

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NOTE TO COMMISSIONER ASSISTANTS
September 15, 1992

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FROM: James L. Blaha, AO/OEDO

SUBJECT: Staff Comments on EPA's High-Level Waste Standards

Enclosed for your information are copies of EPA's draft technical support for high-level waste standards and preliminary comments by the staff on these documents.


James L. Blaha, AO/OEDO

cc: J. Taylor
J. Sniezek
H. Thompson
SECY

9401050209 931116
PDR COMMS NRCC
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SEP 15 1992

MEMORANDUM FOR: James L. Blaha
Assistant for Operations

FROM: Robert M. Bernero, Director
Office of Nuclear Material Safety
and Safeguards

SUBJECT: PRELIMINARY NRC STAFF COMMENTS ON TECHNICAL SUPPORT FOR EPA'S
HIGH-LEVEL WASTE STANDARDS

On September 1 the NRC staff received rough draft copies of several chapters of the technical support for the U.S. Environmental Protection Agency's (EPA's) high-level waste standards, a draft of EPA's Regulatory Impact Analysis, and a request that comments on those drafts be provided to EPA by September 4. Within the limited time available for review, preliminary staff comments have been prepared and transmitted to EPA. These preliminary comments are consistent with previous NRC staff comments on Working Drafts 2 and 3 of EPA's standards. Because the Commissioners have frequently voiced interest in our interactions with EPA, they may wish to receive copies of EPA's draft documents and the staff's preliminary comments. Please distribute as appropriate.

A handwritten signature in black ink, appearing to read "Robert M. Bernero".

Robert M. Bernero, Director
Office of Nuclear Material Safety
and Safeguards

Enclosures:

1. Preliminary Comments
2. Draft EPA Tech. Support
3. Draft Reg. Impact Analysis

~~920924/493~~

J. William Gunter, Director
Criteria and Standards Division
Office of Radiation Programs
U.S. Environmental Protection Agency
Washington, D.C. 20460

Dear Mr. Gunter:

Thank you for the opportunity to review preliminary drafts of Chapters 1-7 and 11 of your Background Information Document (BID) and your draft Regulatory Impact Analysis (RIA) for 40 CFR Part 191. Our review identified a number of minor editorial comments which have been transmitted to your staff by telephone, and will not be repeated here. Our review also indicated that Chapter 6 of the BID (discussion of health effects risks associated with radiation exposure) has not yet been updated to reflect the 1990 BEIR V report. When the BEIR V information has been incorporated, we will solicit a thorough review of Chapter 6 by our Radiation Protection and Health Effects Branch.

Comments on BID

I am concerned that the BID chapters offered to us for review do not contain a discussion of the regulatory significance of potential carbon-14 doses. Chapter 7 describes the method used to project carbon-14 doses, and one can infer from this chapter (as well as Chapter 6) that EPA considers a linear dose-health effect relationship to be applicable even at the microrem/year individual dose rates projected for carbon-14 releases. However, EPA's discussions of the uncertainties associated with the linear hypothesis are concerned with dose rates of millirem/year. I strongly urge EPA to explicitly describe the uncertainties associated with use of the linear hypothesis for microrem/year dose rates, and the basis for EPA's belief that such dose rates require regulatory attention.

I would also like to elaborate on previous NRC staff comments urging comparisons between EPA's high-level waste standards and the risks allowed by other regulations. Chapter 11 of the BID is a step in the right direction, but it would be improved by using consistent units of measurement for the impacts of the various standards and reference points. As now drafted, the impacts are sometimes described in terms of cumulative impacts and sometimes as individual yearly or lifetime risks. Use of consistent units would allow the reader to determine whether EPA's proposed standards are more or less restrictive than the other reference points.

I think it is important that EPA present its own evaluation of the individual risks associated with EPA's containment requirements. Others have made estimates (e.g., by averaging impacts over the entire U.S. population) which suggest that the standards appear overly stringent. The enclosure to this letter demonstrates how EPA's own environmental model can be used to estimate the range of individual risks that would correspond to the release limits of the containment requirements. Whether EPA uses this approach or another, it is imperative that EPA provide its views on the individual risks associated with the containment requirements. Some in the technical community have come to believe that EPA's

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standards are onerously restrictive, in terms of the level of individual risk allowed, and acceptance of EPA's standards will not be achieved until that view has been dispelled. Arguing, as section 11.4 does, that such comparisons are not appropriate does not help the technical community to objectively evaluate the stringency of EPA's standards.

Comments on RIA

Since the document was received from EPA only 4 days before comments were requested, only a cursory review of EPA's draft Regulatory Impact Analysis could be completed. This review identified three major concerns with the document. Two of these, incomplete analyses of potential gaseous carbon-14 releases and inadequate estimates of the costs of demonstrating compliance with the standards, are recognized in the RIA, and EPA indicates its intent to provide improved estimates in later drafts of the document. Since both of these topics have been of significant concern within the technical community, I am pleased that EPA plans to address them before EPA's standards are proposed for public comment.

The third concern involves EPA's application of the basic methodology used for the regulatory impact analysis. Page 80 of the RIA discusses EPA's use of a cost-effectiveness analysis, rather than a cost-benefit approach. In a cost-effectiveness analysis, the incremental costs and benefits of various options are estimated to provide a means for evaluating the merits of the options. In contrast, a cost-benefit analysis would presumably attempt to quantify the total costs and benefits of each option to determine whether the benefits outweigh the costs. The NRC staff agrees that a cost-effectiveness analysis is a more appropriate methodology for the reasons given by EPA (problems with discounting health effects projections and difficulties in equating dollar costs with health effect benefits). Our concern with EPA's application of the methodology involves the range of options considered in EPA's analysis.

The RIA would be substantially improved if EPA would consider a broader range of options in its cost-effectiveness analyses. All of the options evaluated in the current draft of the RIA are relatively minor variations of a deep geologic repository, and the only impact of each option estimated is the statistically expected number of health effects caused by releases associated with each option. A more complete evaluation would include different disposal technologies such as "greater confinement" facilities for transuranic wastes. Additional impacts, including occupational radiation exposure from waste processing and risks associated with transportation of wastes between facilities, should also be included. Finally, EPA should evaluate the cost-effectiveness of alternative formulations of its standards. Standards that emphasize protection of individuals rather than cumulative releases of radioactive materials, standards that are deterministic rather than probabilistic, and standards that address human intrusion qualitatively rather than numerically could have major impacts on disposal facility design and on the costs of demonstrating compliance with the standards. In the NRC staff's view, EPA's RIA cannot be considered adequate without some consideration of a broader range of alternative types of disposal facilities and formulations for the standards.

I look forward to receiving Chapters 8-10 of the Background Information Document and the next iteration of the Regulatory Impact Analysis for review.

Sincerely,

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B. J. Youngblood, Director
Division of High-Level Waste Management
Office of Nuclear Material Safety
and Safeguards

Enclosure:
As stated

DISTRIBUTION

Central File	BJYoungblood, HLWM	JJLinehan, HLWM	HLHP r/f
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DATE	09/3/92	09/3/92	09/3/92	09/3/92	09/3/92

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Determination of Individual Risks
Associated with EPA's Containment Requirements

One of the principal exposure pathways in EPA's environmental model involves transport of radionuclides through an aquifer, discharge into a river, and then withdrawal of river water for consumption by humans as drinking water. The number of fatal health effects per curie discharged to the river is given by the equation on page 7-6:

$$\frac{FHE_{np}}{Q_{np}} = \frac{(P_R)(I_W)(f_{sw})(FCF_{np})(f_{wt})}{R}$$

- where FHE_{np} = time integrated population fatal health effect commitment for radionuclide, n, in the drinking water pathway,
- Q_{np} = integrated release to the river (curies),
- P_R = the number of persons drinking water and water-based drinks,
- I_W = per capita consumption of drinking water and water-based drinks (600 liters/year),
- f_{sw} = fraction of drinking water obtained from river = 0.65,
- FCF_{np} = fatal cancers per curie intake by ingestion,
- f_{wt} = fraction of activity passing through water treatment plant (assumed 1.0),
- R = annual river flow (liters per year).

Dividing both sides of the equation above by P_R and by the time, t , over which a release occurs gives:

$$\frac{FHE_{np}}{(Q_{np})(P_R)(t)} = \frac{(I_W)(f_{sw})(FCF_{np})(f_{wt})}{(R)(t)}$$

The left side of the equation above now represents the average (during time t) yearly individual risk of fatal cancer per curie released. The river flow rate, R , would likely be between 10^{10} and 10^{14} liters/year.¹ We also note, from Table 7.2-2, that FCF_{np} is on the order of 10^2 for many of the longer-lived actinide radionuclides in spent fuel. These data allow estimation of the average yearly individual risk associated with the drinking water pathway.

¹See, e.g., Press and Siever, Earth, W. H. Freeman and Company, San Francisco, 1974, p. 289.

Uniform release over 10,000 years.

For a uniform release over $t = 10,000$ years, the average yearly individual risk per curie released would be:

$$\frac{FHE_{rp}}{(Q_{rp})(P_R)(t)} = \frac{(600)(0.65)(10^2)(1.0)}{(10^{10} \text{ to } 10^{14})(10,000)} = 4X10^{-10} \text{ to } 4X10^{-14}/\text{person-yr-Ci.}$$

EPA's containment requirements allow release of 10,000 curies of most actinides from a 100,000 MTHM repository. The average yearly individual risk for a person near such a repository would then be:

$$\text{Risk} = (10,000)(4X10^{-10} \text{ to } 4X10^{-14}) = 4X10^{-6} \text{ to } 4X10^{-10}/\text{person-yr.}$$

Uniform release over 100 years.

Some types of disruptive events could cause the allowable level of release to occur over times much shorter than 10,000 years. For a uniform release over $t = 100$ years, the average yearly individual risk per curie released would be:

$$\frac{FHE_{rp}}{(Q_{rp})(P_R)(t)} = \frac{(600)(0.65)(10^2)(1.0)}{(10^{10} \text{ to } 10^{14})(100)} = 4X10^{-8} \text{ to } 4X10^{-12}/\text{person-yr-Ci.}$$

Again, for a release limit of 10,000 curies for a 100,000 MTHM repository, the individual risk would lie in the range of:

$$\text{Risk} = (10,000)(4X10^{-8} \text{ to } 4X10^{-12}) = 4X10^{-4} \text{ to } 4X10^{-8}/\text{person-yr.}$$

Other pathways.

Similar analyses could be carried out for the other environmental exposure pathways. Table 7.8-2 suggests that the total risk for longer-lived actinides from all pathways would be about 2-3 times that of the drinking water pathway. If so, the total annual individual risk for a uniform 10,000 year release would be in the range of 10^{-5} to 10^{-9} , and for a 100 year release would range from 10^{-3} to 10^{-7} . Such risks do not seem out of line with the risk benchmarks discussed in Chapter 11.

UNITED STATES ENVIRONMENTAL PROTECTION AGENCY
WASHINGTON, D.C. 20460

AUG 17 1992

*received Sep 1, 1992*OFFICE OF
AIR AND RADIATION

Mr. B. J. Youngblood, Director
Division of High-Level Waste Management
Mail Stop 4H3
Office of Nuclear Materials Safety
and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Youngblood:

Please find the enclosed copy of our latest draft of Chapters 1 through 7 and 11 from our Background Information Document which is being written to support our effort to repromulgate 40 CFR Part 191. I am sending this to you for your information and review. I am also sending a copy to the Department of Energy for the same purpose. If you have comments, please forward them to us by September 4, 1992.

If you have any questions, I may be contacted at 202-233-9290.

Lawrence Stumeter for
J. William Gunter, Director
Criteria and Standards Division, ORP

Enclosure

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Chapter 1: INTRODUCTION

The U.S. Environmental Protection Agency (EPA) is responsible for developing and issuing environmental standards and criteria to ensure that the public and environment are adequately protected from potential radiation impacts. With these objectives in mind, the EPA is proposing generally applicable environmental standards for the management and disposal of spent nuclear fuel and high-level and transuranic radioactive wastes. These standards provide the basic framework to control, in the long-term, the management and disposal of three types of radioactive wastes:

1. Spent nuclear reactor fuel, if ultimately disposed without reprocessing;
2. High-level radioactive liquid or solid wastes from the reprocessing of spent nuclear fuel; and
3. Transuranic wastes containing long-lived radionuclides of elements heavier than uranium; defined as containing more than 100 nanocuries per gram of wastes of alpha-emitting transuranic nuclides, with half-lives greater than 20 years.

1.1 EPA AUTHORITIES FOR THE RULEMAKING

These proposed standards have been developed pursuant to the Agency's authorities under the Atomic Energy Act (AEA) of 1954, as amended, and Reorganization Plan No. 3 of 1970 (NI70). The basic authority under the AEA, as transferred to the EPA by the Reorganization Plan of 1970, includes the mandate of:

"establishing generally applicable environmental standards for the protection of the general environment from radioactive materials. As used herein, standards mean limits on radiation exposures or levels, or concentrations or quantities of radioactive material, in the general environment outside the boundaries of locations under the control of persons possessing or using radioactive materials."

The Nuclear Waste Policy Act (NWPA) of 1982 established formal procedures regarding the evaluation and selection of sites for geologic repositories, including procedures for the interaction of State and Federal Governments; reiterated the existing responsibilities of the Federal Agencies involved in the national program; and provided a time table for several key milestones to be met by the Federal agencies in carrying out the program. As part of this national program, the EPA, pursuant to its authorities under other provisions of law, was required to:

"by rule, promulgate generally applicable standards for the protection of the general environment from off-site releases from radioactive material in repositories."

In December 1987, Congress enacted the Nuclear Waste Policy Amendments Act (NWP87). The 1987 Amendments Act redirects the nuclear waste program to consider Yucca Mountain, located in the State of Nevada, as the prime site for the nation's first high-level waste and spent nuclear fuel repository. All other potential site's activities were to be phased out. If Yucca Mountain is found to be suitable, the President is to submit a recommendation to Congress to develop a repository at this site. The Secretary of Energy is also required to inform Congress and the State if the site characterization activities indicate that Yucca Mountain is unsuitable. The Amendments Act prohibits the Department of Energy from conducting site-specific activities for a second repository unless authorized by Congress. Finally, the Act established a Commission to study the need and feasibility of a monitored retrievable storage facility to complement the nation's nuclear waste management program. The Commission submitted to Congress (as required under the original Act, as amended by Public Law 100-507) a report outlining their recommendations on November 1, 1989 (NWP88, RMRS89).

1.2 HISTORY OF THE EPA RULEMAKING

Since the inception of the nuclear age in the 1940s, the Federal government has assumed ultimate responsibility for the care and disposal of high-level radioactive wastes regardless of whether they are produced by commercial or national defense activities. In 1949, the Atomic Energy Commission (AEC) initiated research and development work aimed at developing systems for the conversion of high-level liquid wastes into a stable form. Then, in 1955, at the request of the AEC, a National Academy of Sciences - National Research Council (NAS-NRC) Advisory Committee was established to consider the disposal of high-level radioactive wastes within the United States. Its report (NAS57), issued in 1957, recommended, that:

1. The AEC continue to develop processes for the solidification of high-level radioactive liquid wastes, and
2. Naturally-occurring salt formations are the most promising medium for the long-term isolation of these solidified wastes.

Project Salt Vault, conducted from 1965 to 1967 by the AEC in an abandoned salt mine near Lyons, Kansas, demonstrated the safety and feasibility of handling and storing solid wastes in salt formations (MC70).

In 1968, the AEC again requested the NAS-NRC to establish a Committee on Radioactive Waste Management (CRWM) to advise the AEC concerning its long-range radioactive waste management plans and to evaluate the feasibility of disposing of solidified radioactive wastes in bedded salt. The CRWM convened a panel to discuss the disposal of radioactive wastes in salt mines. Based on the recommendations of the panel, the CRWM concluded that the use of bedded salt is satisfactory for the disposal of radioactive wastes (NAS70).

In 1970, the AEC announced the tentative selection of a site at Lyons, Kansas, for the establishment of a national radioactive waste repository (AEC70). During the next two

years, however, in-depth site studies raised several questions concerning the safe plugging of old exploratory wells and proposed expanded salt mining activities. These questions and growing public opposition to the Lyons site prompted the AEC in late 1971 to pursue alternatives to the salt site at Lyons (DO72).

In 1976, the Federal government intensified its program to develop and demonstrate a permanent disposal method for high-level radioactive wastes. The Office of Management and Budget (OMB) established an interagency task force on commercial wastes in March 1976. The OMB interagency task force defined the scope of the responsibility of each Federal agency's activities on high-level waste management, including the preparation of environmental standards for high-level wastes by the EPA (LY76, EN77a, EN77b).

A status report on the management of commercial radioactive nuclear wastes, published in May 1976 by the President's Federal Energy Resources Council (FERC), emphasized the need for coordination of administration policies and programs relating to energy. The FERC established a nuclear subcommittee to coordinate Federal nuclear policy and programs to assure an integrated government effort. This report called for an accelerated comprehensive government radioactive waste program plan and recommended the formation of an interagency task force to coordinate activities among the responsible Federal agencies. The EPA was given the responsibility of establishing general environmental standards governing waste disposal activities, including high-level radioactive wastes that must be delivered to Federal repositories for long-term management (FERC76).

In 1976, President Ford issued a major policy statement on nuclear waste. As part of his comprehensive statement, he announced new steps to assure that the United States has the facilities for the long-term management of nuclear waste from commercial power plants. The President's actions were based on the findings of the OMB interagency task force formed in March 1976. He announced that the experts had concluded that the most practical method for disposing of high-level radioactive wastes is in geologic repositories located in stable formations located deep underground. Among the EPA's responsibilities, the Agency was to issue general environmental standards governing nuclear waste facility releases to the biosphere above natural background radiation levels (FO76). These standards were to place a numerical limit on long-term radiation releases outside the boundary of the repository.

In December 1976, the EPA announced its intent to develop environmental radiation protection criteria for radioactive wastes to assure the protection of public health and the general environment (EPA76). These efforts resulted in a series of radioactive waste disposal workshops, held in 1977 and 1978 (EPA77a, EPA77b, EPA78a, EPA78b).

In 1978, President Carter established the Interagency Review Group (IRG) to develop recommendations for the establishment of an administrative policy to address the long-term management of nuclear wastes and supporting programs to implement the policy. The IRG report re-emphasized EPA's role in developing generally applicable standards for the disposal of high-level wastes, spent nuclear fuel, and transuranic wastes (DOE79). In a message to Congress on February 12th, 1980, the President outlined the content of a comprehensive national radioactive waste management program based on the IRG recommendations. The message called for an interim strategy for disposal of high-level and transuranic wastes that

would rely on mined geologic repositories. The message repeated that the EPA was responsible for creating general criteria and numerical standards applicable to nuclear waste management activities (CA80).

In November 1978, the EPA published proposed "Criteria for Radioactive Wastes," which were intended as Federal Guidance for storage and disposal of all forms of radioactive wastes (EPA78c). In March 1981, however, the EPA withdrew the proposed criteria because the many different types of radioactive wastes made the issuance of generic disposal guidance too problematic (EPA81).

In 1982, under the authority of the Atomic Energy Act of 1954, the EPA proposed a set of standards under 40 CFR Part 191, "Environmental Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes" (EPA82). Shortly after the publication of the EPA's proposed rule, Congress passed the Nuclear Waste Policy Act of 1982 (Public Law 97-425), wherein the EPA was to "... promulgate generally applicable standards for the protection of the general environment from off-site releases from radioactive material in repositories..." not later than January 1984 (NWP83).

After the first comment period on the proposed rule ended on May 2, 1983, the EPA held two public hearings on the proposed standards - one in Washington, D.C., on May 12-14, 1983, one in Denver, CO, on May 19-21, 1983 - and during a second public comment period requested post-hearing comments (EPA83a, EPA83b). More than 200 comment letters were received during these two comment periods and 13 oral statements were made at the public hearings. Responses to comments received from the public were subsequently published and released in August 1985 (EPA85a).

In parallel with its public review and comment effort, the Agency conducted an independent scientific review of the technical basis for the proposed 40 CFR Part 191 standards through a special Subcommittee of the Agency's Science Advisory Board (SAB). The Subcommittee held nine public meetings from January 18, 1983 through September 21, 1983 and later prepared and released a final report on February 17, 1984 (EPA83c, SAB84). Although the SAB review found that the Agency's analyses in support of the proposed standards were comprehensive and scientifically competent, the report contained several findings and recommendations for improvement. The report was publicly released on May 8, 1984 and the public was encouraged to comment on the findings and recommendations (EPA84). Responses to the SAB report were subsequently presented and released in August 1985 (EPA85b).

On February 8, 1985, the Natural Resources Defense Council, the Environmental Defense Fund, the Environmental Policy Institute, the Sierra Club, and the Snake River Alliance brought suit against the Agency and the Administrator because they had failed to comply with the January 7, 1984 deadline mandated by the NWP83 for promulgation of the standards. A consent order was negotiated with the plaintiffs that required the standards to be promulgated on or before August 15, 1985. The EPA issued the final rule under 40 CFR Part 191 on that date (EPA85c, EPA85d).

The EPA standards were divided into two main sections, Subparts A and B. Subpart A addressed the management and storage of wastes at any disposal facility operated by the Department of Energy and that is not regulated by the Nuclear Regulatory Commission or by Agreement States. Under Subpart A of the Standard, the exposure limits to any member of the general public are 25 millirem (mrem) to the whole body and 75 mrem to any critical organ. For facilities which are regulated by the Nuclear Regulatory Commission or Agreement States, the Standards were the same and endorsed the annual dose limits given in 40 CFR Part 190, the environmental standards for the uranium fuel cycle, 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to the critical organ.

Subpart B imposed limits associated with the release of radioactive materials into the environment following closure of the repository. The key provisions of Subpart B were:

- Cumulative containment limits over 10,000 years for releases of radioactive materials into the environment;
- Assurance requirements to compensate for uncertainties in achieving the desired level of protection;
- Individual exposure limits based on the consumption of groundwater and any other potential exposure pathways for 1,000 years after disposal; and
- Groundwater protection requirements in terms of allowable radionuclide concentrations and associated doses for 10,000 years after disposal.

Under Sections 191.15 and 191.16 of Subpart B, the annual dose to any member of the general public was limited to 25 mrem to the whole body and 75 mrem to any critical organ. The groundwater concentration for beta or gamma emitters is limited to the equivalent yearly whole body or organ dose of 4 mrem. The allowable water concentration for alpha emitters (including radium-226 and radium-228, but excluding radon) was 15 picocuries/liter. For radium-226 and radium-228 alone, the concentration limit was 5 picocuries/liter. Appendix A of the standards provided acceptable radionuclide cumulative release limits.

In March 1986, five environmental groups led by the Natural Resources Defense Council and four States filed petitions for a review of 40 CFR Part 191 (EPA85c, USC87). These suits were consolidated and argued in the U.S. Court of Appeals for the First Circuit in Boston. The main challenges concerned:

1. Violation of the Safe Drinking Water Act (SDWA) underground injection section;
2. Inadequate notice and comment opportunity on the groundwater protection requirements; and
3. certain aspects of the standards were thought to be arbitrary, not supported in the record, or not adequately explained.

In July 1987, the Court rendered its opinion and noted three findings against the Agency and two favorable judgements. The Court's action resulted in the remand of Subpart B. The Court began by looking at the definition of "underground injection," which is the "subsurface emplacement of fluids by well injection." A "well" is defined by the SDWA and the EPA as a shaft "bored, drilled, or driven where the depth is greater than the largest surface dimension." A "fluid" is a material or substance which flows or moves whether in a semi-solid, sludge, gas, or any other form or state." In the view of the Court, the method envisioned by DOE for disposal of radioactive wastes in underground repositories might fit both of the latter definitions and would "likely constitute an underground injection under the SDWA." Under the SDWA, the Agency is required to assure that underground sources of drinking water will not be endangered by any underground injection. With regard to such potential endangerment, the Court supported part, but not all, of the Agency's approach. A dichotomy appeared in the rationale when endangerment was considered inside the "controlled area" versus beyond the controlled area (i.e., in the accessible environment). Inside the controlled area, the Court ruled that Congress - through the EPA - had allowed endangerment of any present groundwater. Therefore, the EPA's approach of using the geological formation as part of the containment was validated. However, outside the controlled area, the Court found that Section 191.15 would allow endangerment of drinking water supplies. In the context of the SDWA, "endangerment" is considered when doses higher than that allowed by the Primary Drinking Water Regulations may occur. Section 191.15 permits an annual dose of 25 mrem to the whole body and 75 mrem to any critical organ from all pathways. On the other hand, the regulations under the SDWA allow four mrem from drinking water. The Court recognized that less than four mrem may result from the groundwater pathway, however, it rejected this possibility because the Agency stated that radioactivity may eventually be released into the groundwater system near the repository which could result in substantially higher doses. Therefore, the Court decided that it seemed clear that a large fraction of the 25 mrem could be received through the groundwater exposure pathway. Accordingly, the Court found that the high-level wastes standards should have been consistent with the SDWA or the Agency should have explained that a different standard was adopted and its position should have been justified.

The Court also noted that the Agency is not necessarily incorrect in promulgating the proposed standards, however, the Agency never acknowledged the interrelationship of the SDWA and HLW rules nor did it present a reasonable explanation for the divergence between them. The Court also supported the petitioner's argument that the Agency arbitrarily selected the 1,000-year limit for individual protection requirements (Section 191.15) under undisturbed performance. The Court indicated that the 1,000-year criterion is not inherently flawed, but rather that the administrative record and the Agency's explanations do not adequately support this choice. The criterion was remanded for reconsideration and the Agency must provide a more thorough explanation for its basis. Finally, the Court found that the Agency did not provide sufficient opportunity for notice and comments on Section 191.16 (Groundwater Protection Requirements) since that section was added to Subpart B after the standards were proposed. This section was remanded for a second round of notice and comments. There were, however, no rulings issued on technical grounds about Section 191.16.

In August 1987, the Justice Department asked the First Circuit Court to reinstate all of 40 CFR Part 191 except for Sections 191.15 and 191.16, which were originally found defective. The Natural Resources Defense Council filed an opposing opinion. The Court then issued an Amended Decree that reinstated Subpart A, but continued the remand of Subpart B.

1.3 PURPOSE AND SCOPE OF THE BACKGROUND INFORMATION DOCUMENT

This document provides the necessary background information, technical analyses, and justifications in support of the re-promulgation 40 CFR Part 191.

The scope of this Background Information Document (BID) encompasses the conceptual framework for assessing radiation exposures and associated health risks. In general terms, this assessment includes the identification of the potential release mechanisms, radioactive source term characterization, analysis of the movement of radionuclides from the repository through the appropriate environmental exposure pathways, dose estimates received by members of the critical population group and general public, and assessment of the probability of associated health risks.

1.4 ANALYTICAL COMPUTER CODES

A number of analytical computer codes have been used by the EPA to assess the long-term performance of the repository. Two computer codes, namely REPRISK and NEFTRAN, have been used extensively in the current effort. REPRISK, which was developed by the EPA in 1978, is used to assess health effects from radioactive materials released into the environment from a repository. Another code, WESPDOSE, was used to characterize several pathways for the environmental transport of radionuclides (SM85). For calculations involving individual doses, the NEFTRAN code, previously known as NWFT/DVM, was used to model the transport and decay chains of radionuclides in groundwater systems and geological media (CA81, LO87). A more complex groundwater code, SWIFT, has also been used to support the risk analyses, primarily to validate some of the geohydrologic calculations carried out using simpler models (RE81). These computer codes were also used to support the promulgation of the 1985 standards (EPA85c). The following four kinds of "release mechanisms" were addressed in these analyses:

1. Disruption of the repository with associated releases to an aquifer (disruptive events may include, e.g., faulting, breccia pipes, exploratory drilling with no direct hit).
2. Direct impact on a waste package with associated releases to an aquifer (e.g., faulting, breccia pipes).
3. Disruption of the repository with associated releases to the land surface (e.g., exploratory drilling with no direct hit).

4. Direct impact on a waste package with associated releases to the air and/or the land surface (e.g., volcano, meteorite, direct hit by exploratory drilling).

Each release mechanism led to several exposure pathways. The potential consequences of releases to the accessible environment are expressed in terms of: 1) number of somatic health effects, 2) number of genetic health effects, 3) ratio of the amount of radioactivity released to the limit given in 40 CFR Part 191, and/or 4) number of curies released for each radionuclide. The analyses are performed for two time frames. The first time frame gives the dose commitment for each occurrence and release mechanism. The second time frame gives the dose integrated over all times (i.e., beyond the duration of the event).

1.5 PROGRAM TECHNICAL SUPPORT DOCUMENTS

A number of technical support documents have been used, prepared, and published during the history of the rulemaking activities to establish the technical basis of the standards. The following list presents the documents which have been used to support the current rulemaking activities.

1. Technical Support of Standards for High-Level Radioactive Waste Management - Volume A, Source Term Characterization, EPA 520/4-79-007A, March-July 1977.

This report provides a characterization of commercial spent nuclear fuel and high-level wastes, including comparisons of source terms from various fuel cycles and fuel mixes; a characterization of government high-level and transuranic wastes; a comparison with commercial wastes; and an estimate of existing and projected quantities of spent nuclear fuel and high-level and transuranic wastes. The data are presented in several formats and by specific basis (per unit of fuel used or energy generated), as well as on a total basis for a given number of nuclear power plants.

2. Technical Support of Standards for High-Level Radioactive Waste Management - Volume B, Engineering Controls, EPA 520/4-79-007B, March-August 1977.

This report reviews the technology for engineering control of spent fuel and high-level and TRU wastes and projected costs of the various disposal technologies. Analyses include processing and packaging technologies, alternative geologic disposal techniques, effectiveness of engineering controls, and associated cost considerations.

3. Technical Support of Standards for High-Level Radioactive Waste Management - Volume C, Migration Pathways, EPA 520/4-79-007C, March - July 1977.

This report assesses geologic site selection factors; quantification of the potential migration and dispersion of radionuclides through the biosphere; and dose implications of a repository containing radioactive wastes at high concentrations.

4. Technical Support of Standards for High-Level Radioactive Waste Management - Volume D, Release Mechanisms, EPA 520/4-79-007D, March 1980.

This report analyzes the potential radionuclide releases from a generic deep-mined repository for radioactive wastes. Five different geologic media are considered: bedded salt, dome salt, granite, basalt, and shale. A range of potential containment failure mechanisms were evaluated and compared. The results combine radionuclide transport and dose calculations in assessing potential health effects of a repository.

