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Strategy for the Future Use and Disposition of ^{233}U: Overview

C. W. Forsberg
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January 1998

Prepared by the
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37831-6180
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Predecisional Draft

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ACRONYMS AND ABBREVIATIONS

ANL	Argonne National Laboratory
BAPL	Bettis Atomic Power Laboratory
CEUSP	Consolidated Edison Uranium Solidification Program
CH	contact-handled
DNFSB	Defense Nuclear Facilities Safety Board
DOE	U.S. Department of Energy
DP	DOE Office of Defense Programs
DU	depleted uranium
EIS	environmental impact statement
EM	DOE Office of Environmental Management
ES&H	environmental, safety, and health
FSU	former Soviet Union
GCD	greater confinement disposal
GTCC	greater-than-class-C
HEU	high enriched uranium
HLW	high-level waste
IAEA	International Atomic Energy Agency
IAWG	Interagency Working Group
ICPP	Idaho Chemical Processing Plant
INEEL	Idaho National Engineering and Environmental Laboratory
LEU	low-enriched uranium
LLW	low-level waste
LWBR	light-water breeder reactor
LWR	light-water reactor
MSRE	Molten Salt Reactor Experiment
ND	no data
NEPA	National Environmental Policy Act

ACRONYMS AND ABBREVIATIONS (continued)

NRC	U.S. Nuclear Regulatory Commission
NTS	Nevada Test Site
ORNL	Oak Ridge National Laboratory
ppm	parts per million
REN	road map element number
RH	remote-handled
ROD	record of decision
RW	DOE Office of Civilian Radioactive Waste Management
RWMC	Radioactive Waste Management Complex
SNF	spent nuclear fuel
SNM	special nuclear material
SRS	Savannah River Site
t	metric ton
Tl	Thallium
TRU	transuranic
U	Uranium
WACs	waste acceptance criteria
WGP	weapons-grade plutonium
WIPP	Waste Isolation Pilot Plant
Y-12	Y-12 Plant (Oak Ridge)
^{208}Tl	Thallium-208
^{232}U	Uranium-232
^{233}U	Uranium-233
^{235}U	Uranium-235
^{238}U	Uranium-238
^{213}B	Bismuth-213

PREFACE

This report is one of a series of four reports which maps the strategy for the future use and disposition of uranium-233 (^{233}U). A brief description of each report is included herein.

- ORNL/TM-13550—*Strategy for the Future Use and Disposition of ^{233}U: Overview*. This document is a summary of the path forward for disposition of surplus ^{233}U. It includes required activities, identifies what major programmatic decisions will be required, and describes the potential disposition options.

- ORNL/TM-13551—*Strategy for the Future Use and Disposition of ^{233}U: History, Inventories, Storage Facilities, and Potential Uses*. This document includes the historical uses, sources, potential uses, and current inventory of ^{233}U. The inventory includes the quantities, storage forms, and packaging of the material.

- ORNL/TM-13552—*Strategy for the Future Use and Disposition of ^{233}U: Technical Information*. This document summarizes scientific information on ^{233}U. This includes production methods, decay processes, and the material characteristics. The requirements for storage and disposal are also included.

- ORNL/TM-13553—*Strategy for the Future Use and Disposition of ^{233}U: Options*. This document describes the proposed disposition alternatives, identifies what material in inventory could be treated by each disposition option, and provides an initial analysis of each option. A listing of the legislative or regulatory changes required for each alternative is also provided.

EXECUTIVE SUMMARY

A strategy for the disposition of surplus weapons-usable uranium-233 (^{233}U) has been developed. The strategy (1) identifies the requirements for the disposition of surplus ^{233}U; (2) identifies potential disposition options, including key issues to be resolved with each option; and (3) defines a road map that identifies future key decisions and actions. The disposition of weapons-usable fissile materials is part of a U.S. international arms-control program for reduction of the number of nuclear weapons and the quantities of nuclear-weapons-usable materials worldwide. This report identifies and describes alternative pathways that will be evaluated during the NEPA process to determine the preferred disposition option or options.

Uranium-233 is an isotope produced by irradiating thorium with neutrons within a nuclear reactor. Uranium-233, along with high-enriched uranium (HEU) and plutonium, can be produced in significant quantities for the manufacture of nuclear weapons. The United States has investigated using ^{233}U in nuclear weapons, in reactors, and for other purposes. The United States and several other countries have significant quantities of separated ^{233}U. Somewhat < 2 metric tons (t) of separated ^{233}U-containing uranium are in the U.S. inventory. Half of this material is considered high-quality ^{233}U because it has few isotopic impurities.

Uranium-233 has the chemical characteristics of HEU, but its radiological characteristics are closer to plutonium. Like plutonium, ^{233}U must be handled in alpha enclosures which provide containment to minimize the risk of inhalation of toxic materials by workers during processing operations. The nuclear characteristics of ^{233}U, including the quantity of material necessary to build a nuclear weapon, are closer to those of plutonium than to those of HEU. As defined by the International Atomic Energy Agency, a Category I quantity of ^{233}U in the context of safeguards is 2 kg—the same as that for plutonium—whereas a Category I quantity of HEU is 5 kg. Category I quantities of weapons-usable materials are quantities sufficiently large in terms of building weapons that there is a need for careful accounting and special security measures.

Much of the ^{233}U contains variable quantities of the impurity uranium-232 (^{232}U). One of the daughter products of ^{232}U decays with a 2.6-million-electron volts gamma-ray. This emission produces a radiation field such that much of the material must be stored inside shielded vaults. The radiation is sufficient such as to create major handling complications, but it is not sufficient such as to prevent its use as a weapons-usable material. The high-radiation field does prevent the use of existing commercial nuclear-reactor-fuel-fabrication facilities for blending down much (but not all) of the ^{233}U material in inventory with depleted

uranium (DU) so that the material can be used as commercial nuclear reactor fuel. DU is uranium low in fissile isotopes and containing in excess of 99.3 wt % uranium-238 which is a non-weapons-usable isotope of uranium. The variable characteristics of different batches of ^{233}U in inventory suggest a need for different handling requirements for different batches of ^{233}U.

Two basic requirements for ^{233}U disposition are to (1) convert the isotope to a non-weapons-usable material and (2) dispose of it as either a useful material or a waste. Uranium-233 can be converted to non-weapons-usable material (equivalent to 20 wt % ^{235}U) by isotopically diluting it with DU to <12 wt % ^{233}U. Isotopic dilution is the same approach as is used to convert HEU to non-weapons-usable material. Final disposition of ^{233}U is more complex because of technical and institutional factors.

Alternative paths for disposition of surplus ^{233}U have been identified, as shown in Fig. ES.1. There appear to be no technical barriers to disposition. The path forward requires four major policy decisions. Also, a series of assessments, evaluations, and other actions are required before decisions can be made. The policy decisions that must be addressed follow:

- *How much ^{233}U is surplus?* Some limited uses for the ^{233}U material have been identified. Uranium-233 is used in various procedures that analyze for uranium in many fuel cycle facilities and for safeguards measurements. In addition, one of the daughter products of ^{233}U, bismuth-213 (^{213}Bi), is being investigated for use in radioimmunotherapy for cancer patients and is currently being evaluated in clinical trials. The future demand for ^{213}Bi that can be derived from ^{233}U is not currently well defined. However, if the clinical trials prove successful, the annual demand for ^{213}Bi may exceed that which is available from the entire U.S. supply of ^{233}U. The ^{233}U inventory includes some high-quality material and other materials with isotopic impurities (^{232}U and ^{235}U) that limit or encumber any potential non-weapons use of the material. For civilian uses of ^{233}U, some of the material may not be useful even if there is a future demand for all high-quality ^{233}U.

- *Should ^{233}U that is kept for programmatic purposes be blended with DU to eliminate its potential as a weapons-usable material?* For many potential civilian applications of ^{233}U, isotopic dilution does not eliminate its potential usefulness. With enriched uranium, the policy has been to use different enrichments for different purposes. This variation in enrichment levels allows low-enriched (non-weapons-usable) uranium to be used for applications such as fueling commercial nuclear power reactors. Such a policy also may be appropriate for ^{233}U. However, there are a series of complications including the difficulty of handling such materials in alpha enclosures and different potential uses with various isotopic purity levels.

- *Should the surplus ^{233}U be placed in storage rather than permanent disposal? If surplus ^{233}U is placed in storage, should it be isotopically diluted with DU to eliminate its potential as a weapons-usable material?* If, for any reason, the excess ^{233}U can not be disposed of as a waste, the option remains to store material long term. If the material is sent to long-term storage, the first disposition requirement can be met by isotopically blending the ^{233}U with DU to eliminate its potential usefulness as a weapons-usable material before storing it. These two interconnected decisions can be made at any time if it is decided that final disposal as a waste will require too much time and delay programmatic mandates.

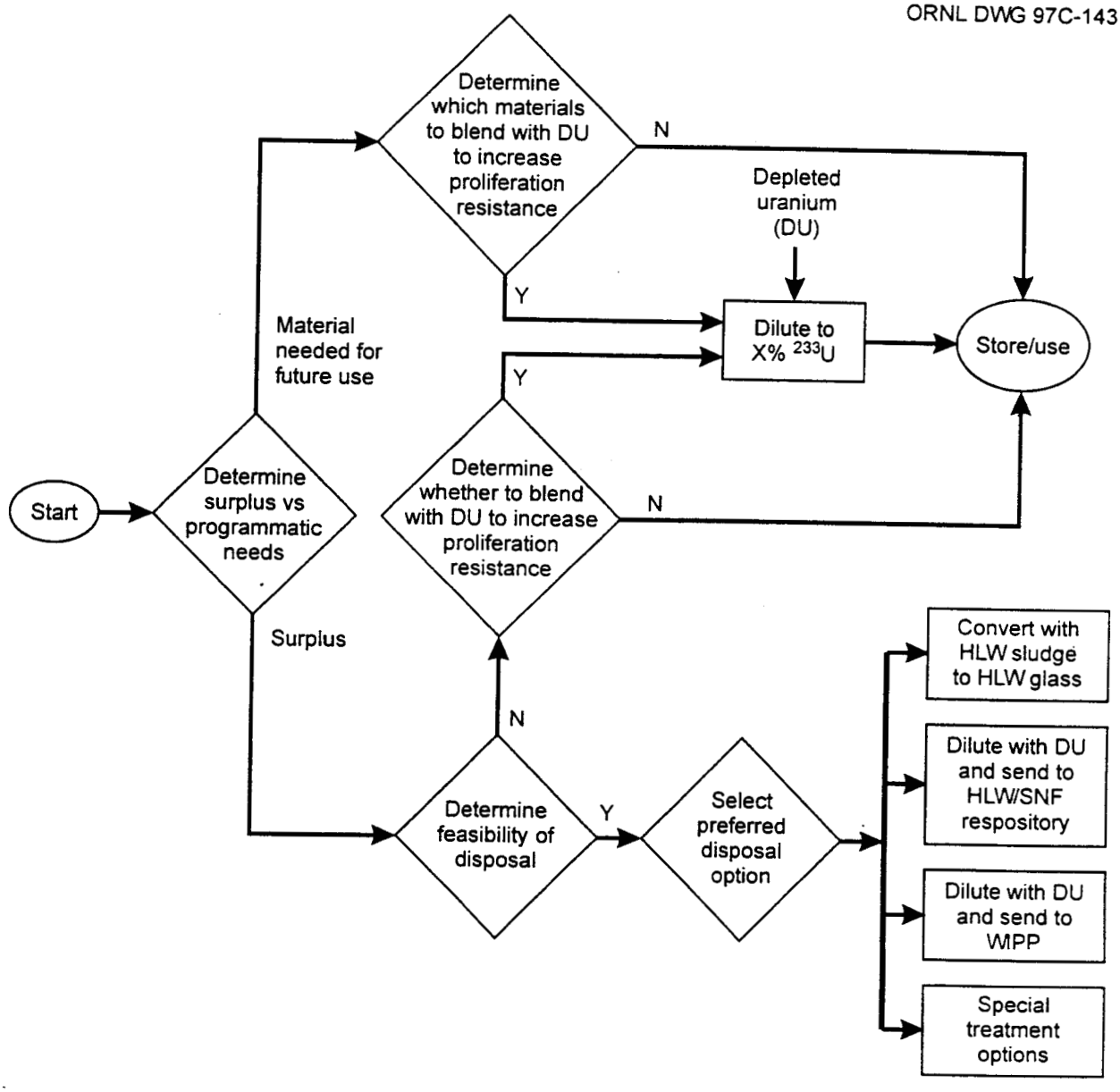

Fig. ES.1. Road map of key decisions and options.

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- *What are the preferred disposition options for surplus ^{233}U?* The options for ^{233}U disposition are significantly different from those for HEU and plutonium for two reasons. First, the quantities of ^{233}U are small compared to those of HEU and plutonium. While new large-scale facilities may be warranted on the basis of economics when processing high volumes of HEU or plutonium, major new facilities are prohibitively expensive for treating the small quantities of ^{233}U materials. Second, some of the ^{233}U inventory contains significant quantities of ^{232}U and its highly radioactive daughter products. The resultant radiation levels prohibit the use of facilities designed for HEU or WGP for processing ^{233}U.

Many disposition options exist. Some of the options can handle only certain batches of ^{233}U in inventory with specific nuclear or chemical characteristics, whereas other options potentially can dispose of the entire inventory. Three major disposition options that can handle the entire inventory have been identified:

- *High-level waste (HLW) glass with DU from HLW sludge.* Uranium-233 is mixed with existing HLW liquids and sludges containing DU. The mixture is then converted to HLW glass for disposal in the spent nuclear fuel (SNF)/HLW repository. This option minimizes waste volumes by beneficial use of DU currently in the HLW sludge. Because HLW facilities are designed for high-radiation levels, the problems with processing ^{233}U with significant radiation levels are minimized. The option also results in a high-quality waste form for ultimate disposal. HLW glass is specifically designed for stable, long-term performance in a repository environment. However, there are engineering issues on how to mix the ^{233}U with the HLW sludge.

- *Special repository waste form for SNF/HLW repository.* Uranium-233 is mixed with DU and converted into a uranium oxide final waste form for disposal in the SNF/HLW repository. This option does not depend upon other programs for treatment of the ^{233}U before its disposal. However, the option results in larger final waste volumes due to the addition of fresh DU to the ^{233}U. This option may require legislative changes.

- *Special repository waste form for the Waste Isolation Pilot Plant (WIPP).* Uranium-233 is mixed with DU and converted into a uranium oxide final waste form for disposal at WIPP. This option does not depend upon other programs for treatment of the ^{233}U prior to disposal. However, the option results in larger final waste volumes due to the addition of fresh DU to the ^{233}U. This option will require legislative changes.

Finally, a series of institutional issues specific to ^{233}U must be addressed because the required institutional structures (that exist for handling HEU and plutonium) are not fully developed for ^{233}U. For example, national and international institutional agreements are needed on the level of isotopic dilution with DU required to convert ^{233}U to a non-weapons-usable material. Such institutional agreements were not created because, for historical reasons, the United States did not choose to use ^{233}U on a significant scale for either weapons or civilian applications (no need). Other countries may make or may have made other decisions. Depending upon ongoing medical research, the United States may eventually use ^{233}U. Establishing an appropriate set of institutional structures for ^{233}U can be a major step toward minimizing the potential use of this material in weapons. Other institutional issues are associated with specific disposal facilities.

Multiple paths for disposition of ^{233}U exist. Moreover, it is likely that the preferred strategy for disposition of ^{233}U will involve more than one option. The path forward for determining the preferred disposition option or options will be to (1) resolve the key technical and institutional issues and (2) initiate a decision-making process pursuant to the National Environmental Policy Act to determine the preferred option or options.

1. INTRODUCTION

1.1 OBJECTIVES

Since the end of the cold war, the United States has been investigating options for disposing of excess fissile materials that can be used to build nuclear weapons. Because Uranium-233 (^{233}U) can be used in the construction of nuclear weapons, it is a candidate fissile material for disposition. Two basic requirements for ^{233}U disposition are to (1) convert ^{233}U bearing materials to non-weapons-usable materials and (2) dispose of it as either useful material or as a waste. This report identifies options for managing ^{233}U, including the options for its disposition. Uranium-233 is a minor fissile material; that is, the quantities of this material are small compared to those of weapons-grade plutonium (WGP) and high-enriched uranium (HEU).

The objectives of this report are to (1) identify the requirements for the disposition of surplus ^{233}U; (2) identify potential disposition options, including key issues to be resolved with each option; and (3) establish a road map that identifies key decisions and actions. The preferred path or paths for disposition of surplus ^{233}U depend upon the weighting of the specific criteria. An initial screening of options was made to identify those that are most feasible and that should be further evaluated as part of the National Environmental Policy Act (NEPA) process before a disposition policy is chosen.

