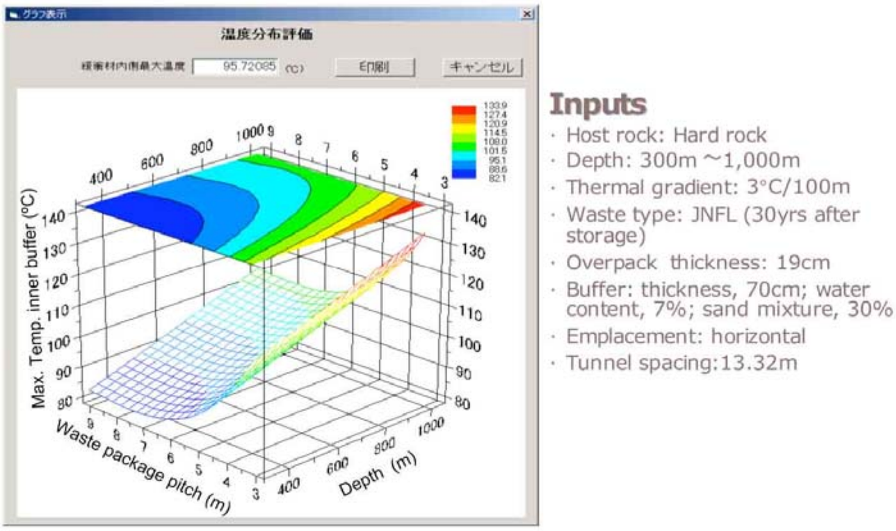


Figure 21. Thermal Analysis for Waste Package Pitch (Min. Footprint) as a Function of Depth

In Japan TRU-waste is officially defined as non-HLW waste generated from the operation and dismantling of reprocessing facilities and MOX fuel fabrication facilities containing long-lived radionuclides such as trans-uranium nuclides. It also includes non-HLW returned from BNG Sellafield (formerly part of BNFL) and AREVA NC (formerly COGEMA). TRU-wastes for geological disposal are categorized into 4 groups as shown in Figure 22 [Ref. 31]. Group 1 includes weak sorbing radionuclides, such as I-129. Group 2 includes hulls and end-pieces, which generate heat and contain large concentrations of C-14. Group 3 includes chemical substances such as sodium nitrate, which have to be further analyzed in terms of impact on radionuclide migration. Group 4 consists of other miscellaneous wastes. Since the heat generation is small, TRU-wastes for deep geological disposal can be emplaced together in tunnels with large cross-sections. However, there is a wide variety of waste materials such as metal, cement, nitrates and organics. For this reason in the basic TRU-waste repository layout, each of the waste groups will be emplaced in separate disposal tunnels. In other words one disposal tunnel or drift will contain waste from one group only.

Disposal tunnels for Group 1 and Group 2 require a buffer consisting of compacted bentonite and sand mixture to maintain a low ground water flow in the repository because of the larger concentration of highly soluble and low sorbing radionuclides in these wastes. On the other hand, disposal tunnels for Group 3 and Group 4 do not require a bentonite buffer. The effects of thermal stress (originating mainly from Co-60 as gamma heat in Group 2), gas generation and extrusion of buffer material have only a small effect on the mechanical stability of the near-field according to the numerical evaluations performed. However, interaction between creep behavior of the host rock and deformation of the EBS could be considerable depending on rock type. Disposal tunnels with horseshoe-shaped cross-sections and circular cross-sections are shown to be mechanically stable in the hard rock dataset but in soft rock dataset only circular disposal tunnels could be used. An advantage of horseshoe-shaped tunnels is that they allow more efficient and cost effective techniques for the emplacement of buffer and waste packages.

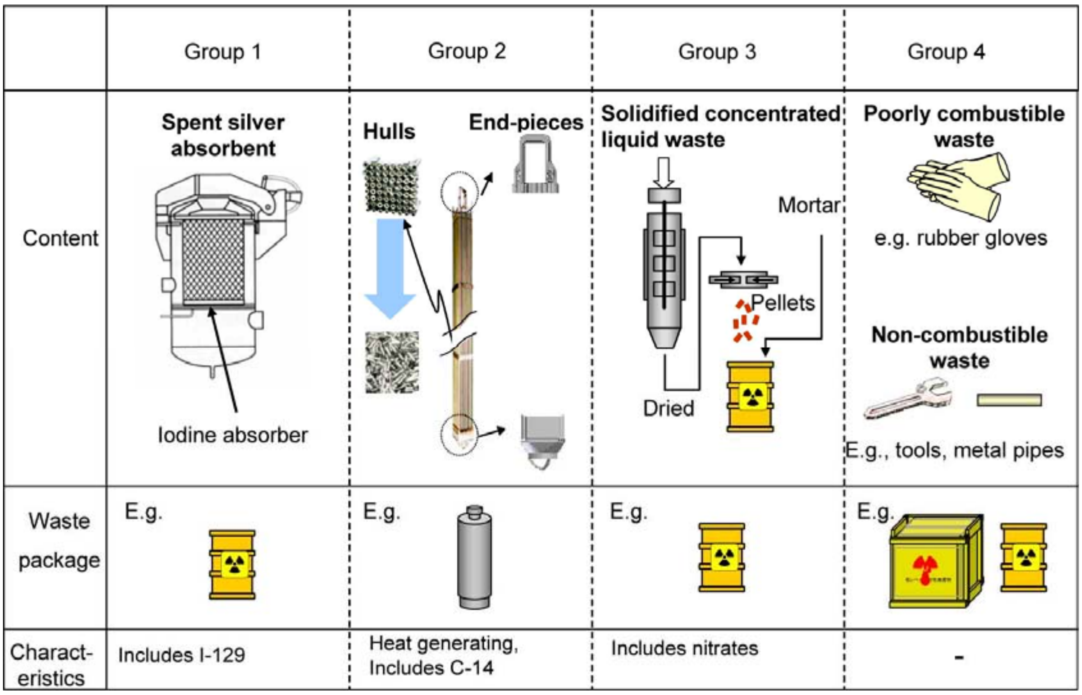


Figure 22. Japanese Grouping of TRU Waste for Deep Geologic Disposal

Deep Borehole Disposal

The disposal of radioactive wastes in deep boreholes is not a new concept, but only began to receive consideration during the 1990s. In the deep borehole concept, waste would be emplaced in the lower part of one or more deep boreholes drilled in tectonically, hydrologically, thermally, and geochemically stable rock formations. Once the emplacement zone of the borehole is filled with materials, the "isolation zone" extending from the top of the emplacement zone to the ground surface is filled and sealed with appropriate materials. A diagram of a deep bore hole disposal facility is shown in Figure 23 [Ref. 32].

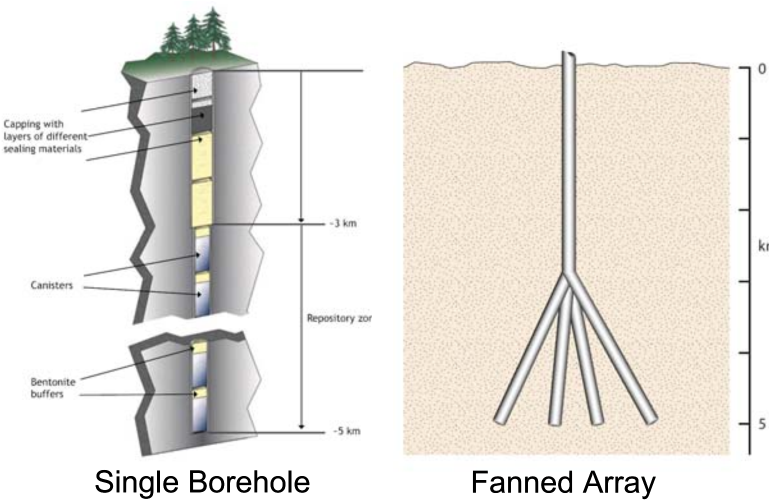


Figure 23. Deep Borehole Concept

At emplacement depths, the groundwater is expected to be relatively stagnant, highly saline, hot (75 to 150°C), and under high pressure. In deep boreholes, there is a large barrier to transport posed by the isolation zone because of its low permeability and high sorptivity, the stability and low-solubility of the disposal form, and high salinity and lack of driving forces for fluid flow. Thus, the disposed material is expected to remain, for all practical purposes, permanently isolated from the biosphere.

Two concepts have been proposed for a deep borehole disposal facility [Ref. 32]. The first is a low temperature concept where wastes are disposed of at low concentrations such that the heat released into the rock is limited. The bedrock surrounding the boreholes will be impacted as little as possible so as to maintain a stable groundwater density stratification. A second concept is a high-temperature facility where the waste is disposed of at larger concentrations with boreholes placed close together. In this concept, the heat generated would partially melt immediately adjacent rock, purging the area surrounding the boreholes of water and gas. As the waste cools, the rock would solidify into dry, newly crystallized rock.

Rather than relying on a combination of engineered and natural barriers to protect the public as does a mined geologic repository, deep boreholes rely on the natural conditions of the site as the only isolation barrier. However, in a high-temperature concept, the waste package must remain intact during the high-temperature stage [Ref. 33]. Lifetimes on the order of 10,000 years are required. Copper or noble metal alloys may be the only metallic materials that could survive the high temperatures and pressures for the requisite time. Mineral-based waste packages may be needed.

As discussed in *Disposition of Excess Weapon Plutonium in Deep Boreholes, Site Selection Handbook* [Ref. 34], ideally, a deep borehole site within the United States would consist of a combination of:

- Crystalline rock at the surface or within 1 km of the surface
- A region that is tectonically stable
- An area located away from population centers
- A region that is not near international borders (~200 km).

Cratons are part of the earth's continental crust that are stable and have been little deformed for a prolonged period, and a pluton is a large mass of igneous material that has intruded into the shallow earth crust. The center of a pluton within a craton that is either exposed or overlaid by a relatively thin layer of sedimentary material would likely be a preferred site for a deep-borehole disposal facility.

As discussed in the *Disposition of Excess Weapon Plutonium in Deep Boreholes, Site Selection Handbook* [Ref. 34], about 90 percent of the United States is underlain by Precambrian rock (>540 million years old), which makes up the continental crust with large areas of this "basement" covered by less than 1 kilometer of sedimentary and volcanic rock. The Midcontinent region (between the Rocky and Appalachian mountains) has an approximate area of at least 2,600,000 km^2 of accessible Precambrian basement within 1 km of the surface and could be a large resource for siting a deep-borehole disposal facility. Within this region, the Canadian Shield is a large tectonically stable area of "basement" rocks that has been exposed by glaciation of a craton (stable continental mass). The Canadian Shield extends into the northern United States where it is either exposed or covered by a thin layer of sedimentary cover. Plutonic rocks within the Canadian Shield area may be ideal sites because they are relatively uniform in nature.

Criteria for developing a deep-borehole disposal facility have been proposed in Final Deposition of High-*level Nuclear Waste in Very Deep Boreholes: An Evaluation Based on Recent Research of Bedrock Conditions at Great Depths* [Ref. 32].

1. The existence of a sufficiently large area at a depth of 3 to 5 km having groundwater, the density-stratification of which is stable

2. The availability of reliable technology for measurements and analyses that can localize areas at -3 to -5 km having groundwater, the density-stratification of which is stable
3. Sufficient knowledge of geodynamic and hydrogeological conditions as to permit the identification of areas at depths of 3 to 5 km
4. The availability of technology for the precision drilling required for both exploration and deposition
5. The ability to deposit filled canisters and, during the period of deposition, to retrieve canisters in order to exchange them or to test materials and technological solutions
6. The feasibility of drilling boreholes, depositing the canisters, and sealing all of the boreholes without corrupting the long-term stability of the density-stratification of the groundwater around the repository
7. The feasibility of storing HLW in canisters for extremely long periods of time so that neither the heat nor the radioactivity generated by the decay process corrupts the stability of the density-stratification of the groundwater around the repository
8. The selection of drilling equipment, canisters, and sealing materials with a view to avoiding chemical reactions that might give rise to gases in the repository area.

In 1996, DOE considered deep-borehole disposal as a method for disposing of surplus weapons-grade plutonium [Refs. 35 and 36]. Although this method was not chosen for managing surplus weapons-grade plutonium, the concept was found to be feasible. As stated in the *Technical Summary Report for Weapons-Useable Plutonium Disposition*:

While no deep borehole disposal facilities for plutonium disposition have ever been developed, many of the technologies needed for this alternative are quite mature; and the basic concept has been considered previously for waste disposal.

It is believed that suitable rock formations can be found in a variety of areas, that they can be adequately characterized, and that the long term evolution of processes can be predicted to assure long term isolation and safety.

Siting guidelines and procedures is the largest area of uncertainty. Site suitability guidelines consistent with the mission and safety concept of deep borehole disposition will require development. A regulatory framework to address this deep borehole disposal does not currently exist. Therefore, regulatory uncertainty was identified as a risk that affects the viability of deep borehole disposition. However, preliminary discussions with licensing experts indicate that a licensing regime can be developed, given sufficient time and a mandate.

The equipment required to implement the deep borehole alternatives are adaptations of equipment designed and used for nuclear weapons testing, geological studies, and the petroleum and gas drilling industries. The equipment requirements with respect to environmental safety and quality are within current capability or are viable extrapolations from existing mechanical engineering designs. An integration and demonstration of the equipment will be required, and the systems engineering must be performed. Notwithstanding, the mechanical design is not expected to be a controlling technical risk for these alternatives.

In 2003, the Massachusetts Institute of Technology released a report entitled *The Future of Nuclear Power, An Interdisciplinary MIT Study* that discussed deep-borehole disposal as an alternative to mined geologic repositories [Ref .37]. The report identified several obstacles as listed below.

- A new set of standards and regulations would have to be developed.
- The difficulty of retrieving waste from boreholes would be difficult should a problem develop.
- Satisfying current U.S. regulations that require a period of several decades during which the waste must be retrievable would be difficult and expensive, but not impossible, in a deep borehole facility.
- The knowledge of *in situ* conditions at great depth would never be as comprehensive as in a shallower mined geologic repository environment.
- Recovery from accidents during waste emplacement would likely be more difficult than in a mined geologic repository.
- It is difficult to predict the impact on public opinion of a shift in siting strategy from a large central repository to perhaps several widely dispersed boreholes.

Despite these obstacles, the report authors state:

Despite these obstacles, we view the deep borehole disposal approach as a promising extension of geological disposal, with greater siting flexibility and the potential to reduce the already very low risk of long-term radiation exposure to still lower levels without incurring significant additional costs.

Other Deep Geologic Disposal Concepts

Other concepts have been proposed for deep geologic disposal systems. However, some have not been fully evaluated and are very conceptual. They are presented here both for completeness and to introduce the concepts for potential future evaluation.

<u>International Repositories—High-Isolation Sites</u>

Preliminary efforts in siting an international mined geologic repository have focused on finding what is termed a high-isolation site. To choose a site for an international repository, an organization called Pangea proposed finding a site that would fulfill the safety requirements of national repository programs, but would also be as simple as possible such that the safety case could be demonstrated with the most transparency [Ref. 38]. This resulted in Pangea identifying a set of attributes where a high-isolation site would have most, but not necessarily all, of them. The attributes are:

- Stable geology (needed because of the extremely long isolation times required)
- Flat topography (reduces driving forces for groundwater movement)
- Near-horizontal sedimentary strata (simpler to investigate and characterize)
- Stable, arid climate with negligible erosion (eases problem of extrapolation into the future)
- Low permeability host rock (reduces groundwater movements)
- Old and saline groundwater (indicates negligible groundwater movement; and non-potable water)
- Stratified salinity (counteracts thermal buoyancy effects)
- Reducing geochemical conditions (reduces solubilities of radionuclides)
- Absence of complex karst systems (simplifies hydrogeologic modeling)
- Low population density (reduces intrusion risks)
- No significant resource conflicts (reduces intrusion risks).

Pangea identified regions where potential high-isolation sites might exist in Western and Southern Australia, Argentina, Southern Africa, and China. Pangea decided to focus its limited resources on Western Australia. However, the premature release of internal documents resulted in political and public resistance. At present, no feasibility results are available, and Pangea has essentially ceased operations. Efforts to advance an international repository continue to be advanced by the Association for Regional and International Underground Storage (ARIUS) [Ref. 39], although they do not specifically promote a high-isolation site concept.

Deep Rock Melting

The deep rock-melting concept involves using the decay heat from the waste to first melt the adjacent rock, and perhaps the waste form itself, so that when cooled, it will produce a solid mass that either incorporates or encases the waste. The waste would be disposed of in either a shaft or excavated cavity at a depth of 2 to 5 kilometers. The high-temperature deep-borehole concept discussed above is an example of a deep rock melting concept.

The technique would only be applicable to wastes that would generate significant amounts of heat. Several of the waste streams generated through the processes being considered would not be amendable to this option because they do not generate significant quantities of heat. The vitrified HLW from COEX processing or the Cs/Sr waste form with little decay could potentially be disposed of using this approach.

As discussed in a NIREX report entitled *Description of Long-term Management Options for Radioactive Waste Investigated Internationally* [Ref. 40], in the late 1970s and early 1980s, a deep rock melting concept was taken to the engineering design stage. The design concept involved a shaft or borehole that led to an excavated cavity at a depth of 2 to 5 km. The designers estimated that the waste would be immobilized in a volume of rock 1,000 times larger than the original volume of the waste.

A variation of the deep rock melting concept is the salt-diver repository where the high-heat generation rates of the waste are postulated to allow disposal at depths up to 10 km underground in salt domes [Ref. 41]. The wastes are packaged into moderately large containers called salt divers that are placed in a salt dome. The high-density salt-diver heat source sinks by heating the salt under the WP until the salt becomes plastic and the salt diver then sinks to the bottom of the salt dome.

There are several obstacles associated with deep rock melting disposal concepts (and variants).

- Establishment of standards and regulations
- Knowledge of *in situ* conditions at depth
- Understanding of high-temperature processes at depth (i.e., rock melting, cooling, and re-crystallization; heating and movement of waste packages through the plastic salt in the salt diver concept).
- Ability to predict characteristics following cooling of the waste to conduct long-term safety assessments of the disposal facility for demonstrating compliance with regulations.
- Wastes disposed of in these concepts would not be retrievable.

Direct Injection

The NIREX report entitled Description of Long-term Management Options for Radioactive Waste Investigated Internationally discusses the direct-injection approach where liquid radioactive waste is directly injected into a layer of rock deep underground that has been chosen because of its suitable characteristics to trap the waste [Ref. 40]. The NIREX report identified a number of geological pre-requisites that are required.

- There must be a layer of rock, the injection layer, with sufficient porosity to accommodate the waste and with sufficient permeability to allow easy injection (i.e., act like a sponge).
- There must be impermeable layers above and below the injection layer that act as a natural seal.
- Additional benefits could be provided from geological features that limit horizontal or vertical migration such as injection into layers of rock containing natural brine groundwater that is stratified.

Direct injection could, in principle, be used on any type of radioactive waste that can be transformed into a solution or slurry. Slurries containing a cement grout that would set as a solid when underground could also be used to help minimize movement of radioactive waste. This would require further processing of solidified wastes at a deep injection facility because it is not likely that it would be permissible to transport wastes in liquid or slurry form. However, deep injection may be possible if a processing facility were located at a site where deep injection would be feasible.

The NIREX report points out that Russia injected some tens of millions of cubic meters (by 2002) of low, intermediate, and HLW into porous sandstones capped by clay at depths of 400 meters and into sandstones and limestones at depths of 1,400 meters.

As discussed by the *National Academy of Science in Disposition of High-Level Waste and Spent Nuclear Fuel: the Continuing Societal and Technical Challenges* [Ref. 42]:

The United States practiced direct injection of low-level liquid waste grouts under high pressure into a shale formation beneath the Oak Ridge, Tennessee site in the early 1970s. This process was abandoned due to uncertainties about how the grout flowed within the fractured shale. In 1972, an NRC study found the option of disposing of HLW at the U.S. Savannah River Site directly into crystalline bedrock beneath the site to be technically feasible. However the report cautioned that public approval for this option would be problematic.

For many years, the former Soviet Union injected intermediate-level liquid waste into the subsurface at sites such as Krasnoyarsk, Tomsk, and Dimitrograd. In these cases, the waste appears to have been contained between geological strata as intended. However, the approach is being phased out because it is not considered to be in line with better practices that include solidifying and packaging the waste.

Although possibly feasible, it is unlikely that direct injection of wastes is a viable option because it involves waste forms that are not in solid form and no waste packaging. In addition, the approach is unlikely to be accepted by the public.

Enhanced Isolation Disposal Facilities

This section summarized concepts for enhanced isolation disposal facilities used primarily to dispose of intermediate-level waste. Most of the concepts presented are either being used, are under consideration, or have been considered in the past by various organizations internationally.

Geologic Repositories

Geologic repositories were discussed above, primarily focused on the disposal of spent nuclear fuel or HLW. However, several organizations are either using or intend to use geologic repositories to dispose of intermediate, and in some cases, LLWs. Several of these facilities, either operational or conceptual, are discussed herein. While several facilities are essentially equivalent to deep geologic facilities in terms of depth and will in fact dispose of HLW, others are at much shallower depths. For completeness, both deep and shallow geologic facilities are summarized.

Switzerland

NAGRA intends to dispose of both low- and intermediate-level wastes in geologic repositories per Switzerland's Nuclear Energy Act, and site selection is underway [Ref. 43]. In the past, a facility constructed horizontally into a hillside has been considered [Ref. 44]. Current activities indicate that the Opalinus clay formation may be suitable for locating a spent nuclear fuel, HLW, and intermediate-level waste repository [Ref. 45].

Finland

Finland disposes of low- and intermediate-level wastes from the operation of the Olkiluoto and Loviisa nuclear power plants in geologic repositories constructed at a depth of 70 to 100 meters in crystalline bedrock at each plant site [Ref .46]. A diagram of the VLJ repository at the Olkilouto site is shown in Figure 24 [Ref. 47].

Sweden

Since 1988, Swedish low and intermediate waste has been disposed of in the SFR repository, located at the Forsmark nuclear power plant [Ref. 46]. The SFR repository is constructed in crystalline bedrock, 60 meters under the Baltic Sea, and consists of five different chambers; four "simple" caverns for LLW and one concrete silo surrounded by a clay buffer for intermediate-level waste [Ref. 49]. A diagram of the SFR repository is shown in Figure 25 [Ref. 50].