5. Technical Support of Standards for High-Level Radioactive Waste Management - Volume E, Addendum to Volumes C and D, EPA 520/4-79-007E, March 1982.

This report updates the information and issues relevant to the conclusions reached in Volumes C and D.

6. Assessment of Waste Management of Volatile Radionuclides, EPA ORP/CSD-79-2, May 1979.

This report reviews waste management technologies in terms of immobilization, containment, and disposal of I-129, Kr-85, H-3, and C-14. Included are alternative disposal options that may be applied to isolate these wastes from human exposures and the environment.

7. Radiation Exposures From Solidification Processes for High-Level Radioactive Liquid Wastes, EPA 520/3-80-007, May 1980.

This report provides an assessment of a generic high-level liquid waste solidification plant and the potential environmental impact of atmospheric discharges during normal operations involving four different solidification processes.

8. A Review of Radiation Exposure Estimates From Operations in the Management and Disposal of High-Level Radioactive Wastes and Spent Nuclear Fuel, EPA 520/3-80-008, August 1980.

This report provides an analysis of the estimated radioactive releases during normal waste management operations (i.e., preparation for storage, disposal, and emplacement) and resulting radiation exposures and doses.

9. Economic Impacts of 40 CFR 191: Environmental Standards and Federal Guidance for Management and Disposal of Spent Nuclear Fuel,

**High-Level and Transuranic Radioactive Wastes, EPA 520/4-80-014,
December 1980.**

This report develops a methodology for examining the potential economic impacts of the proposed environmental standards.

10. **Population Risks from Uranium Ore Bodies, EPA 520/3-80-009,
October 1980.**

This report presents a methodology for estimating the radiological releases and potential health impact of deep-lying uranium ores.

11. **High-Level and Transuranic Radioactive Wastes -Background
Information Document for Final Rule, EPA 520/1-85-023, August
1985.**

This report presents estimates of population doses and risks associated with disposal of radioactive wastes in geologic repositories and describes the methodologies used to derive these estimates.

12. **Final Regulatory Impact Analysis - 40 CFR 191: Environmental
Standards for the Management and Disposal of Spent Nuclear Fuel,
High-Level and Transuranic Radioactive Wastes, EPA 520/1-85-027,
August 1985.**

This report reviews the project costs associated with the management and disposal of high-level radioactive wastes. The reports also addresses the containment, and groundwater and individual protection requirements from such wastes.

13. **High-Level and Transuranic Radioactive Wastes - Response to
Comments for Final Rule, Volume I, EPA 520/1-85-024-1, August
1985.**

This report presents a compilation of public comments and the EPA's responses in support of the promulgation of the proposed environmental standards.

14. **High-Level and Transuranic Radioactive Wastes - Response to
Comments for Final Rule, Volume II, EPA 520/1-85-024-2, August
1985.**

This report presents a compilation of comments generated by the Science Advisory Board and the EPA's responses in support of the promulgation of the proposed environmental standards.

15. **Environmental Pathway Models for Estimating Health Effects From
Disposal of High-Level Radioactive Waste in Geologic Repositories,
EPA 520/5-85-026, May 1986.**

This report presents detailed methodology and models to characterize the mobilization, environmental transport, exposure pathways, and doses associated with potential releases of radioactive materials from high-level waste repositories.

16. **Issues Associated with Gaseous Releases of Radionuclides for a Repository in the Unsaturated Zone, Prepared by SC&A, Inc. and Rogers & Associates Engineering Corp., Contract No. 68D90170, Work Assignment 2-41, June 10, 1992.**

This report evaluates gaseous releases that might be expected to occur from a repository in the unsaturated zone. It examines source terms and pathways, and presents a discussion of the likelihood of gaseous releases exceeding proposed containment requirements.

17. **Application of Containment Requirements of 40 CFR Part 191 to Wastes Other than Spent Fuel: Analysis of Notes to Table 1. Final Draft, Prepared by Neil J. Numark, Associates, Inc., Contract No. 68D90170, December 1991.**

This report provides an analysis of the Notes to Table 1, Appendix A of the proposed 40 CFR 191. Particular attention is given to the application of the release limits to TRU wastes.

18. **Evaluation and Update of Spent Fuel Radionuclide Inventory to be Used in Repository Risk Assessments, Task 4 Report, Draft, prepared by Neil J. Numark, Associate, S. Cohen & Associates, Inc., contract No. 68D90170, April 1992.**

This report evaluates and updates the HLW inventory to be used in the risk assessment for 40 CFR 191. It incorporates 1991 data from the Oak Ridge National Laboratory's Characteristics Data Base for spent fuel.

19. **The Effects of Distance and Simulation Time on Individual Doses and Normalized Releases From A Repository, Task 6 Report, prepared by Rogers and Associates Engineering Corporation, for S. Cohen & Associates, Contract No. 68D90170, April 1992.**

This report evaluates the effects of varying regulatory time periods and distances to the accessible environment on the cumulative releases of radionuclides and groundwater doses to an individual from a repository.

20. **Issues Discussed at Workshops on the High-Level Waste Disposal Regulatory Criteria in 40 CFR 191, prepared by Rogers & Associates Engineering Corporation, under Contract with Sandy Cohen & Associates, Inc., Contract No. 68D90170, Work Assignment 2-78, June 1992.**

This report provides a summary of the issues identified and addressed at two Electric Power Research Institute workshops (held September 1991 and February 1992) on the scientific and technical issues underlying the HLW disposal criteria.

21. Updated Uncertainty Analysis of EPA River Mode Pathways Model Used for 40 CFR 191, Prepared by SC&A, Inc. Contract No. 68D90170, Work Assignment No. 1-107, September 1991.

This report updates the uncertainty analysis of the river mode pathways model developed for the 1985 promulgation of 40 CFR 191.

22. Risk Assessment for TRU Waste Disposal in Bedded Salt; Prepared by Rogers & Associates Engineering Corporation, under contract with Sandy Cohen & Associates, Inc., Contract No. 68D90170, Work Assignment 2-29, March 1992.

This report augments the analysis of TRU waste disposal provided in the 1985 BID for 40 CFR 191. It expands the discussion of uncertainty and sensitivity.

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Chapter 2: CURRENT REGULATORY PROGRAMS AND STRATEGIES**2.1 INTRODUCTION**

Human beings and all other living organisms, have always been exposed to ionizing radiation from cosmic rays and the naturally occurring radioactivity contained in the Earth. These two sources of radiation or radioactivity make up the natural radiation background environment in which all life forms have evolved. Our experience with radioactivity dates back only to the end of the last century, when x-rays were discovered in 1895 and naturally occurring radioactivity was observed in 1896. These discoveries marked the beginning of the deliberate use of radioactivity in science, medicine, and industry.

The findings of radiation science rapidly led to the development of medical radiology, industrial radiography, nuclear physics, and nuclear medicine. By the 1920s, the use of x-rays in diagnostic medicine and industrial applications was widespread. Radium was being routinely used in luminescent dials and by doctors in therapeutic procedures. By the 1930s, biomedical and genetic research scientists were studying the effects of radiation on living organisms and physicists were beginning to understand the mechanisms of spontaneous fission and radioactive decay. In the 1940s, research in nuclear physics had advanced to the point where a self-sustaining fission reaction was demonstrated under laboratory conditions. These events led directly to the construction of the first nuclear reactors and the development of atomic weapons.

Since the end of World War II, research and development activities in all aspects of nuclear physics have been accelerating. Today the use of radiation or radioactivity, be it naturally-occurring or man-made, is widespread and reaches every segment of our society. The uses or applications include:

- Nuclear reactors, which generate electricity and power ships and submarines; produce radioisotopes for research, medical and industrial applications, space, and national defense; and are used as research tools for nuclear engineering and physics.
- Particle accelerators, which produce radioisotopes and radiation, are used to study the structure of matter, atoms, and common materials.
- The radio-pharmaceutical industry, which provides the radioisotopes used in nuclear medicine, biomedical research, and medical treatment.
- Nuclear medicine, which uses radioisotopes for the diagnosis and treatment of numerous diseases.
- X-rays and gamma rays, which are widely used as diagnostic tools in medicine and in diverse industrial applications, such as industrial radiography, luggage x-ray inspections, and non-destructive materials testing.

- Radionuclides, which are used in common consumer products, such as smoke detectors, luminous-dial wrist watches, luminous markers and signs, cardiac pacemakers, lightning rods, static eliminators, welding rods, lantern mantles, and optical glass.

As the use of radioactive materials and radiation became widespread, it was recognized that their use would have to be controlled to protect the users, public, and the environment. The following sections present a brief history of the evolution of radiation protection activities, principles and concepts used in radiation protection, and regulatory programs and strategies. These activities are summarized for two basic types of organizations - those responsible for direct regulation and oversight and those that only provide technical guidance and regulatory recommendations without the force of law.

2.2 THE INTERNATIONAL ATOMIC ENERGY COMMISSION, INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, AND THE NATIONAL COUNCIL ON RADIATION PROTECTION AND MEASUREMENTS

Initially, the dangers and risks posed by x-rays and radioactivity were poorly understood. By 1896, however, "x-ray burns" were being reported in the medical literature, and by 1910, it was understood that such "burns" could be caused by radioactive materials. By the 1920s, sufficient direct evidence (from radium dial painters, medical radiologists, and miners) and indirect evidence (from biomedical and genetic experiments with animals) had been accumulated to persuade the scientific community that an official body should be established to make recommendations concerning human protection against exposure to x-rays and radium.

At the Second International Congress of Radiology meeting in Stockholm, Sweden, in 1928, the first radiation protection commission was created. Reflecting the uses of radiation and radioactive materials at the time, the body was named the International X-Ray and Radium Protection Commission. It was charged with developing recommendations concerning radiation protection. In 1950, to better reflect its role in a changing world, the Commission was reorganized and renamed the International Commission on Radiological Protection (ICRP).

During the Second International Congress of Radiology, the newly created Commission suggested to the nations represented at the Congress that they appoint national advisory committees to represent their viewpoints before the Commission, and to act in concert with the Commission in developing and disseminating recommendations on radiation protection. This suggestion led to the formation, in 1929, of the Advisory Committee on X-Ray and Radium Protection as the advisory group for the United States. This Committee operated until 1964 when it was Congressionally chartered as the National Council on Radiation Protection and Measurements (NCRP).

Throughout their existence, the ICRP and the NCRP have worked together closely to develop radiation protection recommendations that reflect the current understanding of the risks associated with exposure to ionizing radiation (ICRP34, ICRP38, ICRP51, ICRP59, ICRP65).

In 1977, the ICRP released recommendations which are currently in use. In ICRP Publication No. 26 (ICRP77), it adopted the weighted whole-body dose equivalent (defined as the effective dose equivalent) concept for limiting occupational exposures. This change reflected the increased understanding of the differing radio-sensitivities of various organs and tissues, and was intended to sum exposures from external sources and from internally deposited nuclides. (Note: the concept of summing internal and external exposures to arrive at total dose had been mentioned as early as ICRP Publication No. 1 [ICRP59]). The occupational overall annual exposure limit is now 5 rem (as an effective dose equivalent).

The ICRP report also introduced the concept of stochastic and non-stochastic radiation effects, and defined the aim of radiation protection as to "... prevent detrimental non-stochastic effects and to limit the probability of stochastic effects to levels deemed to be acceptable..." The concept of collective dose equivalent for populations was also discussed. Also significant is the fact that the ICRP 26 recommendations represent the first explicit attempt to relate and justify permissible radiation exposures with quantitative levels of acceptable risk. The ICRP concluded that "...the mortality risk factor for radiation-induced cancers is about 10^{-4} per rem, as an average for both sexes and all ages..." Thus, the risks of average occupational exposures (about 0.5 rem/year) are roughly comparable to risks experienced in safe industries, 10^{-4} annually. At the permissible limit of 5 rem/year, the risk is comparable with that experienced by some workers in occupations having higher-than-average risk.

For members of the public, the ICRP considered that an annual risk in the range of 10^{-6} to 10^{-5} would likely be acceptable. This would imply the restriction of the annual dose to an individual of 100 mrem. The existing recommended annual dose limit of 500 mrem, applied to critical groups, was found to provide an adequate degree of safety, even though a few individuals exposed to the limit could have an annual risk in the range of 10^{-5} to 10^{-4} .

The ICRP recommended the continued use of the 500 mrem annual limit for individuals, under specified conditions. No dose limits for populations were proposed; the Commission felt that the system of dose limitation specified in ICRP 26 was "...likely to ensure that the average dose equivalent to the population will not exceed 50 mrem per year..."

In 1979, the ICRP issued Publication No. 30 (ICRP79), which established the Annual Limit on Intake (ALI) system for limiting the intake of radionuclides by workers. The ALI is the activity of a given nuclide which would irradiate a person to the limit set in ICRP No. 26 for each year of occupational exposure. It is a secondary limit, based on the primary limit of equivalent whole-body irradiation, and applies to intake by either ingestion or inhalation. The recommendations of ICRP No. 30 applied only to occupational exposures. In 1983, the ICRP issued a statement (ICRP84) to clarify the use of ALIs and DACs for members of the public. It was recommended that the appropriate authorities should assess each specific situation.

In 1985, the ICRP issued a statement (ICRP85) commenting on dose limits for members of the public. ICRP No. 26 had endorsed an annual limit of 500 mrem, subject to certain conditions. In making this endorsement, it was assumed that the conditions would, in

practice, restrict the average annual dose to about 100 mrem. In the 1985 statement, the Commission stated that the principal limit was 100 mrem, while occasional and short-term exposures up to 500 mrem were thought to be acceptable. More recently, the Commission has published additional guidance for waste disposal (ICRP85b) and for general radiological protection (ICRP91). The first of these "Radiation Protection for the Disposal of Solid Radioactive Waste" emphasizes an individual risk approach that considers both the probability of an event and its consequence.

In 1987, the NCRP issued Report No. 91 (NCRP87), which acknowledged the assumptions and the basic thrust of the recommendations in ICRP Reports 26 and 30. In discussing risk estimates, the NCRP noted that (in 1987) new data were becoming available which might require changes in the current estimates. However, the value of 10^{-4} per rem, recommended in ICRP No. 26, was retained for a nominal lifetime somatic risk for adults.

The NCRP also noted that continuous annual exposure to 100 mrem, which approximates the average whole-body background exposure, gives a person a mortality risk of about 10^{-5} annually, or approximately 10^{-3} in a lifetime. Annual limits of 500 mrem were recommended for infrequent exposures and 100 mrem for continuous (or frequent) exposures. These limits do not include natural background or medical exposures.

In 1989, the International Atomic Energy Agency (IAEA) issued reports 96 and 99 in its Safety Series (IAEA89a, IAEA89b). These documents presented criteria and guidance for the underground disposal of nuclear wastes. Safety Series No. 99, "Safety Principles and Technical Criteria for the Underground Disposal of High Level Radioactive Wastes," sets out basic design objectives to ensure that "humans and the human environment will be protected after closure of the repository and for the long periods of time for which the wastes remain hazardous." It states that for releases from a repository due to gradual processes, the dose upper bound should be less than an annual average dose value of 1 mSv for prolonged exposures for individuals in the critical group (defined as the members of the public whose exposure is relatively homogeneous and is typical of individuals receiving the highest effective dose equivalent or dose equivalent from a given radiation source). It suggests a risk upper bound of 10^{-5} per year for an individual for disruptive events.

2.3 INTERNATIONAL STANDARDS-SETTINGS

As with the United States, countries which are committed to use nuclear power (or in which nuclear power already makes up a significant fraction of the total electrical generating capacity) are establishing long-term programs for the safe management and disposal of spent reactor fuel and high-level radioactive and transuranic wastes (collectively referred to here as HLW). Such programs include adopting a national strategy, assigning the technical responsibility for research and development activities to a state-owned agency, and setting regulatory standards to protect the public and environment. Typically, the HLW management strategy includes, to varying degrees, spent fuel reprocessing, long-term spent fuel storage at and away from reactor sites, and HLW disposal in deep geological media. For illustrative purposes, the program of eight countries is summarized below (NEA86, NEA88, SCH88, IEAL87). These countries are Canada, the United Kingdom, France, the

Federal Republic of Germany, Belgium, Switzerland, Sweden, and Japan. A summary of these countries' planned HLW disposal programs is also provided in Chapter 4.

2.3.1 Canada

In 1986, Canada produced about 15% of its electrical needs through nuclear power (18 pressurized heavy water cooled and moderated reactors). Canada relies on the CANDU reactor design which operates using natural uranium in a once-through fuel cycle, i.e., the fissile material is not recycled or reprocessed. It is estimated that by the year 2000, Canada will have about 34,000 metric tons (heavy metal) of HLW destined for storage or disposal.

The Atomic Energy of Canada Limited (AECL) has the lead role in developing a HLW disposal facility. The AECL has reached a cooperative agreement with Ontario Hydro (a provincially owned utility) for developing interim technologies for the storage and transportation of spent fuel. The Atomic Energy Control Board (AECB) is the lead regulatory agency for assessing and determining the long-term performance of the disposal facility. The AECB also develops and issues policy statements and regulatory guidance for the eventual licensing of the HLW repository.

Since 1984, AECB has issued and proposed criteria addressing the regulatory assessment process, objectives and guidelines for radioactive waste disposal, and facility siting requirements. The overall regulatory objective is to ensure that there is a small probability that radiation doses to the public associated with the repository will not exceed a small fraction of the natural background radiation doses. The concepts of population exposure or dose limits are not used since AECB believes that exposure limits are not enforceable in the long-term. Rather, AECB assigns an acceptable risk of 1.0×10^{-6} per year for serious health effects. For the purpose of demonstrating compliance, the time period need not exceed the first 10,000 years.

2.3.2 United Kingdom

In 1987, the United Kingdom (Britain) produced about 17% of its electrical needs through nuclear power. Britain depends primarily on gas cooled reactors (40 units), but it is also considering other reactor designs, including breeder reactors and pressurized light water reactors. By the turn of the century, Britain plans to have completed the construction of five new power plants and to have placed four more on the drawing boards scheduled for construction early next century. Britain reprocesses spent fuel, and since 1952 nearly 30,000 metric tons (heavy metal) have been reprocessed. Of this total, about 15,000 metric tons have been recycled as new reactor fuel. Britain's current plans are to solidify (in glass) the reprocessing wastes, and then disposal in deep geological media. It is estimated that by the year 2000, Britain will have about 4,000 cubic meters (about 141,000 cubic feet) of HLW wastes destined for storage or disposal due to the reprocessing of some 60,000 metric tons of spent fuel.

The responsibility for the disposal and safeguard of radioactive wastes is shared by several governmental agencies. The regulatory functions are performed by the Radiochemical Inspectorate of the Department of the Environment; Ministry of Agriculture,

Fisheries, and Food; Nuclear Installations Inspectorate, which is part of the Health and Safety Executive; United Kingdom Atomic Energy Authority; and the Secretaries of the State of Scotland and Wales. The government also takes advice from several independent expert and advisory committees, including the Radioactive Waste Management Advisory Committee. In 1982, the government established the Nuclear Industry Radioactive Waste Executive (NIREX) to develop and operate intermediate and low-level radioactive wastes disposal facilities. NIREX was originally established as a partnership consisting of private firms and governmental agencies. In 1985, NIREX was restructured as an independent legal entity as UK NIREX. The current government strategy is to rely on storage at reactor sites and use fuel processing facilities. Reprocessed and solidified wastes will be stored for 50 years prior to disposal. The need for a high-level waste repository is not contemplated until the year 2040.

The Atomic Energy Act of 1946 establishes the authority and responsibility to control and regulate the development of nuclear power in Britain. The Act has since been amended several times to establish new requirements, including those addressing the management and disposal of radioactive wastes. The government has, however, not issued detailed regulations for HLW waste disposal. Current guidance suggests that radiation exposure limits for members of the general public would most likely be based on ICRP guidance, or about 10 mrem per year.

2.3.3 France

In 1987, France produced about 70% of its electrical needs through nuclear power. The French nuclear power program relies primarily on pressurized light water reactors (44 units). Older gas cooled reactors are being phased out, while research and development activities and demonstration projects focus on an alternate reactor design (liquid metal fast breeder reactor) for power production. France reprocesses spent fuel, and since 1976 nearly 10,000 metric tons (heavy metal) have been reprocessed. It is estimated that by the year 2000, France will accumulate about 80,000 cubic meters (nearly 3 million cubic feet) of HLW wastes destined for interim storage and disposal. Current plans are to first solidify (in glass) about 3,000 cubic meters (about 106,000 cubic feet) of the reprocessing wastes before placement and disposal in deep geological formations. In addition to its domestic market, the French spent fuel reprocessing industry also offers reprocessing services to foreign customers.

The French nuclear power industry is controlled by several agencies, some of which are quasi-governmental agencies. The key agencies include the French Atomic Energy Commission (CEA); the Ministry of Industry; the Institute for Nuclear Protection and Safety; the Central Service for the Safety of Nuclear Installations (SCSIN); the Bureau of Geological and Mineral Research; the Electricity de France (the national electric utility); and the CEA's subsidiaries, including the National Radioactive Waste Management Administration (ANDRA), COGEMA (operator of spent fuel reprocessing and HLW immobilization facilities), and SGN (architect and engineering services).

ANDRA was formed in 1979 to be responsible for all radioactive waste disposal activities and long-term management. ANDRA is chartered to design, build, and operate

waste disposal facilities. ANDRA must also comply with the CEA requirements as well as those promulgated by the Central Service for the Safety of Nuclear Installations (SCSIN), which is an independent agency under the Ministry of Industry. The facility siting criteria and system performance requirements have not yet been issued by the government. An advisory committee is providing guidance to the SCSIN on the formulation of technical and safety criteria. Radiation protection limits have not been defined nor have any repository system performance requirements been issued. Long term performance will have to be demonstrated, via modelling, for a 10,000-year period and the assessment may be extrapolated out to 100,000 years.

2.3.4 Germany - Federal Republic

By the end of 1987, the Federal Republic of Germany produced about 34% of its electrical needs through nuclear power. The German nuclear power program relies primarily on pressurized light water reactors (11 units) and boiling water reactors (7 units). Research and development activities and demonstration projects are also evaluating alternate reactor designs (high temperature gas-cooled reactors and liquid metal fast breeder reactor) for power production. Germany is currently shipping its spent fuel to France and Britain for reprocessing. It is estimated that by the year 2000, Germany will accumulate about 8,000 metric tons (heavy metal) of spent fuel to be destined for interim storage and reprocessing.

In Germany, the institutional and legal framework for the regulation of nuclear facilities is based on the joint participation of Federal and State governments. The Atomic Energy Act and the Radiation Protection Ordinance establish the principles and requirements regarding the safe utilization and application of atomic energy and radioactive materials, including the disposal of radioactive wastes. The key agencies include the Ministry for Environmental Protection and Reactor Safety, the Federal Ministry of Research and Technology, the Federal Office of Radiation Protection, the Federal Institute of Geosciences and National Resources, and the host State's Ministry for Environmental Protection. In addition, a consortium of Germany's nuclear utilities and engineering firms have been formed to meet the industry's responsibilities for spent fuel storage, reprocessing, waste management, and waste disposal.

The disposal of radioactive wastes in deep geological media is governed by regulatory requirements issued in 1982. The regulations provide specific objectives to be met for each phase of the development of the repository. Additional licensing procedures and guidance will be issued in support of the licensing activities. The long-term performance objectives for the repository require that doses to members of the general population be limited to 30 mrem per year following closure. It has been proposed that the performance assessment of the repository targets a period of 10,000 years; however, this time duration has not yet been endorsed by the licensing agency.

2.3.5 Belgium

By the end of 1987, Belgium produced about 65% of its electrical needs through nuclear power. The Belgian nuclear power program relies on seven pressurized light water reactors. From 1966 to 1974, Belgium was reprocessing spent fuel at its EUROCHEMIC

facility. Recent efforts have failed to reactivate this facility under a new consortium, SYNATOM, consisting of domestic and foreign firms. Belgium is currently shipping its spent fuel to France for reprocessing. It is estimated that by the year 2000, Belgium will accumulate about 2,500 metric tons (heavy metal) of spent fuel destined for interim storage and reprocessing.

An independent agency (ONDRAF) was established in 1982 for the long-term management and disposal of radioactive wastes, including spent fuel, high-level wastes, and reprocessing wastes returned from the French facility. In addition to ONDRAF, the other key organizations or agencies with direct responsibilities in waste management include the Ministry of Public Health, the Ministry of Employment and Labor, the Ministry of Interior, and the Ministry of Foreign Relations. An inter-ministerial commission was also established to coordinate all related activities with each ministry. The Nuclear Energy Research Center, under the Ministry of Economic Affairs, provides technical assistance in basic and applied R&D in nuclear energy and technology.

There are currently no specific regulatory requirements or criteria governing the disposal of spent fuel and high-level wastes. In the past, Belgium has established radiation safety criteria by organizing committees consisting of international experts as well as adopting U.S. Nuclear Regulatory Commission requirements.

2.3.6 Switzerland

There are currently five nuclear power plants in Switzerland supplying about 39% of the country's electrical power needs. The Swiss nuclear power program relies on a mix of pressurized and boiling light water reactors (3 PWRs and 2 BWRs). Switzerland is currently shipping its spent fuel to France and Britain for reprocessing. It is estimated that by the year 2000, the Swiss will accumulate about 2,000 metric tons (heavy metal) of spent fuel to be destined for interim storage and reprocessing.

A joint government and utility cooperative agency (NAGRA) was established in 1972 to manage the disposal of radioactive wastes, including spent fuel, high-level wastes, and reprocessing wastes returned from the French and British spent fuel processing facilities. In addition to NAGRA, other key organizations or agencies with direct responsibilities in waste management include the Federal Council, the Federal Department of Transport, Communications, and Energy, the Federal Commission for Safety in Nuclear Installations, the Federal Department of Interior, and the Institute of Reactor Research. An interagency working group (AGNEB) was also established to coordinate and support all related activities with each federal agency and the Federal Council.

The Atomic Energy and Radiation Protection Act of 1959 was revised in 1978 to incorporate specific provisions for managing the disposal of radioactive wastes. More recent amendments, published by the Federal Energy Office, have included setting general safety goals. These requirements establish a maximum yearly dose to members of the general public of 10 mrem, following closure. The Federal Energy Office plans to revise its requirements and issue additional guidance in the near future.

2.3.7 Sweden

There are currently 12 nuclear power plants in Sweden supplying about 50% of the country's electrical power needs. The Swedish nuclear power program relies primarily on a boiling light water reactors (9 units) and pressurized water reactors (3 units). Sweden is currently shipping its spent fuel to France and Britain for reprocessing. The Swedish nuclear power program is slated (by a 1980 national referendum) to end by the year 2010. It is estimated that by the year 2010, Sweden will accumulate nearly 8,000 metric tons (heavy metal) of spent fuel and high-level wastes for interim storage, reprocessing, and final disposal.

A joint utility consortium, the Swedish Nuclear Fuel and Waste Management Company (SKB), was established in 1984 to manage the disposal of radioactive wastes, including spent fuel, high-level wastes, and reprocessing wastes returned from the French and British spent fuel processing facilities. In addition to SKB, the other key organizations or agencies with direct responsibilities in waste management, operating under the Ministry of the Environment and Energy, include the Swedish Nuclear Power Inspectorate (SKI), the National Board for Spent Nuclear Fuel (SKN), the National Institute for Radiation Protection (SSI), and the Swedish Consultative Committee for Nuclear Waste Management (KASAM).

The 1956 Atomic Energy Act was revised in 1981 and 1984 to incorporate specific provisions for managing the disposal of radioactive wastes. The 1984 amendment has included setting general safety goals establishing a maximum yearly dose to members of the general public of 10 mrem, following closure. For the purpose of demonstrating compliance with the standards, SKI is currently considering a 1,000-year assessment period, while emphasizing radionuclide release rates rather than individual dose or exposure rates. SKI plans to provide more specific requirements and issue additional guidance in the near future.

2.3.8 Japan

By the end of 1987, Japan produced about 32% of its electrical needs through nuclear power. The Japanese nuclear power program relies on pressurized light water reactors (23 units) and boiling water reactors (27 units). Research and development activities and demonstration projects are also evaluating alternate reactor designs (gas cooled reactor, heavy water moderated reactor, and liquid metal fast breeder reactor) for power production. Japan is currently reprocessing its spent fuel and has also secured reprocessing services from France and Britain. A second spent fuel reprocessing facility is planned for operation by the mid 1990s. It is estimated that by the year 2000, Japan will accumulate about 20,000 metric tons (heavy metal) of spent fuel destined for interim storage, reprocessing, and eventual disposal.

The Japanese institutional and legal framework for the regulation of nuclear facilities is based on the joint participation of the Federal government and high-level waste generators. The Atomic Energy Basic Law of 1955 established the Japan Atomic Energy Commission (JAEC) and the principles and requirements regarding the safe utilization and application of atomic energy and radioactive materials, including the disposal of radioactive wastes. In 1956, the government set up the Japan Atomic Energy Research Institute for the purpose of

conducting safety and technology related research on nuclear energy, including the back end of the nuclear fuel cycle. In addition to the JAEC, other key agencies or organizations include the Nuclear Safety Commission, the Ministry of International Trade and Industry, the Ministry of Transport, the Science and Technology Agency, the Power Reactor and Nuclear Fuel Development Corporation (PNC), the Japan Nuclear Fuel Services Company, the Radioactive Waste Management Center, and the Radiation Council. In addition, the Japanese nuclear utilities and engineering firms have formed two consortia (JAIF and FEPCO) to meet the industry's responsibilities for spent fuel storage, reprocessing, waste management, and waste disposal.

The guidelines governing the Japanese program for the disposal of high-level radioactive wastes in deep geological media were issued in 1985. The Science and Technology Agency and the Power Reactor and Nuclear Fuel Development Corporation were chartered to conduct the necessary research and in-situ experiments to support the development of a repository. The JAERI is to develop regulations and provide specific criteria to conduct safety and performance assessment analyses of the repository. No specific licensing procedures and guidance have been issued in support of the licensing program and activities. No formal individual limits have been issued, but a dose limit of 5 mrem per year has been proposed following closure. The time period for complying with performance objective criteria has not yet been specified.

2.4 FEDERAL RADIATION COUNCIL GUIDANCE

The ICRP and the NCRP function as non-governmental advisory bodies. Their recommendations are not binding on any user of radiation or radioactive materials. The wealth of new scientific information on the effects of radiation that became available in the 1950s prompted President Eisenhower to establish an official government entity with responsibility for formulating radiation protection criteria and coordinating radiation protection activities. Thus, the Federal Radiation Council (FRC) was established in 1959 by Executive Order 10831. The Council included representatives from all of the Federal agencies concerned with radiation protection and acted as a coordinating body for all of the radiation activities conducted by the Federal government (FRC60). In addition to its coordinating function, the Council's major responsibility was to:

"...advise the President with respect to radiation matters, directly or indirectly affecting health, including guidance for all Federal agencies in the formulation of radiation standards and in the establishment and execution of programs of cooperation with States..."