1.2 CAVEATS

There are two important caveats.

- *No decision has been made to dispose of ^{233}U.* The option exists to continue to store the material for possible future use.

- *This report makes no recommendations on the preferred disposition option(s).* It does identify a large number of options, and from those options, it focuses on those that are most practicable. The choice of option(s) will depend upon multiple economic, technical, and institutional considerations, including appropriate NEPA review.

1.3 SPECIAL ISSUES ASSOCIATED WITH ^{233}U

The handling and disposition of ^{233}U are complex issues because of the unique characteristics of ^{233}U and several historical factors. Table 1.1 shows the characteristics of ^{233}U compared to those of the other two weapons-usable materials—WGP and HEU. However, the total material quantity [(<2 metric tons (t)] is small compared to that of WGP and HEU.

Table 1.1. Relative characteristics of weapons-usable materials

Characteristic	Fissile material		
	Plutonium	**HEU**	**^{233}U**
Production	Neutron bombardment of ^{238}U	Separation from natural uranium	Neutron bombardment of ^{232}Th
International Atomic Energy Agency (IAEA) weapons Category I quantity (kg)	2	5	2
Isotopic dilution limit for safeguards[a]	None	<20%	<12%[a]
Isotopic criticality safety limit	Not applicable	1%	0.66%[b]
Chemical properties	Plutonium	Uranium	Uranium
Radiological properties			
Alpha (relative to HEU)	10^4	1	10^3
Gamma	Low	Low	Dependent upon ^{232}U impurity
Containment	Glovebox	Laboratory hood	glovebox/shielded hot cell

[a]No IAEA agreed upon isotopic dilution requirement to convert ^{233}U to non-weapons-usable material. Isotopic dilution is dilution of the ^{233}U with ^{238}U. Calculations indicate that isotopic dilution of ^{233}U with ^{238}U to ~12 wt % is equivalent to 20 wt % ^{235}U in ^{238}U (Forsberg 1998).

[b]Isotopic dilution of ^{233}U with ^{238}U to this limit minimizes the potential for nuclear criticality in disposal facilities (Elam 1997).

As a weapons material, ^{233}U is similar to WGP. The IAEA (1993) defines Category I quantities of weapons-usable materials as 2 kg of WGP, 2 kg of ^{233}U, and 5 kg of HEU. The Category I quantity is that quantity of material requiring nuclear weapons-type security (guns, gates, and guards) to prevent theft of the materials. Uranium-233 can be converted to non-weapons-usable uranium by isotopic dilution with depleted uranium (DU) to <12 wt % (Forsberg 1998). The comparable number for ^{235}U in HEU is 20 wt %.

For geological disposal, the isotopic concentration (Elam 1997) of ^{233}U in ^{238}U to minimize the potential of nuclear criticality (0.66 wt % when isotopically diluted with pure ^{238}U or 0.53 wt % when diluted with DU containing 0.2 wt % ^{235}U) is less than that for ^{235}U (1 wt %) in ^{238}U.

Uranium-233 has radiological characteristics closer to WGP than to HEU. Like WGP, ^{233}U must be handled in an alpha enclosure—a special box in which hazardous materials are processed. The enclosure protects the workers against the toxic material. Unlike ^{235}U and plutonium, most of the ^{233}U contains the uranium impurity ^{232}U in concentrations sufficient such as to require special handing facilities—hot cells or shielded gloveboxes with gamma shielding—because of higher radiation levels. The total quantities of ^{233}U are sufficiently large (~2 t) such that processing is a major concern, but facilities designed to process other fissile materials can not handle ^{233}U. The ^{233}U was produced for weapons and for fuel in advanced nuclear reactors. Because neither of these uses was implemented, large facilities for processing these materials were never built. This situation is in contrast to WGP and HEU, for which facilities exist to process larger quantities of these materials.

The unique history of ^{233}U has implications in terms of institutional issues. U.S. laws, regulations [including U.S. Department of Energy (DOE) Orders], and standards have been developed as needs were identified. The large quantities and large-scale use of WGP, low-enriched uranium (LEU), and HEU have driven the creation of facilities and institutional structures designed to specifically address issues associated with these materials. For example, the Waste Isolation Pilot Plant (WIPP) was designed for disposal of transuranic (TRU) wastes from plutonium and higher actinide processing facilities. In addition to the facility, legal and regulatory structures for the management of TRU waste have been created. The quantities of ^{233}U are sufficiently small such that it would be cost-prohibitive to build special facilities and a separate institutional framework for this material. In many cases, existing and proposed facilities, such as WIPP, are technically usable for disposition of ^{233}U, but the management of ^{233}U was not considered when laws, regulations, and standards for these facilities were developed. Consequently, a set of institutional issues that are unique to ^{233}U must be addressed.

1.4 ASSUMPTIONS

The following assumptions were used in the development of this strategy:

- Uranium-233 determined to be surplus will be offered for international verification and inspection.
- The manner of inspection of ^{233}U inventories under international or bilateral agreements must not yield any data that could be proliferatory.
- Whatever method or methods are ultimately chosen to dispose of and control ^{233}U will take considerable time to decide upon and prepare, so continued interim storage will be required.
- The decision processes will be developed primarily through an open, unclassified process, including appropriate NEPA review.
- Informing and involving the public will be essential factors in planning.
- Budgets will be more constrained than in the past.
- Environmental, safety, and health (ES&H) compliance issues will be a part of all major decisions.
- U.S. actions should be conducted with the view that reciprocal action will be taken by other nations.

1.5 DRIVERS

The following factors will influence the course of action for ^{233}U disposition.

- International treaty environment:

  — Treaty on the Nonproliferation of Nuclear Weapons.
  — Comprehensive test ban treaty.
  — Cutoff of special nuclear material (SNM) production.
  — Safe and secure dismantlement talks.

- Presidential Decision Directive:

  On September 27, 1993, the President issued a decision directive that sets forth a new U.S. initiative for disposition of weapons-usable surplus fissile material. Recent events in Iraq and North Korea, along with the potential availability of nuclear technology from the former Soviet Union (FSU) and the Peoples Republic of China (PRC), have made this a critical issue for the United States and the rest of the world. Specifically, the United States will work to:

  — Reduce proliferation incentives.
  — Strengthen international nonproliferation norms.
  — Undertake global nonproliferation initiatives.
  — Strengthen multilateral export controls.

Key to the success of nonproliferation efforts is broad multilateral support. Our actions must be coordinated with other nuclear nations, where possible, and also be technically effective in stemming proliferation. Disposition-related actions to be taken to implement this policy, directly or indirectly, are to:

— Limit the production of SNM in the United States and discourage it elsewhere.
— Reduce worldwide fissile material stocks.
— Discourage spent nuclear fuel (SNF) reprocessing and uranium enrichment in high proliferation-risk areas.
— Purchase and isotopically dilute HEU inventories from other countries.
— Limit fissile material from U.S. dismantlement to non-weapon uses.
— Place surplus U.S. fissile material under U.S.–IAEA voluntary safeguards.
— Develop verification technologies as appropriate.
— Prevent dissemination of design information during verification.
— Determine surplus U.S. fissile material inventory.
— Achieve multi-nation participation in review and selection of disposition options.

The Presidential Directive assigns the coordination of this initiative to the Interagency Working Group (IAWG) on Nonproliferation and Export Controls. Objectives and milestones established by this IAWG will impact the plans and efforts in this area.

- Emerging proliferants (urgency).
- International limitations for reciprocity.
- Other uses for ^{233}U, including medical applications.
- Budget limitations.
- Current storage and treatment facilities limitations.
- Waste management considerations.
- Existing regulatory context and new laws and compliance rules that may govern the ways in which we can store, process, and ship fissile material.
- U.S. nuclear defense and deterrence strategy.
- Public trust and confidence and evolving public perception and knowledge of risks and benefits of disposition options.

1.6 ISSUES

The following questions must be answered to provide the information needed for decision making.

- What is the appropriate international example we want to set, and what do we want the FSU to implement?
- When and how do we bring in Russian participation in U.S. ^{233}U disposition?
- How do we establish international inspection requirements?
- How do we establish the appropriate international structure for ^{233}U that is comparable to that for HEU and plutonium?
- What is the selection process for disposition options?
- What are the evaluation criteria to support disposition decisions (standards for nonproliferation, cost, ES&H etc.)?
- Should we keep the options for ultimate disposition open? If so, how do we do this?
- How can we best involve stakeholders and build public confidence and trust?

1.7 REPORT ORGANIZATION

This report provides an overview of the strategy for ^{233}U disposition. (As discussed in the preface, three other reports in this series will provide additional detail.) Section 2 of this document describes the history, production, and uses of ^{233}U, while Sect. 3 describes the characteristics of ^{233}U. Section 4 describes the current inventory of materials. Section 5 provides a "road map" to determine what quantities of ^{233}U should be considered as surplus and will require disposition. The road map includes examination of options to convert the ^{233}U into non-weapons-usable material while maintaining its usefulness for civilian application. Section 6 provides a road map to identify disposition options and identifies key decisions that must be made. Section 7 describes the disposition options. Section 8 identifies key schedule considerations that may impact when decisions are made. Section 9 offers some recommendations and presents conclusions.

2. HISTORY, PRODUCTION, AND USES

2.1 HISTORY OF USE IN WEAPONS

Three materials can be produced in large quantities to manufacture nuclear weapons: HEU, WGP, and ^{233}U. Natural, mined uranium contains about 0.7 wt % ^{235}U and 99.3 wt % ^{238}U. Technologies exist to separate the ^{235}U from the other uranium isotopes and produce HEU suitable to use in nuclear weapons. However, a nuclear weapon can not be made from ^{238}U. WGP is primarily ^{239}Pu made by neutron irradiation of ^{238}U. Plutonium is a man-made material. Likewise, ^{233}U is made by neutron irradiation of ^{232}Th, which is the primary isotope in natural thorium. Uranium-233 is a man-made material.

All three fissile materials have been investigated for their use in nuclear weapons. Nuclear weapons in the United States today contain WGP and/or HEU. Uranium-233 was not chosen as a weapons material (Woods 1966; Smith 1963; Rhodes 1995) for several reasons: (1) methods and facilities to make WGP and HEU were developed before methods to make ^{233}U, (2) ^{233}U was more difficult to make with the available technology than were the other materials, and (3) radiation levels from ^{233}U and from associated impurities from the production process are higher (see Sect. 3.2) unless special production techniques are used. The development of such techniques to produce high-purity, lower-cost ^{233}U occurred after major decisions were made as to which weapons materials to use. (Radiation levels in this context refer to penetrating gamma radiation fields to which workers and military personnel are exposed while moving, handling, and storing a weapon containing ^{233}U.)

2.2 PRODUCTION OF ^{233}U

Uranium-233 has been produced by neutron irradiation of thorium at the Hanford Site (Hanford) (Jackson; Atlantic Richfield Hanford Company March 11, 1968) in Washington, the Savannah River Site (SRS) (Severence 1996) in South Carolina, the Indian Point I Nuclear Power Plant in New York, and several other sites (DOE February 6, 1996). The targets or SNF containing the ^{233}U were chemically processed, and the ^{233}U was recovered. This separations operation has occurred at Hanford, SRS, and the West Valley commercial reprocessing plant in New York. Significant quantities (~900 kg) of ^{233}U remain in SNF stored at several sites. Further information on production and history is provided in companion reports (Bereolos 1997a; Bereolos 1997b).

2.3 NON-WEAPONS USES OF ^{233}U

The U.S. government has previously investigated ^{233}U for use in research, navy, space, and power reactors. However, these programs were limited in size. There continues to be a long-term interest in ^{233}U-thorium nuclear fuel cycles because of several characteristics of the ^{233}U fuel cycle. Uranium-233 can be made from thorium, which is about four times more abundant than uranium found in the earth's crust. Also, there are technical advantages of using ^{233}U in some types of nuclear reactors. There may be certain nonproliferation and safeguards advantages with the ^{233}U-thorium fuel cycle if breeder reactors are used. Last, the ^{233}U fuel cycle produces wastes with fewer long-lived actinides. This may simplify disposal of radioactive wastes.

Several countries (India, France, Russia, PRC, and Japan) are continuing to investigate ^{233}U as a nuclear fuel and for other purposes. One small research reactor (in India) and several other special reactors use ^{233}U as a fuel. The Indian program includes separation of pure ^{233}U from irradiated thorium. Except for the Indian research reactor, the other ^{233}U-fueled reactors use fresh fuel containing ^{235}U, generate ^{233}U from thorium in the fuel during irradiation, and burn the ^{233}U as it is generated in the fuel in the reactor. No reactor in the United States today uses ^{233}U, nor are there any plans for use of separated ^{233}U as a reactor fuel (versus ^{233}U generated and used during irradiation of nuclear fuel containing thorium).

There are several past, current, and potential uses for ^{233}U. It has been used in some types of nuclear weapons development tests. Small quantities (<1 kg/y) are currently used in the United States in analytical tests for fissile material safeguards and other purposes. It has also been studied as a fuel for some deep-space and other special-purpose reactor missions. A ^{233}U reactor for low power levels may be made smaller than reactors fueled with HEU or WGP.

One potential large-scale use of ^{233}U is currently being investigated. This is to use the decay product Bismuth-213 (^{213}Bi) from ^{233}U for cancer treatment (Feinendegen and McClure, 1996). Currently, ^{213}Bi is undergoing clinical trials at the Memorial Sloan-Kettering Cancer Center in New York City for treatment of cancer. If this application is fully implemented, all of the pure ^{233}U in inventory would be required. However, some of the isotopically less pure ^{233}U inventory may or may not be suitable for this application. The investigative phase of this program is expected to be completed within several years, and a decision as to whether to approve its use is to be made shortly thereafter.

The ^{233}U inventory includes batches of ^{233}U mixed with different levels of other uranium isotopes. Different applications require different ^{233}U purities; thus, the potential exists to simultaneously have a shortage of high-purity ^{233}U (for medical applications) and surplus low-grade (but weapons usable) ^{233}U.

3. CHARACTERISTICS OF ^{233}U

The chemical, radiological, and institutional characteristics of ^{233}U impact the choice of disposition options. Key properties are summarized herein. More detailed information is available in a companion report (Bereolos 1997a).

3.1 CHEMICAL CHARACTERISTICS

Uranium-233 is chemically similar to natural and enriched uranium. Consequently, the same chemical processes used for natural uranium and HEU are usually applicable to ^{233}U. The ^{233}U has a higher specific radioactivity than natural uranium or HEU (Sect. 3.2); thus, certain radiation-induced chemical reactions are faster in uranium containing significant quantities of ^{233}U. The higher radiation-induced chemical reaction rates are important in certain situations such as long-term storage where the higher-radiation levels of ^{233}U require that storage containers and ^{233}U storage forms do not contain organics (plastics etc.) or water that degrade with higher radiation levels.

3.2 RADIOLOGICAL CHARACTERISTICS

The radiological worker-protection requirements for high-quality ^{233}U are similar to those for WGP. The primary hazard from ^{233}U is alpha radiation, which is also the primary health hazard from WGP. The alpha activity of ^{233}U is three orders of magnitude higher than that of HEU and about one order of magnitude less than that of WGP. Consequently, the handling and containment requirements (alpha enclosures etc.) for ^{233}U are similar to those for WGP.

All ^{233}U contains some ^{232}U. In the production of ^{233}U, some ^{232}U is produced. The concentrations of ^{232}U depend upon the specifics of the production techniques for ^{233}U. The ^{232}U has a decay product, thallium-208 (^{208}Tl), which, in turn, decays to stable lead while producing a high-energy 2.6 MeV gamma ray. The concentration of ^{232}U determines the radiation shielding required to protect workers. Superior-quality ^{233}U contains very low levels [~1 part per million (ppm)] of ^{232}U and has correspondingly low levels of gamma radiation. Low-quality ^{233}U with high concentrations of ^{232}U (tens to hundreds of ppm) and associated radioactive decay products requires heavy radiation shielding and remote-handling (RH) operations to protect workers from gamma radiation.

There is an important radiochemical characteristic of this system. If uranium is chemically purified and the decay products are removed, ^{233}U with significant concentrations of ^{232}U can be processed and converted

into desired forms in unshielded gloveboxes and other alpha enclosures without significant radiation exposure to workers. It takes time (days to weeks) for the ^{232}U radioactive decay products that emit gamma rays to build up to high enough concentrations such as to require thick radiation shielding to protect the workers. The buildup and decay of ^{233}U with ^{232}U impurities are shown in Fig. 3.1 for ^{233}U with high concentrations of ^{232}U. The curve for gamma-ray generation vs time since purification of the uranium shows the buildup of the gamma-radiation doses with time.

The radiological characteristics of ^{233}U have historically determined what uranium was to be managed as ^{233}U. If a mixture of uranium contains several isotopes, the mixture is handled as ^{233}U provided that the ^{233}U is the primary hazard. In practice, this procedure implies that uranium materials containing somewhat >1 wt % ^{233}U would be handled as ^{233}U.