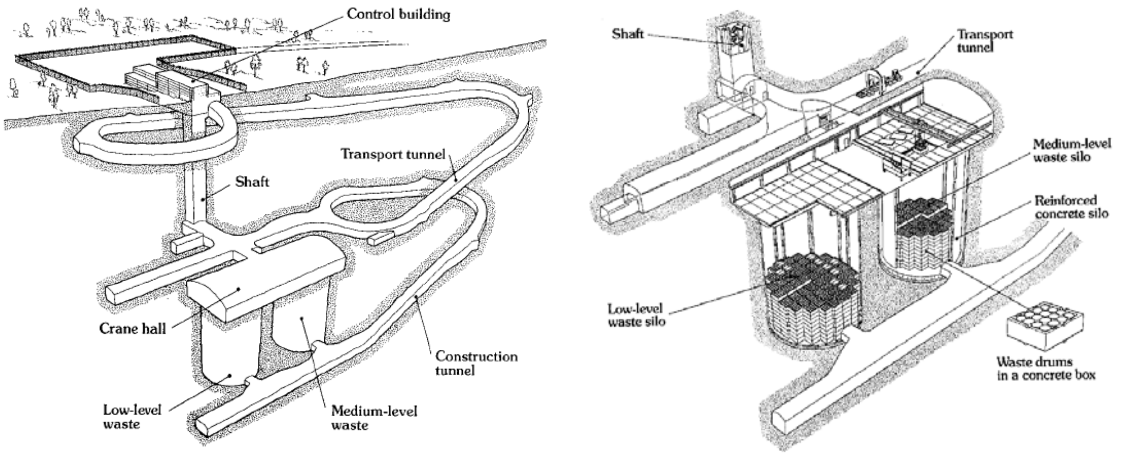


Figure 24. VLJ Repository at Finland's Olkilouto Nuclear Power Plant Site

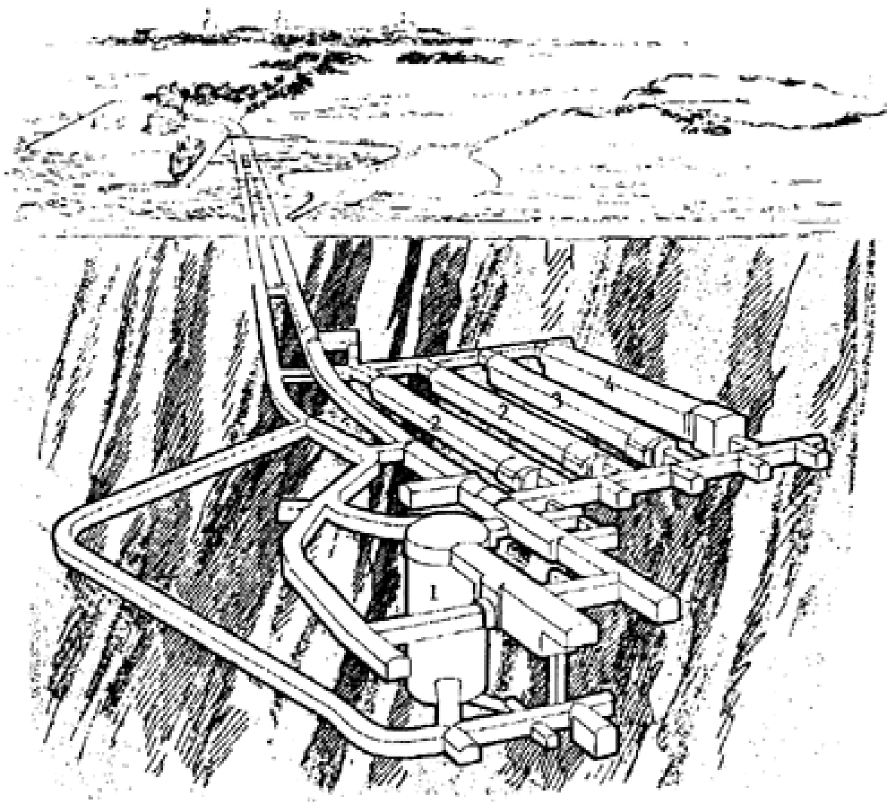


Figure 25. Diagram of the SFR Repository

Canada

Ontario Power Generation intends to dispose of intermediate-level waste in a deep geologic facility in Ontario on the site of the Bruce nuclear power plant [Ref. 51]. The facility would be located 660 meters below the surface beneath thick layers of limestone and shale rock that have remained stable. It is estimated that 160,000 m^3 of LLW and intermediate-level waste will be disposed of in the facility. An artist's rendition of the facility is shown in Figure 26 [Ref. 51].

United Kingdom

The United Kingdom's Committee on Radioactive Waste Management (CoRWM) has recommended that all long-lived radioactive waste be disposed of in a geologic repository [Ref. 52]. This includes HLW, intermediate-level waste, and some LLW that does not meet the acceptance criteria for near-surface disposal. This recommendation and others made by CoRWM regarding the process for siting and developing such a proposal was accepted by the U.K. government [Ref. 53].

Japan

In Japan, an intermediate depth repository concept is considered for the disposal of some LLW as mentioned in section 3.1.2.2.1. Repository design study is being carried out by the Japan Nuclear Fuel Ltd [Ref. 54]. An example is shown in Figure 27 [from Ref. 52].

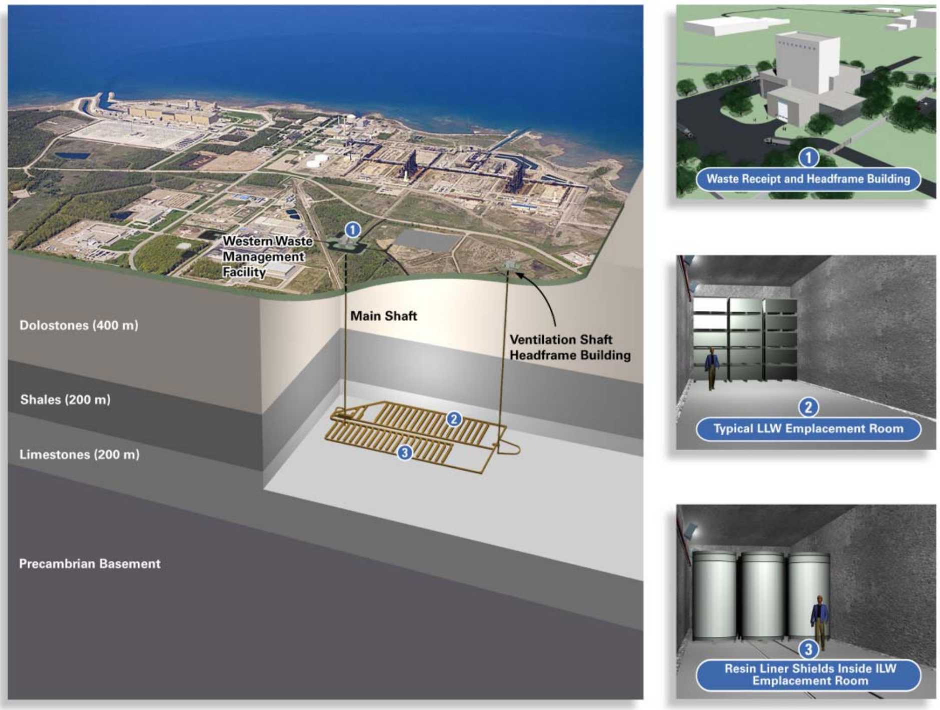


Figure 26. Artist Rendition of OPG's Deep Geologic Disposal Facility

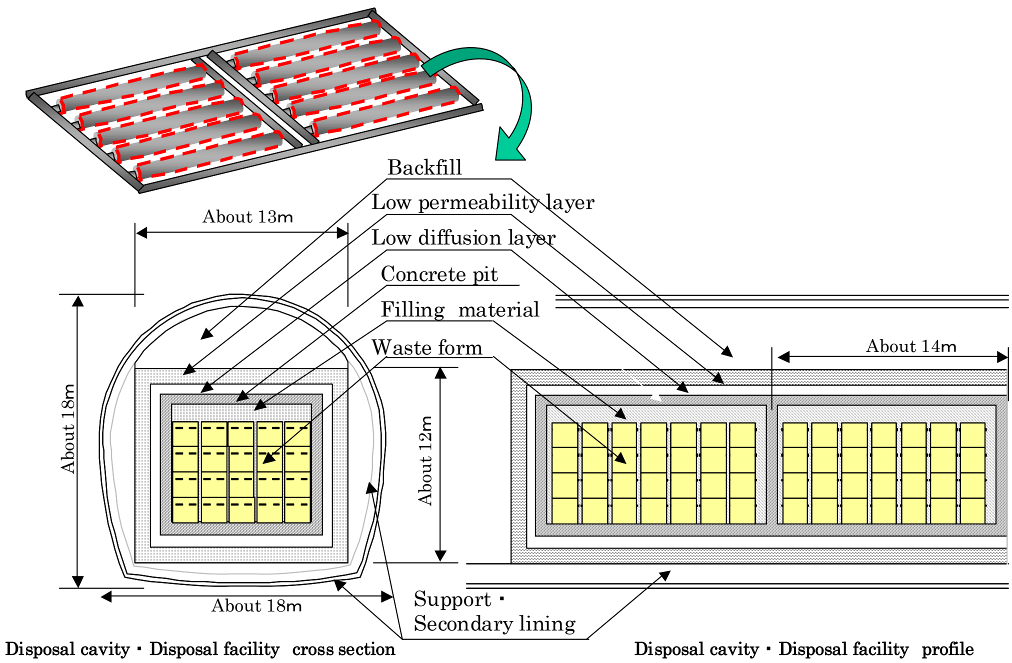


Figure 27. Example of an Intermediate Depth Repository Concept in Japan

Conversion of Mines to Geologic Repositories

Germany has concentrated on converting previously excavated mines into geologic repositories for isolating intermediate-level wastes (non-heat generating) [Ref. 55]. LLW and intermediate-level wastes were disposed of in the Asse (former West Germany) and Morsleben (former East Germany). Operations at Asse, as an experimental repository, were halted in 1978. In 1998, waste emplacement at Morsleben was suspended and will not be resumed. The site continues to be monitored and maintained, and closure activities are scheduled to begin in 2011.

Germany currently plans on using a former iron mine, the Konrad mine, for the disposal of non-heat generating wastes. In 2002, the Konrad mine was licensed to be converted to a repository that could dispose of up to 303,000 m^3 of radioactive waste. The licensing decision was appealed, and the legal process was completed in April of 2007. Wastes will be disposed of at depths ranging from 800 to 1,300 meters.

Intermediate Depth Borehole

On July 23, 2007, DOE issued a notice of intent (NOI) to prepare an environmental impact statement for the disposal of GTCC LLRW [Ref. 56]. In this NOI, DOE proposes to construct and operate a new facility or facilities, or use an existing facility, for the disposal of GTCC LLW and GTCC-like waste. One of the concepts that will be evaluated is intermediate-depth boreholes, shown schematically in Figure 28 [Ref. 57]. The concept involves the construction of a deep borehole (deeper than 30 meters) in the ground. The wastes are then placed in the borehole up to about 30 meters from the surface, and the remaining space is filled with clean soil.

From 1984 through 1989, DOE emplaced high-activity LLRW and some TRU wastes in 13 intermediate-depth boreholes, called greater confinement boreholes at the Nevada Test Site [Ref. 58]. Three-meter-diameter boreholes were constructed to a depth of 36 meters with the bottom 15 meters being used to dispose of wastes. The boreholes were located in unsaturated alluvium with the bottom of each borehole approximately 200 meters above the water table.

Enhanced Near-Surface Disposal

DOE also plans to evaluate the disposal of GTCC LLW and GTCC-like waste in what is termed an enhanced near-surface disposal facility. This involves the placement of the wastes in engineered trenches, vaults, or other similar facilities. The containment characteristics of these disposal facilities are enhanced by incorporating features such as barriers, deeper depth to disposal, and enhanced waste packaging. A schematic diagram of a conceptual enhanced near-surface disposal facility being considered is shown in Figure 29 [Ref. 56].

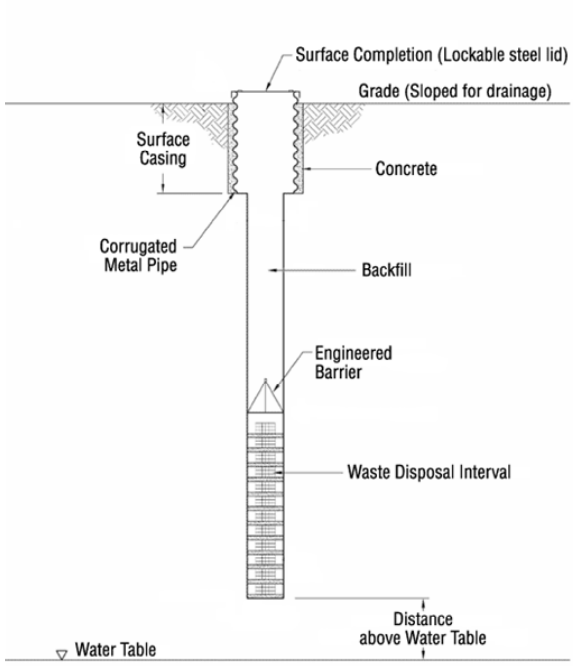


Figure 28. Conceptual Drawing of Intermediate-Depth Borehole

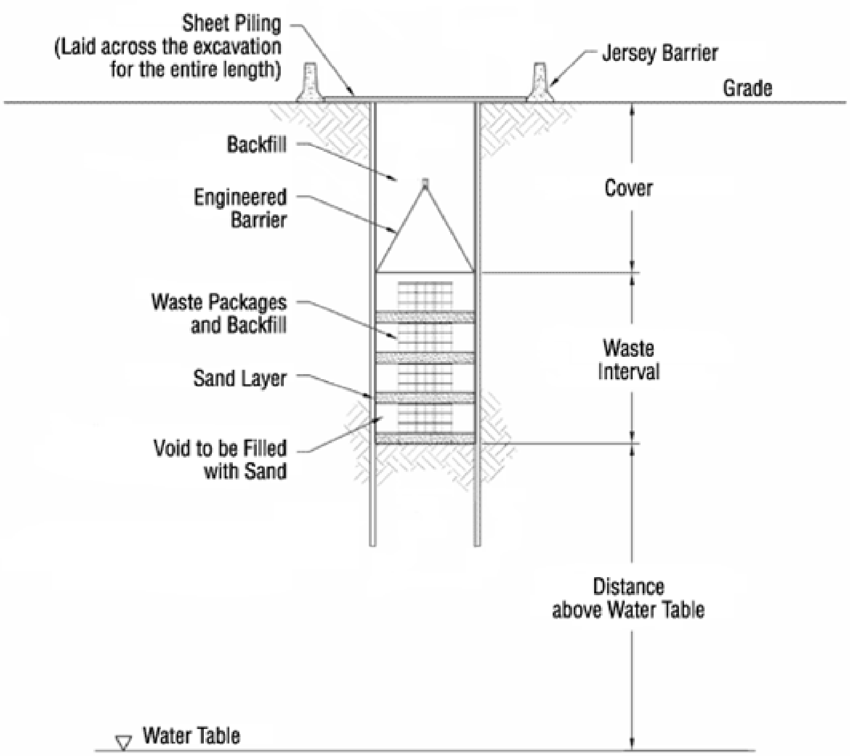


Figure 29. Conceptual Drawing of an Enhanced Near-Surface Disposal Facility

Near-Surface Disposal Facilities

There are different types of near-surface disposal facilities that are being used to dispose primarily of short-lived, LLRW. These include trench facilities, trench facilities with disposal vaults, and above-grade disposal vaults. In 2001, the IAEA released a report entitled *Technical Considerations in the Design of Near Surface Disposal Facilities for Radioactive Waste* [Ref. 59]. This report described technical guidance and information regarding the design objectives and design requirements for near-surface disposal systems. It also described several facilities that are currently in operation. In particular, this report discussed the use of engineered barriers in addition to the natural features of the site to isolate wastes.

For LLRW-containing short-lived radionuclides, the IAEA states that "disposal in trenches with simple engineered barriers might be appropriate, provided that the migration of radionuclides is at an acceptable rate as determined by evaluation of the engineering used." The IAEA also states that "for disposal of LILW (long-lived low or intermediate level waste) with higher levels of radioactivity and/or long lived radionuclides more engineered disposal facilities might be needed (i.e., vaults)."

An example of an above-ground vault design facility is the Centre de l'Aube facility in France, shown in Figure 30 [Ref. 59, Appendix B.1]. The vaults are designed to isolate the waste from groundwater and to have mechanical integrity for 300 years. The base of the vaults is located above the water table. Vaults that dispose of durable waste packages are backfilled with gravel, and those that dispose of less-durable waste packages are backfilled with concrete. Each vault is closed with a concrete slab when full. Final closure will involve construction of a sloped engineered cover composed of several layers of drainage material and clay with a final vegetation cover.

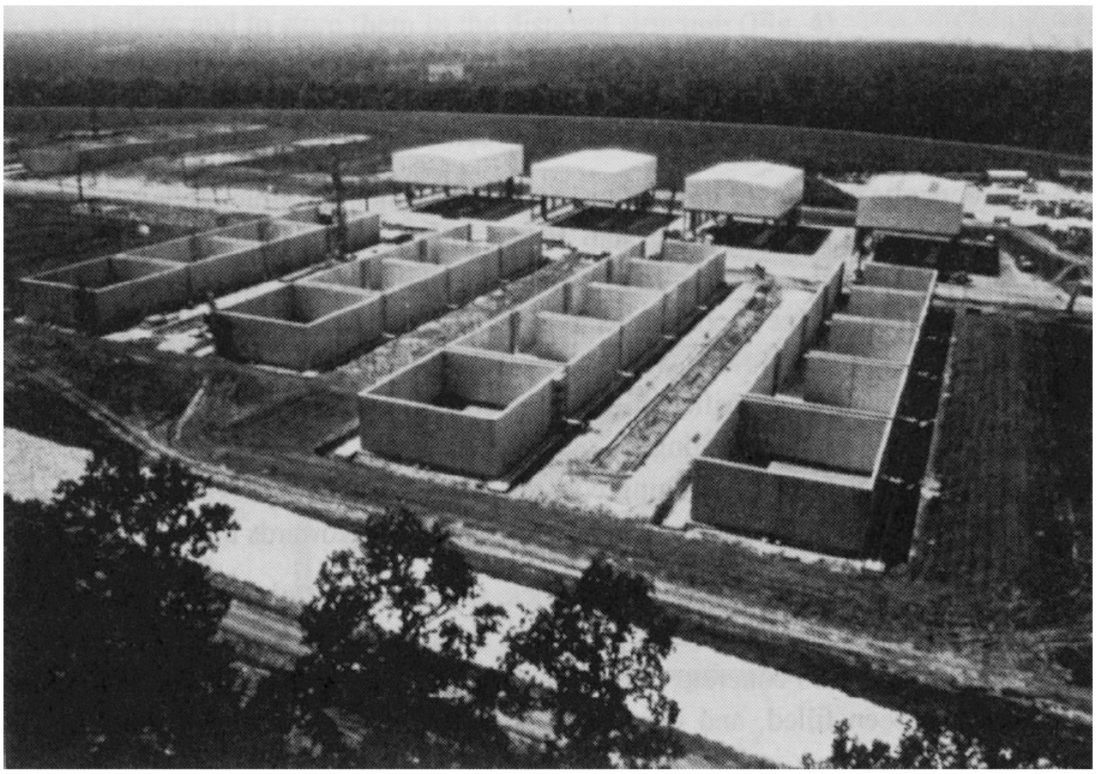


Figure 30. Disposal Vaults at the Centre de l'Aube Facility in France

The Barnwell, South Carolina, site disposes of Class A, B, and C LLRW in vaults located in trenches [Ref. 60]. All LLRW waste containers are disposed of within concrete vaults that are placed in a trench. Different trench designs are used based on the classification of the waste. The Class A trench is approximately 1000 feet (300m)long, 300 feet (90m) wide, and 30 feet (9m) deep. The Class B/C trench is 600 feet (180m) long, 50 feet (15m) wide, and 20 feet (6m)deep. Slit trenches (300 feet (90m) long, 10 feet (3m) wide, and 20 feet (6m) deep) are used to dispose of higher concentrations so as to minimize exposures. The bottom of each trench is located a minimum of 5 feet (1.5m) above the maximum historically measured water-table elevation in the vicinity of the trench. When a vault is full, the space between the vaults is backfilled with clay. Engineered covers are constructed over the backfilled vaults as the trenches fill. The engineered cover consists of a minimum 1-foot-thick clay layer, a geosynthetic clay liner, a high-density polyethylene liner, a sand-drain layer, and a vegetated topsoil cover.

The Richland LLRW disposal facility is a trench design [Ref. 61]. Unstable Class A wastes are segregated from Class B and C wastes and put directly into trenches. Class B and C wastes are typically placed in high-integrity containers or in engineered concrete barriers to achieve required stability and are disposed of. The trenches are typically 45 feet deep (13.5m), 850 feet (25.5m) long, and 150 feet (45m) wide. An engineered cover is placed on the trenches as they are filled. A conceptual drawing of the facility and one of the covers considered in the facility's EIS is shown in Figure 31[Ref. 62].

Near surface repository for LLW waste from nuclear power plants in Japan is operated by JNFL at Rokkasho site [Ref. 63], see Figure 32.

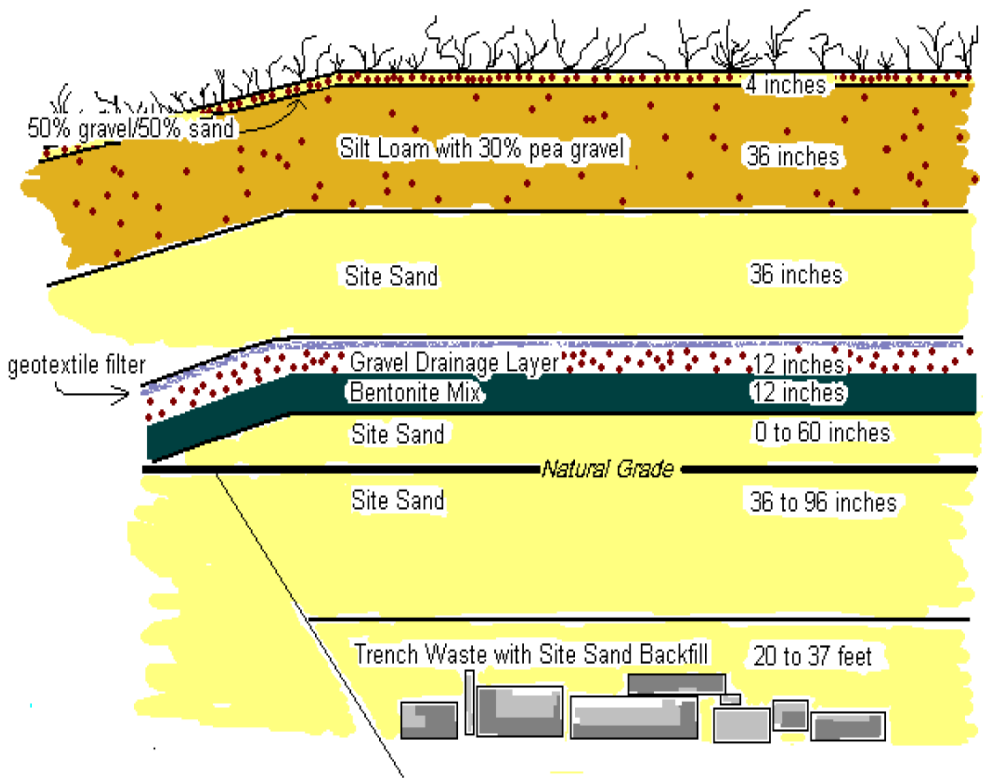


Figure 31. Conceptual Drawing of Richland LLRW Disposal Facility

Figure 32. JNFL near surface disposal facility for LLW from NPP at Rokkasho

3.2.1.2 Factors to Consider in Repository Design Development and Optimization

The overarching factor that must be considered in the development of a repository for disposing of radioactive waste is public safety. A repository must effectively isolate radioactive wastes and be sited and designed such that any releases of radioactive materials that could potentially occur would not lead to the public being exposed to harmful levels of radiation. These exposure levels are set by the regulatory authorities within a country where a repository will be constructed.