The Council's first recommendations concerning radiation protection guidance for Federal agencies were approved by the President in 1960. Based largely on the work and recommendations of the ICRP and NCRP, the guidance established occupational exposure limits, which differed only slightly from those recommended by NCRP and ICRP at the time (NCRP54, NCRP59).

- Whole body, head and trunk, active blood forming organs, gonads or lens of the eyes are not to exceed 3 rem in 13 weeks and the total accumulated

dose is limited to 5 times the number of years beyond age 18, expressed as $5(N-18)$, where N is the current age.

- Skin of the whole body and thyroid are not to exceed 10 rem in 13 weeks or 30 rem per year.
- Hands, forearms, feet, and ankles are not to exceed 25 rem in 13 weeks or 75 rem per year.
- Bone is not to exceed 0.1 microgram of radium-226 or its biological equivalent.
- Any other organs are not to exceed 5 rem per 13 weeks or 15 rem per year.

The guidance also established exposure limits for members of the public. These were set at 0.5 rem per year for the whole body for an individual and an average gonadal dose of 5 rem in 30 years.

In addition to the formal exposure limits, the guidance also established as Federal policy that there should be no radiation exposure without an expectation of benefit, and that "...every effort should be made to encourage the maintenance of radiation doses as far below this guide as practicable..." The inclusion of the requirements to consider benefits and keep all exposures to a minimum was based on the possibility that there is no threshold for radiation. The linear non-threshold dose response relationship was assumed to place an upper limit on the estimate of radiation risk. However, the FRC explicitly recognized that it might also represent the actual level of risk. If so, then any radiation exposure carried some risk, and it was necessary to avoid all unproductive exposure and to keep all productive exposures as "far below this guide as practicable."

2.5 THE ENVIRONMENTAL PROTECTION AGENCY

In 1970, the functions of the Federal Radiation Council were transferred to the U.S. Environmental Protection Agency (EPA). Since then, the EPA has issued Federal guidance for the control of radiation hazards in underground mining (EPA71), for setting occupational exposure limits (EPA81), for occupational exposures of workers subject to federal regulations (EPA87), standards and technical information regarding radionuclide intake and air concentration limits, occupational radiation doses, biological parameters, and dose conversion factors (EPA88).

In addition to the statutory responsibility to provide Federal guidance on radiation protection, the EPA has various statutory responsibilities regarding regulation of exposure to radiation. The standards and the regulations that EPA has promulgated and proposed with respect to controlling radiation exposures and which are related to 40 CFR Part 191 are summarized here.

2.5.1 Atomic Energy Act

The Atomic Energy Act of 1954, as amended, and Reorganization Plan No. 3 granted the EPA the authority to establish generally applicable environmental standards for exposure to radionuclides (AEA54, NI70). Pursuant to this authority, in 1977 the EPA issued standards limiting exposures from operations associated with the light-water reactor fuel cycle (EPA77). These standards, under 40 CFR Part 190, cover normal operations of the uranium fuel cycle, excluding mining and radioactive waste disposal. The standards limit the annual dose equivalent to any member of the public from all phases of the uranium fuel cycle (excluding radon and its daughters) to 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ. To protect against the buildup of long-lived radionuclides in the environment, the standard also sets normalized emission limits for krypton-85, iodine-129, and plutonium-239 combined with other transuranics with a half-life exceeding one year. The dose limits imposed by the standard cover all exposures resulting from radiation and radionuclide releases to air and water from operations of fuel-cycle facilities. The development of this standard took into account both the maximum risk to an individual and the overall effect of releases from fuel-cycle operations on the population, and balanced these risks against the costs of effluent control.

2.5.2 Safe Drinking Water Act

Under the authority of the Safe Drinking Water Act, the EPA issued interim regulations (40 CFR Part 141, Subpart B) covering the permissible levels of radium, gross alpha, man-made beta, and photon-emitting contaminants in community water supply systems (EPA76). The limits are expressed both in terms of average and maximum concentration limits (picocurie/liter) and annual doses to the whole body or organs. The allowable limit for radium-226 and radium-228, combined, is 5 picocuries per liter. For total gross alpha activity, including radium-226 but excluding radon and uranium, the maximum concentration limit is 15 picocuries per liter. The standard also specifies maximum concentration limits for strontium-90 and tritium. The dose limits chosen for man-made beta and photon emitters is 4 mrem/year to the whole body or organ dose for the most exposed individual. The supporting information for the standard justifies the 4 mrem/year dose limit and compares it to the recommended population exposure of 170 mrem/year per capita. The conclusion reached is that when considering all exposure pathways, a 40-fold decrease is appropriate for this single pathway.

In 1991, the EPA issued a Notice of Proposed Rulemaking (NPR) to update the 1976 interim regulations for radionuclide water pollution control (EPA91). The NPR, under the Safe Drinking Water Act, proposed the establishment of Maximum Contaminant Level Goals (MCLGs) and Maximum Contaminant Levels (MCLs). The MCLGs and MCLs target radium-226, radium-228, natural uranium, radon, gross alpha, and gross beta, and photon emitters. As proposed, MCLGs are not enforceable health goals while MCLs are enforceable standards. The EPA concluded that radionuclide MCLGs should be set at zero to avert known or anticipated adverse health effects while providing an adequate margin of safety. In setting the MCLGs, the EPA also committed itself to evaluate the feasibility, costs, and availability of water treatment technologies, as well as other practical considerations. The proposed regulations provide the following MCLs: radium-226,

20pCi/l; radium-228, 20 pCi/l; radon-222, 300 pCi/l; uranium, 20 micro g/l; adjusted gross alpha, 15 pCi/l; and beta and photon emitters, 4 mrem ede/yr. In general, these limits yield doses of between 4 mrem/yr and 20 mrem/yr to individuals drinking the contaminated water.

2.5.3 Clean Air Act

Section 112 of the Clean Air Act (CAA) Amendments of 1977 (Public Law 95-95) directed the EPA Administrator to review all relevant information and to determine if airborne emissions of hazardous pollutants will cause or contribute to air pollution that may reasonably be expected to endanger public health. In December 1979, the EPA designated radionuclides as hazardous air pollutants under Section 112 of the Act (EPA79). In April 1983, the EPA proposed standards regulating radionuclide emissions from four source categories, one of which included DOE facilities. The rule established annual airborne emission limits for radioactive materials and specified that annual doses resulting from such emissions do not exceed 25 mrem to the whole body and 75 mrem to any critical organ to members of the general public. The EPA also proposed not to regulate several other categories of facilities, including high-level radioactive waste disposal facilities.

In October 1984, following a court order to promulgate final radionuclide emission standards or make a finding that radionuclides are not hazardous air pollutants, the EPA withdrew the proposed emission standards based on the findings that the control practices already in effect protected the public from radionuclide releases with an ample margin of safety. The Agency also affirmed its position not to regulate other categories of emission sources, including uranium fuel facilities and high-level radioactive wastes.

In December of 1984, a U.S. District Court found the EPA in contempt of its order and directed the EPA to either issue final radionuclide emission standards or make a finding that radionuclides are not hazardous air pollutants. The EPA complied with the Court in 1985 order by issuing standards for selected sources, National Emission Standards for Hazardous Air Pollutants (NESHAPs) (EPA85a, EPA85b). As a result of the decision in National Resources Defense Council Inc. vs. EPA, which concluded EPA had improperly promulgated vinyl chloride regulations under Section 112 of the CAA by considering cost and technological feasibility, the Agency in November 1987 moved the Court for a voluntary remand of the NESHAPs for the four original categories of emission sources. The EPA agreed to re-examine all issues raised by the parties to the litigation. In December 1987, the Court granted the EPA's motion for voluntary remand and established a schedule to propose new regulatory standards within one year. The Court decision also defined the analytical process under which the EPA was to re-evaluate its standards. Two steps were identified: 1) first determine what is safe, based exclusively on health risk; and 2) adjust the level of safety downward to provide a greater or ample margin of safety.

In March 1989, the EPA issued a proposed rule for regulating radionuclide emissions under NESHAPs following the re-examination of the regulatory issues associated with the use of Section 112 (EPA89). The draft rule proposes four policy alternatives to control emissions and risks from 12 categories of sources, including DOE facilities. Each of the four approaches treats the acceptable risk criterion differently. The four approaches were:

- **Case-by-Case Approach** - Acceptable risk considers all health information, risk measures, potential biases, assumptions, and quality of the information. The preferred level of maximum individual lifetime risk must be 10^{-4} or less.
- **Incidence-Based Approach** - Based on the best estimate of the total incidence of fatal cancer. The proposed acceptable level of incidence must not exceed more than 1 fatal cancer per year per source category.
- **Maximum Individual Risk Approach (10^{-4} or less)** - Only parameter being considered is the best estimate of the maximum individual lifetime risk of fatal cancer. The acceptable maximum individual lifetime risk must not exceed 1×10^{-4} .
- **Maximum Individual Risk Approach (10^{-6} or less)** - This approach is similar to the previous one. The acceptable risk, however, must not exceed 1×10^{-6} .

The definition of the ample margin of safety is established separately after the safe level has been determined based solely on health risks. In reaching its final decision, the EPA must consider all health risk measures as well as technological feasibility, costs, uncertainties, economic impacts of control technologies, and any other relevant information. This decision process may also require the EPA to determine whether or not to require all technologically feasible controls which are affordable, no matter how small the risk reduction.

Based on the comments and the record developed in the rulemaking, EPA selected an approach announced in the notice on benzene standards published on September 14, 1989 (54 FR 38044). Thus, in the first step of *Vinyl Chloride* inquiry, EPA will consider the extent of the estimated risk were an individual exposed to the maximum level of a pollutant for a lifetime. The EPA will generally presume that if the risk to that individual is no higher than approximately 1 in 10 thousand, that risk level is considered acceptable and EPA then considers the other health and risk factors to complete an overall judgement on acceptability. The presumptive level provides a benchmark for judging the acceptability of maximum individual risk, but does not constitute a rigid line for making that determination.

The rule concludes that there is no need to establish NESHAP standards for high-level waste disposal repositories since the releases and consequently the risks are very low and therefore constitute a margin of safety. The reason why the emissions and risks are so low is that radioactive materials received at such facilities are sealed in containers. Normal operations do not require additional processing or handling because spent fuels or high-level wastes are received and emplaced into the ground in their original containers. Operations at the disposal site which may require additional waste processing or repackaging, before the site is declared a disposal facility must comply with NESHAPs Subpart I.

2.5.4 Resource Conservation and Recovery Act

Some of the radioactive wastes covered by this rulemaking also contain hazardous wastes subject to the Resource Conservation and Recovery Act (RCRA); these materials are known as "mixed wastes." RCRA wastes are primarily governed by EPA regulations under 40 CFR Parts 260, 262, 263, 264, 265, 268, and 270. Section 6001 of RCRA explicitly subjects all Federal facilities and their activities to State and Federal regulations under RCRA. However, RCRA Section 1006(a) relieves facilities operating under the authority and control of the Atomic Energy Act of 1954 (AEA) from compliance with RCRA for conditions which could be inconsistent with the requirements of the AEA.

In 1987, the EPA formed the Mixed Energy Waste Study (MEWS) task force to evaluate DOE's proposed option to exempt mixed high-level radioactive wastes (HLW) and transuranic wastes (TRU) from RCRA, Subtitle C (EPA87). The MEWS task force concluded that, with some exceptions, current DOE management of mixed HLW/TRU wastes is equivalent to RCRA requirements. In other words, the management of these wastes would not change significantly if they were required to comply with RCRA Subtitle C requirements for hazardous wastes. The task force, however, noted that there were a few aspects which would not meet RCRA standards. For example, the task force noted that some waste forms do not fit "normal" management practices, particularly when dealing with submarine reactor components, classified TRU wastes, and TRU wastes unacceptable for disposal. For those aspects which do not meet RCRA standards, the task force gave the following examples: waste chemical analyses, groundwater monitoring, TRU waste retrievability, disposal of classified TRU wastes, and self-inspection. Some States were also concerned about the DOE self-regulating its HLW/TRU waste disposal activities under the proposed option, but were willing to consider case-by-case variances with specific requirements.

Since July 1986, the Agency has required states to obtain mixed waste authorization as part of their RCRA programs. Procedures for considering disposal of mixed wastes are now being developed and the Agency is issuing authorizations for States to regulate such types of mixed wastes. The EPA's Office of Radiation Programs and Office of Solid Waste are maintaining cognizance of these developments with the State programs.

2.6 NUCLEAR REGULATORY COMMISSION

Under the authority of the Atomic Energy Act of 1954, as amended, the U.S. Nuclear Regulatory Commission (NRC) is responsible for licensing and regulating the use of by-product, source, and special nuclear material, and for assuring that all licensed activities are conducted in a manner that protects public health and safety (AEA54). The Federal guidance on radiation protection applies directly to the NRC. Therefore, the NRC must assure that none of the operations of its licensees expose an individual of the public to more than 0.5 rem/year from all pathways.

The dose limits imposed by the EPA's standards for uranium fuel-cycle facilities (40 CFR Part 190) apply to the fuel-cycle facilities licensed by the NRC (See Section 2.5 for a summary of EPA regulations). These facilities are prohibited from releasing radioactive effluents in amounts that would result in doses greater than the 25 mrem/year limit imposed

by that standard. Also, NRC facilities are required to operate in accordance with the requirements of the Clean Air Act (40 CFR Part 61), which limits radionuclide emissions to air.

The NRC exercises its statutory authority over licensees by imposing a combination of design criteria, operating parameters, and license conditions at the time of construction and licensing. It assures that the license conditions are fulfilled through inspection and enforcement activities.

2.6.1 Fuel Cycle Licenses

The NRC does not use the term "fuel cycle facilities" to define its classes of licensees. The term is used here to coincide with the EPA use of the term in its standard for uranium fuel cycle facilities. As a practical matter, this term includes the NRC's large source and special nuclear material licensees and production and utilization facilities. The NRC's regulations require an analysis of probable radioactive effluents and their effects on the population near fuel cycle facilities. The NRC also assures that all exposures are maintained as low as is reasonably achievable (ALARA) by imposing design criteria for effluent control systems and equipment. After a license has been issued, fuel-cycle licensees must monitor their emissions and set up an environmental monitoring program to assure that the design criteria and license conditions have been met. For practical purposes, the NRC has adopted the maximum permissible concentrations developed by the NCRP to relate effluent concentrations to exposure.

2.6.2 Radioactive Waste Disposal Licenses

The authority for the NRC to regulate high-level waste disposal originates from Public Law 97-425, also known as the "Nuclear Waste Policy Act of 1982." The Act requires the NRC to promulgate regulations governing 1) construction authorization for a repository, 2) license to receive and dispose of wastes in the repository, and 3) authorization for repository closure (NWPA83).

This Act also requires the EPA to promulgate, "... generally applicable standards for the protection of the general environment from off-site releases of radioactive material in repositories..." The Act also requires that the NRC regulations be consistent with the EPA standards. See Sections 1.2 and 2.5 for a detailed discussion of the EPA's role and responsibilities.

The NRC regulations governing deep geologic disposal are contained in the Code of Federal Regulations, Title 10, Part 60, titled, "Disposal of High-level Radioactive Wastes in Geologic Repositories." These regulations are summarized below. In addition, the NRC certifies (under 10 CFR Part 71) packaging for the transportation of spent nuclear fuel, high-level and transuranic radioactive wastes.

Similar to the licensing of power reactors, 10 CFR Part 60 requires the waste repository operator (DOE) to submit a safety analysis report (SAR) and an environmental report (ER) in order to obtain a license to construct a repository (NRC81, NRC85). The

environmental report must meet the requirements of 10 CFR Part 51, (under NEPA) "Environmental Protection Requirements for Domestic Licensing and Related Regulatory Functions" (NEPA70).

The SAR is required to contain a description of the characteristics of the proposed repository site, including fractures, geomechanics, geochemistry and thermal loading effects. It must also include a description of the natural resources of the site, and an assessment of the waste isolation properties of the proposed site. A program of site characterization field work is required to support the preparation of the SAR. The general plan for this program of characterization is presented in a Site Characterization Plan (SCP). This plan contains the description of the studies to be conducted, their sequencing and possible interferences, and the impacts of the studies on the ability of the site to isolate and contain the waste. Before beginning site characterization, the SCP receives extensive reviews by the NRC, the host state, and other interested parties. Progress during site characterization and any changes to the plans for site characterization are reported in semiannual progress reports which are also reviewed by the NRC and other interested parties. The SAR is then prepared using the information developed during site characterization.

Upon receipt of the SAR, the NRC will conduct a safety review. The planned repository will be evaluated against the technical criteria specified in the NRC regulations in 10 CFR Part 60. If the NRC determines from this evaluation that there is reasonable assurance that the waste can be received, possessed, and disposed of safely, that the common defense and security can be protected, and that environmental values are protected, an authorization will be given to the DOE to begin construction of the repository.

After construction has been completed, the DOE will update the SAR and the environmental report and this information will be reviewed by the NRC to determine if a license to receive, possess, and dispose of waste can be granted. At this stage, the NRC will confirm that construction has been completed in conformity with the license application, and that the repository poses no unreasonable risk to public health and safety. Likewise, at the end of the operating period, the license application and environmental report are updated and an application to amend the license application is submitted by the DOE. This application and the associated updated information are reviewed by the NRC to determine if the repository may be permanently closed.

- **Technical Criteria**

At each stage of the licensing process, the SAR is reviewed to determine if the technical criteria specified in Subpart E of the NRC regulations are satisfied. These technical criteria include performance objectives and other criteria (e.g., requirements on land ownership and control, siting criteria, and design criteria) intended to ensure that the performance objectives are met. The performance objectives are set to ensure radiological safety and waste retrievability during the operating period, waste isolation and containment by the overall system after permanent closure, and adequate performance of particular barriers after permanent closure. These performance objectives require that radiation exposures, radiation levels, and releases of radioactive materials conform to the applicable environmental standards established by the EPA. Therefore, demonstration of compliance

with these standards will be an integral part of DOE's license application. The NRC regulations also specify requirements for monitoring during the institutional control period (NRC83) and provisions for the retrievability of any emplaced wastes. Other requirements deal with land ownership and waste package design criteria.

The performance objective for protection against radiation exposures and releases during the operating period requires that the repository be designed so that radiation exposures, radiation levels, and releases of radioactive materials to unrestricted areas meet the applicable environmental standards; these standards are specified in Subpart A of 40 CFR Part 191. The performance objective for waste isolation containment by the overall geologic repository system requires that releases to the accessible environment following permanent closure conform to environmental standards that apply to this period; these standards in this case are specified in Subpart B of 40 CFR Part 191.

- **Waste Isolation Pilot Plant (WIPP)**

The WIPP project is a DOE facility located near Carlsbad, NM for the disposal of defense-produced transuranic wastes. The NRC has no regulatory authority over the WIPP project; nevertheless, this facility will be required to meet the environmental standards established by the EPA for disposal of these wastes. The DOE currently has the authority to implement the regulation of the WIPP with regard to these standards and the DOE has agreed to the review of compliance with these standards by the State of New Mexico. In addition, it is possible that authority to certify that the WIPP conforms to the applicable environmental standards may be given to an external agency such as the EPA. Furthermore, certification of the containers used to ship the TRU wastes from DOE facilities to the WIPP site is under the authority of the NRC as specified in 10 CFR Part 71. Two types of shipping containers have been designed, one for contact-handled wastes and one for remote-handled wastes. Both designs are currently being reviewed and evaluated by the NRC.

2.6.3 Center for Nuclear Waste Regulatory Analysis

In the fall of 1987, the NRC created the Center for Nuclear Waste Regulatory Analysis to support repository licensing activities. Traditionally, the NRC has relied on the national laboratories for this support. Since the laboratories are largely under DOE control, their involvement in repository licensing could present a potential conflict of interest. The Center is operated by the Southwest Research Institute and is located in San Antonio, Texas.

In supporting the NRC, the Center is charged with providing long-term continuity in technical assistance and research. Also, it is to provide central capabilities for integrating all aspects of the high-level waste licensing program. Current projects at the Center include identifying priority areas of the site characterization plan (SCP) for NRC staff review, analyzing technical uncertainties pertaining to repository siting, recommending candidate areas for additional rulemaking, and assessing the importance of various regulatory requirements.

The Center is currently working on a number of special reports. These include a long-range plan, an open-item tracking system, and an issue resolution monitoring report.

Besides these special reports, the Center is also preparing a number of format and content guides, and standard review plans related to the license application.

2.6.4 Other Activities

The current NRC repository licensing program is divided into two areas - proactive activities and reactive activities. These are described briefly below.

Proactive activities are those that do not depend on DOE action. These include developing and reviewing regulatory requirements and guidance to identify and resolve uncertainties. Regulatory uncertainties exist where regulatory requirements are ambiguous and could be subject to various interpretations. Technical uncertainties are related to demonstrating compliance with a particular regulation. These are currently being addressed so that the NRC can meet the three-year license review schedule mandated by Public Law 97-245 (NWP83).

In another area, the NRC staff is developing and implementing performance assessment models using Yucca Mountain site data. This will help develop technical assessment capability, as well as identify areas of regulatory and technical uncertainty.

These activities have produced licensing review plans in anticipation of the DOE submittals. They include the SCP Review Plan, Study Plan Review Plan, and Quality Assurance Review Plan. The License Application Review Plan is still in preparation.

Other proactive activities include the evaluation of progress on actions required by NWP83. This ongoing evaluation is documented in the Quarterly Progress Reports to the Commission on the High-Level Radioactive Waste Management Program. This evaluation complements other actions and more specific reviews and consultations by taking a broad view of progress and identifying fundamental concerns.

The reactive part of the NRC program consists of pre-licensing reviews that follow DOE's sequence and schedule of activities. To date, this includes reviews of quality assurance programs for DOE and DOE contractors. Quality assurance issues need to be resolved before significant data collection activities are performed at the Yucca Mountain site.

The next major activity will involve the NRC's review of the SCP and will focus on DOE's strategies, assumptions, and programs. For the more detailed Study Plans, prepared by the DOE, the NRC will conduct a completeness review on each. However, a detailed review will be made on only a sample (about 20%) of the hundred or so Study Plans. During site characterization, the NRC will conduct on-site reviews of selected testing activities and selected data.

As site characterization activities proceed, DOE's semiannual progress reports on the site characterization program will be reviewed by the NRC. These reviews will focus on the resolution of previously identified concerns and will evaluate new information about the site and repository design. In addition, the NRC will review selected DOE study reports and

position papers that document the detailed results of work performed to date. The NRC will review DOE's topical reports and issue resolution reports, which summarize the site characterization work for specific licensing topics. These will be used to evaluate compliance with NRC regulations.

All concerns identified by the NRC will be tracked by the staff as open items. The tracking system, presently being implemented, will focus on root causes and DOE's progress toward resolution. The system will also provide and maintain a licensing record of all NRC and DOE actions related to resolving specific issues.

2.7 DEPARTMENT OF ENERGY

The U.S. Department of Energy (DOE) operates facilities for the enrichment of nuclear fuels for commercial and defense reactors, the production and testing of nuclear weapons, the management and disposal of radioactive wastes generated in national defense activities, and research and development, including several national laboratories. In addition, the DOE is conducting several remedial action programs, such as the program for the management of uranium mill tailings and the cleanup of sites formerly used for nuclear activities. These facilities and activities are not licensed by the NRC. However, to protect public health and the environment, the DOE has implemented orders and procedures that are consistent with NRC regulations under 10 CFR Part 20 (NRC60), standards promulgated by the EPA, and other applicable Federal regulations and guidelines.

The DOE is also responsible for the disposal of spent nuclear fuel and high-level radioactive wastes from defense activities and the generation of electricity by commercial nuclear reactors. The facilities, developed by the DOE for the management and disposal of these wastes, will eventually be licensed by the NRC.

2.7.1 DOE Programs for the Environment, Health, and Safety

The DOE is responsible for operating its facilities in a manner that is safe and environmentally sound, as stated in DOE Order 5400.1 (DOE88a). To this end, it has issued a number of orders specifying procedures and standards. (See Table 2.7-1) It should be noted that many of these DOE procedures and standards are currently being reviewed and revised to conform with NRC and EPA regulations and standards (DOE89). Mandatory standards for the protection of public health and the environment are established by DOE Order 5480.4 (DOE84a). These standards apply to all DOE and DOE contractor operations during facility design, construction, operation, modification, and decommissioning. The order mandates compliance with the standards promulgated by the Occupational Health and Services Administration in 29 CFR Parts 1910, 1915, 1918, 1926, and 1928 (DOL74). DOE Order 5480.1B (DOE86a) establishes procedures for the preparation and review of safety analyses for DOE operations, including the identification and control of hazards and risk assessments. DOE Order 5400.2A (DOE87) establishes specific requirements for the coordination of DOE and contractor activities to ensure the timely resolution of significant environmental compliance issues.

DOE Order 5820.2A (DOE88b) establishes policies and guidelines by which the DOE assures that all DOE facilities, including surplus facilities, involving the use of radioactive or mixed waste or waste by-products are operated in a manner that protects the health and safety of the public and the environment. The DOE is developing specific orders for the management of hazardous and radioactive mixed wastes and for environmental surveillance of radioactive effluents.

Under 5482.1B (DOE86b), the DOE established a program for environmental quality assurance; its objective is to ascertain that the DOE's environmental, safety, and health policies are properly interpreted and implemented. The DOE also complies with the national standards established jointly by the American National Standards Institute and the American Society of Mechanical Engineers (ANSI86) for quality assurance in nuclear facilities.

Under the Atomic Energy Act of 1954, as amended (AEA54), the DOE is responsible for keeping radionuclide emissions at its facilities as low as is reasonably achievable (ALARA). Under the authority of the Clean Air Act, the EPA has issued, in 40 CFR Part 61, standards (EPA89a) that limit airborne radionuclide emissions from DOE facilities to an amount that will deliver any member of the public in any year an effective dose equivalent of 10 mrem per year. The current emission levels achieved by emission control technologies and practices at DOE facilities are within these limits. In order to comply with these standards and the maximum permissible concentrations established by the National Council on Radiation Protection and Measurements for radioactive material in air and water (NCRP54, NCRP59, NCRP71), the DOE has issued Orders 5400.3 (DOE89) and 5480.11 (DOE88c) to protect the general population and workers at DOE facilities, respectively, from radioactivity in air and water. These orders set a limit of 10 millirem per year for the effective dose equivalent.

2.7.2 Compliance with Federal Regulations

The DOE has developed orders to ensure the compliance of its facilities and programs with the applicable Federal environmental regulations. (see Table 2.7-2) DOE Order 5440.1C (DOE85) establishes procedures for implementing the requirements of the National Environmental Policy Act of 1970 (NEPA70). New facilities and modifications to existing facilities are subject to extensive design criteria reviews and require the preparation of environmental impact statements. In existing facilities, the DOE has implemented a systematic program for reducing the releases of gaseous and liquid radionuclides to the environment.

In addition, the DOE is subject to the Resource Conservation and Recovery Act (RCRA), which requires that all radioactive wastes containing RCRA-hazardous materials are subject to regulations under both the RCRA and the Atomic Energy Act of 1974 (AEA54). The DOE is also preparing an order for demonstrating compliance with the requirements of the Comprehensive Environmental Response, Compensation, and Liability Act.

2.7.3

Disposal of Spent Nuclear Fuel and High-Level Wastes

The DOE has promulgated, in 10 CFR Part 960 (DOE84b), general guidelines for the recommendation of sites for geologic repositories. The guidelines are compatible with the EPA standards in 40 CFR Part 191 issued in 1982 (EPA82) and the NRC technical criteria in 10 CFR Part 60 (NRC81). They identify the detailed geologic considerations that qualify or disqualify a site for repository development and specify how these considerations are to be implemented. The guidelines were used in evaluating the nine sites that had been identified as potentially acceptable for the first repository, in nominating five of these sites as suitable for characterization, in recommending three sites for characterization, and in an early evaluation of the suitability of the Yucca Mountain site for spent fuel and high-level waste disposal. In addition, the guidelines are followed in a hierarchy developed for the site characterization plan for the candidate repository site (DOE88b).

The DOE has developed its waste-management program, including site characterization, facility development, and facility operation, in compliance with the applicable Federal regulations. The DOE's general plans for compliance with environmental regulations during site characterization at the Yucca Mountain site in Nevada are presented in the Environmental Regulatory Compliance Plan (ERCP) (DOE88d). The ERCP describes a comprehensive program of action to ensure compliance with applicable environmental laws and regulations; it describes site-characterization activities that may trigger environmental regulatory requirements, identifies pertinent environmental laws and regulations, describes the processes used by the DOE for compliance with environmental regulations, and specifies the environmental permits and approvals that must be obtained.

Although the DOE expects no significant adverse environmental impacts to result from site characterization (DOE86c), the DOE plans to monitor site-characterization activities that have the potential for causing adverse environmental impacts and, to the extent practicable, will implement mitigation measures as is appropriate. General plans for monitoring and mitigation are presented in the Environmental Monitoring and Mitigation Plan (EMMP) (DOE88e).

Various environmental studies will be carried out at the Yucca Mountain site to collect the data needed for the EMMP, to obtain various permits, and to gather data that may be needed for the repository environmental impact statement (EIS). Plans and methods for the specific environmental studies to be undertaken will be presented in environmental field activity plans. Plans for EIS data collection will be made after the content of the EIS has been identified and an EIS implementation plan has been issued.

Table 2.7-1. DOE Orders that apply to environmental protection during site characterization.

<u>Number</u>	<u>Date</u>	<u>DOE Orders</u>	<u>Subject</u>
1. DOE 5000.3	11-07-84	Unusual Occurrence Reporting System.	(Safety and Environment)
2. DOE 5400.2	08-13-87	Environmental Compliance Issue Coordination.	(Environment)
3. DOE 5440.1C	04-09-85	Implementation of National Environmental Policy Act.	(Environment)
4. DOE 5480.1B	09-23-86	Environmental Protection, Safety, and Health Protection Program for DOE Operations (Certain Chapters of 5480.1 and 5480.1A are still in effect).	(Both)
5. DOE 5480.3	07-09-85	Safety Requirements for the Packaging and Transportation of Hazardous Materials, Hazardous Substances and Hazardous Wastes.	(Safety)
6. DOE 5480.4	05-15-84	Environmental Protection, Safety and Health Protection Standards.	(Safety)
7. DOE 5480.11	12-21-88	Radiation Protection for Occupational Workers.	(Both)
8. DOE 5481.1B	09-23-86	Safety Analyses and Review System.	(Safety)

Table 2.7-1. Continued.