3.3 NUCLEAR CHARACTERISTICS

The nuclear characteristics of ^{233}U are significantly different than those of HEU. The minimum critical mass of ^{233}U is ~0.52 kg, which is less than that of HEU; thus, facilities designed for HEU may or may not be suitable for storing or processing of ^{233}U.

3.4 INSTITUTIONAL CHARACTERISTICS

Although ^{233}U has been investigated for many applications, it has not been used on a large scale in the United States. The total inventory of separated ^{233}U is (1) small compared to that of HEU and WGP and (2) limited to a few sites. Because there have been no large-scale uses of ^{233}U, the institutional structure required for long-term management of ^{233}U does not fully exist.

3.4.1 Safeguards

National and international safeguards requirements [DOE orders, U.S. Nuclear Regulatory Commission (NRC) regulations, IAEA agreements] for weapons-usable materials have been developed for HEU and WGP; however, the requirements are not developed fully for disposition of surplus ^{233}U. For uranium containing ^{235}U, these regulatory requirements recognize that only HEU can be made into nuclear weapons. Natural uranium, DU, and LEU do not require the safeguards and security required of weapons-usable HEU. For disposition of surplus HEU (DOE June 1996a; DOE July 29 1996), the U.S. policy is to blend HEU with DU to make LEU for use in commercial nuclear power plants. It is universally recognized that this process eliminates the use of this material for nuclear weapons and eliminates the need for weapons-materials-type security.

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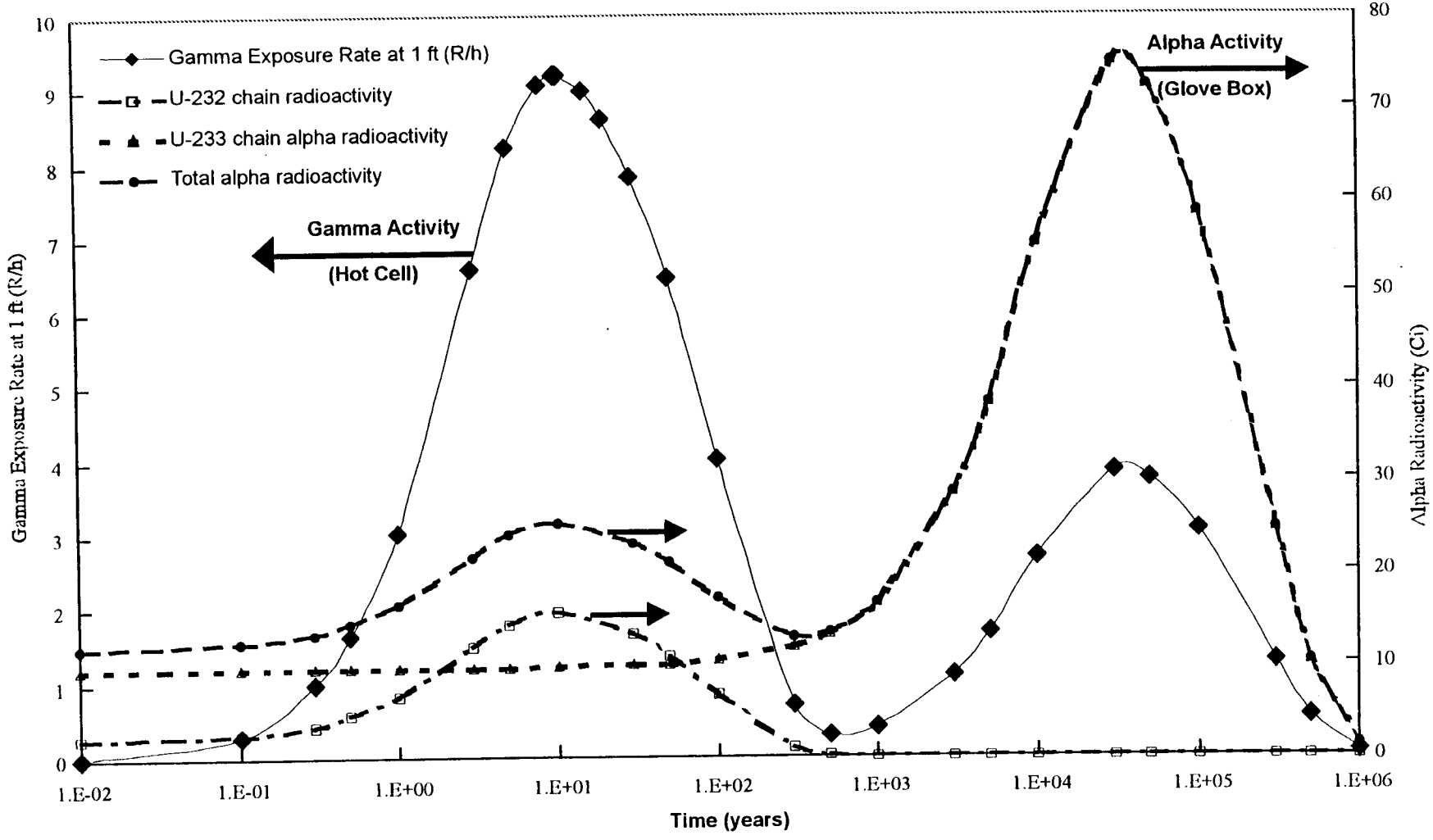

Fig. 3.1. Alpha activity and gamma exposure rate at 1 ft as a function of time calculated for 1 kg ^{233}U (with 100 ppm ^{232}U) as a loose-pour powder (1.5 g/cm^3) contained in a 3-in. diam by 6-in. tall can with 20-mil-thick steel walls.

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The regulations (IAEA 1993) do not recognize that mixing ^{233}U with DU will create a mixture that is unsuitable for use in manufacturing nuclear weapons. However, it is widely recognized within the technical community that isotopic dilution of ^{233}U with DU will eliminate the ^{233}U as a weapons-usable material; however, all materials containing significant quantities of ^{233}U are treated as weapons-usable material. Historically, there never was any consideration of converting ^{233}U to a non-weapons-usable material; thus, the required regulatory structure was not established. The technical basis for converting ^{233}U to non-weapons-usable material by diluting it with ^{238}U is understood (Forsberg 1998), but the regulations and other institutional agreements are not in place. Because one of the goals of the Fissile Materials Disposition Program is to reduce the proliferation risks from these SNM in other countries, institutional agreements on the level of isotopic dilution that eliminates the weapons potential of ^{233}U is required. The isotopic purity that renders ^{233}U non-weapons-usable (<12 wt %) is less than that for HEU (<20 wt %).

3.4.2 Waste Management

In the United States, currently no facilities exist for disposing of wastes containing only significant concentrations of ^{233}U as the predominant radionuclide. Wastes would be generated from any processing or handling operations. Historically, wastes containing ^{233}U have been treated as TRU wastes (i.e., treated similarly to wastes containing plutonium) because the primary hazard—alpha radiation—is identical in both waste types and, thus, the same types of disposal facilities are required. WIPP has been built to dispose of defense TRU wastes. However, the enabling legislation does not permit disposal of wastes containing only ^{233}U or wastes from nondefense activities in these facilities.

4. ^{233}U INVENTORY DESCRIPTION

4.1 CURRENT INVENTORIES

The unclassified, separated inventory of ^{233}U is shown in Fig. 4.1 and listed by site in Table 4.1. Detailed inventory information is available in a companion report (Bereolos 1997b). Uranium-233 in SNF, irradiated targets, and wastes are not included in these numbers. The inventory contains 1,801 kg of uranium in a total of 1505 packages at multiple sites. Most of the ^{233}U and most of the packages are located at Oak Ridge National Laboratory (ORNL) in the National Repository for ^{233}U. The material is primarily in the chemical form of oxides stored in stainless steel or aluminum cans. The ^{233}U is typically packaged in double containers, the inner container being made of stainless steel or aluminum. Figure 4.2 shows the more typical types of inner and outer storage containers at ORNL.

The total inventory of separated ^{233}U is expected to increase by several percent (~31 kg ^{233}U in a total of ~37 kg uranium) over the next several years as material associated with the Molten Salt Reactor Experiment (MSRE) at ORNL is processed (DOE 1997) to resolve safety concerns identified in Defense Nuclear Facility Safety Board (DNFSB) recommendation 94-1. The reactor SNF contains ^{233}U, which will be separated from this fuel to minimize long-term safety concerns (natural processes are slowly separating the ^{233}U from the fuel with the potential to create significant safety problems). There are several other batches of waste from which ^{233}U may be recovered to minimize specific safety concerns. The resultant separated ^{233}U would be added to the national inventory as it is recovered.

4.2 QUALITY

The inventory can be divided into eight major batches based on the characteristics of the ^{233}U. The remainder of the inventory is a 9th batch (of miscellaneous small quantities). The batches and their characteristics are shown in Fig. 4.3 and Table 4.2. The quality of the batches can be measured by two indexes. Most batches are almost isotopically pure ^{233}U—except for Batches 1 and 8, both of which contain significant quantities of ^{235}U. For most applications (except nuclear weapons and power reactors), the high ^{235}U content severely diminishes the value of the ^{233}U. The second index of quality is the concentration of ^{232}U in the ^{233}U. If the ^{232}U concentration is high, the longer-term radiation levels associated with these batches will be high. The concentration of ^{232}U in units of parts of ^{232}U per million parts of ^{233}U is shown in the parenthesis above each column. The first two batches have high ^{232}U concentrations.

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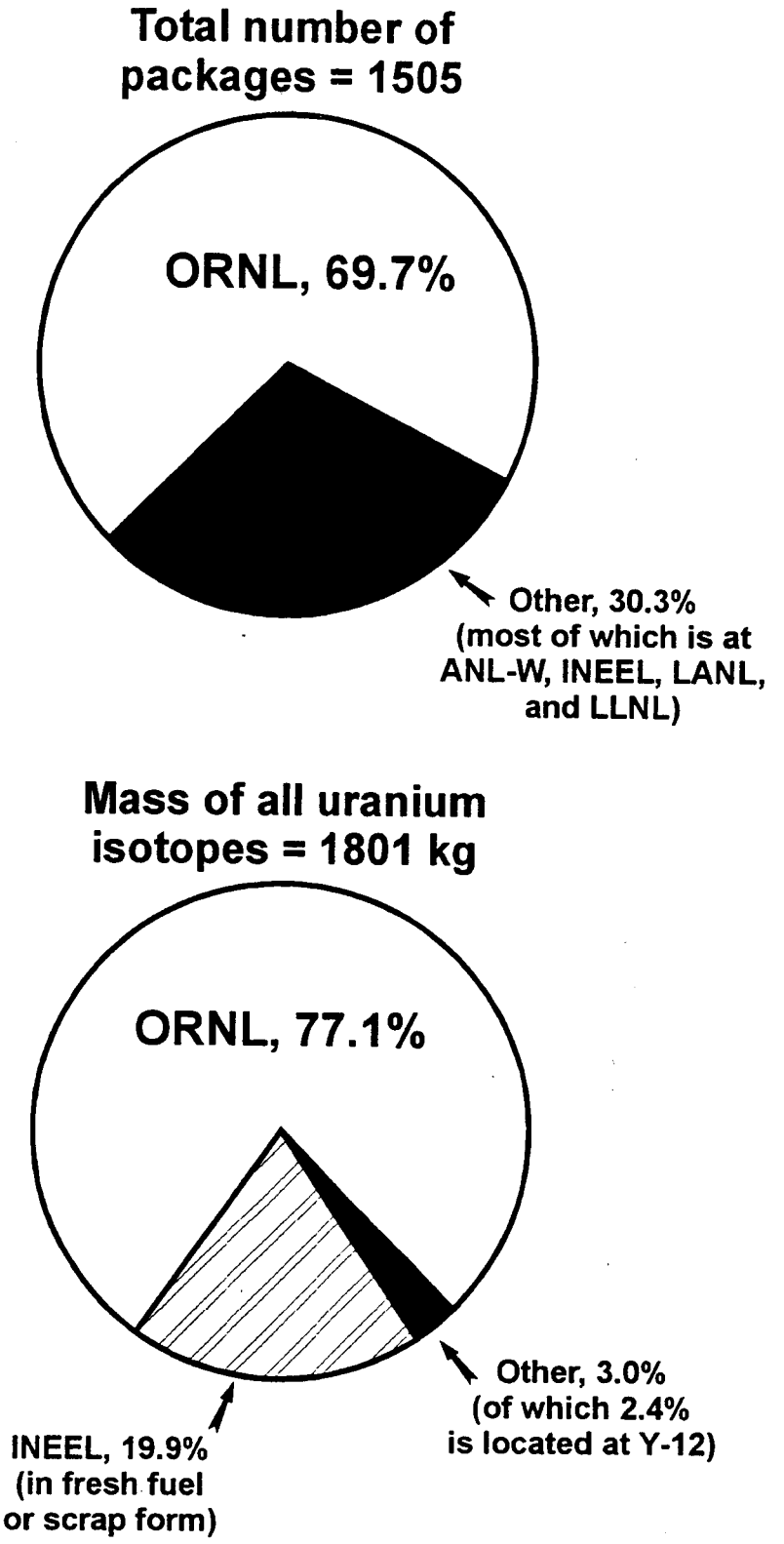

Fig. 4.1. Uranium-233 inventories in the United States.

Table 4.1. ^{233}U Inventories and characteristics[a]

Site	No. pkgs.[b]	Total U[c] (kg)	^{233}U[c] (kg)	^{235}U[c] (kg)
Argonne National Laboratory (ANL)–East	5	*	*	0
ANL–West	63	<0.2	<0.2	0
Bettis Atomic Power Laboratory[d] (BAPL)	13	0.4	0.4	*
General Atomics	2	*	*	*
Hanford	3	0.6	*	0
Idaho National Engineering and Environmental Laboratory (INEEL)/Idaho Chemical Processing Plant[e,f] (ICPP)	186	359	352	0
Lawrence Livermore National Laboratory	50	3	3	0
Los Alamos National Laboratory	109	7.2	7.1	0
ORNL	1049	1388	427	796
Pacific Northwest National Laboratory	15	*	*	0
Rocky Flats Environmental Technology Site	5	*	*	0
Oak Ridge Y-12 Plant (Y-12)[g]	5	43	0.8	39
Total	1505	1,801	790	835

[a]Excludes ^{233}U in materials classified as waste (unless specifically noted), SNF, and irradiated thorium targets.

[b]External packages. In some cases, multiple small packages in a single outer package.

[c]An asterisk (*) is used to represent mass quantities of material <0.1 kg.

[d]Includes TRU waste materials, which are stored in four 55-gal drums. The mass of waste material is currently known to be in surplus of 21 kg.

[e]Some additional materials are categorized as waste or SNF that may be candidate ^{233}U materials.

[f]Includes contributions from 145 drums of unirradiated fuel materials (<35.1 kg U) stored at the INEEL Radioactive Waste Management Complex (RWMC).

[g]May be classified as part of the ^{233}U or HEU inventory.

Fig. 4.2. Storage canisters for ^{233}U at ORNL.