The optimization of a repository design depends on several factors, including the physical extent and characteristics of the host rock, the inventory and types of wastes that will be disposed of, thermal management, waste-form volume, long-term repository performance, and cost.

The size of the physically available host rock plays a significant role in the optimization of a repository. The availability of a large area would allow wastes to be placed as far apart as needed. For example, thermal constraints could be met by placing the wastes very far apart. A limited host rock area would add further constraint to the optimization problem. Note that the “size” of physically available host rock is related to the total inventory that would be disposed of where a smaller inventory can be disposed of in a smaller area. The optimization of a design of a repository with a host rock having a very large area would likely be constrained only by the costs of site characterization, mining, and ground support.

For example, in previous work conducted for a repository in the United States, the horizon for a repository at Yucca Mountain was the Topopah Spring formation with the disposal area having to meet the following requirements

- Desirable properties
- Avoid major faults
- 200 meters below the surface, 160 meters above the water table.

Different layouts for a repository at Yucca Mountain had been developed within the Topopah Spring formation for both high- and low-temperature operating modes. These designs dispose of spent nuclear fuel and HLW ranging from 70,000 to 119,000 MTHM. The required disposal space ranged from 1,150 to 2,500 acres(from 4.65 to 10.1 km^2). The latest design, which was driven by the thermal density of commercial SNF and chosen to satisfy thermal criteria for a low-temperature operating mode, results in a disposal area of 1,150 acres. While the United States is evaluating alternative waste management approaches, all scientific investigation and design work at Yucca Mountain has been halted pending the outcome of this evaluation.

The Electric Power Research Institute completed an evaluation that indicates that the repository could potentially be expanded to 3,200 acres (12.9 km^2) with high confidence and 4,200 to 5,700 acres (17.0 to 23.1 km^2) with less confidence because of the paucity of data north and west of the primary repository area [Ref. 64]. This evaluation concluded that a factor of three increase, to 4,800 acres (19.4 km^2), is credible. Additional site-characterization investigations would be needed to support significant expansion beyond the 1,600 acre area that has been sufficiently characterized.

The characteristics of the host rock would influence the type of repository that is designed. A key characteristic is the hydrologic properties of the site, in particular, saturated versus unsaturated sites. Repository designs in saturated and unsaturated sites differ both to take advantage of the hydrology and geochemical environment, and to address performance-related issues associated with the different repository types. The design of repositories in salt formations may differ from those in saturated and unsaturated rock, again both to take advantage of features offered by salt and to address performance-related issues.

Using long-term ventilation as part of a thermal-management strategy may be more difficult in saturated rock and in salt formations as opposed to unsaturated rock. A decision to use backfill and the type of backfill that could be used would also depend on the host rock condition (saturated, unsaturated, or salt) as would techniques for keeping the repository horizon dry during the construction and operating period.

The properties of the host rock will also have a significant influence on the design of the repository. Thermal properties of the host rock would influence design as related to thermal management. For example, it may be possible to emplace waste at a higher density in rock formations having higher thermal conductivity. Geotechnical properties would also affect the design of a repository, influencing such aspects as whether to use emplacement drifts or borehole emplacement and the size/spacing of tunnels or galleries.

The total inventory of waste and the types of waste that would be disposed of would have a significant influence on several aspects of the repository design. The effect of having to emplace a larger amount of waste is clear; a larger disposal area would be needed. However, the emplacement of a larger inventory could affect the long-term performance of the repository (i.e., a larger inventory could result in a larger estimate of dose to the public). Different strategies may be possible, such as a layered repository, if the lateral extent of suitable host rock is limited. The disposal of heat-generating wastes would invoke thermal-management aspects as an important constraint. It would probably be possible to dispose of wastes emplaced at higher density without thermal constraints being as important by disposing of wastes that do not generate heat. Co-location and segregation of different types of waste (i.e., heat-generating, different waste forms) may be an essential aspect of the optimization problem, and coupled effects would have to be considered.

Thermal management would play a key role in optimizing the repository design for the disposal of any heat-generating waste. Any repository will have thermal constraints related to the performance of the engineered barriers and the near- and far-fields. Thermal constraints (goals), waste-form thermal output, waste-emplacement strategy, and rock properties all play a role in optimizing a repository design.

As shown in Figure 33 [Ref. 65], the thermal density of commercial spent nuclear fuel is dominated by the heat generated mainly by the decay of fission products for the first 60 yr and from actinide elements thereafter. The variables associated with thermally optimizing the sub-surface repository design are illustrated in Figure 34 [Ref. 66]. While this figure was developed for the Yucca Mountain repository, it is applicable to all repositories having thermal constraints. The thermal response of the repository depends on:

- The waste-form thermal density profile (kW/equivalent MTHM), which is a function of the waste-form types, quantities, and half-lives of the immobilized radionuclides.
- The waste package thermal output (kW), which is directly proportional to the amount of waste contained in a waste package.
- The emplacement drift linear heat rate (kW/m), which is controlled by the waste-package thermal output and the waste-package spacing.
- The emplacement drift spacing (which combined with the amount of waste contained in each waste package and the waste-package spacing dictates the areal loading of the repository).

The waste-form thermal output would depend on the thermal density which in turn depends on the separation efficiency and the amount of waste that could be loaded into the waste form. This is shown schematically in Figure 35. Reducing the waste-form thermal output would allow for increased flexibility in the re-optimization of the design and operation of a repository by reducing one of the key variables affecting the repository thermal behavior.

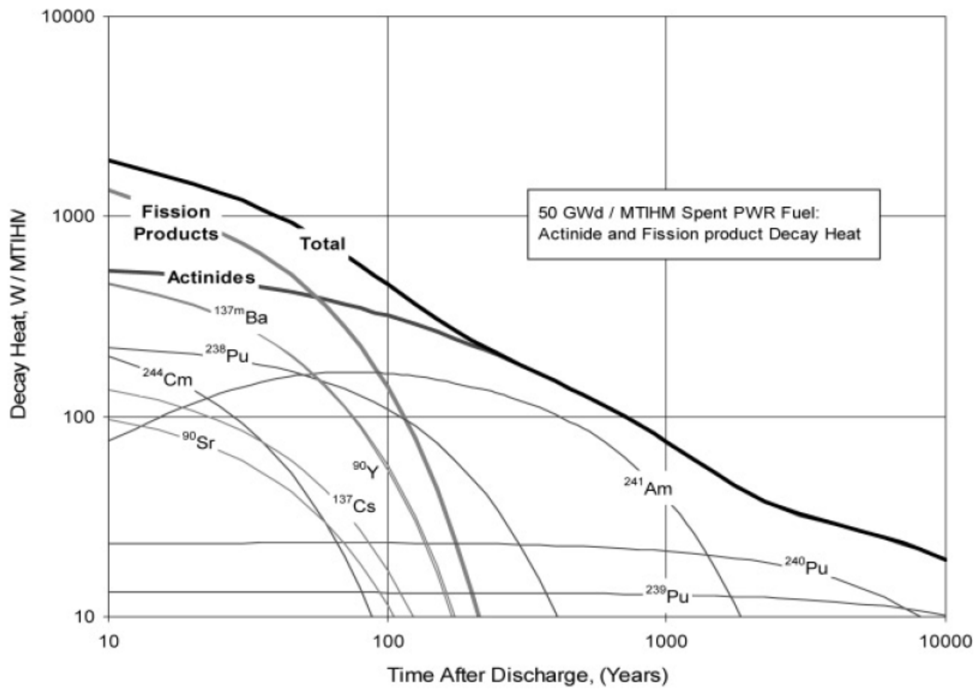


Figure 33. Dominant Decay Heat Contributors in Spent PWR Fuel Irradiated to 50 GWd/MTHM

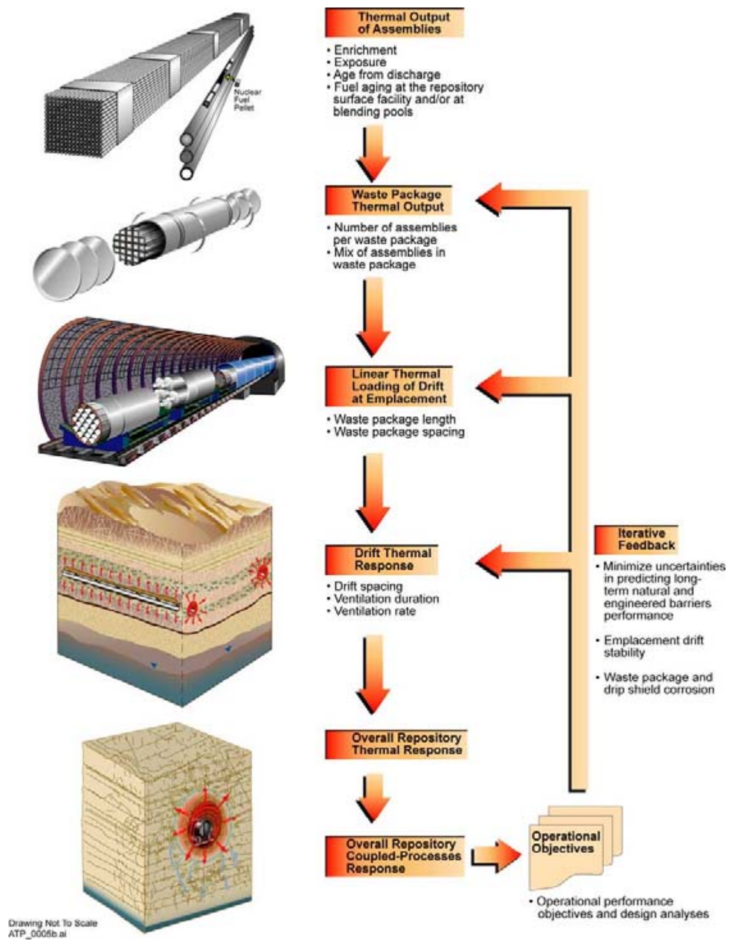


Figure 34. Variables Affecting the Thermal Performance of a Repository

Fuel-cycle scenarios that involve the removal of actinides and key fission products (Cs/Sr) would reduce thermal output, allowing for increased flexibility in re-optimization of repository design. Analyses of a repository at Yucca Mountain have shown that a significantly larger amount of waste could hypothetically be disposed of in the emplacement drifts as currently configured if a significant fraction (99.9%) of the actinides and key fission products (Cs/Sr) are removed. Thermal benefits could still be realized if certain radionuclides remain in the waste (e.g., curium) or if the recovery of radionuclides is not as efficient (e.g., < 99.9%); however, they would be less. These results are shown in Figure 36 [Ref. 65]. Similar results were observed in thermal evaluations of repositories in granite and clay conducted by the NEA and discussed in *Advanced Nuclear Fuel Cycles and Radioactive Waste Management* [Ref. 9].

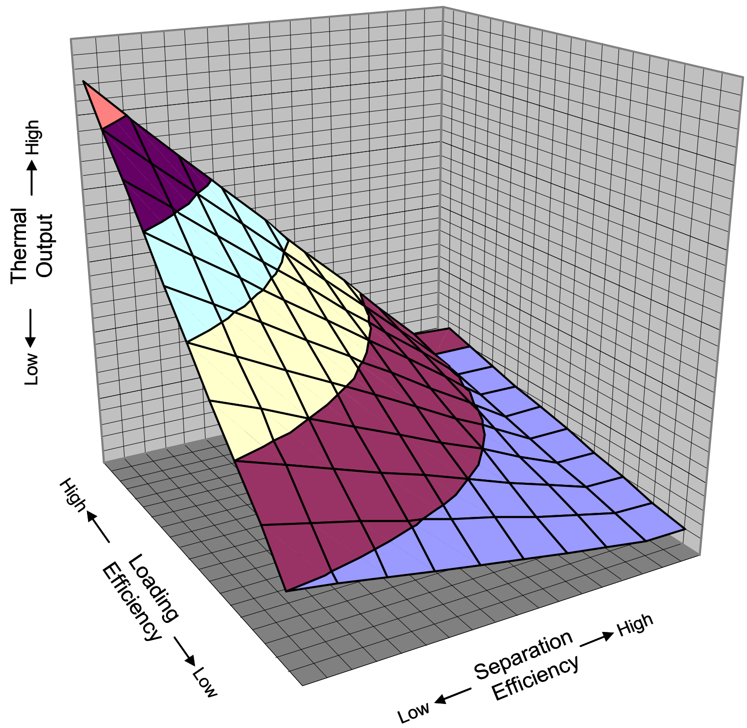


Figure 35. Schematic Showing Dependence of Waste Form Thermal Output, Separation Efficiency, and Waste-Form Loading Efficiency

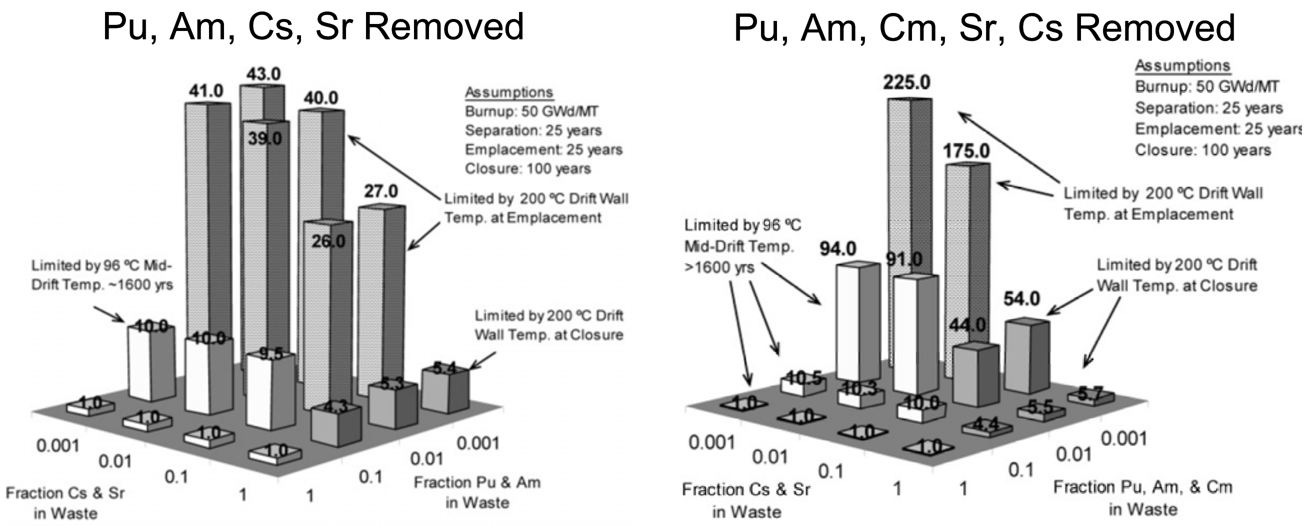


Figure 36. Results of Thermal Analyses for Hypothetical Wastes Disposed of in a Repository at Yucca Mountain: The Effects of Reduction of Waste Amount by Removal Major Heat-generating Radionuclides

The volume of the waste forms that would be disposed of will also play a role in the optimization of a repository design. It is desirable that as much radioactive material as possible be loaded into the waste forms to help reduce the overall cost of the back-end of the nuclear fuel cycle. This applies to storage, transportation, and disposal cost. Increased waste loading densities allow for waste to be disposed of at higher densities such that less repository area is needed for an equivalent amount of energy produced.

Waste-form loading efficiency, which affects waste-form volume, and thermal management are tightly coupled. Whereas it may be possible to efficiently load a waste form, thermal constraints related to storage, transportation, and disposal may limit the size of the waste form and the density that the waste forms could be emplaced in a repository.

The thermal analyses discussed above for a repository at Yucca Mountain have demonstrated that hypothetically greater than 100 times as much process wastes could be emplaced in emplacement drifts spaced 81 meters apart if separation efficiencies of 99.9% could be achieved as compared to the direct disposal of spent nuclear fuel. However, it is highly unlikely that waste forms for disposing of wastes could be developed with sufficient loading densities to realize such a volume reduction. Current estimates indicate that volume reductions on the order of a factor of 10 may be achievable.

Such a volume reduction combined with separation efficiencies greater than 99% would result in a waste form with a very low thermal density, leading to low linear heat rates within the emplacement drifts. Preliminary analyses have shown that further optimization of the sub-surface repository design is possible should a decision be made to dispose of wastes generated from an advanced fuel cycle in a repository at Yucca Mountain. This optimization could potentially include such considerations as1) moving the emplacement drifts closer together, 2) lessening the ventilation requirements (duration and rate), and/or 3) using a multi-level design for the repository.

As discussed above, the protection of public health and safety is the fundamental objective of geologic disposal of radioactive materials. The long-term behavior of the repository following closure will thus have an important role in the design of a repository. The number and types of engineered barriers in the design of the repository would depend on the hazard of the waste disposed and the attributes of the natural system. Examples of potential engineered barriers include waste packages (different materials), drip shields (unsaturated repositories), backfill, bentonite buffers, and radionuclide getters.

The configuration of the repository may also be influenced by long-term performance. For example, the thermal criteria for a repository at Yucca Mountain relate to long-term performance. These thermal criteria affect the spacing of emplacement drifts and the configuration of waste packages in those emplacement drifts. Other factors that influence repository performance may also play a role, including setback from faults, avoidance of fast pathways, and coupled processes.

Although safety is of primary importance, the cost of disposal will play a role in optimizing a repository design. Repositories are valuable "real estate," and efforts should be made to efficiently use repository space (optimize the design). Modular design would allow for future flexibility to take advantage of technological advancements, provide contingency during construction and emplacement phases, and provide flexibility for future expansion. The goal in optimizing the design of a repository should be to reduce total cost to as low as possible while ensuring high level of public and worker safety.

As described above, an important constraint on a repository design is the range of wastes intended for disposal. These are often classified according to their radionuclide content, taking into account the type of radiation which they emit and the half-lives of the constituent radionuclides – but, less logically, may also be defined by the waste source. The separation of disposal projects by waste type reflects the different hazards associated with different wastes – those that are more toxic and longer-lived requiring greater robustness of the engineered and / or natural barriers. Such an approach may however be rather simplistic and miss opportunities for optimization by co-disposing of particular wastes. Further potential for optimization becomes evident if an integrated design procedure is used – the design engineers working closely with the performance assessment, site characterization and public communication teams to ensure that the concepts developed are not only safe, but also practical, acceptable and cost-effective [Ref. 67].

Studies relating optimization of a repository design for HLW disposal have also been conducted in Japan. Regarding defining needed disposal tunnel spacing to emplace HLW from conventional PUREX of UO_2-LWR, minimum spacing was examined to satisfy both thermal constraint (less 100 C in buffer material) and mechanical constraint (stability of drift) [Ref. 29]. In this examination, total excavated volume was also considered as one of indicator for economic efficiency. Those led to optimized disposal tunnel spacing for each combination of rock type (granite, sedimentary rock) and emplacement type (vertical, horizontal) as shown in Figure 37.

The influence of recovering minor actinides and heat generating fission products on repository size was also examined. A comparison of the decay heat characteristics of HLW had been made among the various fuel cycle options [Ref. 68]. It suggested that, for a future fast breeder reactor (FBR) cycle, the removal and burning of minor actinides (MA) would significantly reduce the heat load in waste and would allow for a reduction of repository size by half.

While, to assess the benefit of the partitioning and transmutation (P&T) technology in the future nuclear fuel cycles, the repository area necessitated to dispose of the HLW was discussed quantitatively [Refs. 67 and 68]. Four options of the separation process were assumed in the analysis [Ref. 69]: (1) Conventional PUREX reprocessing, (2) Minor actinide (MA) recycling without partitioning fission product (FP), (3) Partitioning of FP without MA recovery, and (4) Full P&T for both MA and FP. The areas required to emplace waste forms per unit electricity production (m^2/TWeh) were then compared. The results showed that MA recycling significantly reduced the emplacement area for MOX SNFs from both LWR and FBR. The full P&T scheme could give further reduction of the emplacement area (i.e. the enhancement of the capacity of a repository site) independently on the fuel type, the reactor type and the cooling period. Similar conclusions were drawn in U.S. studies.

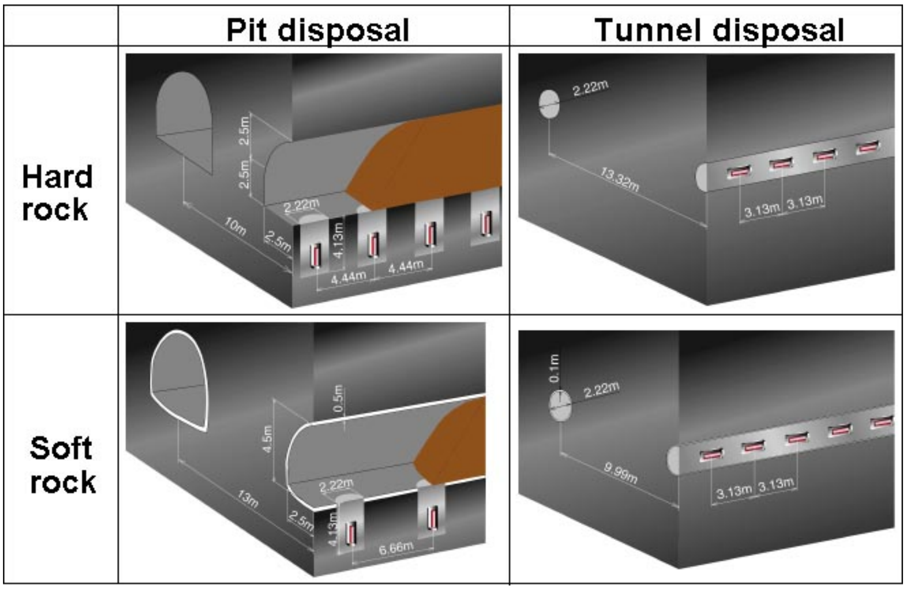


Figure 37. Disposal System Spacing in Japanese Optimization Studies

3.2.1.3 Development of Alternative Repository Design Options in Japan

There exists much international experience in the development of repository concepts for national programs under 20th century boundary conditions. A range of repository concepts proposed so far for deep geological disposal of radioactive waste are illustrated in Figure 38. A “Repository concept” can be defined as a conceptual design of all surface and underground repository structures tailored to a given siting environment, along with a description of how the repository can be constructed, operated and sealed [Ref. 30]. This also includes an evaluation of operational and long-term safety and an assessment of socio-economic impacts. The concept is dynamic, evolving with the national disposal program as it moves from early generic studies through to siting and, eventually, licensing for construction and operation. Indeed, continual evolution during the operational period is also possible, as experience is gained and technology develops.