<u>Number</u>	<u>Date</u>	<u>DOE Orders</u>	<u>Subject</u>
9. DOE 5482.1	09-23-86	Environmental, Safety, and Health Appraisal Program.	(Both)
10. DOE 5484.1	02-24-81	Environmental Protection, Safety, and Health Protection Information.	(Both)
11. DOE 5500.1A	02-26-87	Department of Energy Emergency Management System.	(Both)
12. DOE 5820.2A	09-26-88	Radioactive Waste Management.	(Both)
13. DOE 5480.17	10-05-88	Site Safety Representatives.	(Safety)

Note: Environmental protection orders are undergoing revisions and are scheduled to be issued in late 1989. The following list represents proposed revisions to the environmental protection section of the DOE directives system. The proposed list is as follows:

- 5400.1 Environmental Protection Program Requirements.
- 5400.3 Radiation Protection of the Public and Environment.
- 5400.yy Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) Requirements.
- 5400.zz Hazardous and Radioactive Mixed Waste Requirements.
- 5400.xy Radiological Effluent Monitoring and Environmental Surveillance.

Source U.S. DOE Order 5480.11 (DOE88c).

Table 2.7-2. Environmental statutes applicable during DOE's repository site characterization activities.

- 1. National Environmental Policy Act**
- 2. Federal Land Policy and Management Act**
- 3. Farmland Protection Policy Act**
- 4. Protection of Floodplains and Wetlands, Executive Orders 11988 and 11990**
- 5. Clean Air Act**
- 6. Federal Water Pollution Control Act, a.k.a., the "Clean Water Act"**
- 7. Safe Drinking Water Act**
- 8. Hazardous Materials Transportation Act**
- 9. Resource Conservation and Recovery Act**
- 10. Sikes Act**
- 11. Comprehensive Environmental Response, Compensation, and Liability Act (and EO 12286, 12288, and 12580)**
- 12. Endangered Species Act**

Sources: U.S. DOE Orders 5400.2A and 5400.1 (DOE87, DOE88a).

2.8 DEPARTMENT OF TRANSPORTATION

The U.S. Department of Transportation (DOT) has statutory responsibility for regulating shipments of radioactive materials, including radioactive wastes. DOT its regulatory activities are coordinated with those of the NRC. Its authority includes the responsibility to protect the public from exposure to radioactive materials while they are in transit. The DOT has implemented its authority by specifying performance standards for shipping containers, setting maximum exposure rates for any package containing radioactive materials, and managing the routing of radioactive materials shipments to avoid densely populated areas.

The regulatory authority of the DOT derives from several laws. For the transportation of radioactive waste, the primary laws are the Hazardous Materials Transportation Act of 1974 (HMTA75) and the Federal Railroad Safety Act (FRSA70). These laws authorize the Secretary of Transportation to issue regulations for the safe transportation of hazardous materials, including radioactive materials, and to define the specific relationship between the DOT and state and local authorities.

The regulations promulgated by the DOT are contained in Title 49 of the Code of Federal Regulations. Those directly applicable to the transportation of radioactive wastes are mainly included in 49 CFR Parts 171-177 (DOT83). They define the types of materials that are regulated; specify the DOT's enforcement authority, including potential sanctions; and state specific requirements for materials handling, the marking and labeling of packages, the placarding of shipments, the routing of shipments, and the training of drivers.

Specific provisions cover carriage by rail (49 CFR Part 174), carriage by vessel (49 CFR Part 176), and carriage by public highway (49 CFR Part 177). Transportation by barge is regulated by the standards promulgated under 49 CFR Chapter 2. In addition to the regulations established under the HMTA, the DOT's Federal Highway Administration has established, in 49 CFR Part 300, general standards for highway transportation.

2.9 OFFICE OF THE NUCLEAR WASTE NEGOTIATOR

The Office of the Nuclear Waste Negotiator, created by the 1987 Amendments of the 1982 Nuclear Waste Policy Act, is an independent Federal entity. The Nuclear Waste Negotiator is appointed by the President. The mission of the Negotiator is to seek a dialogue with the Governor of every State and the leaders of all federally recognized Indian tribes to explore upon what terms and conditions, if any, they might willingly host a facility for the permanent or temporary storage of nuclear waste.

The Negotiator is authorized to negotiate with the Governor or tribal leader of the interested potential host jurisdictions to determine the terms and conditions under which they would agree to host either a Monitored Retrievable Storage facility or a repository. Preparation of an environmental assessment and consultation with Federal agencies concerning a site's technical suitability are required when a negotiation begins. The negotiation is to result in a written agreement that will be submitted to Congress and enacted into law before it becomes effective.

2.10 STATE AGENCIES

States have played an important role in protecting the public from hazards associated with ionizing radiation. Twenty-nine States have assumed the NRC's inspection, enforcement, and licensing responsibilities for users of nuclear source and by-product materials and users of small quantities of special nuclear material. These "NRC Agreement States," are bound by formal agreements to adopt requirements consistent with those imposed by the NRC.

2.10.1 Federal Provisions for State Participation

State and public participation in the planning and development of geologic disposal is essential to promote public confidence in the safety of geologic repositories for spent nuclear fuel and high-level radioactive wastes. The Congress has provided for public participation in the NWPA and in the Nuclear Waste Policy Act Amendments Act of 1987 (Amendments Act) (NWPA87). Specific provisions of the NWPA, as amended, govern the notification of potentially affected States and Indian Tribes (Section 116(a)). Other provisions require the Secretary of Energy to hold hearings in the vicinity of the repository before selection takes place (Section 114(a)(10)).

The Secretary of Energy is also required to provide to the Governor and legislature of any affected State, and to the governing body of any affected Indian Tribe, timely and complete information regarding determinations of plans made with respect to site characterization, development, design, licensing, construction, operation, regulation, or decommissioning of a repository (Section 117(a)). The Amendments Act allows a potential host State or Indian Tribe to submit a notice of disapproval to Congress upon notification by the Secretary (Section 116(b)).

Under Section 175 of the Amendments Act, the DOE reported to Congress in December 1988 on the potential impacts of locating the repository at the Yucca Mountain site (DOE88f 1). The report included recommendations for mitigation of the impacts and the responsibility of each government authority (i.e., Federal, state, or combined) for such mitigation.

The NWPA, as amended, also requires the Secretary to seek to enter into binding consultation-and-cooperation agreements with an affected state. If a consultation and cooperation agreement is not signed within six months, the Secretary is required to report to the Congress, in writing, within 30 days on the status of negotiations. The report must also be transmitted to the affected state for review and comment (Section 117(c)).

The NWPA authorizes the Secretary to site, construct, and operate one facility for monitored retrievable storage (MRS). One of the main provisions of the amended Act is Section 141(h), which defines multiple opportunities for states and Indian Tribes to actively participate in the selection of the MRS site. The Act gives a potential host state or Indian Tribe the opportunity to disapprove the selection of an MRS site (Section 146(a)) and provides for the option of a benefit agreement with the state or Indian Tribe in which the site is located (Section 147). An MRS facility would be licensed by the Nuclear Regulatory

Commission under 10 CFR Part 72; this regulation presently does not require compliance with the EPA standards of 40 CFR Part 191 for disposal facilities and would require modification when 40 CFR Part 191 is finalized.

2.10.2 Participation Activities

With the passage of the Amendments Act, the site characterization program now focuses on Nevada. The DOE is continuing to work with representatives of the State of Nevada and affected local governments to identify the types of information exchanges that are desirable and most effective, to identify the program information that is of interest to them, and to establish effective mechanisms for the provision of such information (e.g., briefings, formal or informal meetings and hearings, technical progress reports, project update meetings). Equally important are the mechanisms for receiving and addressing technical comments from the state and local governments on program activities.

2.10.3 State of Nevada Programs

Nevada has been an NRC Agreement State since 1972, when enabling legislation was passed by the State legislature to allow the State to enter into an agreement with the Nuclear Regulatory Commission (NRS72). This law designates the Health Division of the Department of Human Resources as the radiation-control agency for the State, as well as having responsibility for responding to radiological emergencies within the State. Within the Health Division, these responsibilities are vested with the Radiological Health Section (RHS) of the Bureau of Regulatory Health Services, which licenses low-level waste sites and the possession and use of x-ray sources and radioactive materials. Under current law and regulations, the only responsibility the RHS has for high-level wastes is for emergency response in case of an accident on highways or other State property (MAR89). The transportation of radioactive materials and wastes in Nevada is currently regulated by the Public Service Commission. The State regulations are essentially identical to the U.S. Department of Transportation's regulations on the transport of radioactive materials under 49 CFR Part 173.

With the passage of the NWPA, the State of Nevada established the Agency for Nuclear Projects/Nuclear Waste Project Office (NWPO) in order to meet its responsibilities to its citizens, and to fulfill the requirements of the NWPA. The NWPO was originally established by the legislature in July 1983 and was reorganized as a separate State Agency in 1985 (NRS85). The NWPO's basic responsibilities are to (1) advise the Governor and Legislature on matters concerning the disposal of spent fuel and high-level radioactive wastes in the State, (2) develop and administer coordinated programs of planning and evaluation, (3) work closely with, consult with, and assist local government entities in communication with the Department of Energy, and (4) perform the duties and responsibilities of the State of Nevada as described in the NWPA. The NWPO has issued, through these contractors and its staff, over 30 reports on a variety of subjects pertinent to the proposed disposal facility at the Yucca Mountain site. The reports are generally oriented to pointing out the State's specific and unique needs and flagging any shortcomings, from the State's perspective, of the DOE's plans and of the overall HLW disposal program. These reports fall into three broad categories:

- Reviews, comments, and critiques of DOE program documents (e.g., the environmental assessment and the site characterization plan) and the implications of these programs for the State.
- Technical issues regarding the site itself, primarily earth science, and environmental issues.
- Inventory, current status, and project studies regarding socio-economic and transportation issues. These include studies of demographics; business, tourism, taxation, and revenue issues; risk issues; public perception and opinion; transportation routes and needs assessments; and evaluations of past accidents involving nuclear materials.

The 1985 legislation creating the NWPO also established the Commission on Nuclear Projects. The Commission contains seven members appointed by the Governor, with recommendations from the Legislature (NCN86). The Commission reports to the Governor and the Legislature on matters relevant to the repository program. The same legislation also created the Legislature's Committee on High-Level Radioactive Wastes whose seven members are drawn from the Senate and Assembly. Its purpose is to study and evaluate (1) information and policies regarding the location of a repository in Nevada, (2) any potentially adverse effects from the repository in Nevada, (3) any other policies relating to the disposal of spent fuel and high-level wastes, and (4) to recommend appropriate legislation to the legislature (NRS85).

2.10.4 Programs in the State of New Mexico

The New Mexico Environmental Evaluation Group (EEG) was created in 1978 to conduct independent scientific reviews and to evaluate the potential impact on public health and environment from the Waste Isolation Pilot Plant (WIPP) project (EEG88, EEG89, NEI89). The WIPP facility is a repository designed to demonstrate the disposal of national defense-related TRU wastes. The EEG was formed in response to the authorizing legislation for the Waste Isolation Pilot Plant, since Congress specifically excluded DOE from the licensing requirements of the Nuclear Regulatory Commission for the WIPP facility.

The EEG is a full-time, multi-disciplinary group funded entirely by the U.S. Department of Energy for the State of New Mexico. The EEG is the only independent oversight group monitoring the WIPP site and its activities; however, it does not have any regulatory authority on the WIPP facility and it can only recommend actions to DOE for its consideration. In spite of these constraints, the EEG has been influential in making recommendations which led to the relocation of the repository, redesign of the waste shipping containers, consideration and evaluation of transportation issues, and monitoring of WIPP site activities. The EEG has also organized several technical forums to evaluate technical issues and consider alternate approaches.

In October 1988, the EEG was assigned to the New Mexico Institute of Mining and Technology to provide a better climate for technical independence. Up to that point, the EEG had been attached to the Environmental Improvement Division, a component of the

New Mexico Health and Environment Department. The EEG Director is appointed by and reports directly to the President of the Institute and the Director appoints all other EEG staff. Scientific disciplines represented in the EEG group include engineering, geology, hydrology, health physics, environmental monitoring, radiation protection, radiological health, and quality assurance. The EEG has offices both in Carlsbad and Albuquerque, NM.

Since 1978, the EEG staff have conducted several evaluations to assess the suitability of the WIPP site, including identifying potential environmental problems, suitability of facility design, suitability of the proposed waste shipping containers, waste form characterizations and other related technical topics. EEG responsibilities also include the conduct of an environmental radiation surveillance program to establish a background base line for naturally occurring radioactivity present in air, water, and soils for both on and off-site locations and within the surrounding communities. Both EEG and DOE have independent monitoring stations located in the exhaust stacks of the WIPP facility to characterize and document airborne emissions.

EEG disseminates its findings and analyses by publishing reports, articles in professional journals, presentations to scientific society meetings, public hearings, and by issuing pamphlets and brochures to the public. The EEG has published 40 major reports since 1978. It also distributes the results of its analyses to DOE, the Governor's Office, the New Mexico Legislature, Congress, the scientific community, and general public. Typically, EEG reports have addressed the following technical issues: site characterization; performance assessment; facility operations; monitoring; and transportation. Several of these reports present independent evaluations and analyses of DOE studies, models, assumptions, and plans.

2.11 Indian Tribes

Indian Tribes have a unique sovereign status in U.S. law, and this status was recognized by the NWPA and the Amendments Act. This government-to-government relationship between the Federal government and Indian Tribes obligates the DOE to interact directly and specifically with Indian Tribes in areas where repository or MRS siting activities will occur. The NWPA, as amended, under Section 2(2), defines:

"... affected tribe as (1) any Indian Tribe within whose reservation boundaries an MRS, test and evaluation facility, or a repository for high-level wastes or spent fuel is proposed to be located, or (2) whose federally defined possessory or usage rights to other lands outside of the reservation's boundaries arising out of congressionally ratified treaties may be substantially and adversely affected by the locating of such a facility. Provided, that the Secretary of the Interior finds, upon the petition of the appropriate governmental officials of the tribe, that such effects are both substantial and adverse to the tribe..."

As noted above, many of the sections of the NWPA, as amended, that delineate the participation activities and rights of affected States in repository and MRS siting decisions also apply to affected Indian Tribes. The means to disapprove of the site selection and designation process is given in Section 118(a). An affected Indian Tribe is also eligible to

receive the same grants, financial and technical assistance, and payments equal to taxes for which a State is eligible under Section 116(c).

The Yucca Mountain site is not located on nor adjacent to an Indian reservation. However, three American Indian reservations are located in rural parts of the bi-county area. The Moapa Paiute Reservation, which is located in northeastern Clark County, petitioned the Secretary of the Interior for affected status in 1984, but the petition was denied because the Moapa Band was not found to have any claims or usage rights arising out of a Congressionally ratified treaty (DOI84). Two bands of the Shoshone (Yomba and Duckwater) have small reservations in northern Nye County that are over 200 miles from the site (DOE86c, MIL89). Another Indian Tribe, the Western Shoshone, considers the lands around the Yucca Mountain site to be of ancestral and religious significance.

Since the passage of the Amendments Act, no Indian Tribes have been designated as affected tribes. However, the DOE is cooperating with Indian Tribes that may be located near the transportation routes or the WIPP facility. The DOE informs Indian Tribes of the status of the program through a cooperative agreement with the National Congress of American Indians. Finally, to ensure compliance with the American Indian Religious Freedom Act, the National Historic Preservation Act and related statutes, and the National Environmental Policy Act, the DOE will consult with Indian Tribes that have current or traditional religious or cultural ties to the Yucca Mountain site (DOE88g).

Chapter 2 References

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- EEG89** Neill, R. H., Ph.D., Observations on the WIPP Project and Radioactive Waste Disposal in General, Presentation to the Los Alamos National Laboratory Colloquium, Environmental Evaluation Group, Albuquerque, New Mexico, March 7, 1989.
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Chapter 3: QUANTITIES, SOURCES, AND CHARACTERISTICS OF SPENT NUCLEAR FUEL AND HIGH-LEVEL AND TRANSURANIC WASTES**3.1 INTRODUCTION**

This chapter presents current inventories of commercial spent fuels, commercial and U.S. Department of Energy (DOE) high-level radioactive wastes, and DOE transuranic wastes. These inventories were compiled from the most reliable Federal government information sources publicly available (BUR82, DOE88a, DOE88b, EIA88, JAN83, LIT79, STO79). Estimates of generated wastes and spent fuel to the year 2020, based on the latest DOE information and projected U.S. commercial nuclear power growth, are also presented. The spent fuel and wastes are characterized according to their volumes (or quantities) and their nuclear, physical, and chemical properties.

The wastes are broadly characterized as high-level waste (HLW) and transuranic (TRU) waste. In addition, an inventory of commercial reactor spent fuel may also require an expansion of current storage or the construction of additional facilities for interim storage, pending the availability of commercial reprocessing facilities, permanent disposal facilities, or monitored retrievable storage.

Both spent fuel and high-level radioactive wastes from reprocessing are intensely radioactive and generate substantial quantities of heat. The radioactivity and heat production continue for long periods of time because the wastes contain a number of long-lived radionuclides. The transuranic elements in particular have long radiological half-lives, generate very little heat relative to spent fuel, and present a potential health hazard for tens of thousands of years. Transuranic elements are nuclides with an atomic number greater than 92 and include plutonium, curium, americium, and neptunium.

3.2 SPENT NUCLEAR FUEL

In this standard, spent nuclear fuel is defined as fuel that has been withdrawn from a nuclear reactor following irradiation and whose constituent elements have not been separated by reprocessing (EPA85). The generators of spent nuclear fuel are: 1) commercial light-water reactors (LWRs), 2) government sponsored research and demonstration programs, universities, and industry, 3) experimental reactors, i.e., liquid metal fast breeder reactor (LMFBR) and high temperature gas cooled reactors (HTGR), 4) U.S. Government nuclear weapons production reactors, and 5) Department of Defense (DOD) reactors.

Approximately 96 percent of the spent fuel from commercial power reactors is stored in pools at reactor sites. The rest is stored at the West Valley Demonstration Project (WVDP) in New York, and at the Midwest Fuel Recovery Plant (MFRP) at Morris, Illinois. The WVDP facility is currently being decommissioned. All utility-owned spent fuel assemblies previously stored there have been returned to the utilities, and the fuel remaining is DOE-owned material. Spent fuels from one-of-a-kind reactors are currently stored at Hanford (HANF) and the Idaho National Engineering Laboratory (INEL). Spent fuel from the Fort St. Vrain HTGR is stored at the Idaho Chemical Processing Plant (ICPP) at INEL.

Other types of special spent fuel are stored at the Savannah River Plant (SRP) and INEL. These fuels are government-owned and are not scheduled for reprocessing in support of DOE/defense activities.

The fuel currently used in commercial light-water reactors consists of a mixture of uranium-238 and uranium-235 dioxides encased in zirconium alloy (zircaloy) or stainless steel tubes. During reactor operation, fission of the uranium-235 produces energy, neutrons, and radioactive materials. The neutrons produce further fission reactions and thus sustain the chain reaction. The neutrons also convert some of the uranium-238 into plutonium-239, which can fission as uranium-235 does. In time, the fissile uranium-235, which originally constituted some 3 to 4 percent of the enriched fuel, is depleted to such a low level that power production becomes inefficient. Once this occurs, the fuel bundles are deemed "spent" and are removed from the reactor. Typical removal rate is one-third of the fuel, or 30 metric tons per year and per reactor. Reprocessing of commercial spent fuel has been proposed to recover the unfissioned uranium-235 and the plutonium for reuse as a fuel resource, but such reprocessing is not currently taking place.

The radioactive materials associated with spent fuel fall into three categories - a) fission products, b) actinide elements, and c) activation products. Typically, fresh spent fuel contains more than 100 radionuclides as fission products. Fission products are of particular importance, because of the quantities produced, their radiological half-lives, their heat production, and their potential biological hazard. Such fission products include: strontium-90; technetium-99; iodine-129 and -131; the cesium isotopes, such as cesium-134, -135, and -137; tin-126; and krypton-85 and other noble gases.

The activation products include tritium (hydrogen-3), carbon-14, and other radioactive isotopes created by neutron activation of fuel assembly materials and impurities in cooling water or in the spent fuel. The actinides consist of uranium isotopes and transuranic elements (i.e., isotopes with an atomic number greater than 92, including plutonium, curium, americium, and neptunium formed by neutron capture, and their decay products). The exact composition of radionuclides in any given spent fuel sample depends on the reactor type, the initial fuel composition, the length of time the fuel was irradiated, and the elapsed time since its removal from the reactor core.

3.2.1 Spent Fuel Inventory and Projection

By the end of 1988, there were 17,607 metric tons (MT) of spent fuel in inventory from commercial reactor operation (DOE89a). Of this amount, 27 MT are stored at the WVDP facility and 668 MT are stored at the MFRP. The remainder is stored at each reactor site. The historical and projected quantities of the spent fuel inventory and accumulated radioactivity are given in Table 3.2-1.

The radioactivity in spent fuel depends primarily on its age. As the spent fuel ages, many of the short-lived fission products decay away. Calculations of waste activities 10 years after removal from the reactor, with consideration being given only to radionuclides (fission products and heavy elements) with half-lives greater than 20 years, show that the 1988 activity of the 17,607 MT of spent fuel corresponds to about 18.6 billion curies.

The projected inventory of spent fuel (Table 3.2-1) was based on DOE's lower reference case projections for installed nuclear capacities given in Table 3.2-2 and for the burnup rate and duration assumed for those reactors. DOE's projection assumes 15 reactors now in the construction pipeline will become operational at the year 2005 (EIA88). This is in addition to the 107 reactors already operating by the end of 1988. The DOE also assumes that two reactors currently on order will eventually be built and become commercially operable by 2005. The DOE's inventory projections assume the startup of a MRS Facility and a Commercial Repository in the year 2003.

It is estimated that by the year 2020 the total nuclear electrical capacity will reach 122.7 gigaWatts (DOE89a, EIA89). The position that new reactor orders will resume assumes that future changes, driven by political, environmental, and economic issues as well as the decreasing availability of oil, will present nuclear power as a better alternative. For example, it is assumed that clean air standards will become stricter principally in response to the acid rain issue and the uncertainty about the greenhouse effect associated with the build up of atmospheric CO₂ resulting from the combustion of fossil fuels. These factors could enhance the choice of nuclear over fossil-fueled (coal and oil) power plants.

These projections do not include potential contributions from spent naval propulsion reactor fuel. Although the current plans do not include such a possibility, modifications to the nation's strategy for nuclear weapons may result in the availability of fuel-grade material without reprocessing. In this case, disposal of spent propulsion reactor fuel may be considered for the repository.

3.3 HIGH-LEVEL RADIOACTIVE WASTES

The EPA standards (40 CFR Part 191) define high-level radioactive wastes as the highly radioactive materials resulting from the reprocessing of spent nuclear fuel, including liquid wastes produced directly in reprocessing, and any solid material derived from such liquid wastes (EPA85). This definition is the same as that given in the NWPA (NWPA83). NRC regulations require that commercial high-level radioactive wastes generated in the future be converted to a solid form within 5 years (NRC88).

The fission products, actinides, and neutron-activated products of particular importance are the same for HLW as those listed for the spent fuel assemblies (DOE89a, DOE88b, LIT79, STO79).

Weapons program reactors are operated mainly to produce plutonium. Reprocessing to recover the plutonium is an integral part of the weapons program operations. Naval propulsion reactor fuel elements may also be reprocessed to recover the highly enriched uranium that still remains after use.

High-level radioactive waste that is generated by the reprocessing of spent reactor fuel and targets would contain more than 99 percent of the non-volatile fission products produced in the fuel or targets during reactor operation. It generally would contain about 0.5 percent of the uranium and plutonium originally present in the fuel. Most of the current HLW inventory, which is the result of DOE national defense activities, is stored at the Savannah

River Plant (SRP), the ICPP at the Idaho National Engineering Laboratory, and the Hanford sites. A small amount of commercial HLW was generated at the Nuclear Fuel Services Plant at West Valley, New York, from 1966 to 1972. That facility is now referred to as the West Valley Demonstration Project (WVDP) and is under the responsibility of the DOE Idaho Operations, West Valley Project Office. These wastes have been through one or more treatment steps (i.e., neutralization, precipitation, decantation, evaporation, etc.). Their total volumes depend greatly on the steps to which they have been subjected during the various processing stages. Such wastes must be incorporated into a stable solid medium (e.g., glass) for final disposal, and the volumes of these interim wastes will be greatly reduced once this has been accomplished.

The DOE defense HLW at INEL results from reprocessing nuclear fuels from naval propulsion reactors and special research and test reactors. The bulk of this waste, which is acidic, has been converted to a stable, granular solid (calcine). At SRP and HANF, the acidic liquid wastes from reprocessing defense reactor fuel is or has been made alkaline by the addition of caustic soda and stored in tanks. During storage, these alkaline wastes separate into three phases: liquid, sludge, and salt cake. The relative proportions of liquid and salt cake depend on how much water is removed by waste treatment evaporators during waste management operations. The condensed water is currently sent to seepage basins and holding ponds.

The commercial HLW at West Valley consists of both alkaline and acidic wastes. The alkaline wastes were generated by reprocessing commercial power reactor fuels and some Hanford N-Reactor fuels, whereas acidic wastes were generated by reprocessing a small amount of commercial fuel containing thorium.

The inventories of HLW in storage at the end of 1987 are listed in Table 3.3-1 (by volume) and Table 3.3-2 (by radioactivity). Projected volume and radioactivity data for DOE defense, West Valley, and future commercial HLW are given in Table 3.3-3.

3.3.1 HLW Inventories at SRP

Approximately 128,000 m³ of alkaline HLW that has accumulated at the SRP over the past three decades is currently stored underground in high-integrity, double-walled, carbon-steel tanks. The current inventories (Tables 3.3-1 and 3.3-2) consist of alkaline liquid, sludge, and salt cake that were generated primarily by the reprocessing of nuclear fuels and targets from plutonium production reactors. As generated, most of the waste is acid. The sludge is formed after treatment with caustic agents. Salt cake results when the supernatant liquor is concentrated in waste treatment evaporators.

3.3.2 HLW Inventories at INEL

About 11,000 m³ of HLW is currently stored at the Idaho Chemical Processing Plant (ICPP) at INEL; this volume consists of 7,600 m³ of liquid wastes and 3,400 m³ of calcine materials (Tables 3.3-1 and 3.3-2). Liquid HLW is generated at ICPP primarily by the reprocessing of spent fuel from the national defense (naval propulsion nuclear reactors) and reactor testing programs; a small amount is also generated by reprocessing fuel from non-

defense research reactors. This acidic waste is stored in large, doubly contained, underground, stainless steel tanks. The waste is then converted to a calcine, after which it is stored in retrievable stainless steel bins housed in reinforced concrete vaults.

3.3.3 HLW Inventories at HANF

The alkaline HLW (243,500 m³) located at HANF is stored in four phases: liquid, sludge, slurry, and salt cake. This waste, which has been accumulating since 1944, was generated by reprocessing production reactor fuel for the recovery of plutonium, uranium, and neptunium for defense and other Federal programs. Fuel reprocessing was suspended from 1972 until November 1983. Most of the high-heat-emitting isotopes (Sr-90 and Cs-137, and their decay products) have been removed from the old wastes, converted to solids as strontium fluoride and cesium chloride, placed in double-walled capsules, and stored in water basins. The liquid, sludge, slurry, and salt cake wastes (Tables 3.3-1 and 3.3-2) are stored in underground concrete tanks with carbon steel liners.

3.3.4 HLW Inventories at WVDP

About 2,116 m³ of HLW is stored at the WVDP Facility and consists of 2,066 m³ of alkaline wastes and only 50 m³ of acid wastes. The alkaline wastes were generated by reprocessing commercial and a few Hanford N-Reactor spent fuel elements. Initially, all of the wastes were highly acid; treatment with excess sodium hydroxide led to the formation of an alkaline sludge. The acid wastes now in storage were generated by reprocessing a small batch of thorium-uranium fuel from the Indian Point-1 Reactor. The alkaline wastes are stored in an underground carbon-steel tank, and the acid wastes are stored in an underground stainless steel tank. Reprocessing at the WVDP plant was discontinued in 1972, and no additional HLW has been generated since. The current inventories of HLW at WVDP are presented in Tables 3.3-1 and 3.3-2.

3.3.5 Waste Characterization

It is difficult to characterize HLW generically at any site because such wastes have been generated by several different processes and several methods have been used to condition the wastes for storage (e.g., evaporation and precipitation). In some instances, several different wastes have been blended. Nonetheless, representative chemical and radionuclide compositions of the HLW at SRP, ICPP, HANF, and WVDP can be found in other sources (DOE88a, DOE88b).

As with spent fuel, HLW radioactivity levels depend on age. To bring the level of radioactivity into perspective, the activity of fission products and heavy element radionuclides with half-lives exceeding 20 years in existing HLW is estimated to be about 700 million curies.

3.3.6 HLW Projections

Projections for HLW (volume and radioactivity) by source are presented in Table 3.3-3. The projections for SRP are based on the assumption that three reactors will be operating

through the year 2000. After the year 2000, these three reactors are expected to be replaced by a single new production reactor, and the Defense Waste Processing Facility (DWPF) is expected to begin to produce wastes in a glass form by late 1990. The HLW glass will be stored on-site until a national HLW repository becomes available. Current plans call for the DWPF to produce approximately 5,700 canisters of glass between 1990 and the end of 2020.

The ICPP projections are premised on predicted fuel deliveries and estimates of fuel reprocessing and waste management operations. The HANF projections assume that the fuel reprocessing plant will operate through 1993. A Hanford Waste Vitrification Plant (HWVP) may begin operation in 1999.

The projections of HLW for Hanford do not include vitrification, since material balances for such processes are not yet available. At the WVDP, vitrification of the HLW is scheduled to begin in 1994 and to be completed in 1995.

3.4 TRANSURANIC WASTES

The EPA standards (40 CFR Part 191) define transuranic wastes as those wastes containing more than 100 nanocuries per gram of alpha-emitting transuranic isotopes, with half-lives greater than 20 years (EPA85).