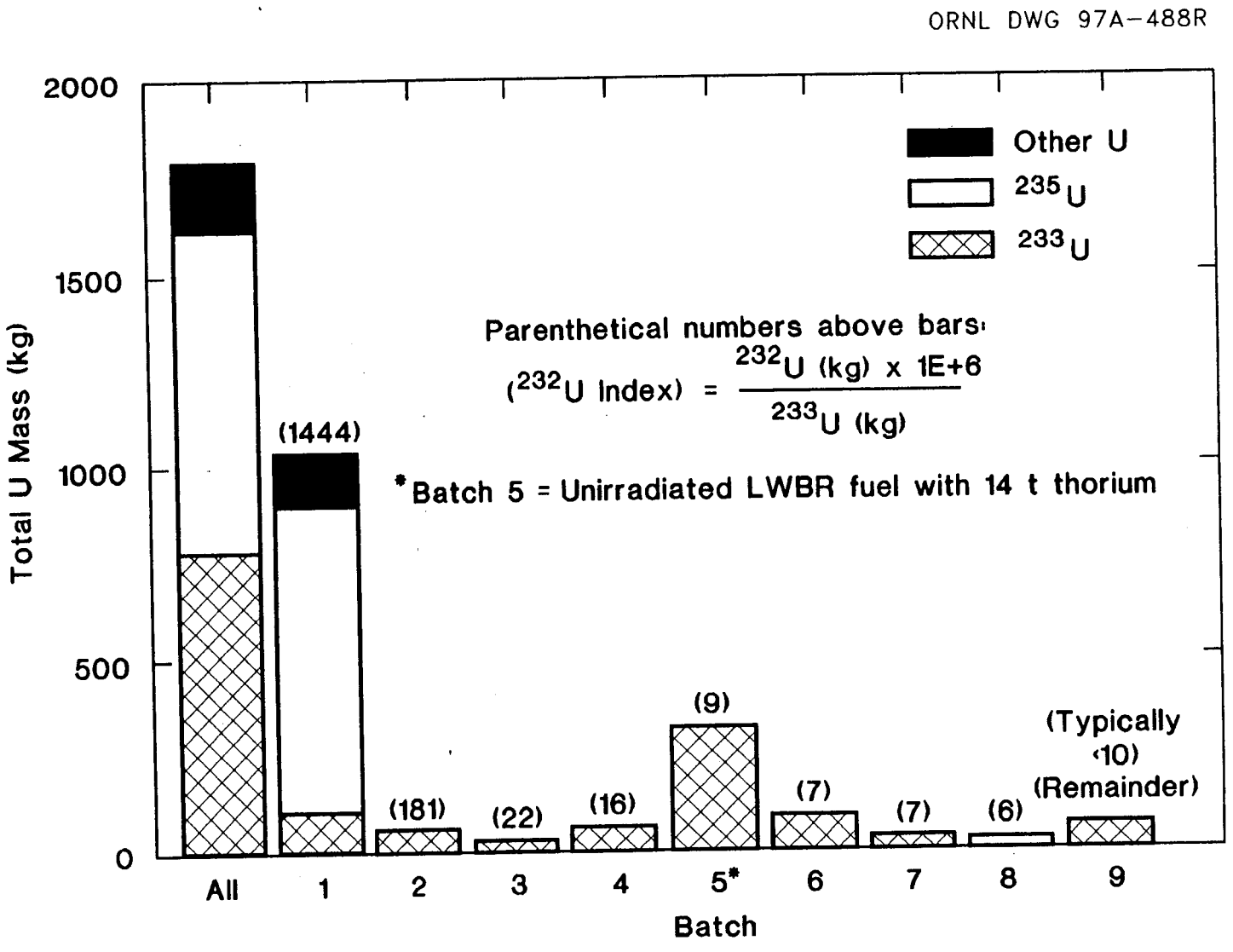

Fig. 4.3. Quality of ^{233}U in inventory, by batch.

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Table 4.2. Quality of major batches of ^{233}U in inventory

Batch No.	Site	Material	Total U (kg)	Uranium isotopic composition			Measures of quality[b]	
				^{235}U (kg)	^{233}U (kg)	^{232}U (ppm)[c]	Total U (kg)/ ^{233}U (kg)	^{232}U (kg) × $10^6/^{233}U$ (kg)
1	ORNL	U_3O_8 monolith in >400 welded stainless steel cans [Consolidated Edison Uranium Solidification Program (CEUSP) material]	1042.6	796.3	101.1	140	10.3	1440
2	ORNL	UO_x powder in 140 welded inner aluminum cans	67.4	0.00	61.6	165	1.1	181
3	INEEL/RWMC	Unirradiated rods and pellets in 145 drums	35.1	0.00	34.2	21	1.03	22
4	ORNL	U_3O_8 monolith in 27 welded stainless steel cans placed in tin-plate cans	65.2	0.00	60.3	15	1.08	16
5	INEEL/ICPP	Unirradiated light-water breeder reactor (LWBR) fuel with 14 MT natural thorium	323.5	0.00	317.4	9	1.02	9
6	ORNL	UO_x powder in 174 stainless steel screw-top cans	96.5	0.00	91.2	7	1.06	7
7	ORNL	UO_x powder in 1743 welded stainless steel plates	45.7	0.00	44.8	7	1.02	7
8	Y-12[d]		42.6	38.70	0.8	0.113	53.2	6
Remainder	Various	Variety of material forms, packages, and compositions	82.1	<0.1	79.4	<10	1.03	<10
Total			1800.7	835.00	790.8			

[a]These data do not represent the entire inventory total because many small batches are not listed.
[b]A low number implies higher quality.
[c]Based on total uranium.
[d]May be classified as impure HEU or within the ^{233}U inventory.

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4.3 OTHER SPECIAL CHARACTERISTICS

Several of the batches have other unusual characteristics related to packaging, material form, and chemical purity that are important in any evaluation of the use, storage, and disposition options for ^{233}U. Two batches are of particular note because of the quantity of material involved, presence of other nuclides, physical form, or packaging; these two batches are described briefly below. Most other ^{233}U materials in inventory are primarily uranium oxides with limited impurities.

4.3.1 CEUSP Material

The material form and packaging system for the CEUSP uranium (Batch 1) is unusual—the ^{233}U oxide is a monolithic block physically bound to the stainless-steel container. The oxide can be removed from the package only by acid dissolution or an equivalent mechanical means. If the container is cut open, the uranium oxide can not be removed physically from the package using conventional techniques (i.e., powder pouring) and must be chemically or mechanically extracted.

A special loading procedure was used for the CEUSP uranium. Each stainless steel container was placed vertically in a high-temperature furnace, and the ^{233}U was added as a concentrated uranyl nitrate solution. In the package, the nitrate decomposed to an oxide. This created a cast-in-place monolithic ceramic in each storage package. This approach results in an extremely stable storage form with minimum potential for creation of hazardous aerosols during handling and storage operations. However, such highly-stable storage forms require additional efforts to recover the uranium from the storage package.

The CEUSP material also contains cadmium and gadolinium that were added as neutron absorbers to prevent nuclear criticality during the prolonged (17 year) storage of this fissile material as a nitrate solution (prior to its solidification).

4.3.2 LWBR Fresh Fuel

The second unusual batch is the LWBR unirradiated fuel (Batch 5). This is most of the ^{233}U inventory at INEEL. This fuel has two unusual characteristics. First, it is unirradiated nuclear fuel pellets packaged in metallic fuel rods. Second, the ^{233}U has been mixed with thorium oxide such that the average ^{233}U content is only about 2 wt %. The ^{233}U content of the fuel rods and pellets varies widely depending upon intended location within the fuel assemblies. The dilute form of this feed requires that if this material is to be disposed of, then either: (1) the disposition option must be able to tolerate large quantities of thorium oxide or (2) separation of the ^{233}U from the thorium is required before the disposition of the ^{233}U. The ^{233}U in this batch of material is of a high quality; thus, it may be a candidate for future use.

4.4 UNCERTAINTIES IN ^{233}U CHARACTERISTICS

The relatively pure ^{233}U in inventory is well characterized. However, there are some packaging and material form uncertainties about the (1) unirradiated rods and pellets (Batch 3) and (2) unirradiated LWBR fuel (Batch 5). In both of these batches, the quantities of ^{233}U are known, but the chemical composition and characteristics of the other materials in the containers are not fully known. These batches resulted from the operation and shutdown of the LWBR program and contained essentially all the residues and scrap from LWBR fuel fabrication operations. Some parts of these batches are very well characterized LWBR fuel rods. Other parts are poorly characterized. In addition, the miscellaneous category (Batch 9) contains packaging and material forms that are not well characterized.

Additional ^{233}U for disposition (in addition to that from the MSRE) may be recovered from SNF and other waste streams sources as part of ongoing safety and environmental management activities. The expectation is that the quantities of such material will be small compared to existing inventories.

5. AMOUNT OF URANIUM-233 THAT SHOULD BE KEPT AND THAT SHOULD BE SURPLUS

5.1 KEY QUESTIONS

Three questions must be answered before disposal of surplus ^{233}U can be considered:

- *How much ^{233}U is needed for future use?* Uses exist for ^{233}U. Not all of the inventory is surplus. Some of the ^{233}U inventory will be reserved for future national needs.

- *Can the ^{233}U be isotopically diluted with ^{238}U (DU) to eliminate its potential use for manufacture of nuclear weapons and still remain useful?* Essentially all of the ^{233}U in the inventory is in the form of weapons-usable materials. For some applications, such as production of medical isotopes, it may be feasible to eliminate the weapons potential of the ^{233}U by isotopically diluting it with ^{238}U while maintaining its future potential usefulness. [In this context, it is noted that in the management of ^{235}U, different enrichments of ^{235}U are used for different purposes. In many applications, weapons-usable HEU is not required. Specifically, the United States has adopted a policy to encourage the use of non-weapons-usable ^{235}U (<20 wt %) for fueling of research reactors. A similar isotopic-dilution strategy may be feasible for ^{233}U that would minimize safeguards associated with future uses of ^{233}U.]

- *If isotopic dilution is established as a method to eliminate its potential use in nuclear weapons, should surplus ^{233}U be immediately isotopically diluted to eliminate its potential as a weapons-usable material?* There are two requirements for disposition of surplus ^{233}U: convert the material to a non-weapons-usable uranium and dispose of the surplus material. Economic and technical reasons support conducting both operations simultaneously. However, the option exists to immediately convert surplus ^{233}U to non-weapons-usable material to meet national nonproliferation goals and then to dispose of the surplus material later in time.

5.2 PLAN FOR RESOLUTION OF KEY QUESTIONS

The plan for resolution of these key issues is shown in Fig. 5.1. A total of 14 steps are identified by road map element number (REN). Most of the steps are needed to provide information to answer the three questions identified above. Current activities and future plans are described in the context of this plan.

5.2.1 Define and Characterize ^{233}U Inventory (REN 1)

The starting point for the disposition of surplus ^{233}U is a determination of the inventories and characteristics of ^{233}U in the United States. For this program, only separated ^{233}U is included (i. e., ^{233}U in irradiated SNF waste, and irradiated targets are excluded). A national inventory of ^{233}U has been created (Bereolos 1997b). The quantities, quality, and locations of the ^{233}U inventory have been identified. Additional information is currently being collected to better define the characteristics of the materials in inventory.

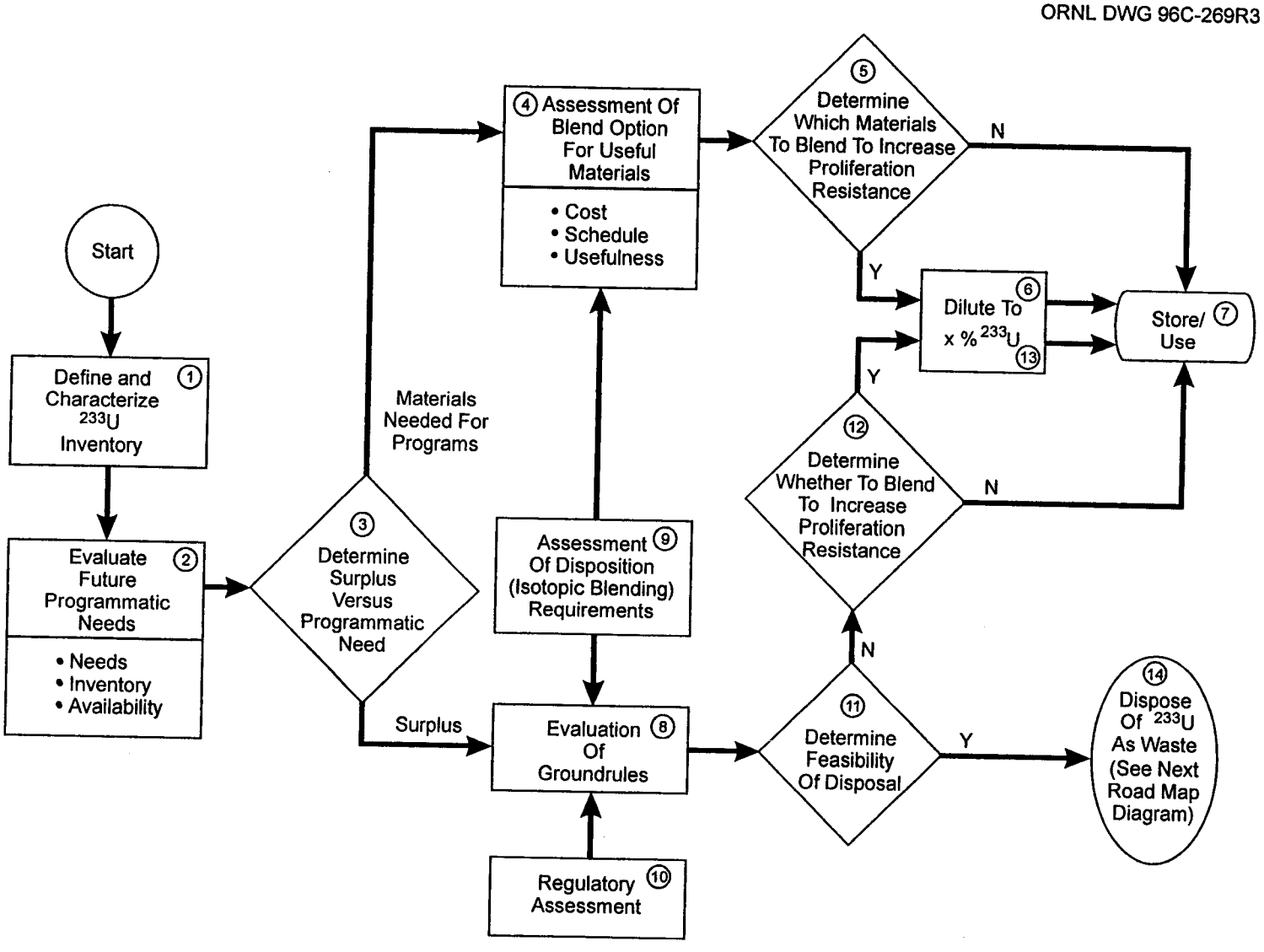

Fig. 5.1. Road map for decisions relating to the storage, use, and conversion of ^{233}U to non-weapons-usable material.

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5.2.2 Determination of Surplus vs Programmatic Needs (REN 3)

The first major programmatic decision for disposition of ^{233}U is to determine what materials should be retained for existing or new programs and what materials should be declared surplus. Evaluations of needs and inventory availability (REN 2) are required before such decisions are made. A major complication is associated with the uncertainties in future demand. In particular, one of the ^{233}U daughter products (^{213}Bi) is being evaluated in clinical trials for the treatment of cancer. Clinical trials (Feinendegen and McClure, 1996) will be conducted over the next several years. Some of the ^{233}U inventory (Batches 1 and 8) may contain sufficient isotopic impurities (^{232}U and ^{235}U) to eliminate or prohibitively encumber any potential non-weapons use of the material.

Future programmatic needs for ^{233}U are currently being evaluated (REN 2) with two activities. The first activity is to survey users and likely users to determine their potential future requirements. The second activity is to determine what inventory should be kept to meet these needs. The ^{233}U inventory includes batches having very different nuclear, radiological, and chemical characteristics (Table 4.2). Not all of the existing inventory may be useful for the projected needs (i.e., it is possible to simultaneously have surplus ^{233}U in inventory but a shortage of ^{233}U with specific characteristics to meet future programmatic needs). Three factors are used to measure the quality of ^{233}U.

- *^{232}U content.* Inventories with high concentrations of ^{232}U indicate low-quality associated with (1) high radiation levels, (2) restrictions on possible future uses because of the radiation levels, and (3) high processing costs to provide protection of workers from the high radiation fields.

- *^{235}U and ^{238}U content.* Inventories with high concentrations of non-^{233}U uranium isotopes indicate high volumes of uranium per unit of ^{233}U. Although the ^{235}U impurity is acceptable for use in nuclear weapons and some power reactors, it may not be acceptable or significantly inconvenience for some other applications. If the other uranium isotope is ^{238}U, the inventory mass is increased, but weapons safeguards issues can be avoided and nuclear criticality issues can be reduced if the fissile concentrations are sufficiently low.

- *Chemical/physical and packaging characteristics.* The ^{233}U inventory is in multiple chemical/physical forms and packaging systems. Large fractions of material are cast-in-place oxide monoliths in welded stainless steel containers; Zircalloy-clad rods of uranium-thorium dioxide fuel pellets made with large amounts of natural thorium; or oxide powders packaged in stainless steel screw-top containers, welded aluminum cans, or welded stainless steel plates. A variety of other chemical forms exist in other, diverse packaging configurations in the inventory. The diversity of chemical and physical material characteristics and packaging systems influences ^{233}U usefulness and complicates the approach to its use or disposition.

The current applications for ^{233}U (safeguards, etc.) indicate that some pure materials will be retained. The very poor quality of other batches of ^{233}U suggest that some material may be declared surplus.

5.2.3 Determination of What Useful Materials to Blend to Increase Proliferation Resistance (REN 5)

The second major programmatic decision is whether to isotopically dilute with DU some or all of the ^{233}U that is kept for future use. Isotopic dilution converts the ^{233}U to a non-weapons-usable material. In this context, it is important to understand the U.S. policy on the use of ^{235}U. For enriched uranium containing ^{235}U, the United States has encouraged a policy of using enriched uranium for civil applications with uranium enrichment levels <20 wt % ^{235}U. At these enrichment levels, the material is not considered weapons-usable material. This historical strategy for ^{235}U may be relevant to ^{233}U for many of its uses. Two activities are required to provide the information needed for such decisions.