Studies carried out over the last couple of decades have shown that, under the constraints set by national programs, many different combinations of waste type / engineered structures and geological settings can provide high levels of safety. In the past, there have been two main types of implementation strategy [Ref. 70]:

- Given a site (e.g. in the vicinity of waste production), tailor a reference disposal concept to it;
- Assuming a reference disposal concept, select a suitable site that will make its implementation easier.

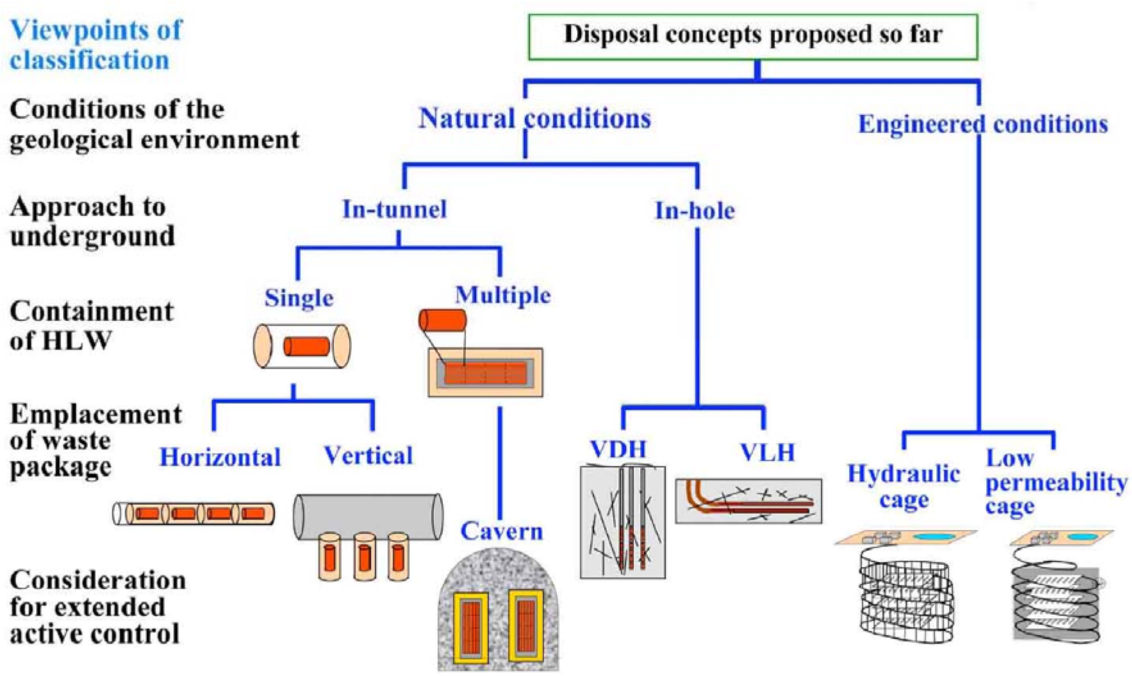


Figure 38. Range of Repository Concepts Considered in Japanese Optimization Studies

A better approach may be to specify key barrier functions, materials and operational goals and encourage flexibility to refine the design as the project moves towards implementation (or even after operation has commenced), building on experience gained. It is still necessary to define some kind of reference design to serve as a focus for planning, but accepting that this has a model nature can encourage repository optimization.

In Japan, such a flexible design process is a particular characteristic of the HLW program, where it is considered essential due to the decision to proceed with siting based on a call for volunteers [Refs. 71 and 72]. Although the original generic H12 concept and its variants, which were established for initial feasibility demonstration [Ref. 29], still remain a focus and define the main engineered barrier system (EBS) components considered, additional design options have been proposed, taking account of international developments. The following discusses the Japanese study of the development of repository design options.

The “Specified Radioactive Waste Final Disposal Act” promulgated in June 2000 sets rather clear boundary conditions on the repository concept to be developed. The Act not only restricts the waste inventory, but also specifies a staged implementation procedure, the specific requirements for a "geologically stable" site, the minimum repository depth and also a siting procedure that takes into account active communication with local communities. This legislative background can be seen to reflect the international experience in siting projects for nuclear waste repositories, where local acceptance is seen as one of the most critical parameters influencing the success of specific projects. Based on this boundary condition, the implementing organization, NUMO, decided to proceed with an approach based on soliciting volunteer host communities. In December 2002, a formal call for volunteer host communities was sent to all (over 3000) municipalities in Japan [Ref. 73]. This call for volunteers included an explanation of the requirement for geological stability in Japan with a special emphasis on avoiding volcanic areas, the vicinity of active faults and regions with high uplift and erosion.

This volunteer approach results in special challenges for the developing repository concepts. NUMO builds on the basis of two decades of work to establish the fundamental feasibility of HLW disposal in Japan which is recorded in the H3 [Ref. 74] and H12 projects [Ref. 29]. These early generic studies involved a "conventional" approach of first defining a fundamental repository concept and then showing that it would reach acceptable safety levels for generic siting environments which were considered typical of those found in Japan. For NUMO, however, the question is different: given a volunteer community that fulfills the geological stability criteria, can an appropriate repository concept be developed? In this case, long-term safety clearly still plays a critical role but, with the special emphasis on maintaining local acceptance, other factors may also be very prominent (i.e., operational safety, QA, ease of understanding of the safety case by a non-technical audience, reversibility at early stages of implementation, cost and resultant flexibility for providing local economic incentives, repository footprint, etc.).

At the first stage of evaluating potential repository concepts at volunteer sites that satisfy the geological stability requirements, NUMO will place particular emphasis on the Design Factors relating to safety and engineering practicality [Ref. 30]. Criteria specifying minimum levels of practicality and safety will be set for relevant geological formations as shown in Figure 39 [Ref. 30] and, if these are clearly not met, NUMO would reject such a site. Thus, NUMO have no intention of trying to “engineer around” problems at a location which is fundamentally unsuitable for the presently considered disposal technology. NUMO accept that, as technology develops in the future, currently exotic concepts might become practical (e.g. rock melting by very hot waste packages) which could allow such sites to be considered, but such speculative considerations are not considered in their present program.

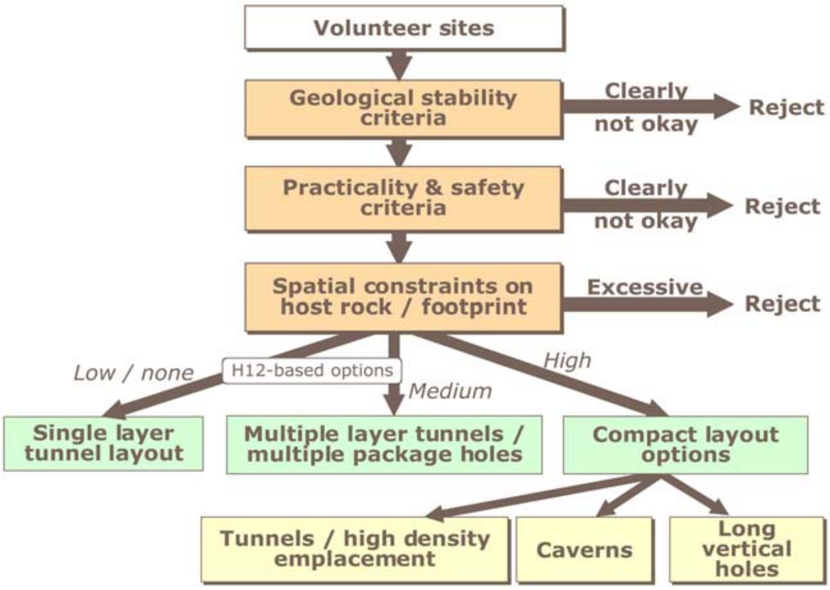


Figure 39. Evaluation of Constraints on a Site and Resultant Design Variants

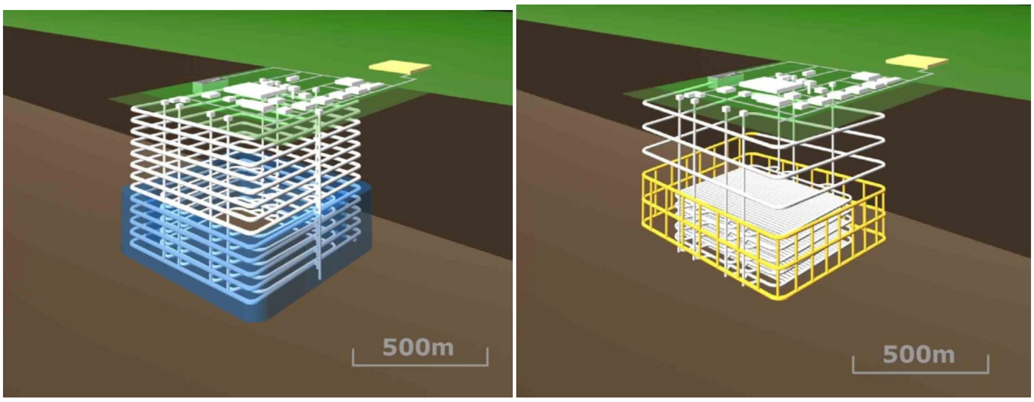
Particular challenges may arise for cases in which the volunteer site may be suitable in terms of geological stability, but contain host rocks of relatively high permeability. Although such permeability may lie in a range which is not critical for long-term safety (as hydraulic gradients may be very low in deep geological environments), inflow of water could present particular difficulties for some disposal options. Nevertheless, low permeability liners and / or the prefabricated EBS modules (PEMs) could be used to design a feasible system. The later has an advantage in terms of minimizing many practical problems with handling bentonite.

For sites where the available area of suitable host rock is limited by either geological or political constraints, multi-level emplacement panel designs can be considered. For a rock with good mechanical properties, 5-10 emplacement levels could be considered within the boundaries set by minimum repository depth (in law, 300 m; in practice probably ~500 m) and maximum depth~1,000 m, often determined by the geothermal gradient (as operations get much more complicated if ambient rock temperatures are significantly above ~50-60ºC). A particular concern which has to be addressed with such options is thermal loading, which can result in temperatures within the bentonite buffer rising above the (rather arbitrary) 100ºC limit chosen for the H12 project. There is, indeed, much evidence that even decades of exposure to temperatures up to ~150ºC cause no significant degradation of the properties of compacted bentonite / sand buffer [Ref. 75] and hence confirmation of this could be a focus for supporting R&D if such an option was under consideration. Alternatively, a PEM design could be produced in which the integrity of the external handling shell is assumed over the critical thermal period (as no alteration of dry buffer would occur). Indeed, such prefabricated modules offer many practical advantages in terms of remote handling, emplacement QA, layout flexibility, etc. and merit more detailed study of the range of designs which could be considered.

Starting from the H12 concept [Ref. 29], alternative repository concepts have been developed that increase flexibility in repository design, given potential constraints from the siting environment and/or social requirements. For the case where very compact repository designs like those described previously are under consideration, it becomes feasible to consider an alternative approach to respond to relatively high water flows by engineering improvements to the rock surrounding the repository.

External Engineered Barrier

An obvious approach for a multi-level repository with external spiral access ramps would be to use existing grouting technology to reduce rock permeability—in effect, forming a hydraulic barrier around the entire emplacement zone as shown in Figure 40a [Ref. 30]. Such a barrier could considerably reduce water inflow into the repository during the operational (and any post-operational, institutional control) phase. Reduced inflow not only makes drainage easier (considering both expected operation and perturbations such as power cuts), but reduces the perturbation of the local groundwater system, which could be caused by decades of such drainage—e.g., forming a drawdown cone that could bring oxidizing, surface waters to the emplacement levels with consequent geochemical alterations of the ambient rocks. The extent to which such a barrier could also perform a role over the long term following repository closure is difficult to assess at present, but, at least in principle, this would be possible. An alternative way to reduce flow through the repository would involve building a high permeability "hydraulic cage" around the repository as shown in Figure 40b [Ref. 30] that is similar, in principle, to the Swedish “WP Cave” concept [Ref. 76].



a) Hydraulic Barrier b) Hydraulic Cage

Figure 40. Rock Engineering Options

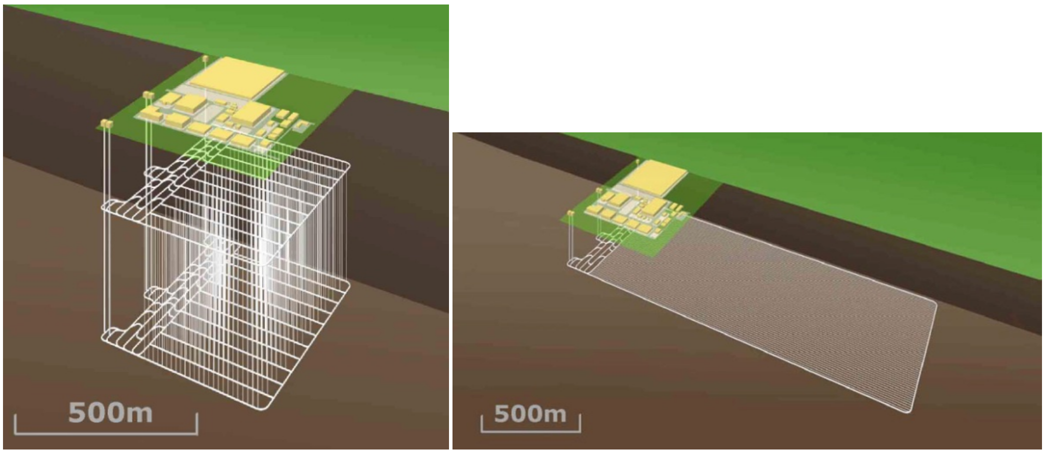
Although perhaps counter-intuitive, such a very high permeability structure diverts flow around the repository in a manner analogous to the function of the high electrical conductivity “Faraday cage” for electromagnetic radiation. This approach does not practically apply to rocks with a generally high permeability, but could be useful in cases where a high water flux is caused by a high hydraulic gradient, or cases where low water flux is associated with high water velocities because the flow is localized (e.g., in a network of fractures). This option may increase the local hydraulic perturbation during the operational phase, but could very well significantly improve long-term repository performance (as the long lifetime of a high-permeability zone may be easier to confirm than that of a low-permeability zone).

Vertical/Horizontal Borehole Emplacement

When high-emplacement densities are considered, it is possible to reduce the footprint of the repository yet further by considering emplacement in vertical boreholes. Such disposal options have been considered by several countries over the last few decades, but are generally dismissed because of risks of operational perturbations (due, for example, to waste packages jamming during emplacement). To take such concerns into account, a hybrid option can be derived in which long (e.g., 500m) emplacement holes are drilled from underground caverns, and waste packages are emplaced as pre-fabricated modules as shown in Figure 41a [Ref. 30]. Preliminary assessment of such options indicates that it may be possible to design around most major operational perturbation scenarios.

Again, high thermal loading becomes a factor that has to be very carefully considered when assessing post-closure safety.

The same basic concept of borehole emplacement can be used for horizontal/sub-horizontal emplacement, shown in Figure 41b [Ref. 30] in locations where more conventional repository operational schemes may be difficult—e.g., in relatively weak rocks in a coastal location. Again, consideration of practicality of emplacement leads to focusing on prefabricated EBS designs.



a) Vertical Boreholes b) Horizontal Boreholes

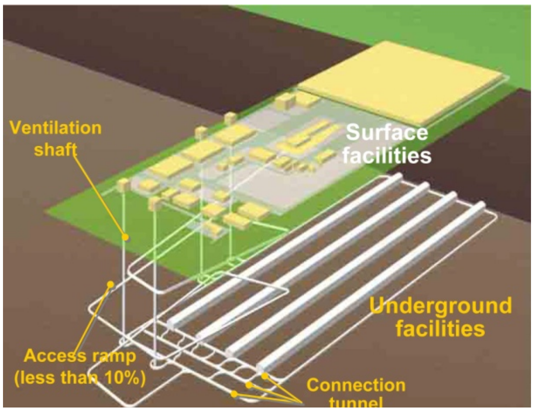
Figure 41. Borehole Options

Given the high weighting assigned to public acceptance, it might well be necessary (or desirable) to move away from the fundamental disposal philosophy contained in all of the examples illustrated above. These concepts are all based on the idea that waste is sealed in place as quickly as possible for the very good reason that a sealed disposal system is inherently more stable mechanically, hydrologically, and geochemically. Any "monitoring" that is included in such designs must avoid perturbing the engineered barriers and hence, with present technology, cannot really measure any parameter relevant to long-term safety.

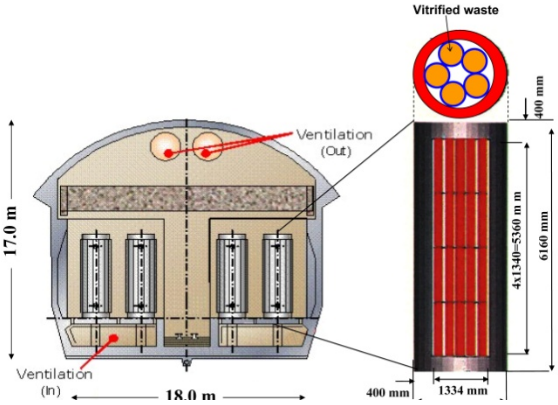
The lack of precedent for HLW disposal has, however, led to a call for slow, staged implementation that is monitored and easily reversible for a long institutional control period. To provide a disposal system that meets this requirement requires alternative repository design options, which could have additional attractions from the viewpoint of long-term safety.

Cavern Extended Storage

An example of cavern extended storage is illustrated in Figure 42 [Ref. 77]. Massive multi-purpose (transport/ storage/disposal) casks containing many HLW waste packages (typically around 20) are, possibly after a period of surface storage, transferred to an underground open cavern for a long-term period (typically up to ~300years) during which they are fully monitored and can be retrieved at any time. When a decision is made to stop monitoring, the open caverns can be simply backfilled to form a permanent disposal facility. This option thus can contribute to verifying the technical feasibility of reversibility/retrievability.



a) Example Layout of CES Repository



b) Cross-Section of CES

Figure 42. Outline of the CES Concept

This long-term institutional control certainly raises issues associated with operational safety during this period (especially considering perturbations such as earthquakes, civil disturbances, etc.) but has also significant advantages due to its small footprint and, possibly, the long-term performance of the sealed system [Ref. 77].

The repository design options discussed above are summarized in a "Repository Concept Catalogue", which illustrates components that can be assembled into complete systems, maximizing design flexibility [Ref. 30]. Some examples from the catalogue are shown in Figure 43 [Ref. 30].

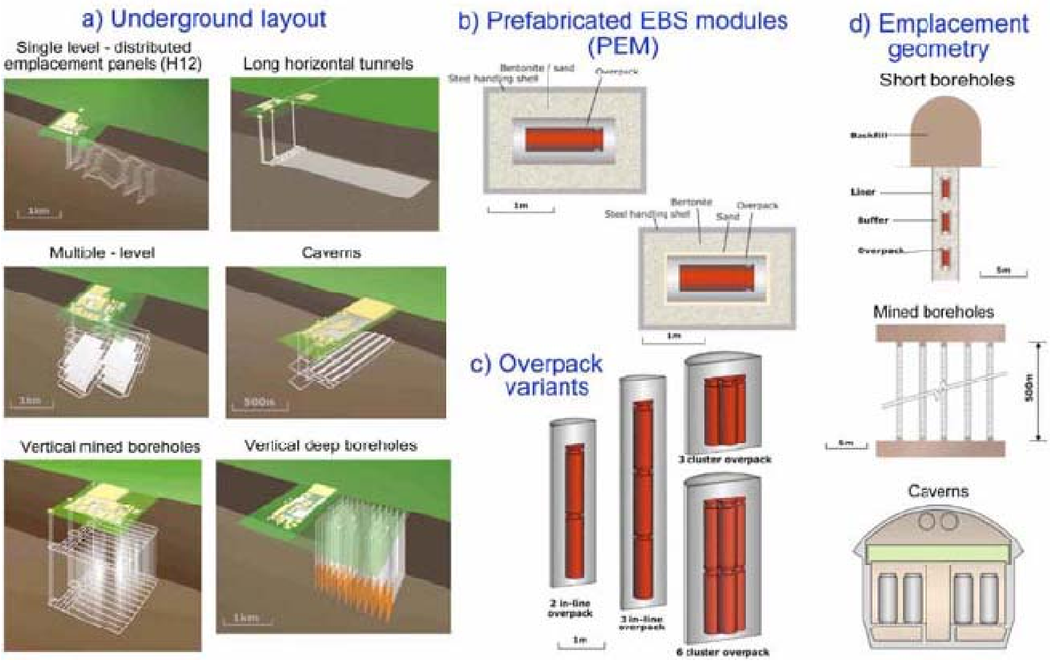


Figure 43. Examples of Repository Design Concepts

3.2.1.4 Framework for Analysis of Repository Concepts

A comprehensive analytical framework, which covers not only long-term safety but also engineering issues, socio-economic aspects, etc., is desired for the development of repository concepts. The development should be efficient and flexible in the iteration during the siting program (or even after operation has commenced). Its process should be carried out in a clear and transparent manner. In this study, a structured approach is taken to satisfy these requirements. It is also expected that this approach will provide a systematic archive of all information supporting the decision-making in the development.

Figure 44 is a schematic view of the proposed idea for an analytical framework of different options in comparison with each other in terms of application to the specific sites. From these comparisons, advantages and disadvantages of each design concept is identified. As the first step for analysis, it is necessary to specify the reference option for comparison, and then various options are analyzed through the filter of design factors. The design factors should be set from the viewpoint of repository implementation.