Alpha-emitting transuranic nuclides present a hazard because of their long radiological half-lives and high chemical toxicity. Most of the radionuclides that are contained in TRU wastes have very long half-lives and are typically present at low concentrations (DOE88a, LIT79, DOE88b, JAN83, BUR82, BRY81). Although a few decay products have energetic gamma emissions, their most significant hazard is due to alpha radiation emissions. Most TRU wastes can be handled with just the shielding that is provided by the waste package itself. These wastes are classified as "contact-handled" TRU wastes. A smaller volume may be contaminated with sufficient beta, gamma, or neutron activity to require remote handling. Heat generation in stored TRU waste is not a factor affecting how closely packages can be stored; however, avoiding the assembly of a critical mass as a result of densely-stored material must always be considered.

Relative to other radioactive wastes, TRU wastes represent a group of liquid and solid materials with widely varying chemical and physical properties. These wastes are categorized as contact-handled (CH), i.e., having a surface dose rate of less than 200 milliRoentgen per hour (mR/h); or remote-handled (RH), i.e., having a surface dose rate of greater than 200 mR/h.

Most TRU wastes are generated in DOE defense-related activities at the Rocky Flats Plant (RFP), Hanford facilities, and the Los Alamos National Laboratory (LANL). Nearly one-half of all TRU waste comes from weapons components manufactured at RFP and subsequent plutonium recovery at all three sites. Smaller amounts are generated at the Oak Ridge National Laboratory (ORNL), SRP, INEL, Argonne National Laboratory (ANL), Mound Facility, Bettis Atomic Power Laboratory, Lawrence Livermore National Laboratory, and Battelle-Columbus Laboratory. It should be noted that TRU wastes originating from the Mound Facility, Bettis and Argonne Laboratory, and from the Rocky Flats Plant are shipped

to INEL for interim storage. The second largest source of TRU wastes is decontamination and decommissioning projects which account for one-fourth of the total. About one-fifth of TRU wastes come from laboratory activities, which can produce exotic TRU isotopes.

The amounts of TRU wastes from fuel cycle activities are in fact quite small because of the current moratorium on reprocessing and plutonium recycle. The Nuclear Fuel Services' reprocessing of nuclear fuel at West Valley, New York, produced some TRU waste that was disposed at that site. A small amount of TRU waste is also being generated in industrial and government-sponsored fuel fabrication and research.

3.4.1 Inventories and Characterization

Before March 1970, TRU wastes were disposed by shallow-land burial at AEC (now DOE) and commercial sites in pits and trenches and covered with soil. Beginning in 1970, the AEC initiated a policy of retrievable storage for TRU wastes since it concluded that such wastes must be disposed using methods which provide greater confinement. Consequently, since 1970 TRU wastes have been stored in facilities for easy retrieval. Storage facilities have been built to suit the needs of each DOE site selecting methods which considered local climate, waste forms, existing volumes, and future generation rates. In addition, a program was established to characterize all previously disposed wastes and to identify long-term waste management options since early burial practices were not governed by current requirements. Such wastes, as well as newly generated wastes, would eventually be disposed at a dedicated transuranic waste disposal site such as is being considered at the Waste Isolation Pilot Plant, located in New Mexico. (DOE89).

The estimated buried volume mass of contained TRU elements and their associated alpha activities for each DOE site are given in Table 3.4-1. Storage facilities and waste disposal containers are designed for a 20-year lifetime, during which time, the necessary measures will be taken regarding the identification of permanent disposal options. According to the DOE, all of the stored retrievable wastes are located at the DOE sites listed in Table 3.4-2. Also given in this table are waste volumes, the mass of TRU elements, and the radioactivity as of December 31, 1988. Estimates of the radioactivity of this waste are based upon emplacement records and a knowledge of the types of operations at each disposal site or for each waste generator.

Over the years, some of the buried waste containers have been breached, and the surrounding soil has been contaminated. Accurately determining the volume of contaminated soil is a difficult task, and the estimated amounts cover a rather broad range (Table 3.4-3). Also, in the early days at HANF, ORNL, and LANL, some liquid wastes containing TRU elements were spilled or drained into the ground. Further characterization studies are needed to provide a better estimate of the total volume of soil that is contaminated with TRU elements.

From ongoing characterization studies, several DOE sites have estimated that their buried and retrievable TRU solid wastes are composed primarily of the physical species given in Table 3.4-4. Most of the storage sites have relatively large fractions of combustible material and contaminated metal.

Estimated isotopic compositions for buried and retrievable wastes at the several DOE sites where TRU wastes are emplaced are given in Table 3.4-5. These estimates reflect information of DOE site operations and commercial TRU waste sources to characterize waste compositions when documented data are not available. Separate data for contact-handled and remote-handled waste were available for all sites that store both types of such wastes; however, composition data were not available for buried TRU waste at ORNL and portions of the waste buried at SRP. The radioactivity of the wastes buried at ORNL was assumed to be the same as that of the contact-handled waste. These data represent the best site estimates of the isotopic compositions of existing TRU wastes at government sites. The mix categories represent variations on major waste stream composition based on the total volume in storage plus the estimated waste volume generation through the year 2013 for each of the listed DOE sites.

3.4.2 TRU Waste Projections

TRU waste inventories and projected accumulations at government sites, of contact and remote-handled wastes from DOE defense activities, are listed in 5-year increments in Table 3.4-6. Projections are given, starting in 1988, for buried and stored wastes up to the year 2013. By 1990, when the Waste Isolation Pilot Plant, located in New Mexico, was expected to start receiving TRU wastes (DOE89), about 190,000 m³ of such wastes were expected to have accumulated at the several DOE facilities.

Table 3.2-1. Historical and projected* mass and radioactivity of commercial spent fuel (DOE89a)

End of Calendar year	Mass accumulated (Mt)	Radioactivity accumulated (10 ⁶ Ci)
1970	55	215
1975	1,556	3,273
1980	6,534	10,159
1985	12,607	14,052
1988	15,607	18,654
1990	21,400	21,400
1995	30,800	25,600
2000	40,200	31,200
2005	48,700	31,900
2010	58,500	37,300
2015	70,200	42,400
2020	84,400	48,400

*Lower Reference Case projected capacity includes all existing reactors completed or under construction plus additional new reactors beyond the year 2005.

Table 3.2-2. Historical and projected* installed nuclear electric power capacity (DOE89a)

End of calendar year	Total GW(e)	End of calendar year	Total GW(e)
1960	0.2	1985	79.3
1965	0.3	1988	93.5
1970	6.4	1990*	99.6
1975	36.9	2000*	103.9
1980	51.4	2010*	100.6
		2020*	122.7

*Lower Reference Case projected capacity includes all existing reactors, completed or under construction, plus additional new reactors beyond the year 2005.

Table 3.3-1. Current volume of HLW in storage by site through 1988 (DOE89a)

Site Capsules ^{a)}	Total	Volume (10 ³ m ³)				
		Liquid	Sludge	Salt cake	Slurry	Calcine
Defense						
Savannah River Plant		64.2	14.1	50.0	™	™™128.43
Idaho Chemical Processing Plant		7.6	™	™	™	3.0™11.0
Hanford		26.8	46.0	93.0	73.4	™0.004243.5
	Subtotal	98.6	60.1	143.0	77.7	3.40.004
382.93						
Commercial						
West Valley Demonstration Project						
Acid waste		0.05	™	™	™	™™0.05
Alkaline waste		2.02	0.046	™	™	™™2.066
	Subtotal	2.15	0.046	™	™	™™2.116
	Grand total	100.67	59.97	143.5	73.4	3.00.004
385.05						

^{a)}Capsules contain either strontium (90Sr-90Y) fluoride or cesium (137Cs-137Ba) chloride.

[™]Not Applicable.

Table 3.3-2. Current radioactivity of HLW in storage by site through 1988 (DOE89a)

Site Capsules ^{a)}	Radioactivity (10 ⁶ Ci)					
	Total	Liquid	Sludge	Salt cake	Slurry	Calcine
Defense						
Savannah River Plant		99.0	400.0	162.1	^{b)}	^{c)} 661.3
Idaho Chemical Processing Plant		10.1	^{b)}	^{b)}	^{b)}	56.9 ^{c)} 67.0
Hanford		23.3	121.4	12.6	111.1	^{c)} 177.1445.5
	Subtotal	132.4	521.4	174.7	111.1	56.9177.1
1,173.6						
Commercial						
West Valley Demonstration Project						
Acid waste		1.84	^{b)}	^{b)}	^{b)}	^{c)} 1.8
Alkaline waste		13.1	13.7	^{b)}	^{b)}	^{c)} 26.8
	Subtotal	14.9	13.7	^{b)}	^{b)}	^{c)} 28.6
	Grand total	147.3	535.1	174.7	111.1	56.9177.1
1,202.2						

^{a)}Capsules contain either strontium (90Sr-90Y) fluoride or cesium (137Cs-137Ba) chloride.

^{b)}Not Applicable.

^{c)}Includes strontium and cesium in capsules and separated concentrates that are awaiting encapsulation.

The quantity of 90Sr-90Y is 61.3 x 10⁶ Ci and that of 137Cs-137mBa is 141.8 x 10⁶ Ci.

Table 3.3-3. Historical and projected volume and associated radioactivity of HLW in storage by site through 2020 (DOE89a)

End of calendar year	Volume (10 ³ m ³)								Radio-activity (10 ⁴ Ci)
	Liquid	Sludge	Salt cake	Slurry	Calcine	Capsules ^(a)	Glass ^(b)	Total	Total
Savannah River Plant									
1980	59.8	10.5	26.4	--	--	--	--	97	699
1985	71.3	13.8	37.6	--	--	--	--	123	841
1988	64.2	14.1	50.0	--	--	--	--	128	661
1990	57.5	14.4	49.9	--	--	--	--	124	664
1995	39.9	10.0	40.2	--	--	--	1.0	94	849
2000	34.7	6.0	29.7	--	--	--	2.0	73	837
2005	41.0	3.4	15.1	--	--	--	2.6	62	863
2010	36.8	1.6	16.4	--	--	--	3.3	58	858
2015	36.8	1.5	10.3	--	--	--	3.4	52	888
2020	36.8	1.5	7.5	--	--	--	3.5	49	884
Idaho Chemical Processing Plant									
1980	9.3	--	--	--	2.1	--	--	11	53
1985	7.1	--	--	--	3.0	--	--	10	69
1988	7.6	--	--	--	3.4	--	--	11	67
1990	7.6	--	--	--	3.9	--	--	12	77
1995	6.3	--	--	--	5.2	--	--	12	85
2000	6.7	--	--	--	6.3	--	--	14	128
2005	5.5	--	--	--	9.5	--	--	15	172
2010	2.0	--	--	--	11.3	--	--	13	182
2015	0.4	--	--	--	14.5	--	--	15	251
2020	0.4	--	--	--	16.8	--	--	17	284

^(a)Includes strontium and cesium in capsules and separated concentrates that are to be encapsulated.

^(b)Glass may be in storage at the site, in transit to a repository, or in a repository.

Table 3.3-3. Historical and projected volume and associated radioactivity of HLW in storage by site through 2020 (DOE89a) (continued)

End of calendar year	Volume (10 ³ m ³)								Radioactivity (10 ⁶ Ci)
	Liquid	Sludge	Salt cake	Slurry	Calcine	Capsules ^{a)}	Glass ^{b)}	Total	Total
Hanford									
1980	39.0	49.0	95.0	4.0	--	0.002	--	187	558
1985	28.1	46.0	93.0	55.1	--	0.004	--	222	554
1988	26.8	46.0	93.0	77.7	--	0.004	--	244	446
1990	25.3	46.0	93.0	81.3	--	0.004	--	246	415
1995	7.9	46.0	93.0	95.6	--	0.004	--	242	392
2000	6.9	46.0	93.0	92.1	--	0.004	--	238	342
2005	6.9	46.0	93.0	94.1	--	0.004	--	240	303
2010	6.9	46.0	93.0	95.7	--	0.004	--	58	858
2015	6.9	46.0	93.0	97.4	--	0.004	--	52	888
2020	36.8	46.0	93.0	97.4	--	0.004	--	49	884
West Valley Demonstration Project									
1980	2.15	0.046	--	--	--	--	--	2.196	35
1985	2.15	0.046	--	--	--	--	--	2.196	31
1988	2.07	0.046	--	--	--	--	--	2.129	29.3
1990	1.44	0.046	--	--	--	--	--	1.596	28.0
1995	--	--	--	--	--	--	0.21	0.21	24.7
2000	--	--	--	--	--	--	0.21	0.21	22.0
2005	--	--	--	--	--	--	0.21	0.21	19.7
2010	--	--	--	--	--	--	0.21	0.21	17.6
2015	--	--	--	--	--	--	0.21	0.21	15.6
2020	--	--	--	--	--	--	0.21	0.21	13.9

^{a)}Includes strontium and cesium in capsules and separated concentrates that are to be encapsulated.

^{b)}Glass may be in storage at the site, in transit to a repository, or in a repository.

Table 3.4-1. Inventories and characteristics of DOE/defense TRU waste buried through 1988 (DOE89a)

Values reported by burial site as of Dec. 31, 1988				
Burial site	Volume (m ³)	Mass of TRU elements (kg)	Alpha radioactivity (Ci)	
HANF	109,000	346	29,200	
INEL	57,100	357	73,267	
LANL	14,000	53.5	9,230	
ORNL	6,200	5.6	270	
SAND	3	<<1	1	
SRP	4,534	9.1	9,831	
Total	190,837	771.2	121,799	

Table 3.4-2. Inventories and characteristics of DOE/defense waste in TRU retrievable storage through 1988^(a) (DOE89a)

Values reported by burial site as of Dec. 31, 1988				
Burial site	Volume (m ³)	Mass of TRU elements (kg)	Alpha radioactivity (Ci)	
Contact-handled				
HANF	15,161	436	35,830	
INEL	63,975	747.8	261,417	
LANL	7,451.6	541.7	187,717	
NTS	596	4.1	705	
ORNL	625.2	26.6	17,505	
SRP	6,489	195.0	653,191	
Subtotal	94,297.8	1,951.2	1,156,365	
Remotely-handled				
HANF	137	6	855	
INEL	53.8	0.42	115	
LANL	11.1	1.8	150	
ORNL	1,304	106.2	2,920	
Subtotal	1,505.9	114.42	4,040	
Total	94,572.9	2,077.62	1,137,081	

^(a)Values cited are total quantities which represent the combined value of certified TRU waste and TRU waste managed as LLW (i.e. waste that is stored as TRU but falls below the 100 nCi/g alpha activity level).

Table 3.4-3. Inventories and characteristics of soil contaminated with DOE/defense TRU waste buried through 1988 (DOE89a)

Values reported by burial site as of Dec. 31, 1988(a)

Burial site	Volume (m ³)	Mass of TRU elements (kg)	Alpha radioactivity (Ci)
HANF	31,960	190.2	16,706
INEL	56,000-156,000	unknown	unknown
LANL	1,140	unknown	unknown
MOUND	300-1,000	0.009-.029	150-526
ORNL	13,000-61,000 ^(b)	unknown	unknown
SRP	38,000	unknown	unknown
Total	140,400-289,100	unknown	unknown

^(a)See text for details.

^(b)If soil containing TRU waste can be isolated from 1,600,000 m³ of soil containing TRU and LLW waste.

Table 3.4-4. Estimated physical composition of retrievably stored, newly generated, and buried TRU waste at DOE/defense sites (DOE89a)

Waste type	Waste composition, vol %				
	Contact-handled		Remote-handled		Buried
	RSW ^(a)	NGW ^(a)	RSW ^(a)	NGW ^(a)	
<u>ANL-E</u>					
Absorbed liquids or sludges		36			
Combustibles		32		50	
Glass, metal, or similar noncombustibles		32		50	
<u>HANF</u>					
Absorbed liquids or sludges					8
Combustibles	43	43	69.5	17	20
Concreted or cemented sludge	6	6	0.1		5
Filters or filter media					1
Glass, metal, or similar noncombustibles	48	48	30.4	75	48
Other					18
Dirt, gravel, or asphalt	3				
<u>INEL</u>					
Absorbed liquids or sludges	12	18.5			23.4
Combustibles	25	41.9	8	8	31.8
Concreted or cemented sludges	13	1.0			3.9
Dirt, gravel, or asphalt		2.3			6.7
Filters or filter media	5	6.9	11.2	11.2	1.3
Glass, metal, or similar noncombustibles	35	24.1	80(d)	80(d)	10.5
Other	10	5.3	0.8	0.8	22.4
<u>LANL</u>					
Absorbed liquids or sludges	22	10			4
Combustibles	8	25	50	50	7
Concreted or cemented sludges	36	15			44
Dirt, gravel, or asphalt		1			30
Filters or filter media	4	1			2
Glass, metal, or similar noncombustibles	30	48	50	50	13
<u>LLNL</u>					
Combustibles		73			
Concreted or cemented sludges		1			
Filters or filter media		7			
Glass, metal, or similar noncombustibles		15			
Other		4			
<u>MOUND</u>					
Combustibles		1			
Concreted or cemented sludges					
Dirt, gravel, or asphalt		89			
Glass, metal, or similar noncombustibles		10			
<u>NTS</u>					
Combustibles	51.5				
Concreted or cemented sludges	1				
Glass, metal, or similar noncombustibles	47.5				

Table 3.4-4. Estimated physical composition of retrievably stored, newly generated, and buried TRU waste at DOE/defense sites (DOE89a) (continued)

Waste type	Waste composition, vol %				Buried
	Contact-handled		Remote-handled		
	RSW ^(a)	NGW ^(b)	RSW ^(a)	NGW ^(b)	
ORNL					
Absorbed liquids or sludges			65		
Combustibles	59	55	20	55	40
Dirt, gravel or asphalt	1	1			
Filters or filter media	5	5	14	1	
Glass, metal, or similar noncombustibles	35	39	1	42	30
Other				2	30
RFP					
Absorbed liquids or sludges					
Combustibles		15.5			
Concreted or cemented sludges		36.3			
Dirt, gravel, or asphalt		0.7			
Filters or filter media		0.7			
Glass, metal, or similar noncombustibles		41.3			
Other		5.5			
SRP					
Absorbed liquids or sludges	(c)	3.5		(c)	
Combustibles	(c)	70		(c)	
Filters or filter media	(c)	5		(c)	
Glass, metal, or similar noncombustibles	(c)	27.5		(c)	
Other	(c)	1.5		(c)	

^(a)Retrievably stored waste (RSW).

^(b)Newly generated waste (NGW).

^(c)Not reported, assumed to be same as stored waste.

^(d)This is alpha hot-cell waste.

^(e)Not reported; assumed to be same as newly generated waste.

Table 3.4-5. Calculated isotopic composition (wt %) of buried and retrievably stored TRU waste for each site^(A) (DOE89a)

Isotopes ^(B)	Retrievably stored and newly generated wastes, wt %									
	Contact-handled							Remote-handled		Buried
	Mix-1 ^(C)	Mix-2	Mix-3	Mix-4	Mix-5	Mix-6	Mix-7	Mix-8	Mix-9	Mix-10
ANLE^(D)										
²³⁵ U							56.0			
²³⁹ Pu	87	85					40.0			
²⁴⁰ Pu	11	8					4.0			
²⁴¹ Am	1	1								
²³⁷ Np	5									
²⁴¹ Pu	1	1								
MFP ^(E)							<1.0			
HANE^(D)										
²³⁹ Pu	2.2						2.2	4.8		2.2
²⁴⁰ Pu	0.1						0.1	0.7		0.1
²⁴¹ Pu								0.1		
²³² Th	3.1						3.1	16.0		3.1
U depleted	72.8						72.8	21.6		72.8
U enriched	1.8						1.8	54.3		1.8
U normal	19.9						19.9	2.4		19.9
Other	0.1						0.1	0.1		0.1
INEL^(D)										
²⁴¹ Am	0.08	5.0		Trace						0.0001
²³⁹ Pu	Trace		80							
²⁴⁰ Pu	92.99	80	12.49	16		5.0	1.35	1.0		0.0017
²⁴¹ Pu	5.8	10	2.5	4		1.0	0.15			0.0001
²⁴² Pu	0.40	0.08								Trace
²⁴³ Pu	0.03									
²³² Th									25.0	
²³⁵ U									5.0	
²³⁸ U			10.19			38.20	39.40	69.0		0.060
²³⁵ U			74.74			55.20	59.10			99.800
MFP						0.6				
Other	0.70	5.00								0.1

Table 3.4-5. Calculated isotopic composition (wt %) of buried and retrievably stored TRU waste for each site^(a) (DOE89a) (continued)

Isotopes ^(a)	Retrievably stored and newly generated wastes, wt %									
	Contact-handled							Remote-handled		Buried
	Mix-1 ^(a)	Mix-2	Mix-3	Mix-4	Mix-5	Mix-6	Mix-7	Mix-8	Mix-9	Mix-10
LANL^(a)										
²³⁵ U							47	47		
²³⁸ U							28	28		5.0
²³⁹ Pu	5	0.5	1.2	0.5						0.01
²⁴⁰ Pu	92	21.5	98.8	93	100		22.7	22.7		91
²⁴¹ Pu							2.1	2.1		
²⁴² Pu							0.2	0.2		
²⁴¹ Am	3	78		6.5			0	0		3.3
MFP										
Other										0.69
LLNL^(a)										
²³⁹ Pu	0.014	0.013	0.011	0.009	0.004					
²⁴⁰ Pu	92.477	86.531	73.657	59.661	24.962					
²⁴¹ Pu	5.965	11.941	24.896	14.915	49.922					
²⁴² Pu	0.532	0.498	0.424	0.343	0.144					
²⁴³ Pu	0.023	0.022	0.018	0.015	0.006					
²⁴¹ Am	0.990	0.995	0.994	25.057	24.962					
MOUND^(a)										
²³⁹ Pu	79.894									
²⁴⁰ Pu	17.1									
²⁴¹ Pu	3.0									
²⁴² Pu	0.006									
NTS^(a)										
²³⁹ Pu	0.5									
²⁴⁰ Pu	93.0									
²⁴¹ Pu	6.0									
²⁴² Pu	0.5									

Table 3.4-5. Calculated isotopic composition (wt %) of buried and retrievably stored TRU waste for each site^(a) (DOE89a) (continued)

Isotopes ^(a)	Retrievably stored and newly generated wastes, wt %									
	Contact-handled							Remote-handled		Buried
	Mix-1 ^(a)	Mix-2	Mix-3	Mix-4	Mix-5	Mix-6	Mix-7	Mix-8	Mix-9	Mix-10
ORNL^(a)										
²³⁵ U	67.34	6.13						94.57	52.55	
²³⁸ U	11.95	2.99								
²³⁴ U	7.69									
²³⁹ Pu	4.58									
²⁴⁰ Pu	20.71	42.44	99.98				98.77	2.57	29.99	
²⁴¹ Am	1.68							2.11		
²⁴² Cm	0.96	0.02						0.75		
²⁵² Cf		0.15								
¹³⁷ Cs										
⁹⁰ Sr									0.95	
²³² Th									15.18	
¹⁵² Eu										
¹⁵⁴ Eu										
²³⁷ Np		13.03								
²⁴⁰ Pu		15.42								
²⁴¹ Pu		3.78								
Other		1.15					1.23		1.33	
RFP^(a)										
²³⁸ Pu	Trace									
²³⁹ Pu	91.00									
²⁴⁰ Pu	5.7									
²⁴¹ Pu	0.3									
²⁴² Pu	Trace									
²⁴¹ Am	1.7									
²³⁵ U	0.6									
SIS^(a)										
²³⁸ Pu	0.02	83.7	80.4							
²³⁹ Pu	93.18	14.0	16.2							
²⁴⁰ Pu	6.0	2.0	2.5							
²⁴¹ Pu	0.5	0.3	0.7							
²⁴² Pu			0.2							
²⁴¹ Am	0.3									
²³⁷ Np				100	100					
²⁴² Cm								100		

Table 3.4-5. Calculated isotopic composition (wt %) of buried and retrievably stored TRU waste for each site^(a) (DOE89a) (continued)

Footnotes

- ^(a) Data from Joseph Lippis, U.S. Department of Energy, Albuquerque Operations Office, WIPP, Carlsbad, New Mexico, memorandum to J.A. Klein, Oak Ridge, Tennessee, "TRU Waste Programs IDB Submittal through December 31, 1988," dated Sept. 15, 1989.
- ^(b) Isotopes listed are those that are either > 1%, by weight, or > 1%, by activity, of the total.
- ^(c) The mixes represent major waste stream composition variations.
- ^(d) At ANL-E, 46 vol % of the contact-handled TRU waste is Mix-1, 54 vol % is Mix-2, and 100 vol % of the remote-handled TRU waste is Mix-7.
- ^(e) Assumed ¹³⁷Cs to determine weight percent. ¹³⁷Cs chosen because it is the longest-lived major isotope in Mixed Fission Product (MFP).
- ^(f) At HANF, 100 vol % of the contact-handled TRU waste is Mix-1, 12.4 vol % of the remote-handled TRU waste is Mix-7, 5.1 vol % is Mix-8, 82.5 vol % is Mix-9 (the composition of Mix-9 is unknown), and 100 vol % of the TRU-contaminated buried waste is Mix-10.
- ^(g) HANF reported isotopic composition of uranium as U depleted, U enriched, and U normal. For radionuclide decay calculations, the data were converted to ²³⁵U and ²³⁸U by assuming 99.5%, 97.0%, and 99.3% ²³⁵U, respectively.
- ^(h) At INEL, 90.7 vol % of the contact handled TRU waste is Mix-1, 3.6 vol % is Mix-2, 0.3 vol % is Mix-3, and 5.4 vol % is Mix-4; 7.9 vol % of the remote-handled TRU waste has the same isotopic composition as Mix-1, 45.9 vol % is Mix-7, and 46.2 vol % is Mix-8. A portion of Mix-8 may also contain some Mix-9; and 100 vol % of the TRU contaminated buried waste is Mix-10.
- ⁽ⁱ⁾ At LANL, 27.5 vol % of the contact-handled TRU waste is Mix-1, 6.6 vol % is Mix-2, 5.2 vol % is Mix-3, 54.0 vol % is Mix-4, and 6.7 vol % is Mix-5; 78.6 vol % of the remote-handled TRU waste is Mix-7, and 21.4 vol % is Mix-8. Also, 100 vol % of the TRU-contaminated buried waste is Mix-10. Mix-2 contains trace weight % MFP but 10.7 activity % MFP.
- ^(j) Trace by weight percent, 85% by activity.
- ^(k) Trace by weight percent, 95% by activity.
- ^(l) At LLNL, 92.4 vol % of the contact-handled TRU waste is Mix-1, 1 vol % is Mix-2, 3.8 vol % is Mix-3, 2.3 vol % is Mix-4, and 0.5 vol % is Mix-5.
- ^(m) At MOUND, 100 vol % of the contact-handled TRU waste is Mix-1.
- ⁽ⁿ⁾ At NTS, 100 vol % of the contact-handled TRU waste is Mix-1.
- ^(o) At ORNL, 25.6 vol % of the contact-handled TRU waste is Mix-1, 38.7 vol % is Mix-2, and 37.5 vol % is Mix-3; 27.6 vol % of the remote-handled TRU waste is Mix-7, 6.4 vol % is Mix-8, and 66 vol % is Mix-9. No information available on buried waste at ORNL.
- ^(p) At RFP, 100 vol % of the contact-handled TRU waste is Mix-1. Weight percent totals less than 100% due to traces and round off.
- ^(q) At SRS, 57.0 vol % of the contact-handled TRU waste is Mix-1, 31.2 vol % is Mix-2, 2.2 vol % is Mix-3, and 8.9 vol % is Mix-4, 0.6 vol % is Mix-5, and 0.1 vol % is Mix-6. No information available on buried waste at SRS.

Table 3.4-6. Current inventories and projections of DOE buried and stored TRU waste from defense activities (DOE89a)

End of calendar year	Volume (10 ³ m ³)	Radioactivity (10 ³ Ci)	Mass (kg)
	Accumulation	Accumulation	Accumulation
<u>Buried^(a)</u>			
1988	190.8	62.3	771.2
1990	190.8	62.3	771.2
1995	190.8	62.3	771.2
2000	190.8	62.3	771.2
2005	190.8	62.3	771.2
2010	190.8	62.3	771.2
2015	190.8	62.3	771.2
2020	190.8	62.3	771.2
<u>Stored^(a)</u>			
1987	57.7	3,871.1	2,064.3
1990	67.8	6,921.7	2,078.9
1995	83.1	12,010.6	3,785.8
2000	99.2	17,118.3	4,873.2
2005	114.6	22,211.9	5,952.7
2010	129.7	27,296.2	7,027.0
2013 ^(b)	138.7	30,346.8	7,671.6

^(a)Certified TRU waste (excludes waste managed as LLW).

^(b)The destination of TRU waste after 2013 will not be defined until 2002.

Chapter 3 References

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Chapter 4: PLANNED PROGRAMS FOR THE MANAGEMENT AND DISPOSAL OF SPENT NUCLEAR FUEL AND HIGH-LEVEL RADIOACTIVE WASTES**4.1 INTRODUCTION**

As discussed in Chapter 1, the U.S. Department of Energy (DOE) is responsible for the care and disposal of government produced TRU wastes and spent nuclear fuel and high level wastes, regardless of their source. The DOE is conducting two programs to this end: (1) the Civilian Radioactive Waste Management Program, which pertains to the management and disposal of spent fuel from commercial nuclear reactors, commercial high-level waste, and any other wastes deemed by the U.S. Nuclear Regulatory Commission (NRC) to require geologic disposal, and (2) a program for the management and disposal of high-level and transuranic wastes generated in DOE atomic energy and defense activities.

The Congressionally selected disposal method for commercial HLW is burial in repositories excavated in geologically stable rock formations. Geologic disposal was selected after an evaluation (DOE80a) of several alternative concepts, including transmutation; disposal in space; the rock-melt concept; disposal in continental ice sheets, very deep holes, or isolated islands; and disposal under the ocean floor (the subseabed concept). In a recent rulemaking, the NRC has concluded that there is reasonable assurance that a repository located in deep geological media can provide safe disposal for spent fuel and high-level wastes (NRC84).

Disposal in deep geologic media has also been selected as the disposal method for much of the defense high-level and transuranic wastes. The DOE plans to dispose of the transuranic wastes in the Waste Isolation Pilot Plant (WIPP), located in New Mexico, if that facility is found suitable (DOE89a), and to a limited extent into the Greater Confinement Disposal Facility, located in Nevada. High-level wastes requiring disposal will be emplaced in the repository developed for commercial wastes

A geologic repository will consist of surface facilities, underground facilities, and shafts or ramps connecting the surface and the underground facilities. When the repository is prepared for permanent closure, seals will be constructed for the shafts, ramps, and exploratory boreholes. The underground facilities is expected to consist of entry drifts and disposal rooms excavated deep (hundreds to thousands of feet) beneath the surface, with boreholes drilled vertically into the floors or horizontally into the walls of the disposal rooms for the emplacement of waste canisters.