First, assessments (REN 4) are required to determine those future applications for which isotopic dilution of ^{233}U with ^{238}U can eliminate the usefulness of ^{233}U as a weapons material while not eliminating its use for such applications. The assessments also are required to evaluate technical and economic trade-offs. Isotopic dilution of ^{233}U will increase future processing costs and storage space requirements for this material, but dilution will (1) reduce safeguard concerns and requirements and (2) eliminate the material as a candidate weapons-usable material (a major disposition program goal).

Second, if ^{233}U is to be isotopically blended with DU to eliminate its potential use as a weapons material, there must be institutional agreement (REN 9) on the level of isotopic dilution required. U.S. regulations and IAEA guidelines define different safeguards and security requirements for different levels of ^{235}U enrichment. Uranium ≥20 wt % ^{235}U is considered HEU that is suitable for weapons manufacture. In contrast, the current IAEA regulations and U.S. regulations do not recognize isotopic dilution as a mechanism for converting ^{233}U into a non-weapons-usable material. However, it is generally accepted as a technical fact that isotopic dilution with DU eliminates the potential for using ^{233}U as a weapons-usable material. Until the end of the cold war, there was no consideration of isotopically diluting ^{233}U to non-weapons-usable material; hence, regulations addressing this issue were not developed.

Technical studies have been completed recently to determine the required isotopic dilution to convert ^{233}U to non-weapons-usable material (Forsberg 1998); however, appropriate rule-making procedures must be implemented to create an official level of isotopic dilution equivalent to that which exists for ^{235}U. Information for this rule-making activity is being assembled. The nuclear characteristics of ^{233}U suggest that ^{233}U isotopically diluted to ~12 wt % is equivalent to ^{235}U isotopically diluted to 20 wt %. The results of this activity are required for the assessment of blend options (REN 4).

If a decision is made to (1) keep some fraction of the ^{233}U and (2) isotopically dilute it with DU, the ^{233}U and DU will be dissolved in nitric acid or an oxide eutectic and thoroughly mixed together (REN 6). Co-dissolution and mixing of the two uranium isotopes in solution ensure complete isotopic dilution. The solution would then be converted to an appropriate chemical and physical form for storage or use. Acid dissolution is the current technology for blending ^{233}U with other uranium isotopes. After its processing and packaging, the ^{233}U will be sent to storage or to users, as is appropriate (REN 7).

5.2.4 Determine the Feasibility of Disposal (REN 11) and Determine Whether to Blend to Increase Proliferation Resistance (REN 12)

The third major programmatic decision is whether surplus ^{233}U should be retained at a high fissile concentration or be immediately isotopically diluted to eliminate its potential as a weapons-usable material. This is a two part decision based on the observation that there are two major objectives for disposition of surplus ^{233}U: (1) convert the ^{233}U to non-weapons-usable material and (2) dispose of the surplus ^{233}U as a waste. These two requirements can be accomplished by a set of sequential actions or simultaneous actions.

- Surplus ^{233}U can be converted to a non-weapons-usable form by isotopically diluting it with DU and placing the resultant material in storage. At a later date, the ^{233}U can be converted to a final waste form (including additional isotopic dilution for criticality safety in a geological repository) for disposal.

- Surplus ^{233}U can be converted simultaneously to a non-weapons-usable material by isotopic dilution while converting the mixture into a final waste form for disposal. Simultaneous conversion minimizes processing and handling operations with significant cost and safety advantages.

However, combining the two requirements may extend the period before the ^{233}U is converted to non-weapons-usable material. There is a policy issue of whether or not to immediately isotopically dilute ^{233}U to render it non-weapons usable and later convert the ^{233}U to an acceptable waste form. The first question is determining the feasibility of disposal (REN 11). This depends upon the requirements. If disposal can be accomplished in a relatively short time, there is no incentive to isotopically blend the ^{233}U with DU to convert it to non-weapons-useable ^{233}U, place the mixture in storage, remove it from storage, and process it a second time as a waste form for disposal. If there are major problems in final disposal of surplus ^{233}U, there is the option to isotopically dilute the material to eliminate its potential as a weapons-usable material and place it in storage for future disposal. Such decisions can be made at any time.

An understanding of the feasibility of disposal of ^{233}U requires the determination of the legal and institutional requirements (REN 10). As discussed in Sect. 3.4, the institutional structure for management of ^{233}U is ill-defined compared to that for HEU and WGP. Initial evaluations of the ground rules identified several issues unique to ^{233}U:

- *Requirements for isotopic blending.* Current laws and regulations do not recognize isotopic blending of ^{233}U with DU as a method to convert ^{233}U into a non-weapons-usable material. Activities are underway (Sect. 5.2.3) to create the appropriate legal structure for conversion of ^{233}U to a non-weapons-usable material by isotopic dilution.

- *Requirements for disposal.* If there are no uses for surplus ^{233}U, it simply becomes a waste. An analysis of the legal definition of ^{233}U as a waste was completed (Bereolos 1997a). Determination of the waste classification determines what the viable disposal options are. If ^{233}U is a waste, it is classified as (1) Greater-Than-Class-C (GTCC) low-level waste (LLW) under NRC waste classification rules or (2) special-case wastes under DOE waste classification rules. In either case, it would probably be unacceptable for shallow land disposal. As a waste, the ^{233}U could also be co-processed with TRU waste or high-level waste (HLW) for deep geological disposal. However, legal and institutional issues are associated with each option (see Sect. 6).

6. APPROACH FOR IDENTIFICATION AND SELECTION OF DISPOSITION OPTIONS

6.1 KEY QUESTIONS

There is one disposition question: What is the preferred method or methods to dispose of surplus ^{233}U? While this is the same question that exists for HEU and WGP, the decision-making process and the types of options may be different because of the characteristics of ^{233}U and the institutional structures:

- *Quantity.* There are incentives to co-process any surplus ^{233}U with existing waste streams. The disposition of large quantities of fissile material (as with WGP and HEU) can justify the construction and operation of large facilities and commitment of significant resources. In contrast, the total quantities of ^{233}U are sufficiently small that it is practical to consider using facilities (e.g., vitrification facilities) designed for other missions to accomplish the task. This possibility has potential cost and environmental advantages, but it also implies that the ^{233}U disposition mission for certain options may be constrained by other waste management activities.

- *Characteristics.* There are incentives to consider multiple disposition options. Uranium-233 inventories can be divided into eight major categories (Fig 4.3). These categories have very different characteristics. One batch (Y-12 material) may not require glovebox handling. Some of the ^{233}U can be handled in gloveboxes. Much of the ^{233}U must be processed in shielded enclosures. In contrast, all HEU can be handled directly in facilities with appropriate ventilation, and all WGP is handled in gloveboxes. If ^{233}U is co-processed with other materials and wastes, there may be economic and environmental incentives to process different batches in different facilities.

6.2 PLAN FOR DETERMINING PREFERRED DISPOSITION OPTIONS

The decision-making flowsheet for determining the preferred options is shown in Fig. 6.1. This flowsheet picks up where the flowsheet in Fig. 5.1 left off—after decisions have been made leading to disposition of ^{233}U as a waste. A total of 12 steps are identified by number. The plan uses an iterative process to identify preferred options. Options are identified (REN 16) and screened (REN 17). The remaining options are further evaluated (REN 19 and REN 24) and screened (REN 20 and REN 25). After the second screening, only a limited number of options remain for consideration. Current activities and future plans are described in the context of this plan. The initial identification of options (REN 16) has been completed with identification of viable options (REN 17). A description of options is provided in Sect. 7. The plan allows for an identification of new options and evaluation of those options if the initial assessments do not prove to be viable disposition options.

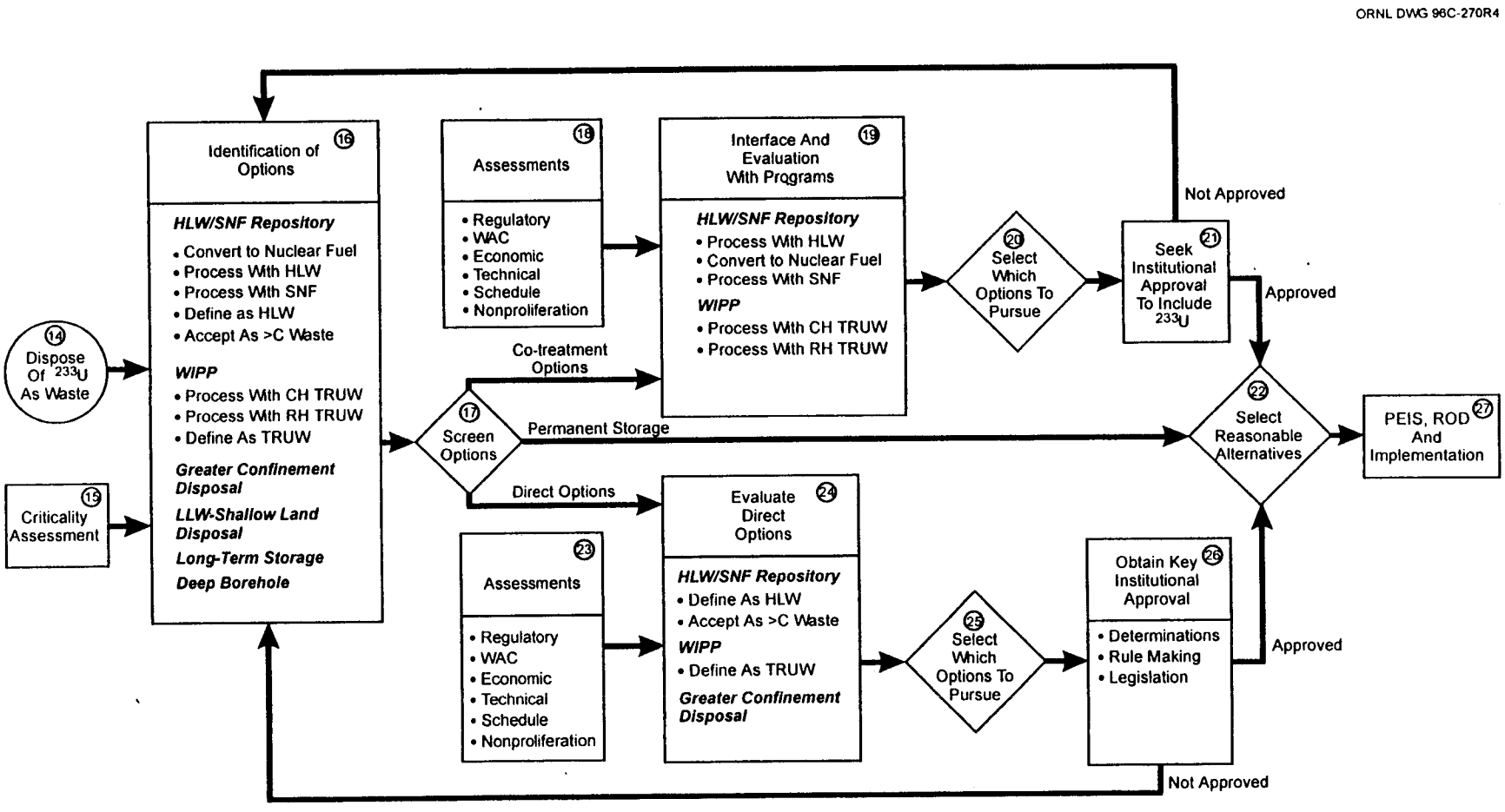

Fig. 6.1. Road map for decisions relating to the final disposition of surplus ^{233}U.

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6.2.1 Identification of Options (REN 16)

Disposition options must be identified. There are two sets of requirements. The first requirement is that the ^{233}U must be mixed with DU to convert it to non-weapons-usable material. The second requirement is that it must be disposed of. Disposition options (within the constraint of the first requirement) were identified considering current and proposed disposal facilities in the United States for radioactive materials. For each type of disposal facility two questions were asked:

- Can surplus ^{233}U be processed into a waste form acceptable for direct disposal at the site?
- Can surplus ^{233}U be coprocessed with other wastes expected at the site?

There are three sets of inputs needed to identify realistic disposal options. These studies have been completed.

- *Isotopic Blending Requirements* (REN 9). Disposition of ^{233}U requires that it be isotopically diluted with DU to eliminate its potential as a weapons usable material. This requirement defines the minimum quantities of DU that must be combined with the ^{233}U during disposition. Initial studies indicate that ^{233}U must be isotopically diluted to <12 wt % with DU to convert it to a non-weapons useable material.

- *Regulatory constraints* (REN 10, Fig. 5.1). The waste management system in the United States was not created as a single system, but rather it evolved over time and expanded to meet new needs as they were identified. One consequence of this history is that no clearly identified disposal pathway exists for wastes containing ^{233}U. A regulatory assessment was completed (Bereolos 1997b) to provide an understanding of the potential options under current law for disposal of ^{233}U. The analysis indicated that ^{233}U wastes, by themselves, would be considered as GTCC (special case) waste unsuitable for disposal in shallow-land disposal facilities.

- *Criticality Assessment* (REN 15). As a waste, a unique characteristic of ^{233}U is that it is a fissile material. A criticality assessment was completed (Elam 1997) to define criticality control requirements for ^{233}U as a waste. Uranium-233 must be diluted to ~0.66 wt % with ^{238}U (0.53 wt % if DU with a ^{235}U assay of 0.2 wt % is used) if the potential of nuclear criticality in a disposal facility is to be minimized. This level of isotopic dilution is equivalent to ^{235}U isotopically diluted to ~1 wt %.

  Criticality control requirements for geological disposal of ^{233}U as a waste may require greater isotopic dilution of ^{233}U than requirements to convert the material to a non-weapons-usable material. This conclusion is identical to that for disposal of HEU as a waste form. However, the required dilution levels for HEU are different than for ^{233}U because of the different nuclear characteristics of the two materials.

Disposition options were identified by the type of disposal site. For each disposal site, different waste forms for ^{233}U were identified:

- *HLW/SNF Repository (Fig 6.2)*

  — *Process with HLW.* In this option, the ^{233}U is blended with HLW sludge containing high concentrations of DU. The ^{233}U is combined with the HLW sludge and processed to an HLW glass. Isotopic blending of the DU from the HLW with the ^{233}U eliminates safeguards and nuclear criticality concerns. Because DU in the waste stream is used for isotopic blending, this option produces the lowest quantity of additional waste of the ^{233}U disposition options. Depending upon the specific suboption, some DU may be premixed with the ^{233}U to minimize nuclear criticality concerns during blending operations or otherwise simplify operations.

  — *SNF.* The ^{233}U is isotopically diluted with DU, fabricated into light-water reactor (LWR) fuel, irradiated, and sent to the repository as LWR SNF. This option includes the suboption of simultaneously blending with excess HEU and DU to produce a LEU more similar to commercial LEU as a power reactor fuel.

  — *Process with Miscellaneous SNF.* In this option, the ^{223}U is co-processed with miscellaneous SNF into a repository-acceptable waste form. The existence of the option is dependent upon future decisions concerning how to treat miscellaneous DOE SNF for disposal.

  — *Process as ^{233}U waste form.* The ^{233}U is isotopically diluted with DU, converted to uranium oxide, and packaged into a form acceptable for repository disposal. Depending upon the regulatory approach used, the final waste form could be defined as either HLW or GTCC LLW.

- *WIPP (Fig 6.3)*

  — *Define as TRU waste.* The ^{233}U is defined as TRU waste, isotopically diluted with DU, converted to uranium oxide, and packaged for disposal at WIPP. This requires a legal change in materials accepted for disposal at WIPP.

  — *Process with RH TRU waste.* In this option, the ^{233}U is blended with RH TRU waste and DU and then is converted into a final waste form for WIPP. The feasibility of this option is dependent upon future decisions as to how to treat RH TRU waste.

  — *Process with contact-handled (CH) TRU waste.* In this option, the ^{233}U is blended with CH TRU waste and DU and then converted into a final waste form for WIPP. The feasibility of this option is dependent upon future decisions concerning how to treat CH TRU waste.

- *Dispose in greater confinement disposal (GCD) facility.* In this option, the ^{233}U is processed with DU and is sent to a GCD facility. Formerly, DOE operated such a facility at the Nevada Test Site (NTS).

- *Deep borehole.* The ^{233}U is disposed of in a deep borehole.

- *LLW-Shallow Land Disposal.* The ^{233}U is disposed of in a shallow land disposal facility for LLW.

- *Permanent storage.* In this option, the ^{233}U is placed in long-term storage canisters and stored for the indefinite future. It is the default option if no other disposition option is implemented.