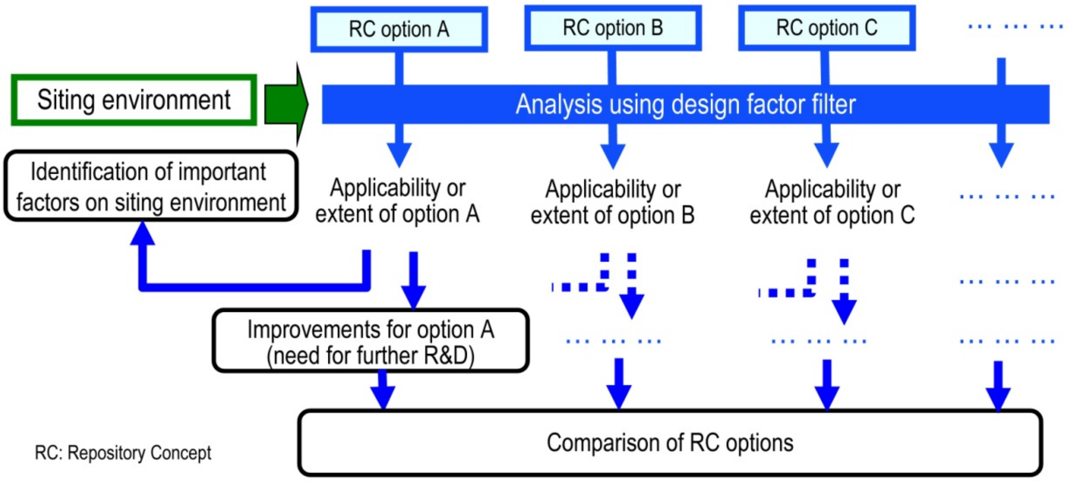


Figure 44. Proposed Idea for Analytical Framework of Repository Concepts

Design Factor Filter

Long-term safety is an essential requirement of all designs, but a set of factors should been taken into account to address issues bearing directly on the chosen design since the repository concept is required to be not only technically feasible but also practical. For this purpose, it is useful to analyze the repository concepts from the viewpoint of the “design factors,” which are explicitly defined by the Nuclear Waste Management Organization of Japan [Ref. 30]. The design factors are classified into:

- Long-term safety:
  - perspectives on robustness of the post-closure safety case
- Operational safety:
  - conventional and radiological safety of construction, operation, closure, and decommissioning
- Engineering feasibility/quality assurance:
  - fundamental feasibility of construction and operation to defined quality levels

- Engineering reliability:
  - ➢ practicality of implementation in view of boundary conditions (e.g., emplacement rate) and robustness with regard to operational perturbations
- Site characterization/monitoring:
  - ➢ effort required to satisfy technical requirements for site characterization and monitoring data
- Retrievability:
  - ➢ ease of waste package retrieval after emplacement
- Environmental impact:
  - ➢ extent of all environmental impacts associated with repository implementation
- Socio-economic aspects:
  - ➢ factors contributing to costs and acceptance by all key stakeholders.

These factors are always explicitly considered when designs are being developed, but it is recognized that the weighting of different factors will change as a project moves from a first conceptual phase towards implementation at a specific site. Their attributes allow the analysis to be well structured and characterized in detail.

As an example of the analysis, the Japanese reference repository concept illustrated in Figure 45 [Ref. 29] can be examined in terms of these design factors. Referring to different type of repository options listed in the NUMO's Repository Concept Catalogue [Ref. 30], some areas where design improvements are possible could be identified discussing pros and cons that are discussed below.

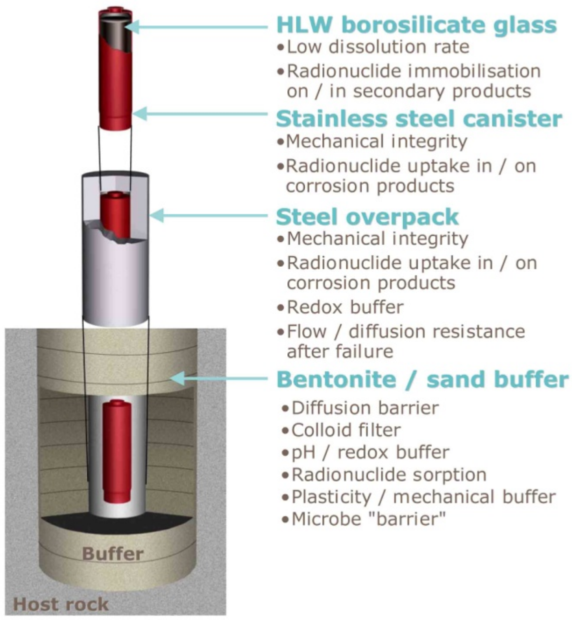


Figure 45. Japanese Reference EBS Concept (H12 concept)

(1) Multiple Overpack Containing Two or Three Containers of HLW

Figure 46[Ref. 30] shows the placing of several vitrified waste packages in a single overpack. Engineering feasibility should not be greatly influenced as designs containing two or three containers of HLW are similar in terms of dimensions, weight, and thermal loading to particular overpacks studied for the direct disposal of SNF[Refs. 78 and 79]. Reducing the number of emplacement operations could ease engineering practicality, and, if the repository could be made more compact, site characterization requirements and environmental impacts could be reduced. Such variants could reduce costs, but a more rigorous analysis would be needed to verify that there was no detriment to either long-term or operational safety.

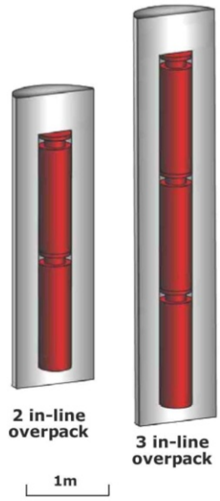


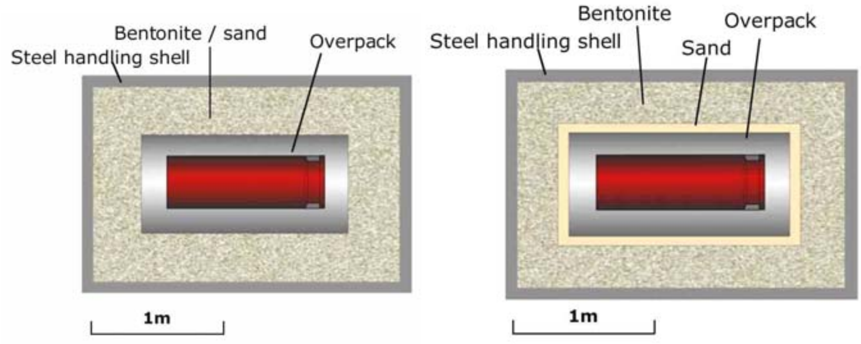
Figure 46. Multiple Overpack Containing Two or Three Containers of HLW

(2) Prefabricating the Main Components of the EBS

Figure 47 [Ref. 30]shows some examples of the PEMs. The difficulty of quality assuring the EBS and, in particular, handling compacted bentonite under high humidity conditions, has indicated that PEMs can improve engineering practicality as noted as an option in H12 [Ref. 27]. SKB and Posiva are testing such a concept at present [Ref. 80] and past work in the United States indicates that such designs may also be easier to retrieve [Ref. 81].

(3) Variable layouts

Figure 48 [Ref. 28] shows some examples of variable layouts to make the best use of available host rock. NUMO has already extended the H12 vertical emplacement concept, which considered only a single waste package in a disposal pit, to "multiplex" options in which two or three packages are stacked in a single pit [Ref. 28]. In principle, this can be extended further to longer boreholes containing many waste packages or, indeed, deep boreholes drilled from the surface [Refs. 37 and 76]. Similarly, horizontal emplacement panels can be stacked to make better use of a thick host rock formation or extended as long tunnels to use a formation with limited access (e.g., an underwater formation accessed from land). For very large, multiple waste container overpacks, a cavern disposal option could be considered. Such layout options have clear pros and cons in terms of the design factors above, but a detailed evaluation to allow their direct comparison can only be done on a site-specific basis.



H12-Based Design

- Waste canister
- Overpack
- Bentonite/sand buffer
- Steel handling shell

Variant 1 with Separated Bentonite and Sand

- 100% bentonite buffer.
- Addition of a sand layer around the overpack:
  - acts to remove the possibility of overpack sinking by creep
  - increases radionuclide transport resistance during the period of H_2 gas generation.

Figure 47. Examples of Prefabricated EBS Modules

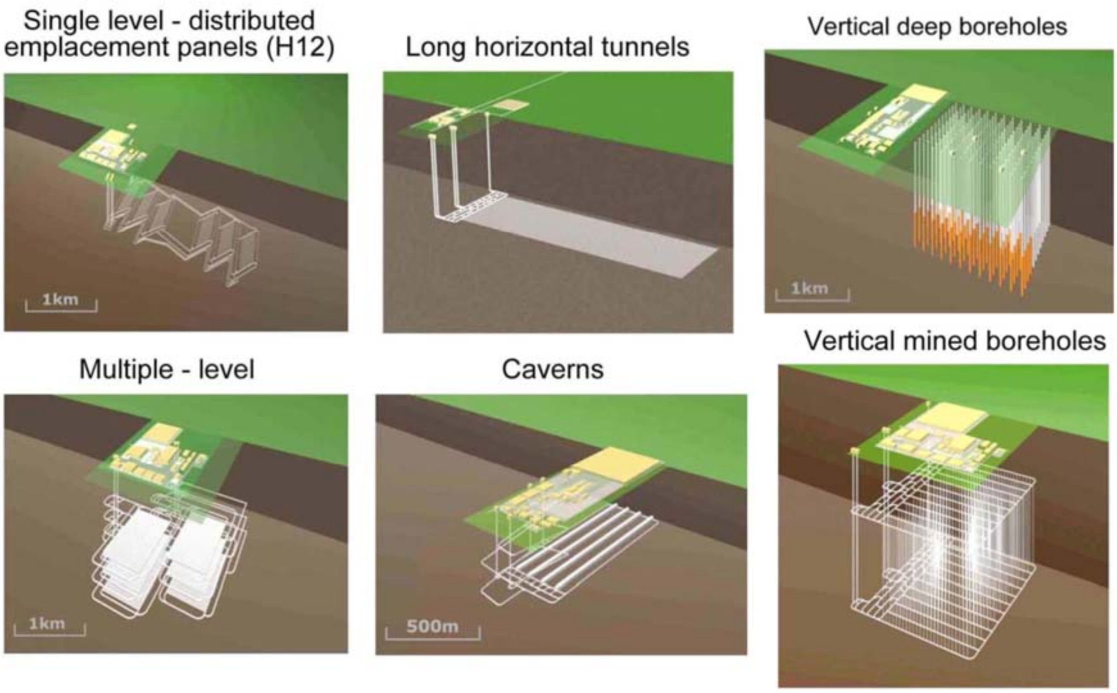


Figure 48. Examples of Variable Layouts

A structured approach for iterative development of repository concepts as a site characterization procedure has been developed taking the “design factors” into account, which not only allows the repository design to be tailored to the site but also feeds back to allow optimization of the characterization program [Ref. 30]. This approach will also include development of top level tools for information collation and synthesis, which will feed back to allow prioritization and optimization of the associated national R&D program. The work involved will be fully documented, not only to provide a mechanism for quality assurance via expert review but also to inform key stakeholders of progress, with special emphasis on the local populations of the areas investigated.

Ideally, the output of such repository concept studies will be the definition of a range of potential designs / layouts for a number of different sites. Sensitivity analysis associated with safety assessments will identify the key site parameters which determine the practicality and long-term safety of each. If the number of sites/designs is impractically large, some form of multi-attribute (or multi-criteria) decision analysis (MAA) may be used to rank options, including consideration of additional factors such as cost and acceptability (e.g. ease of monitoring, reversibility).

The MAA method allows a traceable evaluation of options against a set of agreed criteria. Using such structured methods, it is also possible to include stakeholders with a wide range of skills and experience in identifying and comparing options. The set of criteria should be comprehensive and, to the extent possible, each criterion should be discriminatory, unambiguous and independent. In the analysis, each option is scored against the criteria and the sum of the scores then indicates the preferred option. The analysis may be enhanced by attaching weightings to the scores that account for stakeholders’ values on the relative importance of the criteria. The criteria for evaluating design options include those relevant to factors mentioned above, e.g. long-term safety, operational safety, etc. More detailed criteria may also be considered according to program specific requirements. Examples of further potentially relevant criteria include waste retrievability and repository “footprint”. When using MAA methods, care should be taken not to mix “exclusion” criteria (i.e. those that result in on/off decisions).

In the case where the number of sites is very small, it may be decided to include a step of further evaluation of design variants to determine if other suitable options exist – particularly if some or all of the sites are marginal.

The variety of requirements and constraints inevitably requires a structured approach for developing repository concepts. The repository design process in Japan, for example, is structured to respond to the boundary conditions set by law and a decision to proceed with site selection via a volunteer process. A wide range of options is being considered, which maximizes flexibility. Nevertheless, as volunteer sites are investigated, the range of options should be narrowed down rapidly to allow focusing of limited R&D resources. At present, the focus of work is on the clarification of design options with a high emplacement density and preparation of the information needed to identify relevant repository concepts during the process of literature study of volunteer sites.

3.2.2 Advanced Waste and Storage Form Development (Task 2-2)

This section summarizes advanced waste form development activities (task 2-2). A comparison of waste forms and technologies for the United States and Japan is presented in Section 3.2.2.1. Possible collaborative research and development are discussed in Section 3.2.2.2. Potential collaboration areas and activities for Phase II is then discussed in Section 3.2.2.3.

3.2.2.1 Comparison of Waste Forms and Technologies

This section discusses waste forms and technologies being researched and/or under development in the United States and Japan.

United States

Comparison of waste forms and technologies to form a baseline was complicated by the wide variety of radionuclides considered and the differences between aqueous and electrochemical waste streams. For example, both reprocessing concepts release tritium, but iodine and carbon are released during aqueous dissolution. However, they are retained in the molten chloride salt used by Echem. Also, some judgments are very subjective; powder processing is used commercially every day to fabricate fuel, but it may be unacceptably problematic to consider for remote production of intensely radioactive Cs/Sr waste forms. With those limitations in mind, the processing options were evaluated based on the parameters in

Table 5. Waste-treatment experts from across the United States were enlisted to evaluate the options using these criteria with only slight modifications to fit the wastes being considered.

Potential waste streams that could be generated in an advanced fuel cycle are summarized inTable 6. Note that options are retained in several instances pending process development. Waste forms recommended are based on current assumptions, including the general UREX+1a and Echem process flowsheets, current U.S. regulations, and the overriding assumption that the text in the Nuclear Waste Policy Act will be reconsidered such that the “highly radioactive material resulting from the reprocessing of spent nuclear fuel” will not all have to be disposed of as HLW disposal in a geologic repository. However, in the U.S., the Secretary of Energy will convene a blue-ribbon panel of experts to evaluate alternative approaches for meeting the Federal responsibility to manage and ultimately dispose of spent nuclear fuel and high-level radioactive waste from both commercial and defense activities.

If the waste is not GTCC or is allowed to decay to less than GTCC, it enters the sphere of commercial LLW management. As noted above, the Cs/Sr stream from reprocessing either from UREX+ or Echem would contain barium, which could render this stream a mixed waste. This category could also include decontamination solutions, solvents, contaminated equipment, etc. Commercial power generation has all but eliminated the generation of mixed waste, but when reactors aredecommissioned, there is expected to be more waste containing PCBs and asbestos. After treatment for the RCRA contaminant or condition, these wastes can be disposed of. Currently, there are limited disposal facilities for routine disposition of Class B/C wastes, and they are commonly blended or load averaged with Class A wastes as permits allow. The balance of plant operations for fuel reprocessing will significantly add to the nominal rate of Class A/B/C waste generation today. These wastes include protective clothing, failed equipment, filters, water, and offgas-treatment wastes, containers, etc. This is one penalty that cannot be avoided with fuel reprocessing, but it can be mitigated by judicious design and management. In the future, licensed facilities will be needed to dispose of Class A/B/C wastes.

The Low-Level Radioactive Waste Policy Act (LLRWPA) establishes responsibilities for the disposal of LLWs for both the states and the Federal Government [Ref. 82].Each state, either by itself or in cooperation with other states, is responsible for the disposal of:

- Class A, B, or C radioactive wastes generated within the state
- LLW that is generated by the Federal Government except for waste that is owned or generated by DOE
- Class A, B, or C radioactive waste generated outside the state and accepted for disposal.

Table 5. Waste Form Evaluation Criteria

Commercial Practicability (weighted 40%)
Technical Practicality
Flexibility/Robustness
Complexity of Process
Scalability
Waste Loading
Processing Cost
Transportability
Secondary Waste Generation
Technical Maturity (weighted 10%)
Development Cost
Schedule to Implement
Process/Product Maturity
Waste Form Performance (weighted 40%)
Thermodynamic stability
Rad/Mech/Thermal Stability
Chemical Durability (e.g., TCLP, PCT)
Predictable Performance
Waste Stream Specific Criteria:
Heat transfer and degradation
RedOx Sensitivity
Resistance to degradation by decay, valence change, atomic size, chemistry
Resistance to radiolysis effects gas generation from water, degradation of the waste form
Ability to treat more than one waste
Experience with disposal of similar materials
Stakeholder Acceptance (weighted 10%)
All processes designed to meet standards

Notes: TCLP – Toxicity Characteristics Leaching Procedure
PCT – Product Consistency Test

Table 6. Potential Waste-Form Concepts

Waste Stream	Waste Form(s) UREX+	Waste Form(s) Echem
Tc	Metal alloy, possibly containing UDS and transition metal FP. Alloy may require Zr/Fe, which could come from cladding and hardware.	Metal alloy containing UDS and transition metal FP. Alloy may contain cladding and may require supplemental Zr or Fe, which could come from additional cladding and hardware. Will also contain used metal fuel baskets used in electro-reduction and used niobium crucibles.
Cs/Sr	Glass or ceramic—process design should consider ramifications of high heat, and high radioactivity; powder handling should be avoided.	Glass-bonded sodalite. Will contain ^{129}I and possible ^{14}C carried over from electro-reduction.
Ln	Glass—borosilicate glass if segregated as separate Ln stream. Ln/FP borosilicate glass if Ln and FP streams are combined.	Borosilicate glass if segregated. Glass-bonded sodalite if combined with Cs/Sr.
FP	Metal alloy potentially combined with Tc and UDS. Borosilicate glass if combined with lanthanides.	Included in metal waste form with Tc described above.
UDS	Metal alloy potentially combined with Tc and FP.	Included in metal waste form with Tc described above.
Metals—Cladding/Hardware	Compacted metal. Metal ingot if cost effective.	Compacted metal. Metal ingot if cost effective.
Tritium	Grouted tritiated water (HTO).	Grouted tritiated water (HTO).
Iodine	Grouted or heat-treated silver zeolite.	Glass-bonded sodalite w/Cs/Sr and/or FP.
Carbon-14	Grouted $Na/CaCO_3$.	Grouted $Na/CaCO_3$ with residual in glass-bonded sodalite w/Cs/Sr and/or FP.
Kr	Pressurized gas cylinder w/wo Xe.	Pressurized gas cylinder w/wo Xe.

The Federal Government is responsible for the disposal of:

- LLW owned or generated by DOE
- Any other LLW with concentrations of radionuclides that exceed the limits for Class C radioactive waste.

The issue of adequate commercial LLRW disposal capacity has been recognized by several organizations. In 2004, the GAO evaluated LLRW disposal capacity in the United States and concluded that [Ref. 83]:

Although no shortfall in disposal availability appears imminent, uncertainties remain about future access to disposal facilities. Even with the prospect of new disposal options, there is no guarantee that they will be developed or be available to meet national needs for class B and C wastes disposal. While LLRW generators have options available to mitigate any future disposal shortfall, including storing waste, storage is costly and it can lead to increased safety and security risks. Therefore, continued federal oversight of disposal availability and the conditions of stored waste is warranted.

The GAO recommended that:

> *The Congress may wish to consider directing NRC to report to it if LLRW disposal and storage conditions should change enough to warrant congressional evaluation of alternatives to ensure safe, reliable and cost effectiveness of disposal availability.*

In commenting on the GAO report, NRC stated that the GAO provided an accurate summary of the current LLRW disposal activities at that time, of which there has been no significant change, and potential issues that may arise in the future. NRC stated that given the failure to develop any new sites under the LLRWPA, the GAO should explore alternatives that "would potentially provide a better legal and policy framework for new disposal options for commercial generators of LLRW."In response, the GAO stated that such an evaluation by them was not required "as long as the NRC places no time limits on storage and provides assurance that it is safe and secure, and any shortfalls in disposal capacity would be managed in the short-term."The GAO believes it is NRC's responsibility to report to Congress on when such an evaluation is needed.

Thus, there has been no effort to evaluate alternatives for increasing the commercial capacity for commercially generated LLRW. This is further evident by the NRC staff determination in SECY-07-1080 [Ref. 84] that such activities are of low priority. The issue continues, however, as is evident in an article published in the May/June 2007 issue of Radwaste Solutions [Ref. 85].That article concludes that there is a crisis in regard to commercial disposal capacity.

Individual states and compacts would be responsible for the disposal of LLRW generated by commercial nuclear reactors as established in the LLRWPA. However, the responsibility (commercial or Federal) for disposing of LLRW that would be generated from recycling facilities is unclear. The responsibility for disposing of LLRW generated by uranium enrichment facilities established in the USEC Privatization Act [Ref. 86] may establish precedent for the responsibility of disposing of LLRW generated from recycling facilities. The USEC Privatization Act states that:

- the DOE, at the request of the generator, shall accept for disposal LLRW generated by either United States Enrichment Corporation or any person licensed by the NRC to operate a uranium enrichment facility under applicable sections of the Atomic Energy Act;
- no State or interstate compact shall be liable for the treatment, storage, or disposal of any LLRW attributable to the operation, decontamination, and decommissioning of any uranium enrichment facility; and
- a generator may enter into agreements for the disposal of LLRW with any other person other than the DOE that is authorized by applicable laws and regulations to dispose of such wastes.

LLRW generated by uranium enrichment facilities can be disposed of either in Federal or commercial disposal facilities. However, LLRW disposal is expected to be in commercial disposal facilities. Revision 2 of the Low-Level Waste Disposal Capacity Report [Ref. 87] indicates that the LLRW generated at the Portsmouth and Paducah gaseous diffusion plants will be disposed of in both Federal and commercial disposal facilities. The environmental impact statement for the National Enrichment Facility states that all LLRW (Class A only) would be disposed of in a commercial facility [Ref. 88].

The issue of LLRW disposal capacity will ultimately need to be solved independently of the fuel cycle scenario chosen. The existing 104 nuclear power plants, other generators of LLRW, and any new plants that are constructed will ultimately need capacity to dispose of their LLRW. However, the deployment of an advanced nuclear fuel cycle is predicated on the deployment of new reactors and recycling facilities. The capacity for disposing of LLRW will be needed to support a growing nuclear enterprise. Market forces may solve this issue without intervention, but it may be prudent for the program to be supportive of and involved in efforts to evaluate alternatives for verifying LLRW disposal capacity. It may also be necessary to work to verify that a disposal pathway exists for LLRW generated by new reactors deployed as part of an advanced nuclear fuel cycle and generated at recycling facilities should there be a desire to dispose of the waste in a commercial facility.

The decommissioning of nuclear reactors in the United States generates LLRW of all classes (A, B, C, and GTCC). The spent nuclear fuel is currently stored on-site pending the completion of a geologic repository. Currently 14 nuclear reactors are being decommissioned in the United States [Ref. 89]. Ultimately, the current generation of reactors will all have to be decommissioned. This will generate large volumes of LLRW, potentially further challenging the capacity to dispose LLRW in the United States. As discussed above, 84 of the 104 operating reactors in the United States have either received a 20 year extension to their operating license, submitted an application for license extension, or plan to submit an application. Thus, the majority of the decommissioning wastes will not be generated for an additional 20 years, deferring the issue of disposal capacity.