The repository will be prepared for permanent closure by backfilling the underground areas and permanently sealing the shafts and ramps. The surface facilities will be decontaminated and decommissioned, and the site will be eventually returned to its natural state. Permanent site markers, records, and other passive institutional controls will be erected to warn future generations of the presence of the repository and its contents.

An important element of the overall repository and disposal system is the waste package, which will be designed to meet various functional and regulatory requirements.

Among these are pre-closure requirements for radiation protection, maintaining the option to retrieve the emplaced wastes, and a 300 to 1,000 year canister lifetime. The post-closure requirements include providing substantially complete waste containment and thereafter controlling the radionuclide release rate from the engineered-barrier system.

4.2 CIVILIAN RADIOACTIVE WASTE MANAGEMENT PROGRAM

The DOE's current program for the management of civilian radioactive wastes has evolved from the National Waste Terminal Storage Program, which was established in 1976 to identify suitable repository sites. A two-fold approach was used in the search for suitable sites. The first approach was a systematic survey of regions and areas underlain by bedded salt and locations containing salt domes. This led to the eventual identification of four sites in bedded salt and three salt domes as potentially acceptable sites (see Section 4.4). The second approach was to search for suitable sites on Federal lands dedicated to nuclear activities, which led to the identification of two potential candidate sites: 1) a site in a volcanic rock (tuff), the Yucca Mountain, Nevada, and 2) a site in basalt, located in Hanford, Washington. This approach allowed the consideration of sites in diverse geologic environments and rock types or formations. In addition, the DOE was investigating sites in crystalline rock for the second repository and was conducting literature studies of such alternative host rocks as shales and other argillaceous rocks.

- **The Nuclear Waste Policy Act of 1982**

The DOE HLW disposal program entered a new phase in December 1982, when a political consensus was achieved by the passage of the Nuclear Waste Policy Act of 1982 (NWPA82). The Act authorized the construction of one geologic repository and specified that no more than 70,000 metric tons (heavy metal) of wastes be emplaced in the first repository, at which point a second repository would be developed. The Act also established a process and schedule for siting two geologic repositories and set up the Nuclear Waste Fund for paying the costs of the program. Furthermore, the Act assigned to the DOE the responsibility for managing this program and establishing the Office of Civilian Radioactive Waste Management for that purpose. In addition to the provisions for the repositories, the Act directed the DOE to study the need for, and feasibility of, monitored retrievable storage (MRS) and to submit to the Congress a proposal for the construction of one or more MRS facilities (DOE87a).

The DOE's plans for implementing the requirements of the Act were presented in the Mission Plan (DOE85a), which included a schedule showing that the first repository would start operations in 1998 and the second repository, if authorized by the Congress, would start in 2003. For the second repository, the DOE was conducting preliminary screening for crystalline rock sites in 17 States located in the northeastern, north-central, and southeastern regions of the United States.

- **The Nuclear Waste Policy Amendments Act**

Faced with greatly increasing costs and continued public opposition, the 100th Congress approved legislation amending the Nuclear Waste Policy Act of 1982. Known as

the Nuclear Waste Policy Amendments Act of 1987 (NWPA87), this legislation was signed into law by President Reagan.

Its most significant provisions were the following: site characterization for the first repository was limited to one site, Yucca Mountain in Nevada; only one repository was to be developed at the present time, with the DOE required to report to Congress between the years 2007 and 2010 on the need for a second repository; and the construction of an MRS facility was authorized subject to certain conditions. Additionally, three new institutional entities were established. Those were the MRS Review Commission, the Office of the Nuclear Waste Negotiator, and the Nuclear Waste Technical Review Board. The MRS Review Commission, whose members are appointed by the Congress, is to report to Congress on the need for an MRS facility. The main role of the Negotiator is to attempt to find a state or Indian Tribe willing to host a repository or an MRS facility at a technically-qualified site on reasonable terms. The Nuclear Waste Technical Review Board will provide independent oversight of the DOE's civilian waste management program.

- **Waste Management System**

In response to the Amendments Act, the DOE is developing a waste management system consisting of: 1) a geologic repository, 2) a MRS facility, and 3) a transportation system. The DOE plans for each of these elements are briefly summarized below.

Geologic repository -- The geologic repository program is currently focused on site characterization activities at Yucca Mountain. The DOE is preparing to construct two exploratory shafts and to conduct surface-based and in-situ tests designed to provide information about the suitability of the site. A detailed site characterization plan (DOE88a) based on conceptual repository and waste package designs has been published. This plan is based on a ranked hierarchy system reflecting regulatory requirements and DOE strategies for resolving technical and licensing issues. It provides a framework for conducting the testing needed to resolve issues about the design and performance of the repository and the waste package. It also describes the DOE's plans for assessing the pre-closure and post-closure performance of the repository, including the development and validation of the necessary models. In a parallel effort, the DOE will develop advanced design concepts for the repository and waste packages, including designs or design changes which will be included in the license application to the NRC.

The Yucca Mountain site lies in the southern part of the Great Basin, an arid region with linear mountain ranges and valleys, very little rainfall, sparse vegetation, and a sparse population. At Yucca Mountain, the water table is very deep, lying as much as 2,500 feet below the land surface. The repository is currently planned to be located in the unsaturated zone. The unsaturated zone is the rock mass between the surface of the land and the water table. At Yucca Mountain, the unsaturated zone is thick enough to allow the construction of a repository at a depth of about 1,050 feet while remaining about 660 to 1,300 feet above the top of the water table. The rock formation selected as the potential host media is volcanic tuff, which is a moderately to densely welded and devitrified rock (see Section 4.4 for more details). The host rock formation is known as the Topopah Spring Member of the Paintbrush Tuff.

The unsaturated rock of the Topopah Spring tuff is expected to provide a suitable environment for the long-term performance of the waste package. For example, the pressure exerted on the disposal containers is estimated to be equal to that of atmospheric pressure. There will be no hydrostatic pressure because the repository is to be located above the water table, and the waste packages will not be subjected to loads induced by the creeping (plastic movement) of the rock because the host rock is not plastic enough. Any water available for the corrosion of containers and waste dissolution is expected to be limited to minimal amounts. These and other pertinent features of the site will be subject to thorough investigations during site characterization by the DOE.

MRS facility -- The DOE has conducted a series of systems studies to determine the preferred configuration for the MRS facility (DOE89b). The preferred configuration is that of a facility which receives and temporarily stores spent fuel and initiates spent fuel shipments to an operational repository. The capability of packaging wastes is an option which could be added at some later time. It would consist of a facility and equipment needed for additional functions, such as waste consolidation and repackaging into disposal containers. This option would provide added flexibility and facilitate the operations of the waste management system. The MRS facility should be able to start receiving spent fuel more than 3 years earlier than the repository. The Amendments Act established two different paths for siting the MRS facility: 1) siting through a DOE-directed screening process or 2) siting through the Office of the Nuclear Waste Negotiator.

Transportation -- The transportation system will rely on the use of shipping casks, to be developed by the private sector, and transportation support services, which may include a cask maintenance facility, operational facilities, and support equipment. Specific facility needs will be identified as the designs for other facilities proceed. In order to increase cask capacity and reduce the number of shipments and overall disposal costs, the DOE has embarked on a cask development program designed to support the shipment of the following types of wastes:

1. Spent fuel from reactor sites to the facilities of the Civilian Radioactive Waste Management Program.
2. Spent fuel and other types of radioactive materials from the MRS facility to the repository.
3. Non-standard spent fuel and non-fuel bearing components from reactor sites to the MRS facility or the repository.
4. Defense and commercial high-level radioactive wastes from other storage locations to the repository.

The DOE will use the private industry to the maximum extent possible in both the development and the acquisition of transportation equipment and services. It plans to contract with private industry for cask development, certification, and the fabrication of prototype casks. After the development has been completed, the DOE will contract with the private sector to supply a fleet of casks for its transportation operations.

4.3 PROGRAMS FOR THE MANAGEMENT AND DISPOSAL OF DEFENSE WASTES

High-level wastes are produced during the reprocessing of spent reactor fuel and irradiated targets to recover uranium and plutonium. Most of the high-level wastes produced in the United States through 1987 have originated from atomic energy research and defense activities, such as the manufacture of nuclear weapons, and are stored at three DOE facilities: the Hanford Site in the State of Washington; the Savannah River Plant near Aiken, South Carolina; and the Idaho National Engineering Laboratory (DOE88b). These wastes are in the form of alkaline liquids, salt cake, slurry, sludge, acid liquid, and calcine; they are stored in underground tanks or bins. Most of them have already been subjected to some treatment (e.g., neutralization with caustic soda, which produces the sludges), and most will require incorporation into a stable solid medium, such as glass or ceramic. The solid waste will be packaged in stainless-steel canisters.

4.3.1 Defense High-Level Wastes

In accordance with the provisions of the Nuclear Waste Policy Act of 1982, the DOE performed a comparative evaluation of two disposal options for defense high-level wastes: 1) disposal in a commercial geologic repository and 2) disposal in a geologic repository constructed for defense wastes only. The two options were compared in terms of criteria specified in the Act: 1) cost efficiency, health and safety, regulation, transportation, public acceptability, and 2) national security. The results (DOE85b) indicated that there are no compelling reasons to develop a repository for defense wastes only, but the only factor that showed a clear advantage for disposal in a commercial repository was cost efficiency. The Secretary of Energy recommended to the President, and the President agreed, that a combined repository option could be implemented (DOE85a).

Typically, defense high-level radioactive wastes will be shipped in containers to a commercial repository for disposal after solidification, e.g., in glass or some other solid forms. At the repository, the containers will then be transferred underground for emplacement and final disposal. Transportation to the repository will be conducted by the DOE's Office of Civilian Radioactive Waste Management, and the costs of geologic disposal will be paid by DOE contributions to the Nuclear Waste Fund.

Hanford Site – Since the early 1940s, DOE Hanford operations have resulted in the generation of large volumes of solid and liquid wastes. Solid wastes are routinely disposed of on site by ground burial. Liquid wastes are stored either temporarily or permanently in special tanks, processed, and discharged into surface ponds, cribs, ditches, or released into the Columbia River. There are four major operating areas at the Hanford Site which support a diverse range of production and research activities. They are: the 100-Areas, which include the N-Reactor and eight deactivated production reactors; the 200-Areas (West and East), which include two reactor fuel reprocessing plants and waste treatment facilities; the 300-Areas, which include the reactor fuel manufacturing and research and development facilities; and the 400-Areas, which includes the Fast Flux Test Facility.

The high-level radioactive wastes stored at Hanford may be divided into five groups: liquids (11% by volume), sludges (19%), and salt cake (38%) - stored in single-shell tanks; slurry (32%) - stored in double-shell tanks; and encapsulated wastes which contain heat producing radionuclides (e.g., strontium-90 and cesium-137, and their decay products) (DOE89d). Encapsulated wastes are stored in water basins for cooling. Encapsulated wastes, however, represent a very small fraction of the total waste volume (less than 0.002%), but about 40% of the total activity for the waste forms identified above (DOE89d). Table 4.3-1 presents a summary of the Hanford waste storage methods, volumes, TRU activity, and number of sites (see Chapter 3 for more details).

Much of this waste has been accumulating since the 1940s and was initially stored in single-shell tanks. More recently, double-shell tanks were built to reduce the possibility of leakage into the environment since it was known that a number of the single-shell tanks (26) were leaking (DOE87b). The double-shell tanks provide redundant containment by incorporating primary and secondary carbon steel walls which are supported and encased in a reinforced concrete outer shell. The radioactive decay-heat is removed by a closed-loop cooling coil system within which circulates cooling water. The contents of each tank are continuously mixed by air circulators and air ballast tanks which provide an intermittent flushing action to prevent sediments accumulating at the bottom of the tank.

In total, there are five types of tanks (Types I through IV are single-shell and Type V is double-shell) used in Hanford with capacities ranging from 210 to 3,800 m³. Typically, seven single-shell tanks (3,800 m³) are needed to provide a 5-year storage capacity associated with the processing of about 2,000 metric tons of spent fuel (DOE80a). Currently, there are 28 double-shell tanks in use in addition to the existing 149 single-shell tanks (DOE87b). These tanks are located in 17 tank-farm sites located in both the East and West 200-Areas. The tanks are placed underground and are covered with about 2 meters of soil. On the average, the tanks are located about 60 meters above the local water table.

The wastes, destined for storage in the double-shell tanks, are typically pre-treated prior to being pumped into the tanks. The low specific-activity portion of this waste stream will be separated and disposed in a grouted form in near-surface vaults at the site. The remaining wastes are to be solidified into borosilicate glass before being shipped to a commercial geologic repository. The DOE will build a plant for vitrifying these wastes. Current plans call for the plant to start operating in 1999. Encapsulated strontium and cesium wastes will also be sent to the commercial repository for disposal.

The DOE has identified preferred disposal alternatives for some waste forms while it has deferred its decision for other types of wastes (DOE87b, DOE89c). For example, the DOE currently plans to store, for the foreseeable future, wastes contained in single- and double-shell tanks (DOE87b, DOE89c). The potential disposal methods being considered include in-situ immobilization of the tanks and their contents, disposal of a fraction of the tank wastes (as low specific activity grout) in near-surface vaults, and waste treatment and vitrification for final disposal at the commercial HLW repository. Encapsulated strontium and cesium wastes are also to be sent to the commercial HLW repository once it becomes operational. Transuranic wastes buried at the Hanford facility will be eventually retrieved, processed, and repackaged prior to being sent to a dedicated transuranic waste disposal

facility (DOE89a). For contaminated soils, DOE is evaluating several options which include in-situ stabilization and geologic disposal. In the interim, the DOE will continue maintenance activities of the sites where radioactive materials are presently buried. Current DOE schedules indicate that these disposal plans will be implemented over a 20-year period (DOE89c).

In response to Federal, State and local requirements, the DOE is conducting a comprehensive environmental monitoring program to assess the impact of facility operations in the vicinity of the Hanford Site (JAC88). The results of the environmental monitoring program indicate that on-site radionuclide ground-water concentrations were noted to be above the EPA Drinking Water Standards (DWS) and in some instances above the DOE's Derived Concentration Guides (DCG) (JAC88). The results of the monitoring program indicates that the following radionuclides are present in ground-water: tritium, cobalt-60, strontium-90, technetium-99, ruthenium-106, antimony-125, iodine-129, iodine-131, cesium-137, uranium-234, and uranium-238.

Certain chemicals regulated by the EPA and the State of Washington were also present in the ground-water near the operating areas. The primary source of the ground-water contamination is due to liquid wastes released into the ground by past and on-going site and facility operations. Waste disposal activities, at both active and inactive sites, have also contributed to the current levels of contamination.

Elevated tritium levels are present in all Hanford Site Areas, except in the 300-Areas. The highest concentrations were noted to be in the 200-Areas, in both East and West sections. The tritium plume in the 200-Areas is characterized with peak concentrations ranging from 1 million to nearly 14 million pCi/L. The peak tritium concentration in the 100-Areas was reported to be 1.3 million pCi/L. Other locations on the Hanford Site are characterized by tritium concentrations ranging from non-detectable levels (about 300 pCi/L) to a few hundred thousands pCi/L. In general, the tritium plumes are moving east and southeast following the movement of the groundwater toward the Columbia River. The EPA DWS for tritium is 20,000 pCi/L and the DOE DCG limit is 2 million pCi/L (JAC88). The EPA DWS is based on an organ (whole body for tritium) annual dose limit of 4 mrem while the DOE DCG represents a committed effective dose equivalent of 100 mrem per year (JAC88).

Gross-beta radioactivity was found in wells throughout the Hanford Site. This radioactivity is associated with the presence of cobalt-60, strontium-90, technetium-99, antimony-125, cesium-137, uranium decay products (thorium-234 and protactinium-234), and to a certain extent to iodine-131 and iodine-129. The highest concentrations were noted to occur in the 100 and 200-Areas. The gross-beta radioactivity in ground-water samples is characterized with peak concentrations ranging from 1,000 to nearly 16,000 pCi/L. Other locations on the Hanford Site are characterized by total gross-beta radioactivity ranging from non-detectable levels (about 16 pCi/L) to a several hundreds pCi/L. The radionuclide distribution and ground-water concentrations were reported, in decreasing order, to be technetium-99 (up to 29,000 pCi/L), iodine-131 (up to 28,000 pCi/L), strontium-90 (up to 10,000 pCi/L), antimony-125 (up to 300 pCi/L), iodine-129 (up to 47 pCi/L), ruthenium-106

Table 4.3-1. Summary of waste sites, volume, and activity at the DOE Hanford Facility.^(a)

Type of Wastes	Number of Sites/Tanks	Area (Ha)	Volume (m ³) ^(b)	TRU (Ci) ^(b)
Single-shell tanks	12/149	5.5	1.4E+5	6.1E+4
Double-shell tanks	5/28	1.2	9.7E+4	3.2E+5
Capsules ^(c)	1/--	0.01	minimal ^(d)	minimal ^(d)
Retrievable TRU wastes	7/--	5.0	2.6E+4	9.0E+4
Buried TRU wastes ^(e)	9/--	7.3	1.1E+5	3.0E+4
TRU contaminated soil	24/--	1.2	3.2E+4	2.0E+4

(a) Data Extracted from DOE87b, Volume 1, Table 3.1, page 3.6. Also see Chapter 3 for more details.

(b) Exponential notation, 1.4E+5 means 1.4×10^5 or 140,000.

(c) For this entry and the following ones, the wastes are not stored in tanks.

(d) Presence of TRU material is negligible, most of the activity is due to long-lived fission products totaling of about 203 million curies. The volume of the capsules is less than 0.002% of the waste volume to be treated and disposed.

(e) Wastes buried up to 1970.

(less than 30 pCi/L), cesium-137 (less than or equal to 22 pCi/L), and Co-60 (less than or equal to 20 pCi/L) (JAC88). As with tritium, these radionuclides are also moving east and southeast following the movement of the ground-water toward the Columbia River. Except for technetium and iodine, the ground-water plumes associated with these radionuclides are not as extensively dispersed as the one due to tritium. The reported concentrations for technetium, iodine, and strontium in several wells exceed the EPA DWS.

The presence of alpha-emitting radionuclides were detected in several wells located in the 100, 200, and 300-Areas. The total gross-alpha radioactivity is thought to be due to uranium since plutonium concentrations were noted to be below the limit of detection (about 0.1 pCi/L) (JAC88). The highest concentrations were noted to occur in the 200-Areas (West) while much lower concentrations were detected in the eastern sector of the 200-Areas. The peak concentrations in the 200-Areas (West) were reported to range from 100 to 10,500 pCi/L. Uranium has been also been noted in the vicinity and downgradient of the fuel fabrication facilities (300-Areas) and near inactive waste disposal sites. The average uranium concentrations were reported to range from 2 to 310 pCi/L, with peak concentrations ranging from 100 to nearly 12,000 pCi/L. Other locations on the Hanford Site are characterized by uranium concentrations ranging from non-detectable levels (0.5 pCi/L) to less than 100 pCi/L. The reported gross-alpha concentrations in several wells exceed the EPA DWS.

The Hanford radiological environmental surveillance program also routinely monitors other areas, at both on and off-site locations. These locations include three on-site ponds and one lake, soils at 38 different on and off-site locations, and at upstream and downstream points on the Columbia River.

Radionuclide concentrations in the three ponds and West Lake have been noted to vary (DOE87b). The 1987 survey results indicate that tritium is the dominant radionuclide with peak concentrations ranging from 160 to 9,500 pCi/L. The next predominant radionuclide is cesium-137 which was reported to range from 1.1 to 50 pCi/L. Strontium-90 was also detected in pond and lake water samples with peak concentrations ranging from 0.4 to 2.8 pCi/L. Total gross beta and alpha water sample activity revealed peak water concentrations of 490 and 267 pCi/L, respectively, for West Lake. The gross beta and alpha water activity in the three ponds were typically one to two orders of magnitude lower than those noted for West Lake.

Soil sample analyses at 15 on-site locations revealed four radionuclides have been routinely detected in measurable levels. Strontium-90 is known to be present in concentration varying from 0.02 to 0.38 pCi/g with an average of 0.31 pCi/g. Cesium-137 has been measured at concentrations varying from 0.01 to 16 pCi/g with an average of 2.0 pCi/g. Plutonium-239 and 240 have also been measured at concentrations varying from 0.001 to 0.17 pCi/g with an average of 0.027 pCi/g. Finally, uranium was reported at concentrations ranging from 0.19 to 3.8 pCi/g with an average of 0.58 pCi/g. Typically, the average on-site measurements are higher than those noted off-site by factors ranging from about 2 to 5.

Analyses of water samples taken downstream in the Columbia River indicate that radionuclides identified with Hanford Site operations were noted at very low concentrations, typically well below the applicable drinking water standards (DOE87b). The water samples

were taken at two different locations, one at the 300-Areas Water Intake and the other at the Richland Pumpouse located about 3 km downstream from the site boundary. The Richland Pumpouse is the first downstream point on the river where water is withdrawn for public use. Water sample analyses revealed that tritium is the most predominant radionuclide with a reported peak concentration of 200 pCi/L. Other radionuclides were also reported, including strontium-89 and -90 (0.2 and 0.15 pCi/L, respectively), total uranium (0.61 pCi/L), gross beta (2.8 pCi/L), and gross alpha (0.79 pCi/L). Other radionuclides, including plutonium-239 and -240 as well as other fission products, were reported at lower concentrations, typically ranging from 1.0×10^{-6} to 4.5×10^{-2} pCi/L.

Savannah River Plant – At the Savannah River Plant, high-level wastes, in the form of alkaline liquids, alkaline sludges, and salt cake, are stored underground in high integrity, double walled, stainless-steel tanks. By 1993, hot operation of a waste processing facility to vitrify these wastes into borosilicate glass is scheduled to begin.

Idaho National Engineering Laboratory – In Idaho, high-level wastes, in the form of acidic liquids, are first stored in underground tanks and later converted to calcine. Stainless steel tanks housed in concrete vaults are used to store liquid wastes, and stainless steel bins in concrete vaults are used for the calcine wastes. According to DOE plans, a facility for immobilizing newly generated wastes will start operations early in the next century. It will also process the stored calcine wastes. Evaluations of waste forms and immobilization processes are being pursued.

4.3.2 Transuranic Wastes

- Waste Isolation Pilot Plant (WIPP)

The DOE is developing the Waste Isolation Pilot Plant (WIPP) in a bedded-salt formation near Carlsbad, New Mexico to demonstrate the disposal of defense TRU wastes. The WIPP project was authorized in 1980 by Public Law 96-164 to provide a research and development facility for demonstrating the safe disposal of transuranic wastes produced by national defense activities. If testing proves satisfactory, the DOE is expected to open the site for the permanent disposal of TRU waste.

The WIPP site is in a sparsely populated area on land owned by the Federal government. The WIPP plant consists of surface facilities (mainly a waste-handling building), four access shafts, and underground facilities designed to emplace approximately 6.5 million cubic feet of TRU waste in a 100-acre repository. About 12 acres have also been set aside as an underground test area to conduct experiments and study the behavior and performance of the repository. The repository has been excavated in a bedded-salt formation (the Salado Formation) 2,150 feet beneath the surface.

By mid-1989, the initial major construction activities at the WIPP had been nearly completed (DOE89a). The surface facilities were essentially complete, and most of the underground rooms for experimentation and initial waste emplacement had been excavated. A five-year test phase is planned to develop data for incorporation into the performance assessment. All of the wastes will be retrievably emplaced should the site be declared

unsuitable at the end of the test period. During this phase, the DOE will monitor the site and facility as part of the environmental monitoring programs it has been conducting since 1980.

For shipment to the WIPP site, TRU wastes will be contained in Type B shipping containers (TRUPACT-II) certified by the NRC and carried by truck. The DOE's purpose in using truck transportation for moving waste to the WIPP is to have greater accessibility to the site and greater control of the transportation system, routes, and speed. The proposed routes from the waste storage locations use the interstate highway system to the maximum extent possible (DOE89a). To ensure safe and efficient transport, the DOE will use a transportation tracking and communication system that will combine navigation, satellite communication, and computer network technologies to monitor the movements of TRU waste shipments to the WIPP.

All of the wastes received by the WIPP will have to meet acceptance criteria covering factors such as waste forms and characteristics, gas generation, immobilization, presence of toxic and corrosive substances, and thermal power. All incoming packages will be checked for surface contamination and external radiation exposure rates, and repackaged or repaired if necessary.

If the results of the five-year test phase are satisfactory and the WIPP proceeds to the disposal phase, each disposal room located in the underground facility will be backfilled with crushed salt or other materials as the waste containers are emplaced and permanently sealed. If the DOE decides not to proceed with the disposal phase or when the facility has been filled to capacity with TRU wastes, the WIPP will be decommissioned. The shafts and boreholes will be filled and plugged; the surface facilities will be decontaminated, demolished, or dismantled; and the site will be restored to its original conditions.

- **Greater Confinement Disposal Facility**

In 1981, the National Low-Level Waste Management Program and the DOE's Nevada Operations Office began a project to demonstrate the feasibility of "greater depth" burial in the alluvial sediments of the Nevada Test Site (REY83, EPA87). The purpose of the project, named Greater Confinement Disposal Test (GCDT), was to evaluate the feasibility of disposing of classified TRU wastes and high specific-activity low-level wastes at intermediate depths in large-diameter augered holes. These wastes originate from weapon facilities across the nation. The basic concept involves sinking a shaft 3 meters in diameter and nearly 40 meters deep. The shaft has a capacity of about 1,100 m³. Wastes are then lowered into the hole and stacked up to depth of about 20 meters from the surface. At this point, the hole is backfilled with soil all the way up to the surface. The goal of the GCDT program is collect and analyze data on radionuclide migration and to develop waste handling procedures and equipment. Plans are also being developed to retrieve these wastes after emplacement, if necessary.

4.4 POTENTIAL HOST ROCKS FOR GEOLOGIC REPOSITORIES

Many types of rocks are potentially suitable as host rocks for a repository, depending on the natural attributes of the rock and the geohydrologic setting. Ideally, the host rock should be suitable for the construction of the repository and for waste containment, and the surrounding rock formations should provide adequate isolation (DOE80b). Important natural attributes include thermal, mechanical, hydraulic, and chemical characteristics that affect the response of the host rock to heat, the movement and chemistry of ground water, and the ability to retard the migration of radionuclides. The desirable geohydrologic properties include low rates of ground-water flow, long path lengths to the accessible environment, and evidence of long-term stability (DOE80b).

In the United States, early plans for geologic disposal were based on bedded salt and salt domes. Salt was the rock investigated most extensively as part of a site screening program. Later, when the DOE began to study Federal lands dedicated to nuclear activities, several other host rocks came under investigation. They included argillaceous rocks and tuff in Nevada and basalt in the State of Washington. For the second repository, DOE began to study crystalline rock formations (DOE86a-g). Other rocks that have been considered are limestone, sandstone, anhydrite, chalk, and argillaceous rocks like shale (GON85). The sections that follow briefly review the properties of host rock media most studied in the United States.

4.4.1 Basalt

Basaltic rock masses are among the strongest of common rock types. In addition, basalt has moderate thermal conductivity and a high melting temperature, which enable it to withstand high thermal loads. The basaltic formation that had been investigated in the first repository program was a thick section, about 950 meters below the surface, near the middle of the extensive basalt flows of the Columbia Plateau. The basaltic rock in this section contains openings filled with alteration products (mainly clay minerals), and as a result the rock mass is of low permeability. On the other hand, the basalts of the Columbia Plateau commonly have columnar joints or rubbles that are potential channels for water flow. Water-bearing sedimentary interbeds within the basalt section are also common.

A potential site in basalt is located in the State of Washington. Thick basaltic formations also occur in the States of Idaho and Oregon.

4.4.2 Bedded Salt and Salt Domes

Of the nine sites identified as potentially acceptable for the first repository, seven were in salt: four sites in bedded-salt formations and three in salt domes (DOE85a).

Salt is suitable as a host rock because of its structural strength, radiation shielding capability, high plasticity (which enables fractures to heal or seal themselves at repository depths), low moisture content, and low permeability. In addition, salt deposits are abundant in the United States and are relatively easy to mine. Desirable features of many salt basins are their relatively simple structure and predictable stratigraphy over large areas.

Although salt deposits are widespread, the salt itself and the other deposits with which it is often associated (e.g., hydrocarbons or potash) could increase the probability of human intrusion into a repository. Furthermore, the solubility of salt is greater than that of any other potential host rock. The potential for this failure mode must be carefully assessed in analyzing the long-term performance of a repository sited in salt.

4.4.3 Granite and Related Crystalline Rocks

Granite and related crystalline igneous and metamorphic rocks, such as gneiss, are the most abundant rocks in the upper 10 kilometers of the Earth's continental crust. These rocks underlie virtually all of the United States; they occur at the surface in stable areas, in the cores of many mountain ranges, and beneath all of the younger sedimentary rocks. Their strength, structural and chemical stability, and low porosity make them attractive for geologic repositories. The water content of these rocks is low and is held mainly in fractures and in hydrous silicate minerals. The permeability of these rocks is largely dependent on the presence of fractures, and it is reduced considerably by the closure of fractures, which occurs at depths in excess of several hundred meters. The depth for a repository is likely to vary from region to region, depending on how the permeability is affected by the tectonic history of the region.

Granite as a potential host rock is being investigated in some European countries. In the United States, the DOE had conducted preliminary investigations of near-surface and exposed crystalline rock formations in 17 States in a search for sites for the second repository. However, the Amendments Act directed DOE to terminate site-specific activities for a second repository and limited such activities only to tuff.

4.4.4 Tuff

Tuff is the dominant component of the voluminous and widespread volcanic strata in the Basin and Range province of the western United States. The tuff formation at the Yucca Mountain site, located in southern Nevada, currently being characterized for the first repository, consists of a sequence of welded and non-welded tuffs.

The site selected as the potential host rock is moderately to densely welded and devitrified, with a minor number of cavities. This section of the rock formation has high density, low porosity and water content, good compressive strength, and the ability to withstand the heat generated by radioactive waste. However, the characteristics that affect the thermal and mechanical properties of tuff, such as porosity, degree of saturation, and stress state, are known to vary both laterally and vertically. Consequently, the thermal and mechanical properties are also likely to vary spatially.