ORNL DWG 97C-133R2

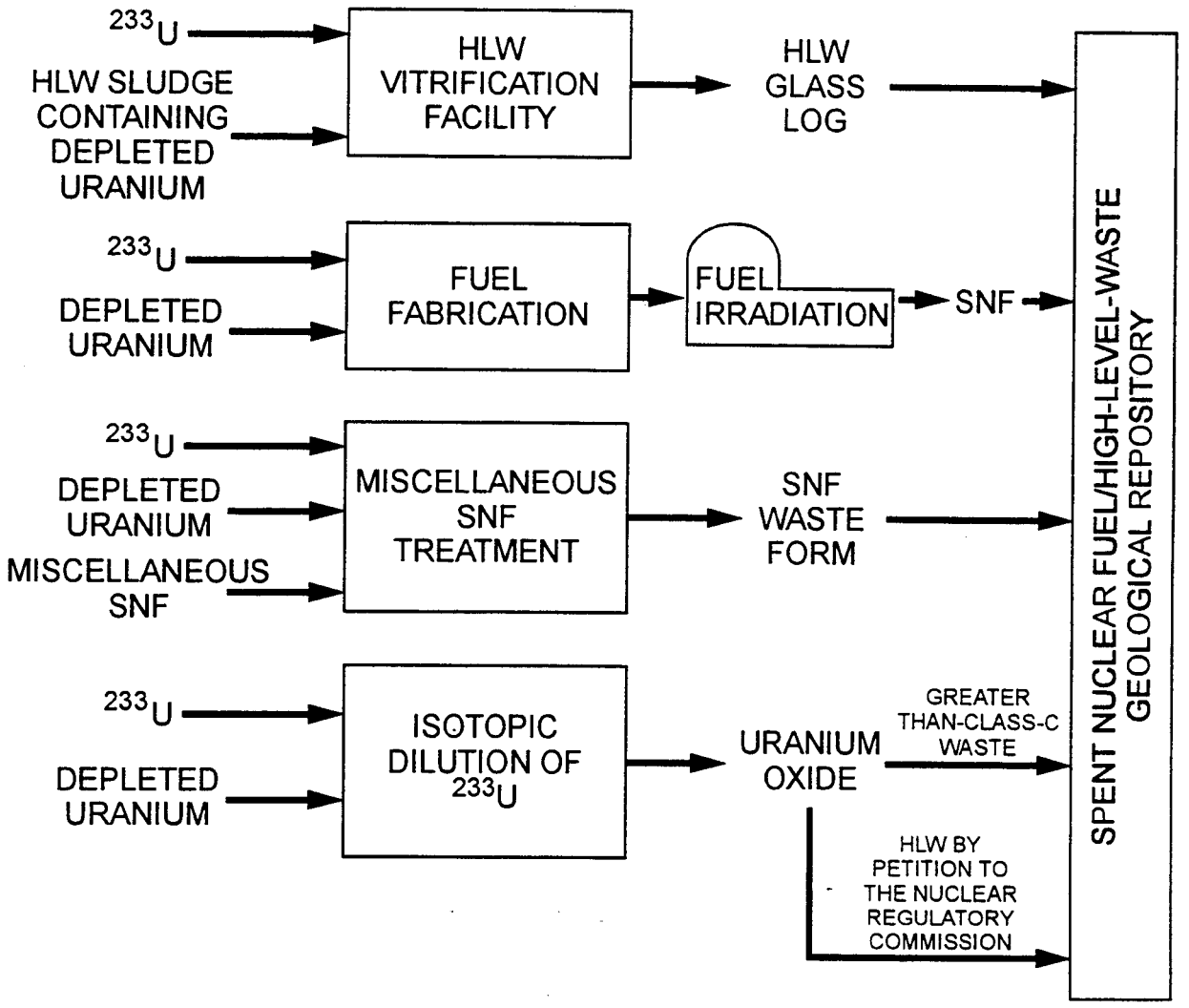

Fig. 6.2. Uranium-233 disposition options for final disposal in an SNF/HLW geological repository.

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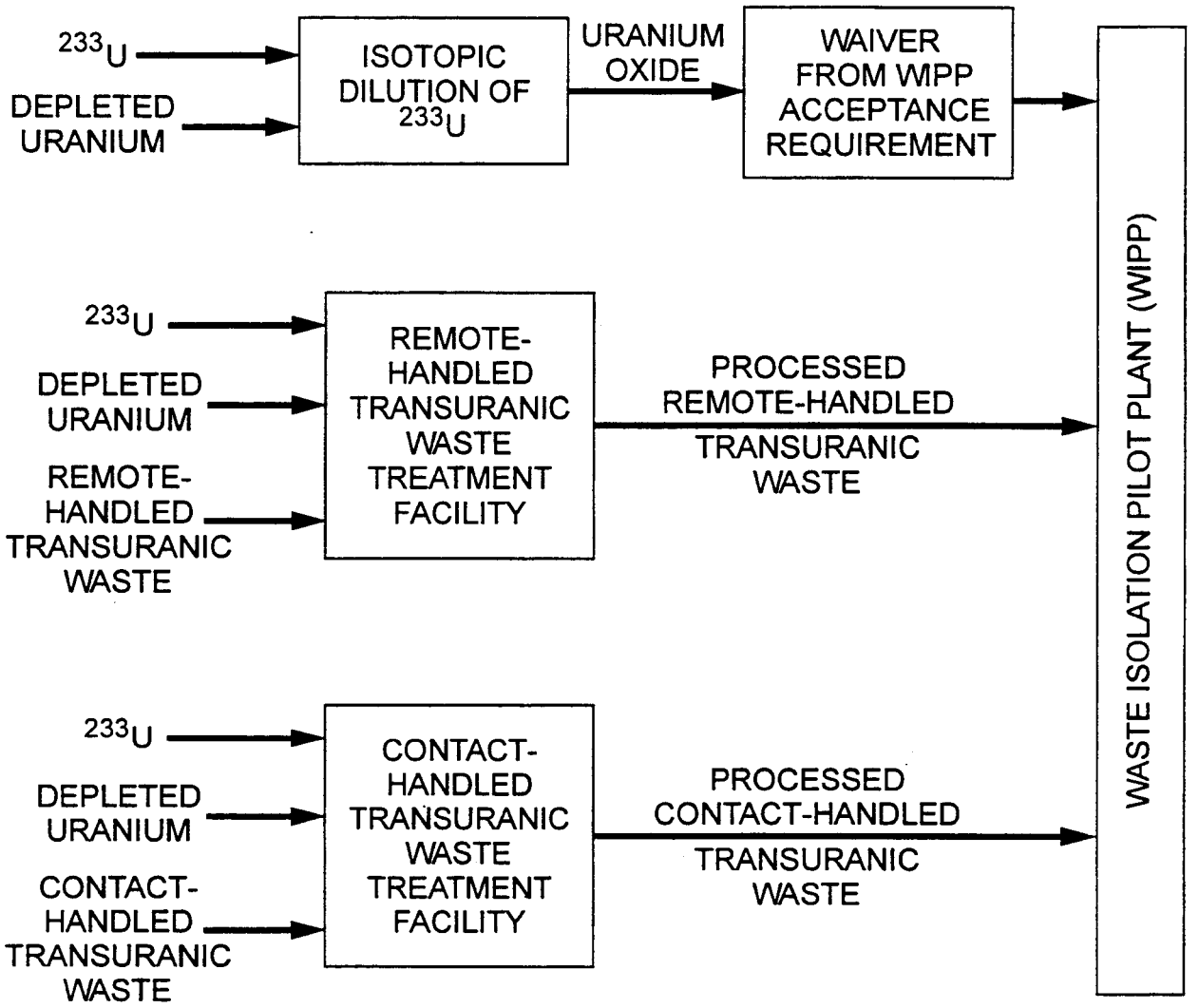

Fig. 6.3. Uranium-233 disposition options for final disposal in the WIPP.

Additional options may be identified during the NEPA process and added for consideration as disposition options.

6.2.2 Screen Options (REN 17)

The options were screened. For example, the allowable concentration of ^{233}U per unit volume of waste in an LLW-shallow land disposal site is so low that it appears unreasonable from a cost and regulatory perspective to further consider such an option. Some of the options, such as co-processing ^{233}U with HLW, appear highly attractive. Additional information is being collected on those that appear most feasible.

The results of these activities have identified two broad categories of disposition options. The first is to dispose of ^{233}U with other wastes. The second set of options is to dispose of the ^{233}U by itself.

6.2.3 Interface and Evaluate Codisposition Options with Other Programs (REN 18–21)

Disposition options exist for co-processing ^{233}U and disposing of it with other wastes. The incentives for such options are that they allow the use of existing or planned facilities for treatment of other wastes for processing and disposal of ^{233}U and, therefore, may result in significant cost savings. However, there is one constraint on such options: The ^{233}U disposition is directly coupled and captive to the schedules and limitations associated with processing the other waste stream. For such options to be considered, strong and effective communication and cooperation between the ^{233}U disposition program and specific waste programs must exist. Five such co-treatment options have been identified and are shown in Table 6.1.

Assessments (REN 18) are required for each option that address regulatory, waste acceptance criteria (WACs), economics, technical constraints, schedules, and nonproliferation issues. These assessments must be performed in cooperation with the organizations that own the existing or proposed facilities. These activities have been initiated. Constraints on specific options will depend upon the specific facilities.

6.2.3.1 Select which Option(s) to Pursue (REN 20)

A decision is required concerning which co-treatment options to pursue further. This decision will depend upon technical, economic, and policy issues associated with both the disposition of ^{233}U and other wastes.

Table 6.1. Co-processing options

Option	Facility, site, owner	Other institutional stakeholders
Process with HLW	Defense Waste Processing Facility, SRS, DOE-EM[a]	DOE-RW[b]/NRC[c]
	Proposed Hanford HLW facility, Hanford, DOE-EM[a]	DOE-RW[b]/NRC[c]
Process with Miscellaneous SNF	Proposed facilities, INEEL and SRS, DOE-EM[a]	DOE-RW[b]/NRC[c]
Process with CH TRU waste	Facility under construction, INEEL, DOE/EM[a]	WIPP
Process with RH TRU waste	Proposed facilities, ORNL, DOE-EM[a]	WIPP

[a]DOE Office of Environmental Management
[b]DOE Office of Civilian Radioactive Waste Management
[c]U.S. Nuclear Regulatory Commission

6.2.3.2 Seek Institutional Approval to Include ^{233}U (REN 21)

For cotreatment options, agreement must be sought with facility owners that co-treatment of ^{233}U with other wastes is both acceptable and reasonable. If such acceptance can not be obtained, a reevaluation of options will be required. The facility operators and key stakeholders are shown in Table 6.1. For the facility owners, there will be several key questions:

- Is the impact on facility operations and throughput acceptable?
- Are any new or different regulatory or institutional issues resolvable?
- Are the financial implications favorable?

6.2.3.3 Decide Upon Reasonable Alternatives (REN 22)

Based on the previous steps, reasonable options for co-treatment of ^{233}U with other wastes can be identified and become formal options for the environmental impact statement (EIS) and the final Record of Decision (ROD) on ^{233}U disposition.

6.2.4 Evaluate Direct-Disposal Options (REN 24)

There are four disposition options (Table 6.2) for managing ^{233}U without combining it with other materials. The ^{233}U is sent directly to the disposal site after processing. In each case, ^{233}U is processed into a form acceptable for disposal. Processing may be identical for the various options and include isotopic dilution of ^{233}U with DU to eliminate potential use as weapons material and nuclear criticality concerns.

Table 6.2. Direct options

Option	Key stakeholder	Other stakeholders
HLW-SNF repository: Define as HLW	DOE-RW	DOE-EM/NRC
HLW-SNF repository: Accept as GTCC LLW	DOE-RW	NRC
WIPP: Define as TRU waste	DOE-EM (WIPP)	DOE-EM
GCD facility	DOE-DP[a]	

[a]DOE Office of Defense Programs.

Assessments (REN 23) are required for each option that address regulatory issues, WACs, economics, technical constraints, schedules, and nonproliferation issues. Unlike the assessments for co-treatment options, it is likely that the results of assessments for different direct disposal options will be almost identical—except for legal and other institutional factors. These assessments must be done in cooperation with the disposal sites because of the legal issues associated with waste classification. This activity has been initiated.

The two primary disposal sites are the HLW/SNF repository and WIPP. The initial regulatory assessment (REN 10) evaluated regulatory issues for direct disposition at these sites (Bereolos 1997a). Several possible waste classifications exist for any ^{233}U sent to the HLW/SNF repository. Similarly, a question exists about what wastes the WIPP site can accept. Although no wastes have yet to be received by these disposal sites, these types of issues have been resolved historically through negotiations with and evaluations by the candidate repository for the specific waste on a case-by-case basis. Such regulatory and institutional issues define the ground rules for sending ^{233}U to these sites. A cooperative activity with the disposal sites is required to obtain agreement on what the ground rules are for acceptance of a ^{233}U waste form.

Direct disposal options may involve more that just the issue of disposal of ^{233}U. For example, because decisions have not been made on disposal of many special-case and GTCC wastes, both the disposal sites and owners of these other wastes will examine any proposal to ship a new waste type (^{233}U) to a disposal facility in terms of implications for management of these other wastes.

6.2.4.1 Decide Which Options to Pursue (REN 25)

A decision is required concerning which direct options to pursue further. This decision will depend upon the assessment of the options—particularly the legal definitions of ^{233}U waste at the time of its disposal.

6.2.4.2 Obtain Key Institutional Approval of ^{233}U (REN 26)

For direct-disposal options, agreement must be reached with the disposal sites. If such acceptance can not be obtained, a reevaluation of options will be required.

6.2.5 Agree to Reasonable Alternatives (REN 22)

Based on the previous steps, reasonable options for direct disposal of ^{233}U can be identified and become formal options for the EIS and the final ROD on ^{233}U disposition.

7. DISPOSITION OPTIONS

Disposition options for ^{233}U were identified by a three-step process: (1) identify acceptable disposal sites for ^{233}U, (2) identify other (non ^{233}U) wastes going to those disposal sites, and (3) identify where the ^{233}U stream could intercept those wastes and be added to those wastes or be a separate parallel waste stream. A description of those options that survived the initial screening is included herein. The summary descriptions herein include (1) description of the option, (2) major advantages, and (3) major unresolved issues. The options—in addition to storage—can be divided into three categories:

- *Comprehensive Options*. Several options exist that can dispose of all potentially surplus ^{233}U. An example is the addition of ^{233}U to HLW sludge containing DU and converting the waste to HLW glass.

- *Specific ^{233}U Category Options*. Several options have the capability to potentially dispose of one or more categories of ^{233}U, but these options are probably unsuitable for disposal of all potential surplus ^{233}U. An example is conversion of the Y-12 ^{233}U to nuclear reactor fuel.

- *Conditional Options*. The United States is planning to build several advanced radioactive waste treatment facilities and disposal sites. If these are built, disposition of the ^{233}U using these facilities may be possible. The use of these options depends upon major decisions on treatment of other waste forms. An example is co-processing ^{233}U with RH TRU waste.

7.1 LONG-TERM STORAGE

Uranium-233 has several current small-scale uses and several potential larger-scale future applications. The option exists to store any surplus material with ^{233}U for future use. Some ^{233}U will be stored for current and projected needs; thus, storage facilities for ^{233}U will continue to be needed.

If storage is selected as the preferred long-term option, there is one policy issue (REN 12 in Fig. 5.1): Should surplus ^{233}U be blended at or below 12 wt % ^{233}U with ^{238}U to eliminate its potential use as weapons-usable material? For most, but not all, applications, isotopic dilution of the ^{233}U would not eliminate its usefulness.

Long-term storage does improve the quality of the ^{233}U. The impurity in ^{233}U that uniquely governs handling practices is ^{232}U and its highly radioactive decay products. However, because the ^{232}U decays with a 70-year half-life, the quality of the ^{233}U improves with time.

DOE currently is considering a program to place existing stocks of ^{233}U into storage containers meeting interim storage standards (DNFSB 1997; Peña 1997). This interim storage program is being integrated with the long-term disposition program to (1) assure the ^{233}U is in a form and container that simplifies disposition activities and (2) allow the interim storage canisters to be used for long-term storage if desired.

7.2 COMPREHENSIVE OPTIONS

Comprehensive options are ^{233}U disposition options that potentially are capable of handling the entire inventory of ^{233}U.

7.2.1 HLW Glass with DU from HLW Sludge

The ^{233}U can be combined with existing HLW liquids and sludges at either the SRS or the Hanford site and converted to HLW glass (Fig. 7.1). Specific approaches are somewhat different at the two sites. In both cases, an HLW glass product is produced that meets all disposition criteria.