Japan

A preliminary survey of studies on advanced waste forms in Japan was carried out. Table 7 summarizes the results. Each study is on a different level of detail; for instance, most ongoing studies with specific activities are addressed in the context of the current LWR-PUREX cycle (Table 7, "Yes" on the fuel cycle (1) column). These studies, aimed at addressing the potentially critical issues for disposal, may have common ground for part of the advanced cycles. A few studies on waste and storage forms specific to advanced cycles (Table 7, without "Yes" on the fuel cycle (1) column) are still basic.An overview of these studies is described below.

It is a matter of concern that I-129 from the spent iodine adsorbent and C-14 from hulls and end-pieces are likely to dominate the dose for TRU waste in deep geological disposal facilities because of their soluble and less-sorbing properties. Supposedly, these wastes have been solidified with cement. Since 2000, alternative waste forms, in order to increase safety margins and flexibility for various geological environments, have been developing in the RWMC/METI R&D program [Ref. 90].

<u>Immobilization of I-129</u>

I-129(half-life = 15.4 million years) cannot be expected significantly to decay by relying on the engineered barrier system containment because of its long half-life. As it could easily migrate along with groundwater flow in the repository and through the host rock, dispersion by natural barrier should be identified as an effective function for reducing the peak dose of I-129.

As an effective countermeasure for reducing the peak dose, the controlled-release of iodine from the waste package was proposed. A releasing period of 100,000years was designed as a target value to reduce the maximum peak dose more than one order of magnitude, considering various geological conditions.

Table 7. Summary of Studies on Advanced Waste Forms in Japan

Category	Waste	Applicability to the fuel cycle		R&D on advanced waste form	Key issues on the waste disposal
		(1)	(2)		
LLW	Iodine adsorbent, Hulls and end-pieces	Yes	Yes (?)	Iodine and carbon confinement[*1]	I-129 and C-14 with highly contribution to the dose due to their properties of soluble and less-sorbing
	low-level concentrated liquid waste	Yes	Yes (?)	Nitrate removal and decomposition [*2]	Uncertainty regarding the behavior of nitrate in the deep geological environment
	Combustible/ incombustible waste	Yes	Yes (?)	Decontamination and volume reduction	Large volume Unsorted (e.g., metal and organic materials mixed)
	Waste specific to FBR and/or P-T technology		Yes	Under survey	Under survey
HLW	HLW / Vitrified waste :Borosilicate-glass	Yes	Yes	Volume reduction and increasing waste loading ratio, melter life-time enhancement [*3]	Future variable in volume, heat, inventory, physical-chemical property, etc. Compatibility with repository system concepts
	Waste specific to FBR and/or P-T technology		Yes	Cs and Sr stabilization	

(1): the current envisaged cycle (LWR+PUREX)
(2): the reference advanced cycle and its options
*1: RWMC/METI R&D program since 2000[Ref. 90].
*2: JAEA/METI R&D program since 2007 [Ref. 91].
*3: JAEA/METI R&D program since 2006 [Ref. 92].

So far, eight candidate concepts have been investigated as shown in Figure 49 (modified from [Ref. 31]). Four of these, rock matrix solidification (quartz and/or alumina), BPI glass, hardened alumina cement solidification, and Cu matrix solidification, have been chosen to further continue R&D. Fundamental R&D (pretreatment condition, treatment condition, mechanical properties research, and modeling) and preliminary evaluations for the four concepts have been already finalized. Each model (e.g., leaching model for rock matrix, glass, and Cu matrix; sorption model for hardened alumina cement) indicated that the target releasing period would be possible. Engineered-scale test and improvement of modeling reliability are expected as further issues.

Long-Term Containment of C-14

A leaching experiment using activated cladding materials showed that C-14 took organic chemical form and less sorbing property. The chemical behavior of C-14 released from metal waste is not well understood; however, a realistic model and reliable data are required to find an effective method of C-14 containment and to improve the evaluation.

The purpose of R&D is alternative package development for C-14 containment for a long enough time for the peak dose to effectively be reduced by decay. The target was determined as over 60,000years, which is ten times of half-life. Two alternative packages have been developed as shown in Figure 50 (modified from [Ref. 31]).

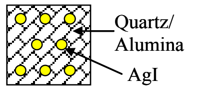
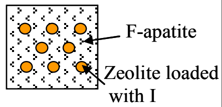
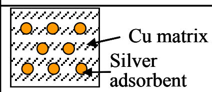
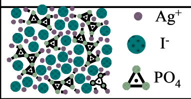
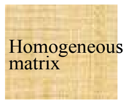
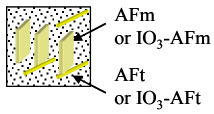
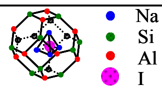
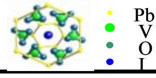
immobilization		Chemical composition: material	Chemical composition: Waste form	Concept of iodine immobilization	Immobiliza-tion concept	Release period expected
Crystalline matrix	Rock	Spent silver adsorbent (Silica gel/Alumina+ AgI)	SiO_2(quartz), AgI (I content:14wt%)		Diffusion model: D=1E-20(m^2/s)	>1.0E+5(y) size: ϕ25.5×24cm
	Apatite	Fluoro-apatite ($Ca_{10}(PO_4)_6F_2$): Zeolite loaded with I = 85:15 (wt%)	Fluoro-apatite ($Ca_{10}(PO_4)_6F_2$): Zeolite loaded with I (I content: 2wt%)		-	-
	Copper matrix	Cu powder: Spent silver sorbent = 50:50 (vol%)	Cu,powder :Spent silver adsorbent (I content: 0.7wt%)		Leaching model: Rate= 9E+8(Bq/y)	~1.0E+5(y) size: ϕ82×173cm
Glass	AgI glass	AgI: $Ag_4P_2O_7$ =3:1 (mole ratio)	$3AgI\text{-}2Ag_2O\text{-}P_2O_5$ glass (I content: 30wt%)		Leaching model: Rate= 1E-8~-7 (g/cm2/d)	>1.0E+5(y) size: ϕ43×32.5cm
	BPI glass	Glass frit: $BiPbO_2I$ =91:9 (wt %) ・Glass frit; $PbO:B_2O_3:ZnO$ =65:30:5 (mol %)	$6.5PbO\text{-}3B_2O_3\text{-}0.5ZnO$ glass, $BiPbO_2I$ (I content: 2wt%)		On-going	-
Cement	Hardened cement	・Alumina cement/ $CaSO_4 2H_2O$=100/15.5 ・$NaIO_3$ (0.4mol/L) sol./cement=0.56	AFm:10wt% AFt :46wt% $Al(OH)_3$:44wt% (I content:1.85wt%)		Kd model : Kd >1000ml/g	>1.0E+5(y) (Reference case)
Synthetic mineral	Synthetic sodalite	$NaAlO_2:SiO_2:NaI$ = 3:3:1 (mole ratio)	$Na_8(AlSiO_4)_6I_2$ (I content: 11wt%)		-	-
	Synthetic apatite	$PbO:V_2O_5:PbI_2$ = 9:3:1 (mole ratio)	$Pb_{10}(VO_4)_6I_2$ (I content: 7.2wt%)		-	-

Figure 49. Summary of Iodine Immobilization Concepts Aiming at a Releasing Period of 100,000 years

The so-called "ultra high performance concrete package (UHPC)" employs high-strength and low-permeable concrete (compressive strength 200MPa, permeability Kw=10^{-19} m/s). As joints and cracks could be pass-way for groundwater intrusion, a monolithic manufacturing method has been developed. As an evaluation, a chemical-hydrological coupling analysis showed that the water infiltration would be limited within 14cm, and the loss of thickness by chemical degradation was within 4 cm after 60,000years. This means if the thickness around 20cm is specified to the package wall, the target would be possible. For the other one, a titanium alloy-carbon steel composite package, a Ti-alloy containing Pd is used as a corrosion-resistance layer. The life-time of the package over 60,000 years was estimated by considering the corrosion behavior of the Ti-alloy.

An engineered scale test and improvement of the model are required as further R&D.

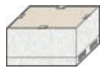
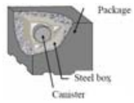
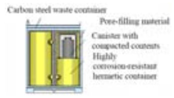
Specification	The current reference package	Alternative package for long-term containment of C-14	
	"Package B"	Concrete package	Metal package
Basic concept			
	Canister storage package	Package sealed with ultra high performance (high-strength and low-permeable) concrete	Package sealed with Ti-alloy container
	Not considered	60,000 years by prevention of water infiltration	60,000 years by corrosion prevention
Size	1.5m×1.5m×1.1m	1.8m×1.8m×1.6m	1.038m×1.038m×1.726m
Material	Main body: carbon steel Filling: cement mortar	High-strength and ultra-low-permeable concrete	Handling package: carbon steelmain body: Ti-alloy / carbon steel composite Filling: silica sand
Weight	About 7 ton	About 13 ton	About 6.5 ton
Number of stored waste	4 (canister)	6 (canister)	4 (canister)

Figure 50. Summary of C-14 Containment Concepts Aiming at the Life-Time over 60,000 Years

Nitrate Decomposition

A current reprocessing process generates wastes including large amount of nitrate salts (mainly $NaNO_3$). If the wastes are not treated specifically, the nitrate salts would be brought into a TRU waste geological disposal facility. The total amount of nitrate to be disposed of was estimated to be about 3.25 kton. Because nitrate acts as oxidizing reagent and its reduced product (e.g., NH_3) is complexed with a metal ion, the solubility and sorption of radionuclides could be affected by the nitrate.

JAEA/METI R&D program has started since 2007 in order to develop a nitrate removal method [Ref.91]. In the R&D program, beaker scale tests of removal process using an anion-exchange membrane and a Na-selective membrane have been conducted, shown in Figure 51. JAEA has also developed methods to decompose the nitrate ion by chemical reduction using a catalyst and a reductant, and by microbe.

Furthermore, chemical evolution model in a repository considering interactions of nitrate with metal, microbes and rock has been developed in order to reduce the uncertainty regarding the chemical behavior of nitrate in the deep underground[Ref. 91].

Advanced Technologies for LLW Treatment

The large volume of LLW is a matter of concern. Advanced technologies for waste treatment are needed to lower the cost and enhance the safety of disposal. JAEA has been developing technologies dealing with decontamination and waste sorting to apply to the advanced LLW treatment facility under construction; for example, a method of decontamination using supercritical carbon dioxide fluid and a pre-treatment method to remove packing materials that contain organic materials and substances harmful to the environment are being developed. The first one is expected to have an advantage of less generation of secondary waste because of its dry system (no acid-leaching, CO_2 re-use), and an extraction of over 99% of U from a used HEPA filter was confirmed by a laboratory experiment. Engineering-scale tests and application to other nuclides, e.g., Pu, are required.

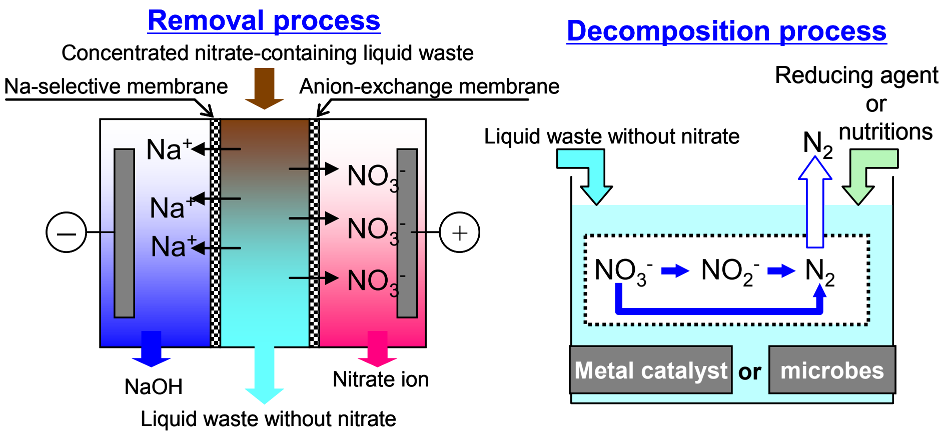


Figure 51. An Example of the Nitrate Decomposition Technology Concept

R&D on Vitrification Technology

HLW borosilicate-glass vitrification technology employing the Liquid Fed Joule-heated Ceramic Melter (LFCM) has been developed in JAEA for applicability and flexibility for the future advanced fuel cycle. Major efforts have been made to address technical issues for volume reduction and increasing the waste-loading ratio as well as extending the melter life-time.

At present, the vitrified glass loads about 25wt% of HLW (which is composed of 10% sodium oxide and 15wt% of the metal element of the waste). If the waste-loading ratio increases to 30wt%, the number of canisters would decrease by 23%. Critical issues to determine the waste-load ratio are the heat generation of products and the molybdenum concentration in a glass matrix. Regarding the vitrification process and the melter operation, the influence of the waste-loading ratio on glass properties, e.g., viscosity and the electric resistance of melted glass as well as noble-metal accumulation, are important research issues. Focusing on these points, the limitation of waste loading has been investigated by a cold engineering-scale test [Ref. 93]. As a result, a stable operation over 30wt% waste loading was confirmed without generation of yellow-phase, which mainly composed of molybdate, or noble-metal accumulation.

A new long-life vitrification melter has been developing in the JAEA/METI program [Ref. 92] since 2006. This program is aiming at extending the melter life-time from the current 5 years up to 20 years, and attaining a stable operation under variable HLW conditions (fuel type, volume reduction) in regards to the future fuel cycle. In addition to the waste-loading issue, a method to prevent corrosion of the refractory and the electrode has been developed. Also, a melter design including the bottom shape and the drain structure to prevent accumulation of noble-metal particles etc. has been developed. An advanced simulation tool to evaluate stable conditions in the long-term operation of the melter has also been developed.

Separation of Sr and Cs

An advanced fuel cycle and its options, including partitioning and transmutation, will generate various types of wastes different from those of the current LWR-PUREX cycle. A group of JAEA personnel has been investigating the concept of partitioning and transmutation and their impact on the HLW disposal (e.g., [Ref. 94]). In their concept, shown in Figure 52 (modified from [Ref. 94]), the partitioning process separates the conventional HLW into four fractions, MA, Tc and platinum group metals, Sr and Cs, and the rest, which include Zr, Mo, Fe, and rare earth elements.

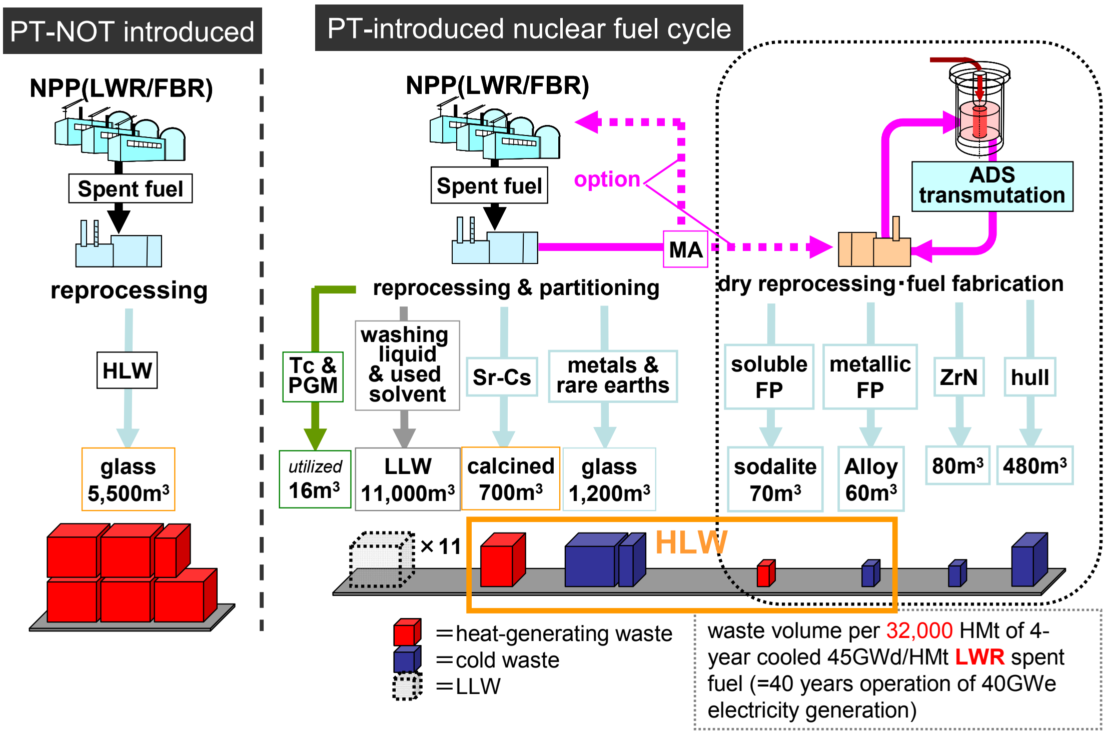


Figure 52. An Example of the Waste Flow in a P-T Cycle

Regarding Sr and Cs waste, the followings are discussed in the study [Ref. 94].

- Sr and Cs are stabilized into a calcined waste form (e.g.,Ref. 95). In Ref. 95, Sr and Cs were absorbed onto orthotitanic acid to 5.39 wt% and natural mordenite to 13.2 wt%, respectively. The grain sizes of ion-exchangers were from 20 to 50 mesh. Then they were calcined in air with an electric furnace for 3 hrs at 1000~1200°C. The calcined sample with Sr was found as SrTiO3, and for the one with Cs, the vitrification was found to proceed. Specific surface areas for both calcined samples significantly decreased to less than 0.1 m2/g. The leaching rates for both calcined samples tended to decrease with leaching time and the rates of 10-5 g/m2/day were lower than that of the borosilicate glass waste form by 2 orders of magnitude.
- The concept of a Sr-Cr calcined waste form is compatible with the partitioning process using the column packed with hydrous titanium oxide and zeolite.
- The Sr-Cs calcined waste form can be expected to stand against high temperatures of about 1200°C because of decay heat in storage. The scale of its storage facility can be estimated about one-fifth of that for conventional glass waste.
- 320-year storage may be required for compact disposal (voluminous disposal in a large cross-section disposal tunnel) of Sr-Cs waste forms to be feasible. The amount of whole HLWs in this concept (Sr-Cs calcined waste form, vitrified waste form including metals and rare earths, and sodalite and alloy waste forms as Accelerator Driven System (ADS) wastes) was calculated per 32,000 HMt of LWR spent fuel reprocessed and partitioned. As a result, it was estimated the emplacement area reduced to 15% or 1% after 130 or 320 years storage for Sr-Cs.

3.2.2.2 Possible Collaborative Research and Development

As discussed in the previous section, many of the waste forms under consideration are still conceptual. A significant amount of research and development is needed to define waste form characterization needs, identify and develop processes and techniques for waste form production, and determine relevant waste form characteristics in various environments (storage, transportation, and disposal).

Research Needs Identified by the U.S.

For all of the waste streams expected to result from aqueous and electrochemical reprocessing, an initial waste form, treatment technology, and disposal/storage path have been identified [Ref. 96]. In most cases, the waste-form chemistry and performance and the process-technology efficiency have been demonstrated and validated on an engineering-scale, and in some cases, two or more options have been suggested. However, some of these waste forms are only conceptual, and substantial R&D is necessary to determine if they can be made in a remote environment and if they perform as well as expected. Data are needed to evaluate waste-treatment technology options that will be considered to produce acceptable waste forms to meet the likely disposal strategy. The following is not a comprehensive list of required R&D; rather, it is a list of topics identified in the course of developing an integrated waste management strategy.

<u>General</u>

1. Evaluate all waste/byproducts for beneficial reuse.
2. Conduct a comprehensive technology readiness assessment on potential processes to manufacture waste forms to prioritize R&D.
3. Characterize the durability of recommended waste forms for eventual acceptance to planned or conceptual disposal facilities.
4. Evaluate the potential for cross-contamination in waste streams and potential ramifications to disposition strategy to prioritize research.

<u>Technetium</u>

1. A method is needed to effectively capture the soluble Tc and transition-metal elements from the expected aqueous solutions in forms amenable to alloying. Studies of the potential methods should address incorporating the capturing substrate into potential waste-form alloys, including Fe-based and Zr-based systems.
2. Optimize alloy formulations for waste loading to immobilize transition metal fission products, including Zr, Mo, Ru, Rh, Pd, and Tc. Niobium from spent crucibles from Echem processing should also be considered.
3. Evaluate alternative reductants to ferrous sulfamate in the separations process to minimize the concentrations of Fe and S in the FP stream.
4. The capacity of alloys to accommodate non-metallic feed materials must be determined. These should include $Zr(MoO_4)_2$ and TcO_2. Slag formation and effects on overall durability should be characterized.
5. A mechanistically based model for alloy degradation and the release of radionuclides must be developed. This is needed to calculate the long-term performance of the alloy waste form in a disposal system.

Cs/Sr

1. Determine the probable effects of transmutation on the solids and the fate of the stable decay daughters.
2. Optimize glass and/or ceramic formulations for waste loading, considering the effects of radiolysis, transmutation, thermal output, and durability during decay storage.
3. Develop strategy and technical data supporting the disposal of a waste form following decay storage, considering the RCRA requirements for Ba and ^{135}Cs content.
4. Optimize the process for removing Cs and Sr from Echem salts and optimize the waste form for waste loading, considering the effects of radiolysis, transmutation, thermal output, and durability during decay storage.
5. Develop preliminary flowsheets that consider the effects of processing high decay-heat waste streams.
6. Evaluate the benefits of heat-management concepts, including incorporating a waste form in a metal matrix, internal or external fins, wet vs. dry storage, etc. to determine if the added thermal conductivity and heat transfer is cost-effective for decay storage.
7. Consider the effects of iodine in Cs/Sr waste from the E-chem process.

Ln/FP

1. Optimize glass/ceramic formulations for waste loading of the combined Ln/FP waste stream.
2. Optimize glass formulations for waste loading of the separate Ln stream.
3. Develop an operating envelope for processing glasses/ceramics in cold crucible induction melter (CCIM), including operating techniques to maximize tolerance for noble metals.
4. Evaluate alternative reductants to ferrous sulfamate in the separations process to minimize the concentrations of Fe and S in the FP stream.
5. Optimize ceramic formulations for waste loading for processing in the CCIM.
6. Optimize the process for partitioning of Ln and FP from Echem salts, and optimize the waste form for waste loading.

Volatile Radionuclides

1. Characterize iodine loading on tritium beds and develop methods to minimize or selectively desorb.
2. Evaluate methods to maximize the long-term retention of iodine on silver sorbent.
3. Evaluate the pressure swing Kr/Xe absorption system patented by MHI and compare this to the temperature-swing process for selectivity and Xe/Kr separation.
4. Conduct optimization studies for Xe/Kr separation using solid sorbents.
5. Determine Tritium DF (decontamination factor) on a 3A molecular sieve from very low dew point gas streams.
6. Quantify Rb corrosion issues in Kr storage cylinders.
7. Develop noncement-based ^{14}C waste form, if required, for a repository.