Lying beneath the welded tuff are non-welded tuffs containing zeolite, a hydrous silicate. These tuffs are characterized by low density, moderate compressive strength, moderate thermal conductivity, and excellent capability for sorption. The latter is important to the waste isolation performance of a repository because it would allow these rocks to significantly retard the migration of radionuclides into the accessible environment.

4.5 SCIENTIFIC INVESTIGATIONS AND RESEARCH

The DOE is conducting major programs of scientific investigations and research for the management and disposal of both civilian and defense wastes. In large part, these programs are being conducted by the DOE National Laboratories, i.e., Argonne, Brookhaven, Idaho National Engineering, Lawrence Berkeley, Lawrence Livermore, Los Alamos, Oak Ridge, and Sandia.

DOE is also assisted by the U.S. Geological Survey and by several contractors with special expertise. A brief summary of these programs, which includes by reference the activities of selected laboratories, is given below. In this respect, it is important to note that the following summary is not meant to be an exhaustive discussion of the role and research activities of each national laboratory.

4.5.1 Civilian Radioactive Waste Management Program

- Site characterization

The dominant scientific efforts in the Civilian Radioactive Waste Management Program are related to the characterization of the Yucca Mountain site, the design of the waste package, and the long-term performance assessment of the repository. As described in a multi-volume plan (DOE88a), the DOE's site characterization program includes investigations directed at collecting geologic, hydrologic, and geochemical geoengineering information and other additional data needed to determine whether or not the site is suitable to host a repository and to provide further technical input for the repository and waste package designs.

One of the objectives of the geologic and hydrologic investigations is to collect the data needed to develop geohydrologic models for the unsaturated zone, the saturated zone, and the surface water system. The models for the unsaturated and saturated zones will be used to simulate ground water flow paths, to calculate ground water fluxes and velocities, and to evaluate potential radionuclide transport under varying site and repository conditions. The results will be used to develop conceptual and numerical models that can be used to assess the combined effects of heat, water, and gas flow under present conditions and for conditions expected to span the next 10,000 years. The models, coupled with site-specific data and parameters, such as geochemical, thermal, and mechanical properties, will be used to support the site and repository performance assessment activities.

Site geochemical investigations consist of laboratory tests and experiments. The objective of these investigations is to determine the capacity of the rock formation to retard radionuclide migration by various mechanisms, such as sorption; precipitation; and by dispersive, diffusive, and advective processes. In addition, these studies are addressing the retardation of gaseous radionuclides.

The geoengineering investigations include studies of rock properties and studies of thermal-mechanical properties. The geologic and geophysical data needed to develop a computer-based three-dimensional physical-property model that will generate a representation

of the physical properties of the formation at the Yucca Mountain site will be collected. The model will be used in the design of the underground facilities and predict ground water travel times, the expected lifetime of the waste packages, and the rates of radionuclide releases from the engineered barrier system to the accessible environment. The thermal-mechanical investigations will also provide information to support the development of design criteria for the repository system and to conduct performance assessment activities.

- **Repository design**

The repository design will in part be based on performance assessment. The performance assessment will include assessments of the potential effects of shaft or repository construction on the integrity of the site and analyze the importance of limiting water usage and infiltration in characterizing and constructing the repository. The results of these analyses will also be used to establish the configuration of the waste emplacement boreholes.

- **Repository Seals**

Since the post-closure seals for the shafts, ramps, and boreholes would not be installed until the underground repository is closed, the design and testing of seals will take place over an extended period. Current work is aimed at developing design concepts, establishing requirements for the seal systems, and evaluating potential seal materials. Design considerations and trade-off studies will be performed to select the appropriate configurations of seal components, placement methods, and materials. The trade-off studies will evaluate the quantities of ground water, routes of infiltration, and the potential for drainage through drift floors, shafts, and ramps. Various seal installation methods will be evaluated as design concepts and selection of seal components and materials are narrowed down to a few candidates.

- **Waste Package**

The waste package development program includes materials testing, design analysis, conceptualization of a reference design, and waste package performance assessment. This program includes studies directed at characterizing the interaction of the host rock environment with waste packages. Laboratory tests and model analyses will be conducted to evaluate geochemical changes induced by the presence of the waste and the materials used in constructing the repository (e.g., grout). In addition, thermal and mechanical analyses of the waste package and the host rock around the emplacement holes will be performed to evaluate temperature distribution profiles and the stability of the emplacement holes as a function of time.

- **Performance Assessment**

The performance assessment program is responsible for developing analytical techniques to assess the pre-closure and post-closure performance of the repository as a total system, which includes the site, the underground facilities, and waste packages. Both conceptual and numerical models are being developed by the program. The assessment models will be validated, using both field and laboratory data. The analytic techniques for

performing the calculations will be tested to ensure that they correctly perform the calculations and the results will be evaluated to ensure that they properly represent site conditions and empirical and theoretical relationships. The performance assessment models will be used to support the NRC license application.

4.5.2 Defense Waste Programs

The research and development (R&D) efforts for defense wastes are divided into three major categories: 1) the immobilization of high-level wastes, 2) the preparation of transuranic wastes for shipment to the WIPP facility, and 3) investigations to demonstrate the performance of the WIPP site. Also these R&D efforts include the development of technology for in-place immobilization of wastes stored in tanks and evaluation of methods for immobilizing wastes stored at the Idaho National Engineering Laboratory.

The DOE has proposed a performance assessment test phase for the WIPP facility to determine whether or not the WIPP site can meet the post-closure standards of 40 CFR Part 191 (DOE89a). These tests, which would be performed with transuranic wastes in place, are designed to provide information on the behavior of the repository, including disposal room closure rates, brine migration, waste corrosion processes, gas generation, gas composition, and gas depletion rates. This performance assessment program will be conducted by the WIPP Project Office under the direction of Sandia National Laboratories.

4.6 INTERNATIONAL ACTIVITIES

Countries which are committed to use nuclear power or in which nuclear power already makes up a significant fraction of the total electrical generating capacity are establishing long-term programs for the safe management and disposal of spent fuel and high-level radioactive wastes. Such programs include adopting a national strategy, assigning the technical responsibility for research and development activities to designated agencies, selection of disposal technologies and geological media, and setting the appropriate regulatory standards to protect the public and environment.

Typically, the objective of a geological disposal program is to immobilize and isolate radioactive wastes from the environment for a sufficient period of time under conditions such that any radionuclide releases from the repository will not result in unacceptable radiological risks. For illustrative purposes, the disposal programs of eight countries are summarized below (NEA86, NEA88, SCH88, IEAL87). These countries are Canada, the United Kingdom, France, the Federal Republic of Germany, Belgium, Switzerland, Sweden, and Japan. A summary of these countries' institutional and regulatory programs is also provided in Chapter 2, Section 2.3.

4.6.1 Canada

The Atomic Energy Control Board of the Canadian Federal Government has been assigned the responsibility for the permanent disposal and isolation of radioactive wastes in Canada. Currently, the program considers only direct disposal without reprocessing, although the reprocessing option has not been completely ruled out. Until a repository is

available, spent fuel will initially be stored at each reactor site and, later, possibly at a central facility. Under a joint agreement, Ontario Hydro (a provincially owned utility) has been mandated to develop the technologies needed for the interim storage and transportation of spent fuel.

The Canadian disposal concept considers siting a repository in a granitic formation located in the Canadian Shield. The repository will be located at depths of 500 to 1,000 meters. The spent fuel canisters will be inserted in floor cavities located in excavated disposal rooms. Once filled, the floor cavities and room excavations will be backfilled and sealed using engineered barriers. The AECB facility design is already well defined (Concept Assessment Documentation) and the design has already begun a phase of public and regulatory review.

In 1986, an underground research laboratory was established in undisturbed granitic rock at depth of 240 meters. The site is located at Lac du Bonnet, in the Province of Manitoba. The purpose of this test facility is to develop methodologies and analytical techniques to evaluate the geomechanical and geohydrological properties of granitic rock and characterization and sealing of rock fractures. Testing was initiated in late 1986 and the construction of the facility was completed in 1988 down to a depth of 420 meters. The AECB intends to submit the facility's final conceptual assessment report by 1991, receive government approval by the mid 1990s, and complete the facility development and site characterization by the 2000. It is expected that the repository will become operational by 2015.

4.6.2 United Kingdom

In the United Kingdom, the responsibility to develop a national strategy for radioactive waste management lies with the Office of the Secretary of State for the Environment, which cooperates with its Scottish and Welsh counterpart offices. The organizations which produce the wastes have the direct responsibility for their safe management and funding. An industry consortium, the UK NIREX Ltd, has been established to develop and operate new radioactive waste disposal facilities in England.

The United Kingdom's radioactive waste disposal program strategy has postponed the development of a disposal facility in deep geological media. Rather, the current plans call for reprocessing of spent fuel, solidification, and surface storage for about 50 years. The United Kingdom has also adopted a policy of monitoring the results of research activities being conducted by other countries. Depending on the outcome of research being conducted abroad, Britain would then identify a high-level waste disposal strategy and a repository program development activities using concepts that best fit British needs.

However, recently some in situ research has been conducted by UK Atomic Energy Authority and UK NIREX Ltd in heat transfer properties of Cornish granite, statistical analysis of fracture occurrence, orientation, and aperture in granite, and fractured flow in Cornwall shale. Other research activities have included geohydrological and geophysical measurements, geochemistry, radionuclide migration and transport, integrated site

characterization and model validation, and characterization of model parameters and measurement methods.

4.6.3 France

The French radioactive waste disposal program considers spent fuel reprocessing, interim storage, and recovery and re-use of plutonium in breeder and light-water reactors. The nuclear waste program has been entrusted to the National Waste Management Agency (ANDRA), an arm of the French Atomic Energy Commission (CEA). Since 1969, low-level, intermediate, and reprocessing wastes have been emplaced in engineered-barrier surface disposal facilities at the La Hague site, near Cherbourg on the English Channel. This facility will reach its design capacity in the early 1990s.

New facilities are planned for the disposal of both low-level and intermediate-level radioactive wastes, and spent fuel and high-level wastes. The radioactive waste disposal program is currently investigating four geological media which include clay, salt, granite, and schist, down to depths of 2,000 meters. Four sites have been identified where characterization activities or investigative work are in progress. These sites are located in the Department of Aisne for clay, Ain for salt, Maine et Loire for schist, and Deux-Sevres for granite. An underground research laboratory has also been established in a former uranium mine, near the town of Limoges, in order to study the properties and behavior of fractured granite. In 1990, a recommendation will be made to select a new site for an underground laboratory to validate the selection of a final repository site. If the site proves to be suitable, the repository could become operational by the end of this century and start receiving transuranic wastes by the year 2000 and high-level wastes by 2010.

4.6.4 Germany - Federal Republic

Germany has adopted an approach which considers spent fuel reprocessing, interim storage, and direct disposal in deep geological formations. The Federal government, Physical Technical Institute (PTB), is responsible for the design and operational management of the future repository. The German program is based on three sites each with a specific complementary role in the overall radioactive waste management program. The sites are the Gorleben, Asse, and Konrad. All three are located in the northeast part of the country.

All radioactive wastes are currently being stored, some with further processing, until the high-level wastes repository becomes operational. In the interim, some wastes may eventually be stored at the Gorleben facility. The licensing process of the Gorleben site is currently on hold pending current litigations. Another storage facility is planned at the Ahaus site. As with the Gorleben site, work at Ahaus has also stopped, pending resolution of legal disputes. Research on repository design has been underway since 1965 at a former salt mine in Asse. The Asse site, until 1978, served as a prototype repository for the disposal of 125,000 containers of low-level and smaller quantities (1,300 drums) of intermediate-level radioactive wastes. In the future low-level and intermediate-level radioactive wastes will be disposed at the Konrad iron-ore mine located at a depth of 1,000 meters in a shale formation.

For high-level wastes and heat-producing wastes, a repository is planned at the Gorleben site in a salt dome beginning at a depth of 250 meters and extending down to 3,000 meters. The geology of the site has been widely investigated by exploratory drilling and by geophysical measurements. In 1986, the construction of an underground laboratory research laboratory was initiated. In 1987, all work was stopped for over one year because of a fatality. Pending completion of the site characterization studies in the mid 1990s, the construction of the repository could start at the turn of the century. Once opened, the facility is planned to be remain operational for as long as 60 years.

4.6.5 Belgium

The Belgian research and development program to establish a radioactive waste repository was initiated in 1974. A national agency, ONDRAF, was established to take the responsibility for implementing and managing a multi-year national program. The Belgian waste management program includes spent fuel reprocessing in domestic and foreign facilities, interim storage at reactor sites, long-term (50-75 years) storage of solidified wastes, and eventual siting and construction of a repository located in a deep clay formation.

In 1975, the Mol-Dessel site was investigated as a candidate site for the Belgian radioactive waste repository. The Mol-Dessel site is situated in a deep clay formation and is the only suitable geological strata identified in Belgium. By 1980, a repository conceptual design was developed for a clay site. By 1985, an underground research laboratory, located at Mol-Dessel in the Boom Clay Formation, was declared operational. The underground laboratory is located a depth of 224 meters and since 1987, a new experimental gallery has been added to the original facility. The purpose of the additional gallery is to conduct high specific-activity disposal experiments and pilot studies (HADES Project). The studies include experiments in corrosion properties of containers and engineered barriers, geochemistry and radionuclide migration, backfilling and sealing technology, and near-field effects of heat and radiation on clays. Based on the outcome of these studies, a larger underground facility will be constructed for a full-scale demonstration project.

The proposed schedule currently considers the continuation of the HADES experiments until 1995; the preparation of a repository safety assessment demonstration report before the turn of the century; the conduct of a full scale demonstration project over several years; and the initiation of operation of the repository by about the year 2025.

4.6.6 Switzerland

The responsibility for establishing a radioactive waste disposal facility has been assigned to the National Cooperative for the Storage of Radioactive Waste (NAGRA). All radioactive wastes will be subjected to the necessary processing before final emplacement in deep geological formations. Because of the country's high population density, shallow-land disposal concepts are not being considered, even for short-lived radionuclides. Two types of waste repositories are being evaluated - one for high-level wastes and the other for radioactive wastes with short-lived radionuclides. Two options are also being considered for transuranic wastes (TRU), one in which TRU wastes would be buried together with HLW

wastes, and another in which they would be buried in a separate facility (e.g., cavern) at the site of the short-lived waste disposal repository.

For the HLW/TRU repository, crystalline rock and impermeable clays or other types of sediments are being considered as host media. In 1985, Project Gewähr was initiated, during which a system of tunnels and silos were mined in crystalline rocks at a depth of 1,200 meters. This site, located in northern Switzerland, was selected as the reference repository. In addition to these activities, investigative techniques and equipment are being tested in an underground laboratory at the Grimsel Pass Test Site. The Grimsel test site is also in crystalline rock and is located in the central part of the country. Since 1987, NAGRA has also been evaluating disposal sites in other types of geological media, which include sedimentary rock. The results of these site investigations will be integrated by 1992, after which time a decision will be made to select one candidate host site and to initiate detailed site characterization studies. Commissioning of the repository is not scheduled to occur before 2020. In the meantime, waste management practices will include interim storage and spent fuel reprocessing. Disposal in foreign countries is also being considered.

4.6.7 Sweden

Following a 1980 national referendum, the Swedish Parliament has decided to phase out nuclear power plants by the year 2010. Consequently, the Swedish Nuclear Fuel and Waste Management Company (SKB) is considering interim storage followed by direct disposal of spent fuel without reprocessing.

The interim storage facility (CLAB), commissioned in 1985, is located in underground granite cavern, at a depth of 30 meters. The CLAB site is located near an existing nuclear power plant (Oskarshamn). The facility's storage capacity will be increased to accommodate the nation's entire inventory of spent fuel, about 8,000 metric tons.

A final repository concept was proposed in 1984. Supporting research was initiated in 1976 and an underground research laboratory was established in 1977, known as the Stripa Project, which is expected to conclude by 1991. Based on the evaluation of the results of the Stripa project, a second generation underground laboratory may be built. It is anticipated that the second underground laboratory, the Swedish Hard Rock Laboratory, will become operational in 1993 if the pre-selected site also proves to be acceptable for a repository. This site will be located in granite at a depth of 400 to 500 meters. Eventually, spent fuel will be disposed of in vertical holes located in a granite or gneiss formation at a depth of about 500 meters. The spent fuel will be contained in high-integrity copper canisters. The individual holes holding canisters will be then be backfilled with natural materials and sealed with engineered barriers. All up front site characterization and engineering studies will be completed by the turn of the century, at which point a license application will be filed for the proposed site. The construction of the repository is expected to start in 2010 and waste emplacement activities are envisioned to begin in 2020.

In Japan, the guidelines and requirements for siting and operating a radioactive waste repository were established in 1985 by the Japanese Atomic Energy Commission. Under the guidelines of a five-year plan announced by the Science and Technology Agency, the Power Reactor and Nuclear Fuel Development Corporation has been given the responsibility of performing in situ experiments, while the Japanese Atomic Energy Research Institute is responsible for developing the regulatory and performance assessment criteria. The current waste management strategy includes spent fuel reprocessing using domestic and foreign facilities, on-site spent-fuel storage, waste solidification followed by long term storage (30-50 years), and eventual disposal in a deep suitable geological formation.

Several in situ experimental tests have been conducted since 1978 in galleries of existing mines and in quarry chambers. The tests have targeted granite, gabbro, diabase, and tuff formations. These field tests were not intended to characterize potential sites, rather they were conducted for characterizing and understanding near-field and short-term phenomena in rock formations. Other test sites have included the Hosokura Mine in tuff, the Tono Site in a uranium mine, Akenobe and Shimokawa mines in gabbro, and granite at the Kasama site. The in-situ tests have been conducted at a depth of 340 meters below ground surface and 186 meters below sea level. The purpose of these tests includes the development and validation techniques and methodologies for site selection, site characterization, conduct of various experiments, facility model validation, and development of facility design, construction, and operation.

More extensive in-situ tests are planned in underground research facilities located in crystalline and sedimentary rock. A diabase site in northern Hokkaido (Horonobe) has been selected following additional exploratory drilling and site investigations; these activities are scheduled to begin in 1989. The development of regulatory requirements for siting a geologic repository is not scheduled until after the turn of the century and the operation of a repository is not contemplated before 2030.

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Chapter 5: RADIATION DOSIMETRY

5.1 INTRODUCTION

The setting of standards for radionuclides requires an assessment of the doses received by individuals who are exposed by coming into contact with radiation sources. Two forms of potential radiation exposures can occur from these sources --internal and external. Internal exposures can result from the inhalation of contaminated air or the ingestion of contaminated food or water. External exposures can occur when individuals are immersed in contaminated air or water or are standing on contaminated ground surfaces. Internal or external doses can result from radionuclides at the site area or from radionuclides that have been transported from these sites to other locations in the environment. The quantification of the doses received by individuals from these radiation exposures is called radiation dosimetry. This chapter highlights the internal and external dosimetric models used by EPA to assess the dose to individuals exposed to radionuclides.

The models for internal dosimetry consider the quantity of radionuclides entering the body, the factors affecting their movement or transport through the body, and the energy deposited in organs and tissues from the radiation that is emitted during spontaneous decay processes. The models for external dosimetry consider only the photon doses to organs of individuals who are immersed in air or are exposed to a contaminated ground surface. In addition, the uncertainties associated with each model will be discussed.

5.2 BASIC CONCEPTS

Radioactive materials produce radiation as their constituent radioactive nuclides undergo spontaneous radioactive decay. The mechanisms of emitting this energy are characteristic of the decay process and include energetic charged particles (alpha and beta particles) and photons (gamma rays and x-rays). Alpha particles are nuclei of helium atoms and carry a positive charge two times that of an electron. These particles can produce dense ionization tracks in the biological material that they traverse. Beta particles are electrons or positrons emitted in radioactive decay. Their penetration power in material is greater than that of alpha particles. Gamma and x-rays are electromagnetic radiation and are distinguishable from alpha and beta particles by their greater penetrating power in material.

This section introduces some terminology used in Chapters 5 and 6 to describe internal and external dosimetry. For a more detailed explanation, the reader is referred to reports published in this area by the International Commission on Radiation Units and Measurements (ICRU80), International Commission on Radiological Protection (ICRP84), and National Council on Radiation Protection and Measurements (NCRP71).

5.2.1 Activity

The activity of a sample of any radionuclide of species, i , is the rate at which the unstable nuclei spontaneously decay. If N is the number of unstable nuclei present at a certain time, t , its activity, $A_i(t)$, is given by

$$A_i(t) = -dN/dt = \lambda_i^R N, \quad (5-1)$$

where λ_i^R is the radioactive decay constant. The customary unit of activity is the curie (Ci); its submultiples, the millicurie (mCi), the microcurie (μ Ci), and the picocurie (pCi), are also often used. The curie, which is defined as 3.7×10^{10} disintegrations per second, is the approximate activity of 1 gm of radium-226.

The time variation of the activity can be expressed in the form:

$$A_i(t) = A_{oi} \exp(-\lambda_i^R t). \quad (5-2)$$

A_{oi} is the activity of nuclide i at time $t=0$. For a sample of radioactive material containing more than one radionuclide, the total activity is determined by summing the activities for each radionuclide:

$$A(t) = \sum_i A_i(t) \quad (5-3)$$

5.2.2 Radioactive Half-Life

From the above equations, it is apparent that the activity exponentially decays with time. The time when the activity of a sample of radioactive material containing species i becomes one-half its original value (i.e., the time t that $A_i(t) = A_{oi}/2$) is called its radioactive half-life, T_i^R , and is defined as:

$$T_i^R = (\ln 2) / \lambda_i^R \quad (5-4)$$

The unit for the radioactive half-life is any suitable unit of time such as seconds, days, or years. The specific activity of a radionuclide (the activity per unit mass) is inversely proportional to the half-life.

5.2.3 Radionuclide Chains

Radionuclides decay either to stable atoms or to other radioactive species called daughters. For some species, a decay chain of daughter products may be produced until stable atoms are formed. For example, strontium-90 decays by emitting a beta-particle, producing the daughter yttrium-90, which also decays by beta emission to form the stable atom zirconium-90:



5.2.4 Biological Half-Life

The biological half-life of radionuclides is the time required for biological tissues to eliminate one-half of the activity by elimination processes. This time is the same for both stable and radioactive isotopes of any given element.

5.2.5 Internal and External Exposures to Radionuclides

The term "exposure," in the context of this report, denotes physical interaction of the radiation emitted from the radioactive material with cells and tissues of the human body. An exposure can be "acute" or "chronic" depending on how long an individual or organ is exposed to the radiation. Internal exposures occur when radionuclides, which have entered the body through the inhalation or ingestion pathway, deposit energy in organ tissues from the emitted gamma, beta, and alpha radiation. External exposures occur when radiation enters the body directly from sources located outside the body, such as radiation from material on ground surfaces, dissolved in water, or dispersed in the air.

In general, for sources of concern in this report, external exposures are from material emitting gamma radiation. Gamma rays are the most penetrating of the emitted radiations, and external gamma ray exposure may contribute heavily to radiation doses to the internal organs. Beta and alpha particles are far less penetrating and deposit their energy primarily in the skin's outer layer. Consequently, their contribution to the absorbed dose to the total body, compared to that deposited by gamma rays, is negligible and will not be considered in this report.

5.2.6 Absorbed Dose and Absorbed Dose Rate

The radiological quantity absorbed dose, D , denotes the mean energy imparted $\Delta\bar{\epsilon}$, by ionizing radiation to a small finite mass of organ tissue with a mass, Δm , and is expressed as

$$D = d\bar{\epsilon}/dm = \lim_{\Delta m \rightarrow 0} (\Delta\bar{\epsilon}/\Delta m). \quad (\text{rad}) \quad (5-6)$$

Internal and external exposures from radiation sources are not usually instantaneous but are distributed over extended periods of time. The resulting time rate of change of the absorbed dose to a small volume of mass is referred to as the absorbed dose rate, \dot{D}

$$\dot{D} = dD/dt = \lim_{\Delta t \rightarrow 0} (\Delta D/\Delta t). \quad (\text{mrad/y}) \quad (5-7)$$

The customary unit of absorbed dose rate is any quotient of the rad (or its multiple or submultiple) and a suitable unit of time. In this report, absorbed dose rates are generally given in mrad/yr.

5.2.7 Linear Energy Transfer (LET)

Linear energy transfer, L_{∞} , is the loss of kinetic energy, by collision, by charged particles per unit length of an absorbing medium. The increment of the mean energy lost, ΔE , to tissue by a charged particle of specified energy in traversing a distance, ΔX :

$$L_{\infty} = dE/dX = \lim_{\Delta X \rightarrow 0} (\Delta E/\Delta X) \quad (\text{keV } \mu\text{m}^{-1}) \quad (5-8)$$

For photons, L_{∞} represents the energy imparted by the secondary electrons (electrons that are knocked out of their orbitals by primary radiation) resulting from secondary interactions between the photons and tissue material. High-LET radiation (alpha particles) imparts more energy per unit length of organ tissue than does low-LET radiation (x-rays, gamma rays, and beta particles). Consequently, the former are more effective per unit dose in causing biological damage.

5.2.8 Dose Equivalent and Dose Equivalent Rate

Dose equivalent is a special radiation protection quantity that is used to express the absorbed dose in a manner that considers the difference in biological effectiveness of various kinds of ionizing radiation. The ICRU has defined the dose equivalent, H , as the product of the absorbed dose, D , the quality factor, Q , and all other modifying factors, N , at the point of interest in biological tissue (ICRU80). This relationship can be expressed in the following manner:

$$H = D Q N. \quad (\text{rem}) \quad (5-9)$$

The quality factor is a dimensionless quantity that depends on the collision stopping power for charged particles. It accounts for the differences in biological effectiveness found among varying types of radiation. By definition, it is independent of tissue and biological endpoint. The generally accepted values for quality factors for high- and low-LET radiation, which are used by EPA, are given in Table 5-1. The product of all other modifying factors, N , such as dose rate, fractionation, etc., is taken as 1.

Table 5-1. Quality factors for various types of radiation (ICRP77).

Radiation Type	Quality Factors (Q)
x-rays, gamma rays, and electrons	1
alpha particles	20

The dose equivalent rate, \dot{H} , is the time rate of change of the dose equivalent to organs and tissues and is expressed as:

$$H = dH/dt = \lim_{\Delta t \rightarrow 0} (\Delta H/\Delta t). \quad (\text{mrem/yr}) \quad (5-10)$$

5.2.9 Effective Dose Equivalent and Effective Dose Equivalent Rate

The ICRP has defined the effective dose equivalent, H_E , as:

$$H_E = \sum_T w_T H_T, \quad (\text{rem}) \quad (5-11)$$

where H_T is the dose equivalent in tissue and w_T is the weighting factor, which represents the estimated proportion of the stochastic risk resulting from tissue, T , to the stochastic risk when the whole body is uniformly irradiated (ICRP77). The weighting factors recommended by the ICRP are listed in Table 5-2.

Table 5-2. Weighting factors recommended by the ICRP for stochastic risks (ICRP77).

Organ or Tissue	w_T
Gonads	0.25
Breast	0.15
Red Bone Marrow	0.12
Lung	0.12
Thyroid	0.03
Bone Surfaces	0.03
Remainder	0.30

The effective dose equivalent rate is the time rate of the delivery of the dose equivalent and is expressed as \dot{H}_E , where:

$$\dot{H}_E = \sum_T w_T \dot{H}_T. \quad (\text{mrem/yr}) \quad (5-12)$$

5.2.10 Relationship of the Dose Equivalent and the Effective Dose Equivalent to Risk

The dose equivalent was introduced by the ICRP to allow one to combine and compare - on the basis of biological effects - absorbed doses of different types of radiation. Subsequently, the effective dose equivalent was introduced to provide a single-valued indicator of risk for dose equivalents distributed nonuniformly in the body. By convention, these concepts, in combination with the ICRP-recommended quality factors and organ-weighting factors, are widely used in radiation protection. These recommended factors, however, are based on dose response models that differ significantly from those used by EPA to estimate risk (see Chapter 6).

To calculate risk, EPA first calculates age-specific, high- and low-LET absorbed dose rates, by organ, for a uniform intake or external exposure rate. The risk from each year's

dose is then calculated using the life table procedure in conjunction with age- and organ-specific risk models adapted from the BEIR III report (NAS80).

These models (see Chapter 6) assume a linear dose-response relationship and a lifetime relative risk projection for cancers other than bone cancer and leukemia, for which absolute risk projection is employed. Finally, the risks from each year's dose are summed to arrive at the risk from lifetime exposure.

In calculating dose equivalents and effective dose equivalents, the ICRP Publication 30 convention was employed, including the same quality factors and organ-weighting factors. Nevertheless, in calculating the risk from a given absorbed dose of alpha particle irradiation, RBEs of 8 and 2.7 were used for the induction of cancers and genetic effects, respectively (see Chapter 6). Since these RBEs are lower than the assumed alpha quality factor ($Q=20$), EPA's estimates of the risk per unit dose equivalent (mrem) will be lower for alpha particles than for x-rays or gamma rays. Likewise, the ICRP organ-weighting factors shown in Table 5-2 do not stand in the same proportion as the organ risks calculated using the EPA models for cancer induction or genetic mutations. Furthermore, EPA considers somatic and genetic risks separately. Thus, even if attention was restricted to low-LET radiation, the estimated risk from a given effective dose equivalent will vary, depending on how the absorbed dose is distributed within the body.

To summarize, because EPA risk models differ from those underlying the ICRP recommendations, the risks calculated directly by EPA are not strictly proportional to the effective dose equivalents derived using ICRP quality factors and organ weighting factors.

5.2.11 Working Levels and Working Level Months

The working level is a unit that has been used as a measure of the radon decay-product activity in air. It is defined as any combination of short-lived radon daughters (though polonium-214) per liter of air that will result in the ultimate emission of 1.3×10^5 MeV of alpha energy. An activity concentration of 100 pCi/L of radon-222 in equilibrium with its short lived daughters gives rise to a potential alpha-energy concentration of approximately 1 WL. The WL unit could also be used for thoron daughters. The potential alpha energy exposure is commonly expressed in units of working level month (WLM). One WLM corresponds to an exposure to a concentration of 1 WL for the commonly used reference period of 170 hours.

5.2.12 Customary and SI Units

The relationship between the customary units used in this text and the international system of units (SI) for radiological quantities is shown in Table 5-3. While the SI radiological units are almost universally used in other countries for radiation protection regulation, the United States has not yet officially adopted their use for such purposes.