With appropriate selection of HLW liquids and sludges, the ^{233}U will be isotopically diluted with DU currently in the HLW sufficient to meet both safeguards (<12 wt % ^{233}U) and repository nuclear criticality criteria (≤0.53 wt % ^{233}U when diluted with DU containing 0.2 wt % ^{235}U). This option is expected to produce the minimum waste volumes of any option—potentially only a few canisters of HLW glass. Minimum waste production is possible because the DU required for isotopic dilution is currently in the HLW tanks and will be converted into glass. In principle, no added DU is required (see below). HLW at both SRS and Hanford contain significant quantities of DU. The tanks at SRS contain about 160,000 kg of U with the expected HLW glass to have ~2 wt % uranium. The tanks at Hanford contain about 1.4×10^6 kg of U, most of which is DU. In either case, the ^{233}U can be processed with the HLW sludge to produce an HLW glass. Because the HLW sludge already contains significant quantities of uranium, addition of the ^{233}U will not significantly alter the chemical characteristics of the HLW glass. This ensures that the final waste form is acceptable for the repository.

Converting ^{233}U to glass also ensures that all chemical and mechanical repository waste form criteria are met for the repository. Specifically, some of the current inventories of ^{233}U contain cadmium and other potential Resource Conservation and Recovery Act waste materials. Existing HLW at the SRS and Hanford site also contain these materials. The HLW glass is designed to accept these materials and produce a chemically nonhazardous glass acceptable to the repository. This implies that removal of cadmium and other neutron absorbers from ^{233}U is not required if it is to be converted to HLW glass.

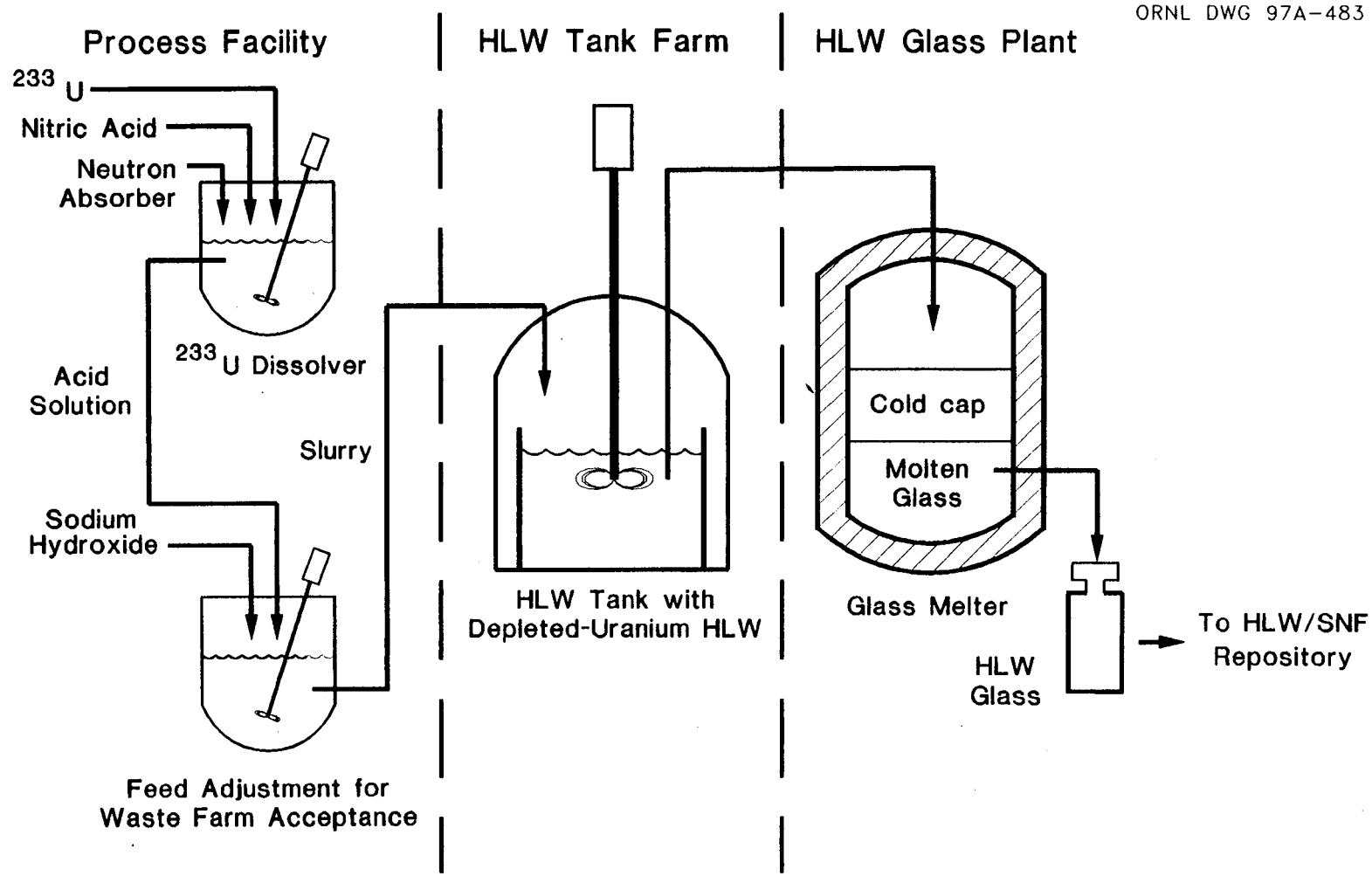

Fig. 7.1. Conversion of surplus ^{233}U and HLW (containing DU) to HLW glass.

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Within this option, there are a series of suboptions that include the addition of some clean DU to the ^{233}U before addition of the mixture to the HLW sludge that contains DU. Dilution of ^{233}U with added clean DU is to be minimized because the added DU generates additional HLW glass logs at a cost of ~$250,000/log; however, there are process criticality safety and operational constraints.

- To minimize ^{233}U processing time and, hence, ^{233}U operational costs, it is desirable to add the ^{233}U to the minimum amount of HLW. However, to minimize HLW glass production, the ^{233}U should be mixed with sufficient HLW sludge that no clean DU needs to be added to the waste. There is a tradeoff between these two strategies to minimize costs. Fortunately, at both SRS and Hanford, most of the HLW sludge with DU is in a limited number of tanks that may allow most of the DU needed for isotopic dilution to be obtained from the HLW sludge.
- There are multiple engineering options for addition of the ^{233}U to the HLW sludge. The mixing can occur in the main HLW feed tank in the tank farm, or in the smaller process feed tanks in the vitrification facilities. For some equipment options, the ^{233}U may need to be partially diluted with DU to assure that nuclear criticality does not occur during process operations. In existing facilities, such as those at SRS, this is more of a constraint (or a higher cost for modifying an existing facility) than for future facilities—such as those proposed for Hanford—that have not yet been built.

This option, like all coprocessing options, does impose schedule constraints on the disposition of ^{233}U. The ^{233}U must be processed when the HLW sludge containing the DU is to be processed. Effectively, there is one window of time when the option exists at SRS and a second window of time when the option exists at Hanford.

7.2.2 Special Repository Waste Form

The ^{233}U can be isotopically diluted with DU (containing 0.2 wt % ^{235}U) to ~0.53 wt % ^{233}U and converted to an oxide to meet chemical, safeguards, and nuclear criticality criteria for acceptance in a SNF-HLW repository (Fig. 7.2). Because of the similarities of the oxide final waste form to LWR SNF, it should be technically acceptable for disposal in an SNF HLW repository. The preliminary criteria for disposal in this repository have been defined (TRW, June 1996) by the repository design team. One category of the ^{233}U inventory contains cadmium added for control of nuclear criticality. The cadmium would have to be removed from the ^{233}U during processing, or it would have to be demonstrated that the material meets repository WAC with respect to hazardous materials.

Several institutional uncertainties are associated with this option. For disposal in the proposed SNF HLW repository, the ^{233}U may be classified as GTCC LLW or require reclassification of the material as HLW by petition to the NRC. Current regulations allow repository acceptance of the ^{233}U under either of these waste classifications. However, the proposed HLW SNF repository is being designed for HLW and SNF. There have been no previous official requests to dispose of GTCC LLW or wastes categorized by petition as HLW by the NRC in a repository. As a new waste type in the context of regulations, there are institutional uncertainties.

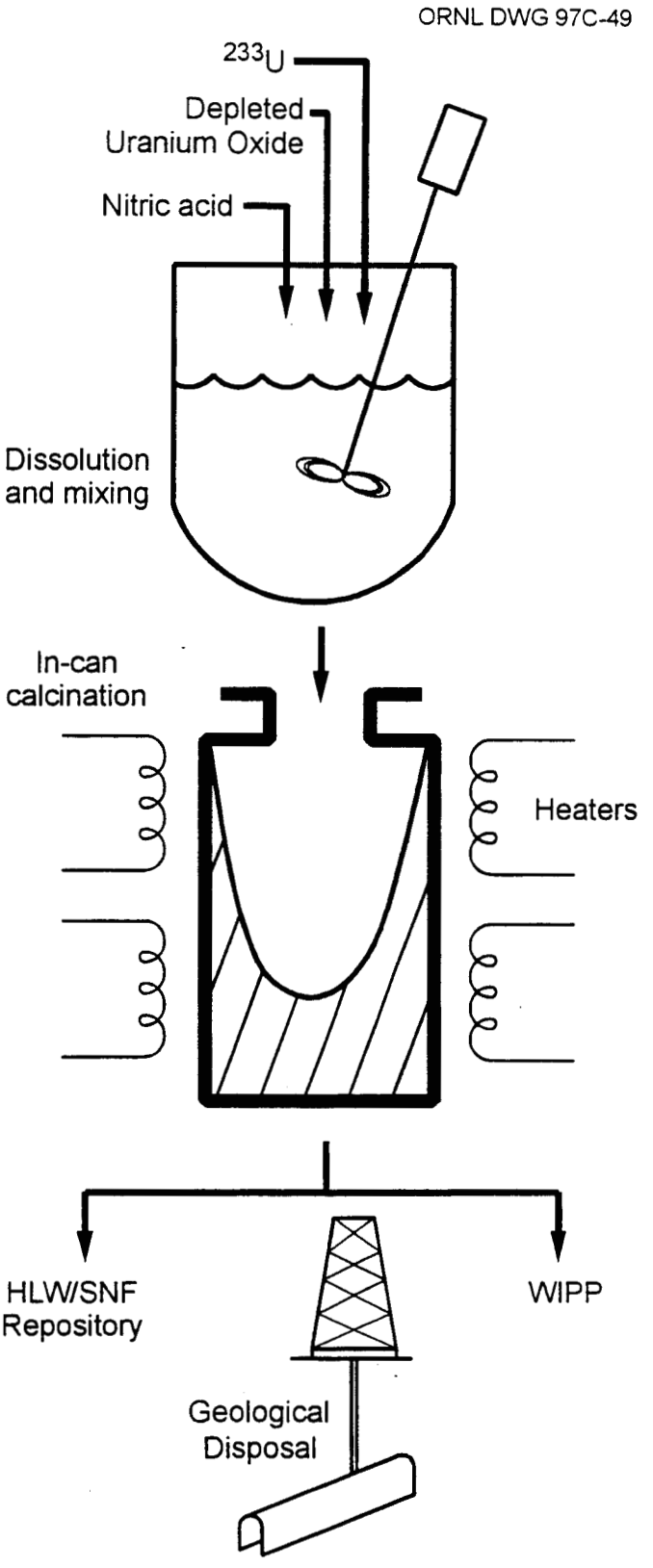

Fig. 7.2. Conversion of high-isotopic-purity ^{233}U and DU oxide to non-weapons-usable uranium oxide.

7.2.3 Customized ^{233}U Waste Form for WIPP

The ^{233}U can be isotopically diluted with DU (containing 0.2 wt % ^{235}U) to ~0.53 wt % ^{233}U and converted to an oxide as a final waste form for WIPP (Fig. 7.2). The processing is similar to that required for disposal of the ^{233}U in an SNF HLW repository. The waste form would be expected to meet or exceed all technical WIPP WACs. The WIPP WACs allow cadmium in the final waste form. This option could be used to dispose of all ^{233}U. The final volume of the waste is increased significantly by the addition of the DU when compared to the coprocessing options.

This option would require a change in the WIPP enabling legislation. The definition of TRU waste in the current legislation does not allow for disposal of wastes containing only ^{233}U at WIPP or wastes from non-defense programs. Historically, DOE has treated wastes with ^{233}U as TRU wastes because the radiological characteristics of these wastes are similar to those of other TRU wastes. Because of these similarities, any facility designed for TRU waste would be expected to be technically suitable for ^{233}U materials—provided that nuclear criticality issues are addressed.

7.3 SPECIFIC ^{233}U CATEGORY OPTIONS

Several options appear feasible for disposal of specific categories of ^{233}U, but these options would be either difficult or very expensive if applied to all ^{233}U.

7.3.1 Low-Enriched SNF

One option for disposition of ^{233}U is to use it in the fabrication of LWR fuel and its subsequent irradiation in a commercial nuclear reactor. Under such a disposition option, the surplus ^{233}U would be purified; blended down with DU; converted to UO_2; pressed into pellets, sintered, inserted into fuel pins; assembled into fuel bundles; and then irradiated in a nuclear reactor. Once irradiated, the SNF would be destined for permanent disposal in the SNF/HLW repository.

7.3.1.1 Past Experience

The United States has experience with the manufacture and irradiation of ^{233}U-based fuels. The Shippingport Atomic Power Station, located near Beaver, Pennsylvania, was the first large-scale central-station nuclear power plant in the United States. It was started up in 1957, was shut down in 1982, and was decommissioned over the ensuing few years (DOE 1992). Shippingport was initially an enriched ^{235}U-fueled pressurized LWR, but was converted in 1977 to an LWBR based on the ^{233}U/thorium fuel cycle.

To minimize the radiological dose to workers during Shippingport fuel fabrication, the ^{233}U nitrate solution at ORNL was purified by ion exchange to remove the ^{232}U decay products and provide a temporary window of time during which the material produced had a significantly reduced radiation field. The resulting solution was then converted to oxide and shipped, not more than 2 weeks following purification, to BAPL for fuel fabrication in a relatively low radiation field. It is noteworthy that the largest radiation exposure of personnel under the Naval Reactors Program occurred during the year 1975—the year of peak ^{233}U fuel fabrication operations. The exposure level was 15 times greater than the amount reported in 1994—primarily resulting from to fabrication of the LWBR core for Shippingport (Mangeno 1995). Some of the ^{233}U in inventory today is of better quality than the Shippingport material; however, a significant quantity of the inventory has higher ^{232}U impurities and higher radiation levels than the material that was processed.

7.3.1.2 Issues Related to LEU SNF Disposition Option

Use of this disposition option poses a number of potential radiological risks that must be addressed. These issues include the following:

- If the fuel is fabricated with current techniques, the ^{233}U must be purified, converted to oxide, and manufactured into LWR fuel in as short a time as possible to minimize worker exposure. Radiation fields from the ^{232}U decay products build up over time.

- To remove the need for expedient processing (and to further reduce the radiological risk), the fuel could be fabricated by semiremote or remote fuel fabrication techniques somewhat similar to those employed for the fabrication of mixed-oxide fuel in Europe. Such a facility may (with modifications) be used for fabricating ^{233}U LWR fuel.

The experience with ^{233}U fuel fabrication has shown that it is difficult to fabricate fuel and minimize radiation exposures to workers without a remote, automated, fuel-manufacturing facility. Development of the technology to dispose of the entire ^{233}U inventory by this option would require significant time and resources compared to other options for limited amounts of material.

7.3.1.3 Y-12 ^{233}U Disposition

One category of ^{233}U potentially may be disposed of via conversion to reactor fuel without great difficulty. The ^{233}U at Y-12 is unusual in that the material is only about 2 wt % ^{233}U, and the remainder is HEU. The ^{233}U is of a high quality with low levels of ^{232}U and its daughter products. Consequently, the radiation levels associated with this material are about a factor of 100 less than a similar quantity of typical ^{233}U in storage. Because of the very special characteristics of this material, it may be possible to blend it with other HEU and down-blend this mixture with DU to LWR fuel oxide.

DOE and the Tennessee Valley Authority are currently evaluating down-blending of isotopically off-specification weapons-useable HEU to reactor grade uranium. This off-specification material contains HEU with high concentrations of ^{236}U and other uranium isotopes in concentrations not found in fresh HEU. A special processing operation and fuel fabrication campaign would be used to fabricate fuel from this HEU. These special operations may allow disposition of the Y-12 ^{233}U with off-specification HEU as LWR fuel. The option is currently being investigated.