Waste Metals

1. Characterize the activation of the hardware components to refine the technical basis for co-disposal versus segregation of hardware and TRU-contaminated cladding.
2. The conditions required to melt the collected hardware, including the needed additives, melting temperature, and cost, should be determined to provide a technical basis supporting the decision to compact or melt.

The discussion abovesummarizes a much more detailed study for selecting the most technically effective waste forms for stabilizing the primary waste streams for their planned disposition as summarized in Tables 1 and 2. In most cases, recommendations were made based on extrapolations of existing data available for similar materials, usually by researchers having extensive experience with those materials. However, little if any data exist for several of the envisioned waste forms, including if they can be made using the waste stream, if the expected waste loading can be achieved, the impact of contaminants, and how practically the manufacturing processes can be engineered for application in a hot-cell. Results from the recommended R&D may well result in other waste forms being recommended. Additional regulatory analyses are needed, and process engineering analysis will almost certainly affect these recommendations. Some processes, particularly the conveying, manipulating, and containing of highly radioactive powders and self heating solutions, are not readily designed for implementation in a remote environment. Although those factors were taken into account in selecting the baseline waste forms, more detailed facility engineering studies could lead to different assessments. Waste-acceptance criteria for disposal facilities, many of which do not yet exist, will also affect these recommendations.

Research Needs Identified by Japan

As similar to the waste form baseline concept presented by U.S., the waste form from both advanced aqueous reprocessing and pyro/electrochemical-reprocessing, which have been considered in the FS study in Japan, is still conceptual and substantial R&D is necessary to determine if they can be made in a remote environment and if they perform as well as expected. While the on-going waste form R&D in Japan is being performed mainly in the context of the current LWR-PUREX cycle, it could provide useful information in future R&D for advanced fuel cycles by demonstrating feasible and practical waste form and treatment technology. For instance, on-going R&D on iodine/C-14 confinement may have potential to be integrated into such planning for future fuel cycles. On the iodine/C-14 confinement R&D, engineered scale test and improvement of modeling reliability are identified as further issues for exploration.

Compared with the U.S. waste form baseline concept that is based on advanced separations processes, it also may be useful to conduct R&D to improve vitrification technology for the composite HLW. As shown above, HLW borosilicate-glass vitrification technology employing LFCM has been developed in terms of applicability and flexibility for a future advanced fuel cycle. R&D efforts focused on addressing technical issues for volume reduction, increasing waste loading ratio, and extending melter life-time.

Regarding LLW, advanced aqueous reprocessing in the FS study in Japan is considering the possibility for the employment of a simple process, using a salt-free reagent, thereby polarising between HLW and VLLW / clearance. While there is on-going R&D on methods of nitrate decomposition, decontamination, etc. future engineering scale tests are required for these LLW treatment technologies.

Test Methods and Release Models

Protocols are being established under the AFCIU for a variety of waste forms to identify testing methods that can be used to provide information needs in three areas: waste-form acceptance, waste-form production, and waste-form performance. Waste-form performance includes an understanding of radionuclide release modes, degradation mechanism of the waste-form matrix, effects of environmental variables, etc. needed to calculate waste-form behavior over long times in performance-assessment calculations. Waste-form production issues include understanding how processing variables affect waste-form behavior and how process control limits can be used to control waste-form product consistency. Waste-form acceptance issues include intrinsic characteristics and performance measures that must be demonstrated to qualify waste packages for disposal. Testing protocols are being developed for waste forms arising from waste streams resulting from the aqueous (aq) or electrochemical (Echem) processing options: iodine, ^{14}C, and ^{3}H recovered from vol-oxidation and dissolution (aq and Echem); Tc-bearing undissolved solids (aq); recovered soluble Tc (aq); separated Cs/Sr (aq); separated transition-metal fission products (aq); separated lanthanide elements (aq); hulls and hardware (aq and Echem); metallic wastes (Echem); salt wastes (Echem); and separated lanthanide elements (Echem). The approach taken is to identify specific test methods appropriate to provide information needs for each waste form. The current completeness of the protocols reflects how well the production and performance of a waste form is understood.

3.2.3 Definition of Optimization Problems (Task 2-3)

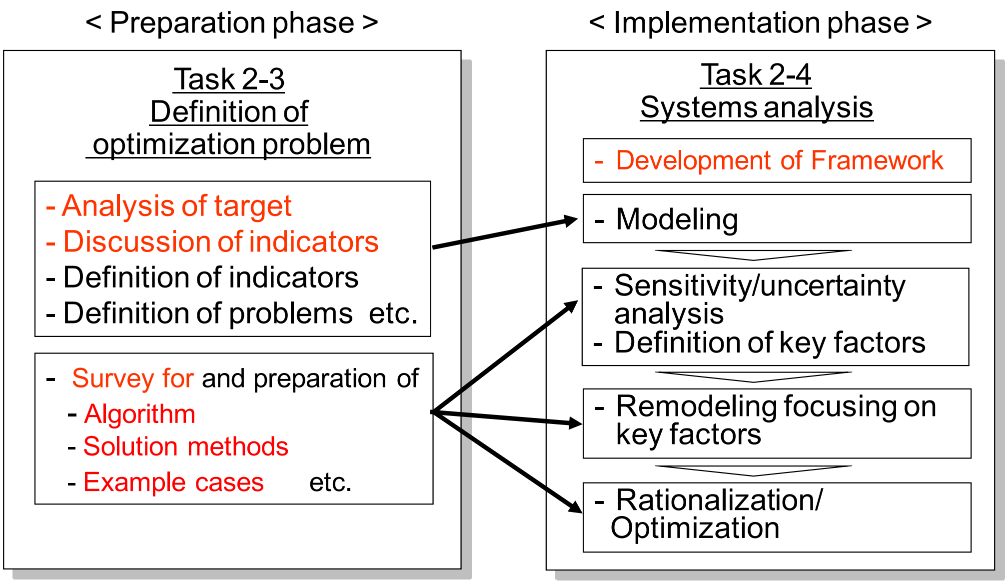
This section discusses the definition of optimization problems (task 2-3). Optimization processes and techniques are discussed in Section 3.2.3.1 followed by a discussion of factors to consider in repository design optimization in Section 3.2.3.2.Potential collaboration areas and activities related to optimization are then presented for Phase II.

3.2.3.1 Optimization Processes and Techniques

Each nation having an active geologic repository program has had to pursue design optimization. The degree of optimization depends on the status of the geologic disposal program. Optimization can be performed at a conceptual stage to provide information regarding the effects of fuel-cycle scenarios on the disposal of radioactive waste. For example, “generic” conditions can be assumed, based on literature information or data obtained from underground research laboratories to conduct the optimization of conceptual designs in a variety of potential geologic formations. Such information could help inform future siting studies, waste-form development, and engineered barrier material development activities. However, repository design optimization can only begin to truly be applied once a program is actively engaged in designing a repository at a specific site and knows the types of quantities of waste that would be disposed of. As discussed above, the actual characteristics of the host rock where the repository will be constructed play a very important role in the optimization.

A variety of tools exist that can be applied to the optimization problem. These include thermal, geotechnical, geochemistry, hydrologic, and material-performance models. These models are state of the art and are currently being used in repository development programs world-wide. These tools could be used in the optimization of conceptual repository concepts for advanced nuclear fuel cycles. In addition, efforts are underway to expand and improve on the existing suite of tools.

To consider optimization problems, Tasks 2-3 and 2-4 should be linked as shown in Figure 53 to understand the complex structure of multiple factors in waste management and fuel cycle. In this frame, Tasks 2-3 and 2-4 could be identified as a preparation phase and an implementation phase for system optimization, respectively.



NOTE: Items in red character are main activities in the Phase I.

Figure 53. Relationship Between Tasks 2-3 and 2-4

For Task 2-3 in Phase I, the main activities are an analysis of the target, which should be considered in system optimization, and a discussion of indicators, which can be used to describe system optimization problems from different aspects. To support these activities, optimization methods in other areas than waste management would be surveyed and evaluated with case examples.

In defining optimization problems, it is important to translate questions to obtain insights for waste-management and fuel-cycle systems into optimization problems using indicators. For example:

- What disposal systems are most appropriate for different scenarios of nuclear energy use?
- How much capacity of what kind of repositories will be needed for each scenario?
- Which nuclear scenario/fuel cycle options are most favorable in terms of reducing the requirements on the repository development program?
- What R&D could lead to more cost-effective waste management for the scenarios that are considered more likely?

The approach and main activities at JAEA in Phase I are:

- Literature survey for existing optimization methodology and case examples in other fields;
- Discussion of optimization problem at different levels;
- Listing of potential indicators for each level.

JAEA has done a literature survey for existing optimization methodologies and case studies in other fields. General optimization problems in other fields applied to characteristics and features of radioactive waste disposal are summarized in Table 8.

Here, a combined methodology would be helpful, e.g., multi-attribute analysis (MAA). Examples of MAA are NUMO's two exercises that serve as preparation for future comparison of different site-specific repository concepts [Ref.30] :

1. the feasibility or suitability of a simplified H12 “generic repository” in different siting environments;
2. the suitability of different repository concepts for a simplified “generic site.”

In the first case, the pros and cons of eight “Potential Siting Environments” (PSEs) as potential hosts for an H12-type repository were identified and compared using MAA. The same MAA approach was used in the second exercise to gain experience in ranking different repository concepts for a specific site. Even on the basis of rather simple preliminary analyses, NUMO found that the subjective weightings placed by different stakeholders on the various site or concept attributes could significantly alter rankings.

3.2.3.2 Factors to Consider in Repository Design Optimization

In discussing repository design optimization problems, the following viewpoints should be considered:

- Variation of nuclear-scenario and fuel-cycle options, which would be input from Tasks 1-1, 1-2, and 2-2
- Variation of optimization problems in different levels with relevant constraints.

The system that should be considered in optimization problems is complex as shown in Figure 54. The system has multi-objectives and multi-variables; therefore, a systematic and structured approach is needed for optimization.

To deal with this complex system effectively, an idea is to classify levels for optimization problems introducing some indicators to represent relationships between levels. It is proposed to classify optimization problems into three levels. These levels were characterized by the extent and degree of details regarding waste in the viewpoints of waste management, as shown in

Figure 55:

Level 1: Fuel cycle systems (including upstream processes);

Level 2: Disposal system (including variation of waste types);

Level 3: Individual disposal systems (for individual waste type).

In Level 1, combinations of fuel cycles and processes could be reflected. Figure 56 shows an example of structuring of processes in Level 1.

In Level 2, disposal systems for classified different waste types are discussed. Figure 57 shows an example of structuring of the Level 2.

Table 8. Correspondence of General Categorization of Optimization Problems in Other Fields to Characters and Features of Subject in Waste Disposal Filed

Design Factors / General categorization of optimization problems	Long Term Safety	Environmental Impact	Operational Safety	Engineering Reliability	Socio-Economic Aspects	Engineering Feasibility / QA	Site Characterization & Monitoring Requirements	Reversibility / Retrievability
Design Support & Structural Optimization	○	-	●	○	○	○	-	-
Layout Optimization	○	○	○	○	○	○	-	-
Resource Allocation & Transportation Planning	-	○	○	-	●	●	-	-
Site Selection	○	●	○	-	●	○	-	-
Decision Making & Consensus Building	○	○	●	○	●	○	○	○
Economic Evaluation	○	○	○	○	●	○	-	-
Dynamic Environment Problem	○	-	○	○	○	○	-	-
Function Optimization	○	-	○	-	-	○	-	-
Combinatorial Optimization	-	○	○	-	○	○	○	-
Others ・ Waste Management ・ Scheduling	○	○	○	○	○	○	○	○

Some cases with the HIGH relation to design factors, ○ : Some cases with the relation to design factors, - : No Case in research to date

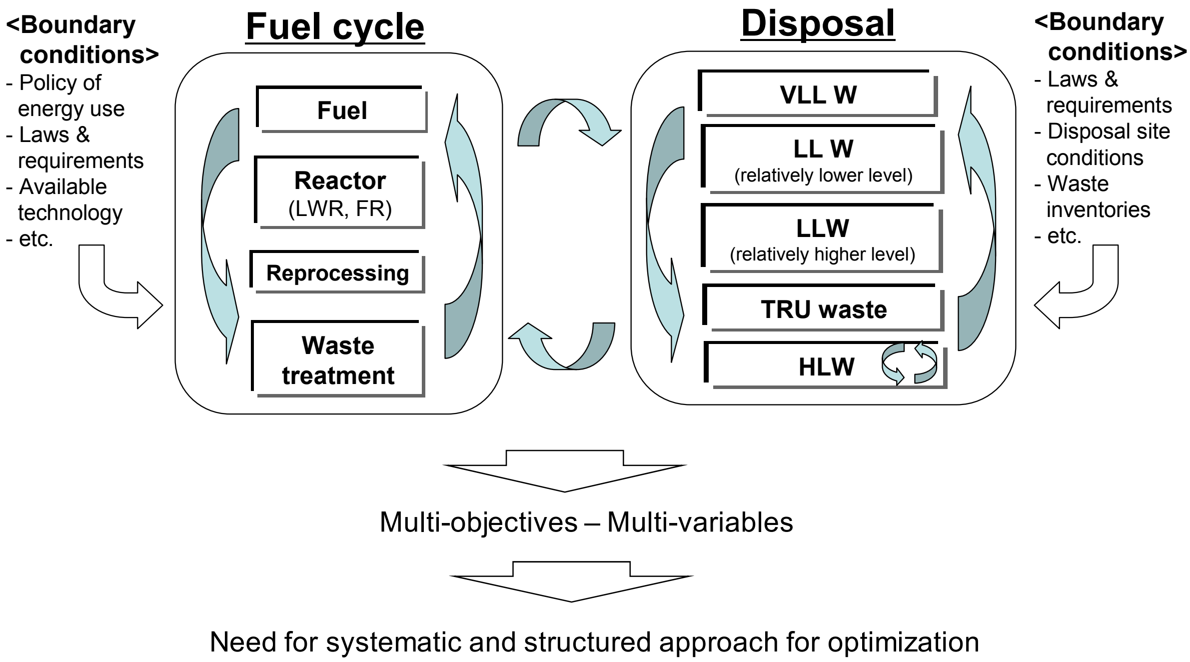
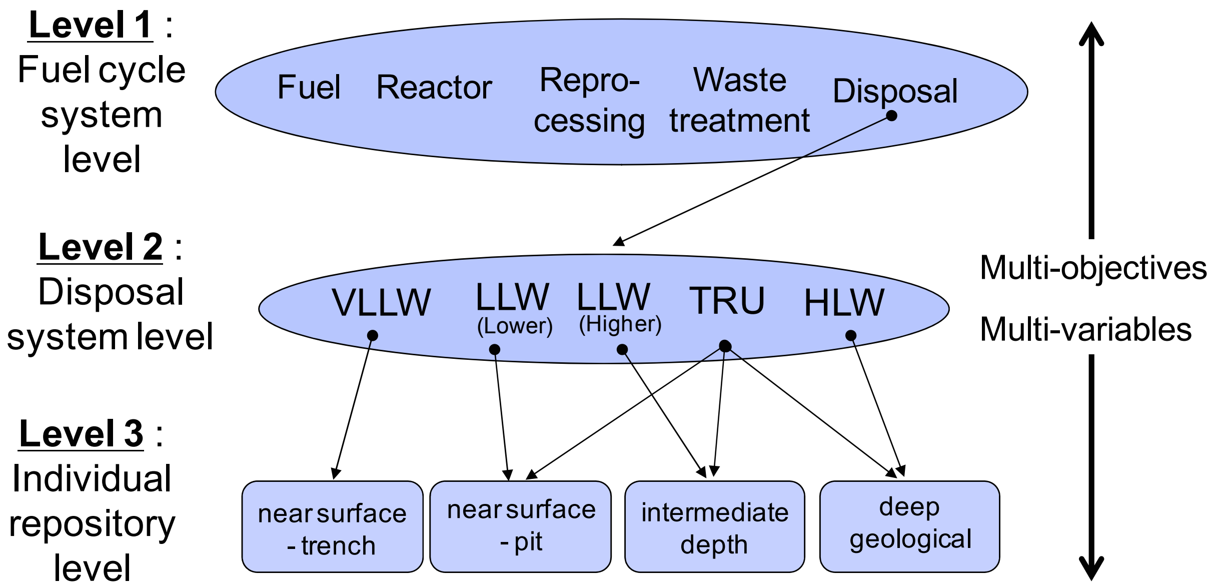


Figure 54. Complexity of the System



Note: For the U.S., the TRU classification does not exist (TRU is captured in GTCC) and LLW (Higher) may be disposable in either near surface or intermediate depth facilities.

Figure 55. Classified Three Levels of Optimization Problems for Waste Management

Fuel	•MA loading •Amount of fuel •Component of fuel •Heat generation			•Classification of waste •Amount of waste •Component of waste •Heat generation
•Expected SF storage period	Reactor (LWR,FR)	•MA burning •Amount of SF •Component of SF •Heat generation		•Classification of waste •Amount of waste •Component of waste •Heat generation
•Expected separation & storage of Am/Cm		Reprocess-ing	•MA recovery •FP separation •Salt free	•Reduction of MA, FP in waste •Classification of waste •Amount of waste •Component of waste •Heat generation
		•Expected component of reprocessing products	Waste treatment	•Classification of waste •Amount of waste •Component of waste •Heat generation
•Expected waste volume reduction •Expected waste characteristics	•Expected waste volume reduction •Expected waste characteristics	•Expected waste volume reduction •Expected waste characteristics	•Expected waste form performance	Waste disposal

Figure 56. An Example of Structuring of Level 1

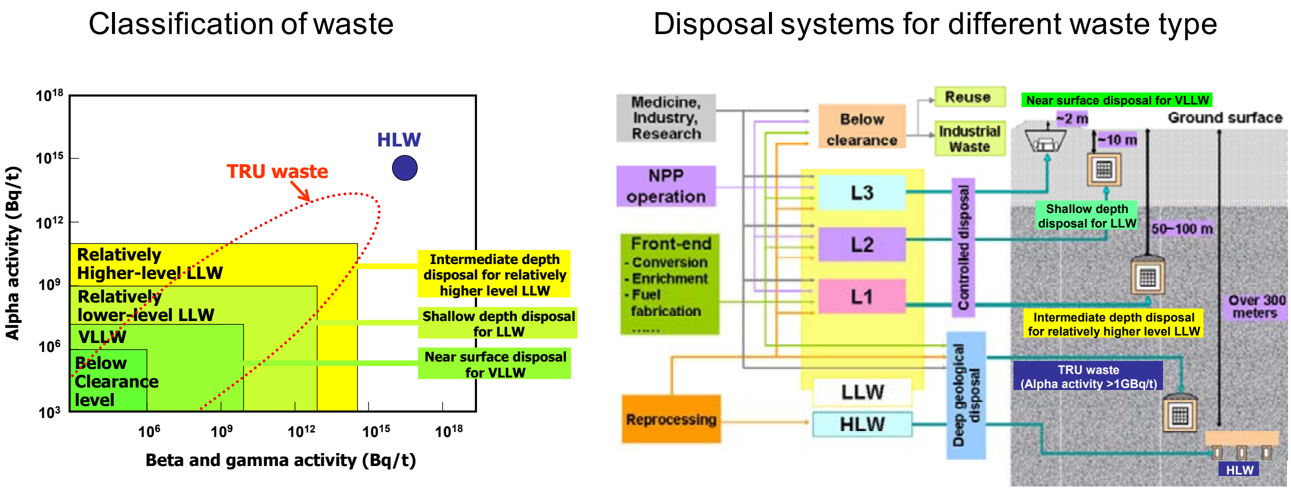


Figure 57. An Example of Structuring of Level 2

In Level 3, the individual disposal system is discussed, mainly in the viewpoints of repository design optimization, for example, by "design factors" developed by NUMO for HLW [Ref. 30] (see also section 3.2.1.4.1).

- Long-term safety

- Operational safety
- Engineering feasibility and QA
- Engineering reliability
- Site characterization and monitoring requirements
- Retrievability
- Environment impact
- Socio-economic aspects.

2) Listing of factors to the repository design optimization challenge

Roughly defined optimization issue for each level would be as follows,

- Level 1 : Fuel Cycle Systems
  - Minimize waste volume
  - Boundary conditions: Technical feasibility for fuel cycle systems, etc.
- Level 2 : Disposal Systems
  - Optimize the classification of waste in different disposal systems
  - Boundary conditions: Characteristics of waste, etc.
- Level 3: Individual Disposal Systems
  - Optimize a disposal system, taking siting environments and design factors into account
  - Boundary conditions: Site conditions, etc.

Using these issues as a starting point, factors that can be used to describe optimization problems for each level have been examined.

To facilitate the description of optimization problems using factors, factors are categorized into the following three types:

- Optimization indicators
- Controlled variables
- Constraints.

Table 8 shows a preliminary list of factors. The list should be discussed in more detail in Phase II to improve sufficiency and then to define important factors that should be used to define optimization problems.

The importance of factors could be discussed, for example, by combining siting environments (site–specific) with generic analysis using the design factors to give perspectives on modification on host rock characteristics, including thermal gradient, saturation vs. unsaturation, etc.

Consideration and treatment of the linkage between levels are important. Here, Level 1 should be studied carefully because it includes issues covered by other WG, for example, fuel fabrications in terms of waste; therefore, specifying the linkage between Levels 1 and 2 would be a challenge. In this WG, it has been agreed to precede studies for Levels 2 and 3, taking account of linkages between these.

Table 8. A Preliminary List of Factors

	Optimization indicators	Controllable variables	Constraints
Level 1: **Fuel cycle systems**	□ Characteristics of generated waste (including easiness of handling of wastes , simplicity of waste characteristics) □ General traceability and traceability of mass flow □ Safety □ Economic competitiveness □ Reduction of environmental burden □ Efficient use of nuclear resources (e.g., recycle of U, Pu) □ Enhancement of nuclear nonproliferation □ Social acceptance	□ Nuclear scenario □ Fuel cycle option • MA recovery rate • FP separation rate • Storage period, etc. □ Results of comprehensive assessment for overall disposal systems	□ Politics for nuclear energy use □ Regulations □ Electronic needs □ Technical feasibility
Level 2: **Disposal systems**	□ Comprehensive index for overall disposal systems □ Characteristics of wastes for each disposal way □ Needed total number of repositories □ Needed total footprints of repositories	□ Options of overall disposal systems □ Disposal system concept for each disposal option	□ Politics for waste management □ Regulations, technical requirements • Classification of wastes • Classification of disposal options □ Characteristics of wastes (activity etc.)
Level 3: **Individual disposal systems**	□ Long-term safety • Dose • Risk • Release rate (from waste, EBS, NB) • *Role of individual safety function and inter-relationship* □ Operational safety • *General impact on workers* • Radiological impact on workers □ *Engineering feasibility, reliability* • *Experience of application* • *Flexibility for variation of waste characteristics, design of repository, site condition, and social changes* • *Confidence in quality assurance* • *Equipment operability, robustness, understandability etc.* □ *Retrievability* □ *Environmental impact* □ Socio-economic aspects • Cost • Capacity and/or footprint of a repository • Public acceptance, etc.	□ Options of disposal-system concepts for each disposal alternative	□ Politics for waste management □ Regulations, technical requirements • Requirements for site selection • Technical standards • Requirements for waste • Dose standard • Safety target □ *Site condition* □ Characteristics of waste form (number, inventory, composition, heat generation, performance, etc.)

Italics: Basically independent of fuel cycle system

3.2.4 Systems Analysis (Task 2-4)

This section discusses the use of systems analysis within the context of the WMWG (task 2-4). The need for systems analysis for integrated waste management is first presented in Section 3.2.4.1 followed by the definition of system analysis in this working group in Section 3.2.4.2. A summary of available system analysis tools is then given in Section 3.2.4.3. The development of a preliminary working scenario for system analysis within this working group is finally summarized in Section 3.2.4.4.

3.2.4.1 Need for Systems Analysis for Integrated Waste Management

Advanced separations techniques could potentially enable more efficient management of wastes that would be generated in a closed nuclear fuel cycle. It may be possible to match the waste forms to the waste stream and to match the disposal environment and the waste form. This could allow efficient processing techniques and improved long-term repository performance.

The advanced separations techniques may result in a variety of waste streams, as discussed in section 3.2.2.1, including reduced metals, glasses, ceramics and oxides. The efficient management of these wastes would involve assessing a variety of options and alternatives related to storage, transportation, and disposal. Potential benefits of utilizing an advanced fuel cycle include increased waste loadings resulting in increased repository capacity, and improved waste form durability in disposal system environments.

To fully accomplish the overall goals of a sustainable advanced fuel cycle, a systems approach to developing the new fuel cycle must include reevaluation of waste forms (see section 3.2.2) and disposition plans for radioactive waste and byproducts (see section 3.2.1). A comprehensive waste-management strategy is absolutely essential to achieving these goals, and to be accepted, this strategy must be both practical and support a cost-effective life-cycle. To this point, this document has described key aspects of how waste could be effectively managed to make efficient use of resources. This section is focused on technical studies and research needed to support decisions that must be made to implement any strategy and identifies key data gaps and alternatives to investigate.

Demonstrating a commercially viable fuel cycle will necessarily drive the research and development program to demonstrate an optimized waste-management strategy that considers the scale and dynamics of complex systems (see section 3.2.3), including fuel fabrication, reprocessing, storage, disposal, and the associated ancillary infrastructure (e.g., transportation) and material flow through the system. Feedback amongst fuel fabrication and recycling and waste and byproduct management is essential to optimize the fuel cycle. The research and development program must effectively coordinate and integrate research and development in all of these areas. Systems analyses are essential to evaluate alternatives balancing technology maturity, environmental impact, social values, and of course cost. An example described earlier in this document is evaluating the many ways to manage thermal effects. Some combination of fuel storage and separations as well as waste stabilization and storage, followed by ultimate disposition of all materials, will be the most practical and cost-effective. Simply because technology is available to make an option possible does not necessarily infer that the option is the most feasible. Partitioning used fuel into many streams and stabilizing the wastes into many forms maximizes the options for waste management, but at a cost of more complex operations, more supporting systems and facilities, and likely more secondary wastes. Equally credible is the scenario described above, combining the waste streams based on target element chemistry, with all of the easily oxidized elements stabilized as oxides in glass or ceramics and all readily reducible elements combined in a metallic alloy. If these waste forms can be practically made, the systems required could be simpler and less costly. Appropriate systems analyses must be done to determine the most practical commercial approach, considering all of the goals for a sustainable fuel cycle that can be implemented internationally.

3.2.4.2 Definition of Systems Analysis in the Waste Management Working Group

Systems Analysis in the context of the research and development phase (discovery phase) is defined as:

> The diagnosis, formulation, and solution of problems that arise out of the complex forms of interaction in systems from hardware to corporations, that exist or are conceived to accomplish one or more specific objectives. Systems analysis provides a variety of analytical tools, design methods and evaluative techniques to aid in decision making regarding such systems.

In the context of advanced fuel cycle research and development, Systems Analysis provides the overall approach for implementing a closed fuel cycle by coordinating and analyzing the fuel cycle from a high-level systems perspective. Systems analyses are performed on systems or subsystems to understand their integrated behavior and performance and the resulting impacts on objectives, including overall program objectives, under various scenarios. This includes applying models, tools, and analyses to evaluate potential configurations of the fuel cycle components, as well as implementing key system demonstrations to validate the analyses. Results are targeted to aid decision-makers in selecting the best fuel cycle and reactor technologies and configurations.

Expected roles of systems analysis would be defined as follows:

- Analysis and understanding of system response to changes in nuclear scenarios, fuel-cycle options, etc, in term of type, amount, and characteristics of waste.
- Use of these results to define key factors to be focused on in optimization.
- Optimization of integrated waste-management system in term of nuclear scenarios, fuel-cycle options, waste types, design options of a disposal system, sitting environments, etc.

Systems analysis would play two roles within the context of the U.S.–Japan Waste Management Working Group. The first role would be to perform subsystem analyses in the area of waste management, in particular radioactive material transportation, storage, and disposal. At this early stage of the research and development program, repository optimization analyses and system analyses are synonymous. Several of the Phase II optimization activities described above are system analyses.

The second role would be to develop models that could be used in the higher-level system analyses tools. Typically, these higher-level system analysis tools use simple models of the various subsystems. These simple models must be based on a detailed understanding of the key factors and coupling of the factors at the subsystem level.

To discuss a framework for systems analysis, tasks that should be included in systems analysis have to be analyzed, taking into account linkages to other tasks in this WG.

For analysis regarding scenario levels 1 and 2, shown in Figures 56 and 57, fuel cycle options, models, and toolkits for inventory development would be selected based on results of Tasks 1-1, 2-2, and 1-2. Optimization problems are then defined and suitable methodologies and toolkits for the problems are selected from results of Task 2-3. For analysis regarding level 3, design options for certain waste types would be identified based on results of Task 2-1,and models and toolkits for design and performance assessment would be selected. Then, optimization problems are defined and suitable methodologies and toolkits for the problems are selected from results of Task 2-3. A first draft of a schematic view of the framework, including those tasks, is shown in Figure58.

This framework has been agreed upon by this WGand as an early activity in Phase II, a preliminary check in each task providing a basis for analysis has been proposed. This WG also pointed out that simplified analyses should keep in mind detailed analyses, but should not become overly complicated.

3.2.4.3 Description of Systems Analysis Tools

The U.S. uses the VISION system analysis tool [Ref. 10] for the Fuel Cycle Research and Development program. The objective of VISION is to evaluate the elements of the fuel cycle that discriminate the different advanced fuel cycles. VISION is used specifically to:

- Perform dynamic scoping trade studies of alternative fuel cycles to obtain qualitative and quantitative comparisons of resource requirements, reactor types and mix, sequencing and timing, waste streams, and geologic repository requirements, with capability to provide cost estimates of levelized cost of electricity, and cash flow/funding requirements.
- Model the nuclear fuel cycle such that dynamic changes in process capability including transition from "design and construct to startup to equilibrium to final D&D" states as well as material, capital, and operating costs can be factored in to a Levelized Life Cycle Cost or other benefit comparison.
- Quickly assess relative differences in fuel cycle strategies and timing with reasonable accuracy.
- Provide a range of model outputs that can support both technical and management review.
- Interact (in some fashion) with higher-level models, e.g., that compare among energy source options.
- Interact (in some fashion) with lower-level modules, e.g., those providing detailed cost and process estimations for individual facilities.
- Provide parameters that are critical to comparing AFCI options, including repository capacity and performance, separation capacity, interim estate storage, energy recovery, proliferation resistance, and safety. Specific waste parameters include waste mass, wasteform mass, wasteform volume, long-term radiotoxicity, and long-term heat commitment to a repository.

In Japan, for example, the FAMILY tool has been applied to examine the state of future reactor types or recycling facilities [Ref. 97] and the SCENARIO tool has been applied to examine contribution of the Partitioning-and-Transmutation (P&T) cycle to HLW disposal [Ref. 98]. The FAMILY tool can calculate the transient mass balance characteristics (e.g., the amount of natural uranium demand, Pu mass balance, and environmental load reduction) and the SCENARIO tool can treat the double-strata fuel cycle consisting of the commercial fuel cycle and P&T cycle. These tools could be applied to a limited number of systems analysis, but lack capabilities to calculate amounts and characteristics of low level waste.

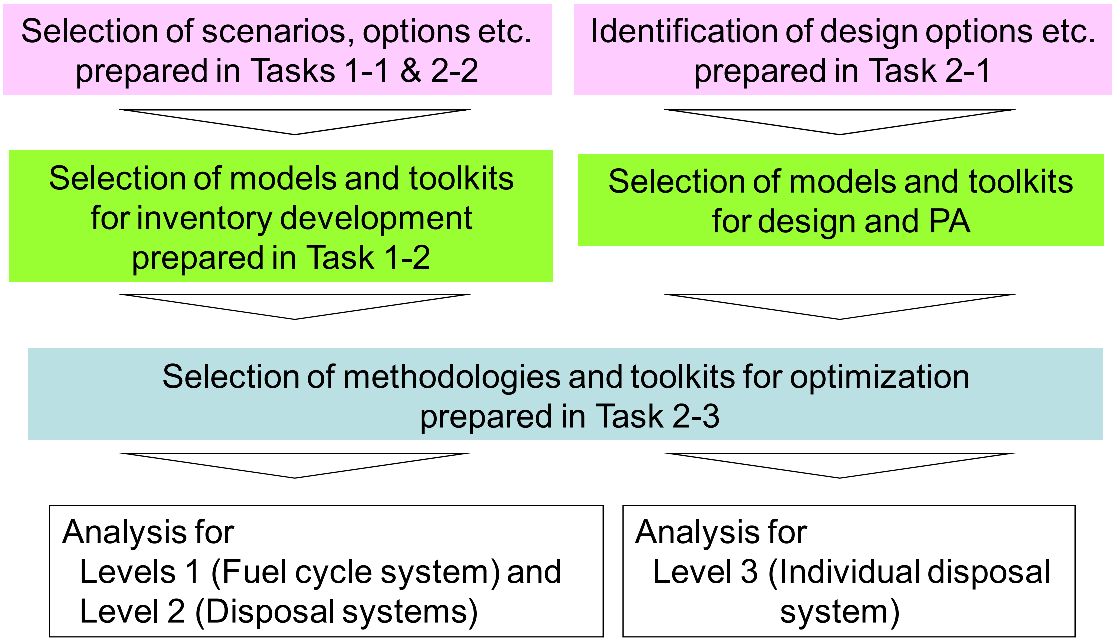


Figure 58. Schematic View of the Systems Analysis Framework

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国際単位系（SI）

表１．SI 基本単位

基本量	SI 基本単位	
	名称	記号
長さ	メートル	m
質量	キログラム	kg
時間	秒	s
電流	アンペア	A
熱力学温度	ケルビン	K
物質量	モル	mol
光度	カンデラ	cd

表２．基本単位を用いて表されるSI組立単位の例

組立量	SI 基本単位	
	名称	記号
面積	平方メートル	m^2
体積	立法メートル	m^3
速さ，速度	メートル毎秒	m/s
加速度	メートル毎秒毎秒	m/s^2
波数	毎メートル	m^{-1}
密度，質量密度	キログラム毎立方メートル	kg/m^3
面積密度	キログラム毎平方メートル	kg/m^2
比体積	立方メートル毎キログラム	m^3/kg
電流密度	アンペア毎平方メートル	A/m^2
磁界の強さ	アンペア毎メートル	A/m
量濃度 (a)，濃度	モル毎立方メートル	mol/m^3
質量濃度	キログラム毎立法メートル	kg/m^3
輝度	カンデラ毎平方メートル	cd/m^2
屈折率 (b)	（数字の）　1	1
比透磁率 (b)	（数字の）　1	1

(a) 量濃度（amount concentration）は臨床化学の分野では物質濃度（substance concentration）ともよばれる。
(b) これらは無次元量あるいは次元１をもつ量であるが、そのことを表す単位記号である数字の１は通常は表記しない。

表３．固有の名称と記号で表されるSI組立単位

組立量	SI 組立単位			
	名称	記号	他のSI単位による表し方	SI基本単位による表し方
平面角	ラジアン(b)	rad	1 (b)	m/m
立体角	ステラジアン(b)	sr(c)	1 (b)	m^2/m^2
周波数	ヘルツ (d)	Hz		s^{-1}
力	ニュートン	N		$m\ kg\ s^{-2}$
圧力，応力	パスカル	Pa	N/m^2	$m^{-1}\ kg\ s^{-2}$
エネルギー，仕事，熱量	ジュール	J	N m	$m^2\ kg\ s^{-2}$
仕事率，工率，放射束	ワット	W	J/s	$m^2\ kg\ s^{-3}$
電荷，電気量	クーロン	C		s A
電位差（電圧），起電力	ボルト	V	W/A	$m^2\ kg\ s^{-3}\ A^{-1}$
静電容量	ファラド	F	C/V	$m^{-2}\ kg^{-1}\ s^4\ A^2$
電気抵抗	オーム	Ω	V/A	$m^2\ kg\ s^{-3}\ A^{-2}$
コンダクタンス	ジーメンス	S	A/V	$m^{-2}\ kg^{-1}\ s^3\ A^2$
磁束	ウエーバ	Wb	Vs	$m^2\ kg\ s^{-2}\ A^{-1}$
磁束密度	テスラ	T	Wb/m^2	$kg\ s^{-2}\ A^{-1}$
インダクタンス	ヘンリー	H	Wb/A	$m^2\ kg\ s^{-2}\ A^{-2}$
セルシウス温度	セルシウス度(e)	℃		K
光束	ルーメン	lm	cd sr(c)	cd
照度	ルクス	lx	lm/m^2	$m^{-2}\ cd$
放射性核種の放射能 (f)	ベクレル (d)	Bq		s^{-1}
吸収線量, 比エネルギー分与, カーマ	グレイ	Gy	J/kg	$m^2\ s^{-2}$
線量当量, 周辺線量当量, 方向性線量当量, 個人線量当量	シーベルト (g)	Sv	J/kg	$m^2\ s^{-2}$
酸素活性	カタール	kat		$s^{-1}\ mol$

(a)SI接頭語は固有の名称と記号を持つ組立単位と組み合わせても使用できる。しかし接頭語を付した単位はもはやコヒーレントではない。
(b)ラジアンとステラジアンは数字の１に対する単位の特別な名称で、量についての情報をつたえるために使われる。実際には、使用する時には記号rad及びsrが用いられるが、習慣として組立単位としての記号である数字の１は明示されない。
(c)測光学ではステラジアンという名称と記号srを単位の表し方の中に、そのまま維持している。
(d)ヘルツは周期現象についてのみ、ベクレルは放射性核種の統計的過程についてのみ使用される。
(e)セルシウス度はケルビンの特別な名称で、セルシウス温度を表すために使用される。セルシウス度とケルビンの単位の大きさは同一である。したがって、温度差や温度間隔を表す数値はどちらの単位で表しても同じである。
(f)放射性核種の放射能（activity referred to a radionuclide）は、しばしば誤った用語で"radioactivity"と記される。
(g)単位シーベルト（PV,2002,70,205）についてはCIPM勧告2（CI-2002）を参照。

表４．単位の中に固有の名称と記号を含むSI組立単位の例

組立量	SI 組立単位		
	名称	記号	SI 基本単位による表し方
粘度	パスカル秒	Pa s	$m^{-1}\ kg\ s^{-1}$
力のモーメント	ニュートンメートル	N m	$m^2\ kg\ s^{-2}$
表面張力	ニュートン毎メートル	N/m	$kg\ s^{-2}$
角速度	ラジアン毎秒	rad/s	$m\ m^{-1}\ s^{-1}=s^{-1}$
角加速度	ラジアン毎秒毎秒	rad/s^2	$m\ m^{-1}\ s^{-2}=s^{-2}$
熱流密度，放射照度	ワット毎平方メートル	W/m^2	$kg\ s^{-3}$
熱容量，エントロピー	ジュール毎ケルビン	J/K	$m^2\ kg\ s^{-2}\ K^{-1}$
比熱容量，比エントロピー	ジュール毎キログラム毎ケルビン	J/(kg K)	$m^2\ s^{-2}\ K^{-1}$
比エネルギー	ジュール毎キログラム	J/kg	$m^2\ s^{-2}$
熱伝導率	ワット毎メートル毎ケルビン	W/(m K)	$m\ kg\ s^{-3}\ K^{-1}$
体積エネルギー	ジュール毎立方メートル	J/m^3	$m^{-1}\ kg\ s^{-2}$
電界の強さ	ボルト毎メートル	V/m	$m\ kg\ s^{-3}\ A^{-1}$
電荷密度	クーロン毎立方メートル	C/m^3	$m^{-3}\ sA$
表面電荷	クーロン毎平方メートル	C/m^2	$m^{-2}\ sA$
電束密度，電気変位	クーロン毎平方メートル	C/m^2	$m^{-2}\ sA$
誘電率	ファラド毎メートル	F/m	$m^{-3}\ kg^{-1}\ s^4\ A^2$
透磁率	ヘンリー毎メートル	H/m	$m\ kg\ s^{-2}\ A^{-2}$
モルエネルギー	ジュール毎モル	J/mol	$m^2\ kg\ s^{-2}\ mol^{-1}$
モルエントロピー, モル熱容量	ジュール毎モル毎ケルビン	J/(mol K)	$m^2\ kg\ s^{-2}\ K^{-1}\ mol^{-1}$
照射線量（Ｘ線及びγ線）	クーロン毎キログラム	C/kg	$kg^{-1}\ sA$
吸収線量率	グレイ毎秒	Gy/s	$m^2\ s^{-3}$
放射強度	ワット毎ステラジアン	W/sr	$m^4\ m^{-2}\ kg\ s^{-3}=m^2\ kg\ s^{-3}$
放射輝度	ワット毎平方メートル毎ステラジアン	$W/(m^2\ sr)$	$m^2\ m^{-2}\ kg\ s^{-3}=kg\ s^{-3}$
酵素活性濃度	カタール毎立方メートル	kat/m^3	$m^{-3}\ s^{-1}\ mol$

表５．SI 接頭語

乗数	接頭語	記号	乗数	接頭語	記号
10^{24}	ヨタ	Y	10^{-1}	デシ	d
10^{21}	ゼタ	Z	10^{-2}	センチ	c
10^{18}	エクサ	E	10^{-3}	ミリ	m
10^{15}	ペタ	P	10^{-6}	マイクロ	µ
10^{12}	テラ	T	10^{-9}	ナノ	n
10^{9}	ギガ	G	10^{-12}	ピコ	p
10^{6}	メガ	M	10^{-15}	フェムト	f
10^{3}	キロ	k	10^{-18}	アト	a
10^{2}	ヘクト	h	10^{-21}	ゼプト	z
10^{1}	デカ	da	10^{-24}	ヨクト	y

表６．SIに属さないが、SIと併用される単位

名称	記号	SI 単位による値
分	min	1 min=60s
時	h	1h =60 min=3600 s
日	d	1 d=24 h=86 400 s
度	°	1°=(π/180) rad
分	’	1’=(1/60)°=(π/10800) rad
秒	”	1”=(1/60)’=(π/648000) rad
ヘクタール	ha	$1ha=1hm^2=10^4m^2$
リットル	L，l	$1L=1l=1dm^3=10^3cm^3=10^{-3}m^3$
トン	t	$1t=10^3$ kg

表７．SIに属さないが、SIと併用される単位で、SI単位で表される数値が実験的に得られるもの

名称	記号	SI 単位で表される数値
電子ボルト	eV	$1eV=1.602\ 176\ 53(14)\times10^{-19}J$
ダルトン	Da	$1Da=1.660\ 538\ 86(28)\times10^{-27}kg$
統一原子質量単位	u	1u=1 Da
天文単位	ua	$1ua=1.495\ 978\ 706\ 91(6)\times10^{11}m$

表８．SIに属さないが、SIと併用されるその他の単位

名称	記号	SI 単位で表される数値
バール	bar	$1\ bar=0.1MPa=100kPa=10^5Pa$
水銀柱ミリメートル	mmHg	1mmHg=133.322Pa
オングストローム	Å	$1\ Å=0.1nm=100pm=10^{-10}m$
海里	M	1 M=1852m
バーン	b	$1\ b=100fm^2=(10^{-12}cm)2=10^{-28}m^2$
ノット	kn	1 kn=(1852/3600)m/s
ネーパ	Np	SI単位との数値的な関係は、対数量の定義に依存。
ベル	B	
デジベル	dB	

表９．固有の名称をもつCGS組立単位

名称	記号	SI 単位で表される数値
エルグ	erg	$1\ erg=10^{-7}\ J$
ダイン	dyn	$1\ dyn=10^{-5}N$
ポアズ	P	$1\ P=1\ dyn\ s\ cm^{-2}=0.1Pa\ s$
ストークス	St	$1\ St=1cm^2\ s^{-1}=10^{-4}m^2\ s^{-1}$
スチルブ	sb	$1\ sb=1cd\ cm^{-2}=10^4cd\ m^{-2}$
フォト	ph	$1\ ph=1cd\ sr\ cm^{-2}\ 10^4lx$
ガル	Gal	$1\ Gal=1cm\ s^{-2}=10^{-2}ms^{-2}$
マクスウェル	Mx	$1\ Mx=1G\ cm^2=10^{-8}Wb$
ガウス	G	$1\ G=1Mx\ cm^{-2}=10^{-4}T$
エルステッド (c)	Oe	$1\ Oe \triangleq (10^3/4\pi)A\ m^{-1}$

(c) ３元系のCGS単位系とSIでは直接比較できないため、等号「 ≙ 」は対応関係を示すものである。

表10．SIに属さないその他の単位の例

名称	記号	SI 単位で表される数値
キュリー	Ci	$1\ Ci=3.7\times10^{10}Bq$
レントゲン	R	$1\ R=2.58\times10^{-4}C/kg$
ラド	rad	$1\ rad=1cGy=10^{-2}Gy$
レム	rem	$1\ rem=1\ cSv=10^{-2}Sv$
ガンマ	γ	1γ=1 nT=10-9T
フェルミ		1フェルミ=1 fm=10-15m
メートル系カラット		1メートル系カラット = 200 mg = 2×10-4kg
トル	Torr	1 Torr = (101 325/760) Pa
標準大気圧	atm	1 atm = 101 325 Pa
カロリー	cal	1cal=4.1858J（「15℃」カロリー），4.1868J（「IT」カロリー）4.184J（「熱化学」カロリー）
ミクロン	µ	$1\ \mu=1\mu m=10^{-6}m$

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