7.3.2 Direct Disposal in WIPP or the SNF HLW Repository

Direct disposal of ^{233}U in either WIPP or a SNF HLW repository is not generally practicable for two reasons: (1) the material remains as weapons-usable material and (2) nuclear criticality issues are not addressed. Isotopic dilution with DU is required. There is one potential exception: some of the LWBR fresh fuel. The ^{233}U concentration in the assemblies averages 2 wt % with concentrations as high as 12 wt % ^{233}U. The higher assay materials are unacceptable for direct disposal; however, the lower assay materials may be candidates for direct disposal if (1) they are placed in waste packages with DU to address nuclear criticality issues and (2) safeguards termination limits are defined. Activities are under way to address these issues. Such an option may significantly reduce the quantities of materials to be processed.

The feasibility of this option depends upon the rules for the termination of safeguards and control of nuclear criticality in geological repositories. These rules are currently being revised. Depending upon the final rules, this option may dispose of a significant fraction of the LWBR fresh fuel assemblies containing ^{233}U. There are also a set of institutional issues associated with each site.

7.4 CONDITIONAL OPTIONS

Conditional options are disposition options that depend upon some other program within DOE making a set of decisions to build specific facilities to treat a specific waste that could also be used to co-process ^{233}U for disposal.

7.4.1 Miscellaneous SNF Processing

DOE is evaluating options for the disposal of highly enriched SNF. The chemical and nuclear characteristics of this SNF make it uncertain whether direct disposal in a repository is feasible. From a technical perspective, the ^{233}U in its stainless steel or aluminum cans has similarities to SNF: (1) the primary component is uranium, (2) there is a clad (can) around the uranium, and (3) there is some radiation associated with the material. These similarities create the potential for use of future SNF facilities for disposition of ^{233}U.

DOE has recently completed a study on the disposition of aluminum clad SNF (DOE June 1996*b*) which evaluated multiple packaging, treatment, and disposal options. Aluminum clad SNF has similarities to ^{233}U in aluminum cans. Some of the aluminum-clad SNF will be reprocessed at the SRS and the HEU that is recovered will be converted to LEU. Recommendations have been made to investigate two other treatment options: (1) direct disposal and (2) press and dilute (or melt and dilute). The second option has two suboptions. Each of these options was evaluated in terms of disposition of ^{233}U.

- *Direct disposal.* Direct disposal is not viable for ^{233}U because it does not convert the material into non-weapons-usable uranium. Because the SNF is highly radioactive, there are limited concerns about the potential use of such materials in weapons. The material is considered to be non-weapons-usable by the presence of a high-intensity radiation barrier. The primary unresolved issue associated with this option for disposal of SNF is repository nuclear criticality.

- *Press-and-dilute(or melt-and-dilute).* The press-and-dilute option refers to pressing SNF into sheets intermixed with sheets containing DU. The melt-and-dilute option refers to melting SNF with DU to create a semi-homogeneous mixture. In both cases, the objective is to create a consolidated aluminum waste form with sufficient DU to avoid potential safeguards and repository nuclear criticality issues. The ^{233}U in aluminum cans could be co-processed with the aluminum SNF.

  There are limitations on the use of this option for ^{233}U. First, only a small fraction (<10%) of the ^{233}U inventory is in aluminum cans. The other ^{233}U inventory would have to be packaged into acceptable packages for this option. Second, decisions have not been made as to whether to implement this option for SNF. Given these caveats, it is noted that if the press-and-dilute (or the melt-and-dilute) options are implemented for the miscellaneous SNF program, those options may provide viable options to dispose of the ^{233}U inventory already stored in aluminum cans and other ^{233}U if repackaged.

DOE is currently identifying options for treatment of other types of miscellaneous SNF to meet disposal requirements. These include potential treatment options for stainless-steel-clad fuel that is somewhat similar to ^{233}U in stainless steel containers. When this work is completed, the options should be examined for applicability to disposal of ^{233}U.

7.4.2 RH TRU Waste Forms

Most of the inventory of RH TRU waste generated in the United States is at ORNL and Hanford. ORNL is considering several options for newly generated wastes, including a small vitrification system for liquid wastes. The ^{233}U may be added to this waste stream with DU and co-processed with future RH TRU wastes (Fig. 7.3). The ^{233}U has many of the characteristics of RH TRU waste, and, historically, DOE has treated wastes with significant quantities of ^{233}U in a manner similar to TRU wastes. The co-processed final waste form would be classified as TRU waste because of the other TRU isotopes present from other waste streams.

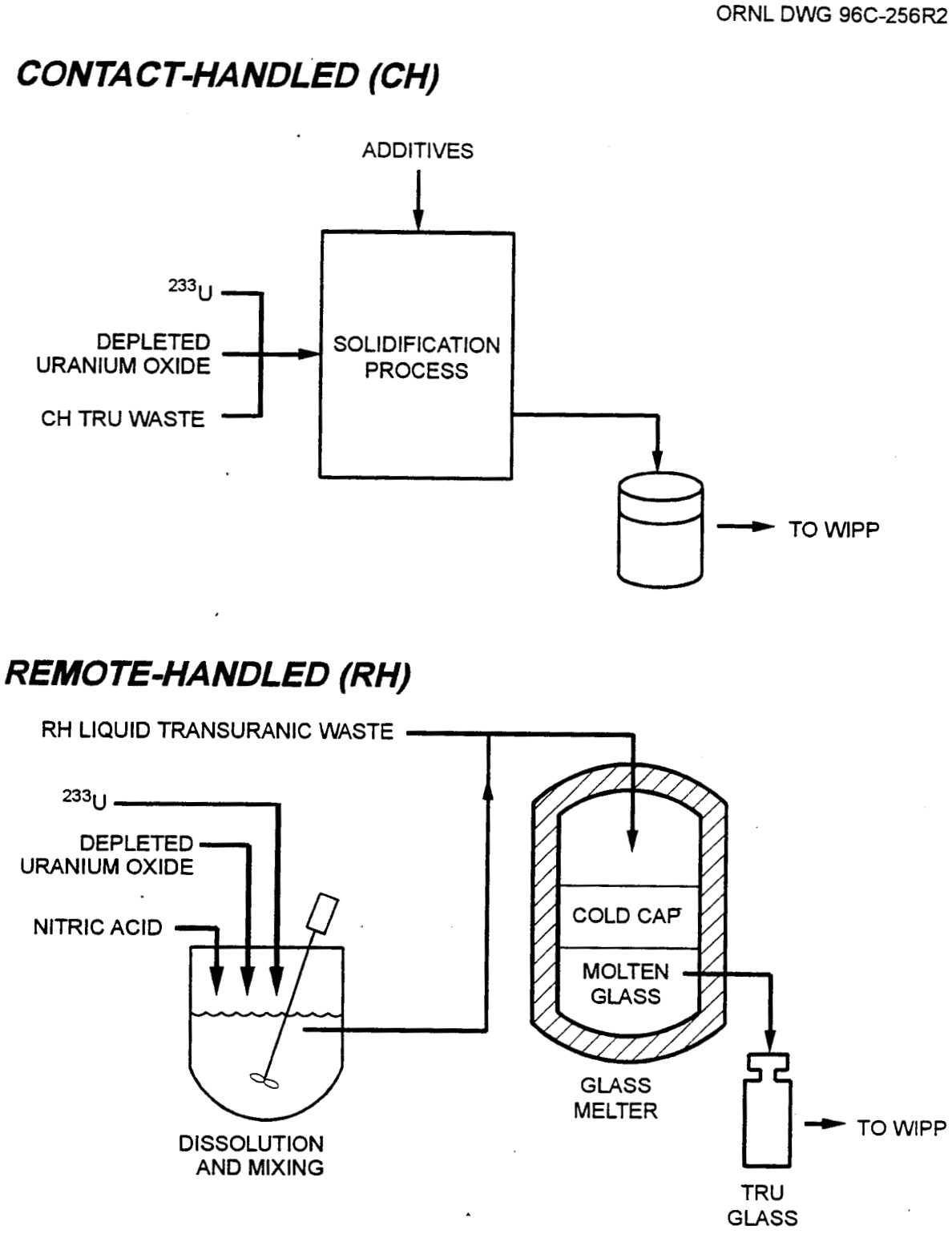

Fig. 7.3. Conversion of ^{233}U and TRU waste to WIPP final waste form.

One specific option deserves special note. Because uranium can form part of the glass structure, the possibility exists to use the uranium (DU and ^{233}U) as a glass former to incorporate other wastes into the glass. If this can be accomplished, the volume of the waste taken up by the ^{233}U would be reduced. The uranium would make up part of the waste form, not just the waste. However, the small annual RH TRU waste generation rate would lead to a protracted ^{233}U disposal scenario. Two uncertainties are associated with this option: (1) the allowable loading of uranium in glass and (2) whether any facilities will be built that are compatible with disposition of ^{233}U.

Studies are planned to evaluate RH waste treatment options for Hanford. No information is currently available on the characteristics of such facilities.

7.4.3 CH TRU Waste

INEEL is planning to treat 45,000 m^3 of CH TRU waste before the waste is shipped to WIPP. This plan creates the option to blend the ^{233}U and DU with the CH TRU waste fed to the process (Fig.7.3). In effect, this process would reduce radiation levels for much of the candidate surplus ^{233}U by about four to five orders of magnitude down to CH levels. The feasibility of this option depends upon the specific technology chosen to treat CH TRU waste, the characteristics of the CH TRU waste to be treated, and the final CH TRU waste form.

7.5 OTHER DISPOSITION OPTIONS

Other disposition options have been identified (DOE 1995), but have not been examined in the level of detail as the previously described options. These are briefly described herein. Each of these options involves major technical or institutional challenges that would require a large national program of extended length to be successfully implemented.

7.5.1 Borehole

The fissile materials disposition program considered deep boreholes for disposal of excess plutonium. This option was eliminated based on multiple technical and institutional considerations. Those same considerations apply to ^{233}U. The basic problem is that borehole is an undemonstrated method to geologically dispose of radioactive waste. It is in technical and institutional terms where conventional geological disposal was 25 years ago. Massive, long-term development would be required to implement this option.

7.5.2 Can-In-Canister

One option under evaluation for plutonium disposition is to (1) convert plutonium into a ceramic form in small cans, (2) place the small cans inside empty HLW canisters, (3) pour HLW glass into the HLW canisters, and (4) embed the cans in the HLW glass. Applying the option to ^{233}U would be difficult compared to direct addition of the ^{233}U to the HLW sludge.

- The plutonium can-in-canister option is not directly applicable to ^{233}U. The incentive of the can-in-canister option for plutonium disposition is that it allows the use of existing vitrification facilities without strongly impacting the operations of those facilities. For the vitrification facility, the only changes in operation are: (1) the "empty" HLW canisters have some additional weight because of the small cans containing plutonium that are inside the canisters, (2) the canisters accept slightly less glass, and (3) added security is needed during operations to protect the plutonium in the cans before the HLW glass is added to the canisters. The large vitrification plant does not require major modifications.

  If ^{233}U cans are added to the HLW canister, major changes in operations are required. The ^{233}U cans will have significant radiation from the ^{208}Tl decay product of ^{232}U. This also requires significant modifications of the vitrification plant. These facilities were designed on the assumption that the "empty" HLW canisters had low radiation levels and could be manually prepared and fed into the plant. None of the areas on the cold-side of the plant have appropriate radiation shielding for canisters with ^{233}U containing significant quantities of ^{232}U with decay products. In contrast, the areas of the plant designed to process HLW sludges are designed for high radiation levels and can handle radioactive ^{233}U without facility changes.

- The proposed plutonium can-in-canister facilities to fabricate the plutonium ceramic in small cans could not be used for ^{233}U. The plutonium processing facilities would be in gloveboxes; whereas, RH facilities are required for processing most of the ^{233}U inventory.

- The can-in-canister requires development and qualification of a new ^{233}U waste form in the can. It must meet all requirements imposed by vitrification operations and all requirements imposed by the repository. This requires major resources and time. Direct addition of the ^{233}U to the HLW avoids these issues because the HLW glass is already qualified for uranium in the glass. The current HLW glass is qualified for ~2 wt % uranium in glass. In a can-in-canister option the uranium concentration would be much higher to minimize the number of cans and the associated cost.

- The can-in-canister option implies a fundamentally different approach to the disposition of ^{233}U compared to all other ^{233}U options. The basic, accepted strategy for disposition of ^{233}U is to follow the disposition strategy developed for ^{235}U—isotopic dilution of ^{233}U with DU. There is full technical and institutional agreement (including agreement with Russia) that isotopic dilution is an acceptable method to convert weapons-useable uranium isotopes to non-weapons-useable materials. The ROD on HEU endorsed this strategy.

7.5.3 Shallow-Land Disposal

The radiological characteristics of ^{233}U are similar to TRU elements such as plutonium. The primary hazard from both is alpha radiation. United States policy requires geological disposal of wastes containing significant quantities of plutonium to minimize the risks from alpha radiation. The United States has built the WIPP facility for disposal of these wastes. By definition, plutonium wastes containing in excess of 100 nCi of plutonium per gram require geological disposal. Based on health-physics considerations, a similar policy would be expected for ^{233}U. Such a policy would prohibit shallow-land disposal of ^{233}U.

The radioactivity of ^{233}U is 9.6×10^6 nCi/g. To dilute ^{233}U to alpha-radiation levels allowed in shallow-land disposal facilities, the ^{233}U would have to be diluted by about a factor of 100,000. Dilution of radioactive waste to change their waste classification and allow shallow-land disposal is usually prohibited unless there are very-special conditions. Furthermore, accomplishment of such a large dilution would present serious engineering problems. Shallow-land burial of ^{233}U is not practicable.

7.5.4 GCD Facility

In principle, the ^{233}U may be disposed of in a GCD facility. These facilities have waste-isolation capabilities that functionally are between those of shallow-land disposal of LLW and deep geological disposal. Depending upon design details, such a facility might be able to accept a ^{233}U waste form. No such facilities currently exist. If such facilities were developed, the likely acceptable form for ^{233}U, after its dilution to meet safeguards and nuclear criticality requirements, would be uranium oxide (Sects. 5.2.4). As with other new waste management options (borehole, etc.), significant resources are required to develop the technology.

8. SCHEDULE

No decision on ^{233}U disposition has been made nor has a disposition schedule been prepared; however, there are schedule constraints for several of the options.

- Coprocessing options only exist at specific times. For example, if ^{233}U is to be codisposed with SRS HLW sludge, the ^{233}U must be processed when the SRS sludge containing the DU is being vitrified. The advantages of codisposal options come with the time constraints associated with those options.

- DOE is currently developing plans to address safety concerns with the storage of ^{233}U. These plans include consideration of new storage facilities. There are economic incentives to dispose of excess ^{233}U before such facilities are built to minimize the size of storage facilities for the ^{233}U that is retained.

Studies are currently underway to define these schedule factors that may influence the timing of decisions within the disposition program.

9. CONCLUSIONS

Alternative paths for disposition of surplus ^{233}U have been identified. There appear to be no technical barriers to disposition. The path forward requires 4 major policy decisions:

- How much ^{233}U is surplus?
- Should ^{233}U that is kept for programmatic purposes be blended with DU to eliminate its potential as a weapons-usable material?
- Should the surplus ^{233}U be placed in storage rather than permanent disposal? If surplus ^{233}U is placed in storage, should it be isotopically diluted with DU to eliminate its potential as a weapons-useable material?
- What are the preferred disposition options for surplus ^{233}U?

There are several disposition options. Some of the options can handle only certain batches of ^{233}U in inventory, whereas other potential options can dispose of the entire inventory. Three major disposition options that can handle the entire inventory have been identified:

- *HLW glass with DU from HLW sludge.* Uranium-233 is mixed with existing HLW sludges containing DU. The mixture is converted to HLW glass for disposal in the SNF/HLW repository. This option minimizes waste volumes by beneficial use of DU currently in the HLW sludge. It also results in a high-quality waste form for ultimate disposal. HLW glass was specifically designed for long-term performance in a repository environment. This option depends upon other programs (and is captive to their schedules) for treatment of the ^{233}U before disposal.
- *Special repository waste form for SNF/HLW repository.* Uranium-233 is mixed with DU and converted into a uranium oxide final waste form for disposal in the SNF HLW repository. This option does not depend upon other programs for treatment of the ^{233}U before disposal.
- *Special repository waste form for WIPP.* Uranium-233 is mixed with DU and converted into a uranium oxide final waste form for disposal at WIPP. This option does not depend upon other programs for treatment of the ^{233}U prior to disposal.

Last, there are a series of institutional issues specific to ^{233}U that must be addressed because the required institutional structures (that exist for handling HEU and WGP) do not fully exist for ^{233}U. For example, there must be national and international agreement on the level of isotopic dilution required to convert ^{233}U to a non-weapons-usable material. Those institutional structures were not created, because for historical reasons, the U.S. did not choose to use ^{233}U on a significant scale for either weapons or civilian applications. Other countries may make or may have made other decisions on ^{233}U use. Depending upon ongoing medical research, the U.S. may use ^{233}U in the future. Creating the institutional structures for ^{233}U can be a major benefit to minimize the potential use of this material in nuclear weapons.

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