Table 5-3. Comparison of customary and SI special units for radiation quantities.

Quantity	Customary Unit		Special SI Unit	
	Name	Definition	SI Unit	Definition
Activity (A)	Curie (Ci)	$3.7 \times 10^{10} \text{ s}^{-1}$	becquerel (Bq)	1.0 s^{-1}
Absorbed dose (D)	rad	$10^{-2} \text{ J kg}^{-1}$	gray (Gy)	1.0 J kg^{-1}
Dose equivalent (H)	rem	$10^{-2} \text{ J kg}^{-1}$	sievert (Sv)	1.0 J kg^{-1}
Linear energy transfer (L_{∞})	keV μm^{-1} (kiloelectron volts per micrometer)	$1.602 \times 10^{-10} \text{ J m}^{-1}$		

5.3 EPA DOSIMETRIC MODELS

The EPA dosimetric models, to be discussed in the following sections, have been described in detail in previous publications (Du80, Su81). Information on the elements treated in these sections was taken directly from those documents or reports. In most cases, the EPA models are similar or identical to those recommended by the ICRP (ICRP79, ICRP80, ICRP81). However, differences in model parameters do exist for some radionuclides (Su81). The basic physiological and metabolic data used by EPA in calculating radiation doses are taken from ICRP reports (ICRP75, ICRP79).

5.3.1 Internal Dose Models

EPA implements contemporary models to estimate absorbed dose rates as a function of time to specified organs in the body. Estimates of the doses resulting from the deposition and retention of inhaled particulates in the lung and their subsequent absorption into the blood and clearance into the gastrointestinal (GI) tract are made using the ICRP Task Group Lung Model (ICRP66).

5.3.1.1 Generalized Scheme for Estimating Organ Absorbed Dose Rates

5.3.1.1.1 Distribution of Activity of Radionuclides in the Body

The complex behavior of radionuclides is simplified conceptually by considering the body as a set of compartments. A compartment may be any anatomical, physiological, or physical subdivision of the body throughout which the concentration of a radionuclide is assumed to be uniform at any given time. The terms "compartment" and "organ" are often used interchangeably, although some of the compartments considered in this report may represent only portions of a structure usually considered to be an organ, while some compartments may represent portions of the body usually not associated with organs. Examples of compartments used in this report are the stomach, the pulmonary region of the lung, the blood, or the bone. Within a compartment, there may be more than one "pool" of activity. A pool is defined to be any fraction of the activity within a compartment that has a biological half-life which is distinguishable from the half-time(s) of the remainder of activity within the compartment.

Activity entering the body by ingestion is assumed to originate in the stomach compartment; activity entering through inhalation is assumed to originate in a compartment within the lung (the tracheo-bronchial, pulmonary, or naso-pharyngeal region). From the stomach, the activity is viewed as passing in series through the small intestine, the upper large intestine, and the lower large intestine, from which it may be excreted. Also, activity reaching the small intestine may be absorbed through the wall into the bloodstream, from which it may be taken in parallel into any of several compartments within the skeleton, liver, kidney, thyroid, and other organs and tissues.

The list of organs or regions for which dose rates are calculated is found in Table 5-4. Activity in the lung may reach the bloodstream either directly or indirectly through the stomach or lymphatic system. The respiratory system and gastrointestinal tract models are discussed further in later sections. Figure 5-1 illustrates the EPA model used to represent the movement of radioactivity in the body.

EPA models separately consider the intake and subsequent behavior of each radionuclide in the body. The models also allow for the formation of radioactive decay products within the body, and it is assumed that the movement of internally produced radioactive daughters is governed by their own metabolic properties rather than those of the parent. This contrasts the ICRP assumption that daughters behave exactly as the parent.

Table 5-4. Target organs and tissues used for calculating the ICRP effective dose equivalent and the EPA cancer risk.

ICRP effective dose equivalent	EPA cancer risk
Ovaries	
Testes	
Breast ^a	Breast
Red marrow	Red marrow
Lungs ^b	Pulmonary lung ^c
Thyroid	Thyroid
Bone surface	Bone surface (endosteum)
Stomach wall	Stomach wall
Small intestine wall	Intestine ^d
Upper large intestine wall	
Lower large intestine wall	
Kidneys	Kidneys
Liver	Liver
Pancreas	Pancreas ^e
Brain	
Spleen	
Thymus	
Uterus	
Adrenals	
Bladder wall	

a) Dose to breast is assumed to equal dose to muscle.

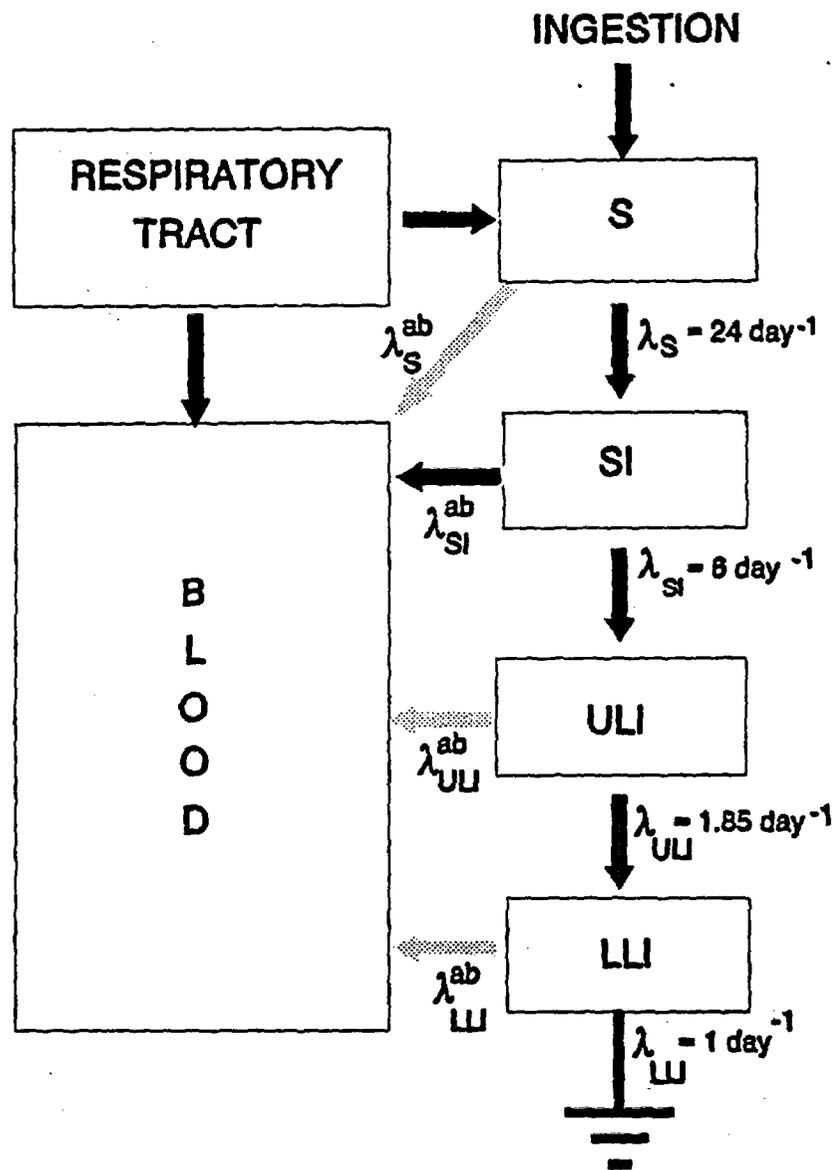
b) The ICRP considers the lungs to be a composite of the trachiobronchial region, pulmonary region, and the pulmonary lymph nodes with a combined mass of 1,000 g (ICRP79).

c) The EPA calculates lung cancer risk on the basis of the dose to the pulmonary lung. The mass of this region, which does not include venous or arterial blood, is considered to be 570 g.

d) The EPA averages the values for the small, upper large, and lower large intestine using weights of 0.2, 0.4, and 0.4 respectively for calculating the risk of bowel cancer.

e) The pancreas is also used as a surrogate organ for calculating the cancer risk for all other organs and tissues.

Figure 5-1. A schematic representation of radioactivity movement among respiratory tract, gastrointestinal tract, and blood.



S = stomach
 SI = small intestine
 ULI = upper large intestine
 LLI = lower large intestine
 λ = elimination rate constant

If $A_{ik}(t)$ denotes the activity of the i th species of the chain in organ k and if that activity is divided among several "pools" or "compartments" indexed by subscript l , then the time rate of change of activity can be modeled by a system of differential equations of the following form:

$$A_{ik} = -(\lambda_i^R + \lambda_{ik}^B) A_{ik} + C_{ik} (\lambda_i^R \sum_{j=1}^{l-1} B_{ij} \sum_{r=1}^{L_k} A_{jr} + P_{ik})$$

$$l = 1, \dots, L_{ik} \quad (5-14)$$

where compartment l is assumed to have L_{ik} separate pools of activity, and where:

- A_{ik} = the activity of species i in compartment l of organ k ;
- λ_i^R = $(\ln 2) / T_i^R$ where T_i^R = radioactive half of species i ;
- λ_{ik}^B = rate coefficient (time^{-1}) for biological removal of species i from compartment l of organ k ;
- L_{ik} = number of exponential terms in the retention function for species i in organ k ;
- B_{ij} = branching ratio of nuclide j to species i ;
- P_{ik} = inflow rate of the i^{th} species onto the organ k ; and
- C_{ik} = the fractional coefficient for nuclide i in the l^{th} compartment of organ k .

The subsystem described by these L_{ik} equations can be interpreted as a biological compartment in which the fractional retention of radioactive species is governed by exponential decay. Radioactivity that enters an organ may be lost by both radioactive decay and biological removal processes. For each source organ, the fraction of the initial activity remaining at any time after uptake at time $t = 0$ is described by a retention function consisting of one or more exponentially decaying terms:

$$R_{ik}(t) = \sum_{l=1}^{L_{ik}} c_{ik} \exp[-(\lambda_i^R + \lambda_{ik}^B)t] \quad (5-15)$$

The subscript l in the above equation represents the l^{th} term of the retention function, and the coefficients c_{ik} can be considered as "pathway fractions."

5.3.1.1.2 Dose Rates to Target Organs

The activity of a radionuclide in a compartment is a measure of the rate of energy being emitted in that compartment, at any time, t , and can be related to the dose rate to a specific organ at that time. This requires estimating the fraction of the energy emitted by the decay of the radionuclide in each compartment that is absorbed by the specific organ.

The absorbed dose rate, $\dot{D}(X;t)$ to target organ X at time t due to radionuclide species i in source organs Y_1, Y_2, \dots, Y_M is estimated by the following equation:

$$D_i(X;t) = \sum_{k=1}^M D_i(X \leftarrow Y_k;t) \quad (5-16)$$

where: $D_i(X \leftarrow Y_k;t) = S_i(X \leftarrow Y_k) A_{ik}(t)$; and $A_{ik}(t)$ is the activity, at time t of species i in source organ Y_k ; $S_i(X \leftarrow Y_k)$, called the S-factor, represents the average dose rate to target organ X from one unit of activity of the radionuclide uniformly distributed in source organ or compartment Y_k . It is expressed in the following manner:

$$S_i(X \leftarrow Y_k) = c \sum_m f_m E_m \phi_m(X \leftarrow Y_k) \quad (5-17)$$

where:

c = a constant that depends on the units of dose, energy, and time being used;

f_m = intensity of decay event (number per disintegration);

E_m = average energy of decay event (Mev); and

$\phi_m(X \leftarrow Y_k)$ = specific absorbed fraction, i.e., the fraction emitted energy from source organ Y_k absorbed by target organ X per gram of X ,

where the summation is taken over all events of type m . The units for S-factors depend on the units used for activity and time; thus, the S-factor units may be rad/Ci-day. The S-factor is similar in concept to the SEE factor (specific effective energy) used by the ICRP Committee 2 in Publication 30. However, the SEE factor includes a quality factor for the type of radiation emitted during the transformation.

The above equations are combined to produce the following expressions for the absorbed dose rates to target organs at any time due to one unit of activity of radionuclide species, i , uniformly distributed in source organs $Y_1 \dots Y_k$:

$$D(X;t) = \sum_k \sum_m A_{ik}(t) S_{im}(X \leftarrow Y_k) \quad (5-18)$$

The corresponding dose equivalent rate, $H_i(X;t)$, can be estimated by inclusion of the quality factor, Q_m , and the modifying factor, $N_m(Y_i)$:

$$H_i(X;t) = \sum_i \sum_m A_{i,m}(t) Q_m N_m(Y_i) S_m(X \leftarrow Y_i) \quad (5-19)$$

Implicit in the above equations is the assumption that the absorbed dose rate to an organ is determined by averaging absorbed dose distributions over its entire mass.

Alpha and beta particles are usually not sufficiently energetic to contribute a significant cross-irradiation dose to targets separate from the source organ. Thus, the absorbed fraction for these radiations is generally assumed to be just the inverse of the mass of organ X, or if the source and target are separated, then $\phi_m(X \leftarrow Y) = 0$. Exceptions occur when the source and target are in very close proximity, as is the case with various skeletal tissues. Absorbed fractions for cross-irradiations by beta particles among skeletal tissues were taken from ICRP Publication 3 (ICRP80). The energy of alpha particles and their associated recoil nuclei is generally assumed to be absorbed in the source organ. Therefore, $\phi_m(X \leftarrow X)$ is taken to be the inverse of the organ mass, and $\phi_m(X \leftarrow Y) = 0$ if X and Y are separated. Special calculations are performed for active marrow and endosteal cells in bone, based on the method of Thorne (Th77).

5.3.1.1.3 Monte Carlo Methodology to Estimate Photon Doses to Organs

The Monte Carlo method uses a computerized approach to estimate the probability of photons interacting within target organ X after emission from source organ Y. The method is carried out for all combinations of source and target organs and for several photon energies. The body is represented by an idealized phantom in which the internal organs are assigned masses, shapes, positions, and attenuation coefficients based on their chemical composition. A mass attenuation coefficient, μ_o , is chosen, where μ_o is greater than or equal to the mass attenuation coefficients for any region of the body. Photon courses are simulated in randomly chosen directions, and potential sites of interactions are selected by taking distances traversed by them as $-\ln r/\mu_o$, where r is a random number distributed between 0 and 1. The process is terminated when either the total energy of photons has been deposited or the photon escapes from the body. The energy deposition for an interaction is determined according to standard equations (ORNL74).

5.3.1.1.4 Effects of Decay Products

In calculating doses from internal and external exposures, the in-growth of radioactive decay products (or daughters) must be considered for some radionuclides. When an atom undergoes radioactive decay, the new atom created in the process, which may also be radioactive, can contribute to the radiation dose to organs or tissues in the body. Although

these decay products may be treated as independent radionuclides in external exposure, the decay products of each parent must be followed through the body in internal exposure situations. The decay product contributions to the absorbed dose rates, which are included in EPA calculations, are based on the metabolic properties of the individual daughters and the organ in which they occur.

5.3.1.2 Inhalation Dosimetry - ICRP Respiratory Tract Model

As stated earlier, individuals immersed in contaminated air will breathe radioactive aerosols or particulates, which can lead to doses to the lung and other organs in the body. The total internal dose caused by inhalation of these aerosols can depend on a variety of factors, such as breathing rates, particle sizes, and physical activity. Estimating the total dose to individuals over a specific time period requires specifying the distribution of particle depositions in the respiratory tract and the mathematical characteristics of the clearance parameters. The EPA currently uses assumptions established by the ICRP Task Group on Lung Dynamics (TGLM)(ICRP66). This section will summarize the essential features of that model. For a more comprehensive treatment, the reader is referred to the actual report.

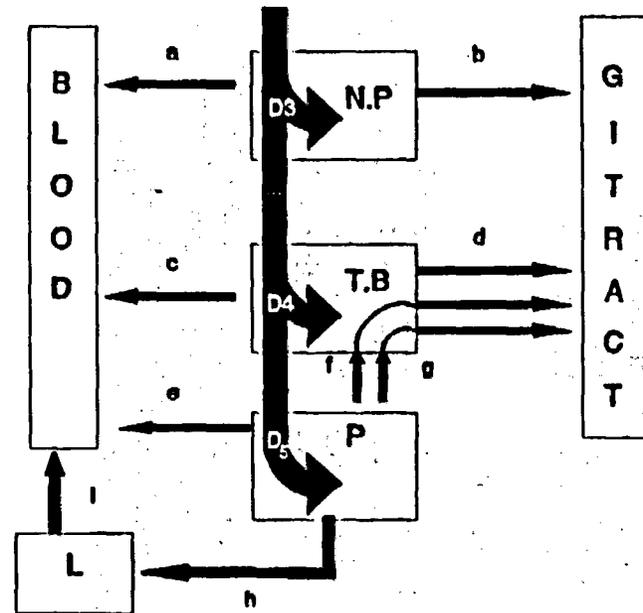
The basic features of the ICRP lung compartmental model are shown in Figure 5-2. According to this model, the respiratory tract is divided into four regions: naso-pharyngeal (N-P), tracheo-bronchial (T-B), pulmonary (P), and lymphatic tissues.

In the model, the regions N-P, T-B, and P are assumed to receive fractions D_3 , D_4 , and D_5 of the inhaled particulates, where the sum of these is less than 1 (some particles are removed by prompt exhalation). The values D_3 , D_4 , and D_5 depend on the activity median aerodynamic diameter (AMAD) of the inspired particles. For purposes of risk calculations, EPA uses AMADs of 1 micron. The lung model employs three clearance classes, D, W, and Y, corresponding to rapid, intermediate, and low clearance, respectively, of material deposited in the respiratory passages. The clearance class depends on chemical properties of the inhaled particles.

Like the ICRP, EPA assumes that the absorbed dose rate to the N-P region can be neglected. Unlike the ICRP, however, EPA averages the dose over the pulmonary region of the lung (compartments e through h), to which is assigned a mass of 570 g, including

Figure 5-2. The ICRP Task Group lung model for particulates

COMPARTMENT		CLASS					
		D ₃		D ₄		D ₅	
		T	F	T	F	T	F
N - P (D ₃ - 0.30)	a	0.01	0.5	0.01	0.1	0.01	0.01
	b	0.01	0.5	0.4	0.9	0.4	0.99
T - B (D ₄ - 0.08)	c	0.01	0.95	0.01	0.5	0.01	0.01
	d	0.2	0.05	0.2	0.5	0.2	0.99
P (D ₅ - 0.25)	e	0.5	0.8	50	0.15	500	0.05
	f	n.a.	n.a.	1.0	0.4	1.0	0.4
	g	n.a.	n.a.	50	0.4	500	0.4
L	h	0.5	0.2	50	0.05	500	0.15
	i	0.5	1.0	50	1.0	1000	0.9



The columns labeled D, W, and Y correspond, respectively, to rapid, intermediate, and slow clearance of the inspired material (in days, weeks, or years). The symbols T and F denote the biological half-time (days) and coefficient, respectively, of a term in the appropriate retention function. The values shown for D₃, D₄, and D₅ correspond to activity median aerodynamic diameter, AMAD = μm , and represent the friction of the inspired material depositing in the lung regions.

capillary blood (ICRP75). In addition, it is assumed that the total volume of air breathed in one day by a typical member of the general population is 22,000 liters. This value was determined by averaging the ICRP-23 adult male and female values based on 8 hours of working "light activity," 8 hours of nonoccupational activity, and 8 hours of resting.

5.3.1.3 Ingestion Dosimetry - ICRP GI Tract Model

According to the ICRP-30 GI tract model, the gastrointestinal tract consists of four compartments: the stomach (S), small intestine (SI), upper large intestine (ULI), and lower large intestine (LLI). The fundamental features of the model are shown in Figure 5-1. It is assumed that absorption into the blood occurs only from the small intestine (SI).

This model postulates that radioactive material entering the compartments of the GI tract is exponentially removed by both radioactive decay and biological removal processes, and that there is no feedback. Absorption of a particular nuclide from the GI tract is characterized by f_1 , which represents that fraction of the nuclide ingested which is absorbed into body fluids if no radiological decay occurs:

$$f_1 = \lambda_{SI}^{ab} / (\lambda_{SI}^{ab} + \lambda_{SI}) \quad (5-20)$$

where

λ_{SI}^{ab} = the absorption coefficient (s^{-1})

λ_{SI} = the transfer coefficient from the small intestine to the large intestine (s^{-1})

Figure 5-1 graphically presents the role of these coefficients in the gastrointestinal model. The kinetic model, as formulated by the ICRP, does not permit total absorption of a nuclide ($f_1 = 1$).

5.3.1.4 Dose Rate Conversion Factors

EPA uses the computer code RADRISK (Du80) for calculating radiation doses and risks to individuals resulting from a unit intake of a radionuclide, at a constant rate, for a lifetime exposure (50-yr dose commitment). These calculations are done for the inhalation and ingestion pathways to individuals who are exposed by immersion in contaminated air or by contaminated ground surfaces.

RADRISK computes doses for both chronic and acute exposures. Following an acute intake, it is assumed the activity in the body decreases monotonically, particularly for radionuclides with rapid radiological decay rates or rapid biological clearance. In the case of chronic exposure, the activity in each organ of the body increases monotonically until a steady state is achieved, at which time the activity remains constant. The instantaneous dose rates at various times after the start of chronic exposure provide a reasonably accurate (and conservative) estimate of the total annual dose for chronic exposure conditions. However, the instantaneous dose rates may err substantially in the estimation of annual dose from an acute exposure, particularly if the activity levels decrease rapidly.

Since the rate of change in activity levels in various organs is more rapid at early times after exposure, doses are computed annually for the first several years and for progressively longer periods thereafter, dividing by the length of the interval to estimate the average annual dose. This method produces estimates of risk that are similar to those computed by the original RADRISK methodology for chronic exposures and provides a more accurate estimate of the risks from acute intakes.

5.3.1.5 Special Radionuclides

The following paragraphs briefly summarize some of the special considerations for particular elements and radionuclides.

5.3.1.5.1 Tritium and Carbon-14

Most radionuclides are nuclides of elements found only in trace quantities in the body. Others like tritium (hydrogen-3) or carbon-14 must be treated differently since they are long-lived nuclides of elements that are ubiquitous in tissue. An intake of tritium is assumed to be completely absorbed and to be rapidly mixed with the water content of the body (Ki78a).

The estimates for inhalation include consideration of absorption through the skin. Organ dose estimates are based on the steady-state specific-activity model described by Killough et al. (Ki78a).

Carbon-14 is assumed to be inhaled as CO₂ or ingested in a biologically bound form. Inhaled carbon-14 is assumed to be diluted by stable carbon from ingestion (Ki78b). This approach allows separate consideration of the ingestion and inhalation pathways. The specific-activity model used for organ dose estimates is also that of Killough et al. (Ki78a). Short-lived carbon radionuclides (e.g., carbon-11 or carbon-15) are treated as trace elements, and the organ doses are calculated accordingly.

5.3.1.5.2 Noble Gases

Retention of noble gases in the lungs is treated according to the approach described by Dunning et al. (Du79). The inhaled gas is assumed to remain in the lungs until lost by radiological decay or respiratory exchange. Translocation of the noble gas to systemic organs is not considered, but doses due to translocated decay products produced in the lungs are calculated. The inhalation of the short-lived decay products of radon is assessed using a potential alpha energy exposure model (see Chapter 6) rather than by calculating the doses to lung tissues from these radionuclides.

5.3.1.5.3 Uranium and Transuranics

The metabolic models for transuranics elements (polonium, neptunium, plutonium, americium, and curium) are consistent with those used for the EPA transuranic guidance (EPA77). A GI-tract-to-blood absorption factor of 10⁻³ is used for the short-lived nuclides of plutonium (plutonium-239, -240, and -242), while a value of 10⁻⁴ is used for other transuranics. For soluble forms of uranium, a GI tract to blood absorption factor of 0.2 is

used in accordance with the high levels of absorption observed for low-level environmental exposures (Hu73, Sp73).

5.3.1.6 Uncertainties in Internal Dose Estimates

Estimates of radiation dose in risk assessment studies have traditionally been based on dosimetric models derived in the context of radiation protection for adult workers. Despite the obvious differences between risk assessment and radiation protection, the dosimetric formulations of the latter have been generally adopted, often with no modifications, in risk assessment activities. This approach permits use of a substantial body of information assembled by international experts from the occupational setting and provides models that avoid the complex problems encountered in biokinetic modeling of radionuclides for the general public in an age-dependent sense. However, the continued use in risk assessment of dosimetric data derived for workers, which neglects organ-specific biokinetics and age dependence, is becoming increasingly difficult to justify. One major limitation of the current ad hoc dosimetric formulations is the great difficulty in making informed estimates of the uncertainties in the estimated dose.

All dosimetry models are inherently uncertain. At best, these models can only approximate real situations in organs and tissues in humans. Consequently, without extensive human data, the uncertainties associated with their use for risk assessment purposes is extremely difficult, and in some cases impossible, to quantify. However, consideration of their limitations in estimating doses to an average member of the general population is essential.

In applying the dosimetric models in current use, as discussed in the previous sections, the primary sources of uncertainty are attributed to ICRP model formulation and parameter variability produced by measurement error or natural variation. The purpose of this section is to provide a general but limited discussion of these sources and to introduce an uncertainty scheme for classifying radionuclides. The authors gratefully acknowledge Dr. Keith Eckerman of Oak Ridge Laboratory for discussions with respect to implementation of ICRP models and for guidance regarding the magnitude of uncertainties. However, the conclusions presented here are those of the Agency.

5.3.1.6.1 Uncertainties Due to ICRP Model Formulation

Uncertainty in calculations based on ICRP models arises primarily from five sources: (1) the uncertainty in the Reference Man data; (2) the uncertainty in the lung and GI-tract model describing the translocation and absorption of inhaled or ingested activity into the blood; (3) the uncertainty associated with the formulation of the ICRP Publication 30 biokinetic models describing the distribution and retention of the activity among the various organs in the body; (4) the uncertainty in the dose models to calculate the absorbed dose to organs from that activity; and (5) the uncertainty in the model parameters.

5.3.1.6.2

Reference Man Concept

To establish a degree of consistency in occupational dosimetry calculations, the ICRP developed the concept of Reference Man (ICRP75). Reference Man is a conceptual individual who has the anatomical and physiological characteristics of a healthy 20 to 30 year old male with a total body mass of 70-kg. The anatomical and physiological data of Reference Man have been embedded in many computational models for estimating organ doses and applied in radiation protection and in some calculations for medicine.

Although these data have been extensively applied in calculating doses, the approach in which Reference Man data is used to represent average individuals in a specific population introduces bias from the outset. The uncertainties in this approach are primarily due to age- and sex- specific differences in the anatomical and physiologic parameters. Biological and ethnic variability also contribute. In addition, the Reference Man data do not always represent data for a 70-kg man. Many of the data found in ICRP Publication 23 were from adults who had anatomical or physiological characteristics significantly different from those of a 70-kg man.

Due to the many parameters involved and the quality of the data available to define the numerical values, it is very difficult to establish the level of uncertainty in using Reference Man data to estimate doses to the average individual in the U.S. population. Furthermore, the Reference Man concept was not formulated so as to facilitate a quantitative analysis of the uncertainty in the dose estimates. Finally, Reference Man is not intended to be representative of the U.S. population.

5.3.1.6.3

ICRP Respiratory Tract Model

When individuals inhale radioactive aerosols, the dose to the lungs and other organs in the body depends primarily on how the aerosols are deposited in and cleared from the airways of the respiratory tract. Mechanisms involved in the deposition of inhaled aerosols and gases are affected by physical and chemical properties, including aerosol size distribution, density, shape, surface area, electrostatic charge, chemical composition and gas diffusivity and solubility. Deposition is also affected by respiratory physiology, morphometrics and pathology.

The ICRP modeling system assumes that deposition rates for aerosols in the respiratory tract are controlled primarily by three mechanisms: sedimentation, impaction and Brownian diffusion. The major uncertainties associated with the ICRP deposition models for the lungs are: (1) the uncertainty in the anatomical model of the respiratory tract, (2) the uncertainty in the effective aerodynamic diameter of the inhaled particles, (3) the uncertainty in the breathing patterns and rates, and (4) the questionable validity of the fluid dynamic models used for all exposure situations.

The number of particles deposited in the lung essentially depends on physiologic, morphometric and anatomical properties, such as airway dimensions and numbers, branching and gravitational angles of airways, and distances to the alveolar walls. The ICRP respiratory tract model (ICRP66) uses the anatomical model devised by Findeisen (Fi35) in

its dosimetric calculations. This model assumes that lung airways are rigid tubes with symmetric dichotomous branching patterns and that their morphometric properties are those of an adult male. In reality, however, the airways have circular ridges or longitudinal grooves (FRC67), and many airways, like the trachea, are irregular in shape (Br52). In addition, airways change in diameter and length during inspiration and expiration (Ho75, Hu72, Th78), which affects gravitational and branching angles (Ph85). Since many of these properties depend on age and sex, using the anatomic and morphometric lung properties of an adult male for estimating doses to other members of the population is likely to introduce considerable bias.

Clearance of particles from the respiratory tract depends on many factors, such as site of deposition, chemical composition, physical properties of the deposited material, and mucociliary transport rates. The uncertainties associated with using the values provided by the ICRP are due primarily to the sparseness of data on lung clearance mechanisms, in general, and secondarily to age, activity levels and general health status of the individual at the time of exposure. Furthermore, as stated earlier, most of the lung deposition data and models are derived from studies of healthy adults. Studies have shown, however, that children's lungs differ from adults' with respect to anatomical, physiological, and morphological properties. As a consequence, particle deposition in the respiratory tract is expected to be higher in children than in adults.

5.3.1.6.4 ICRP GI-Tract Model

The ICRP GI-tract model assumes that ingested material (radionuclides) moves in sequence through the stomach, small intestine, upper large intestine, and lower large intestine. The model depicts an exponential removal from each compartment, characterized by a single removal rate that depends only on the compartment. The model has no provision for addressing endogenous secretion. In addition, it is assumed that radionuclides are absorbed into the blood from the small intestine (SI).

Uncertainties arise when applying these assumptions to the estimation of doses to average individuals. Although radionuclides transported through the GI tract are primarily absorbed into the blood stream from the SI, fractions can be absorbed from the other compartments. Furthermore, the removal rates, which are model parameters, vary among different individuals in the population. Considerable differences can exist depending on the type of radionuclide ingested, its chemical form, the amount and composition of food in the stomach at the time of intake and other factors which vary because of nutritional status, age, and the sex of the individual. The f_1 factor, which represents the fraction of material absorbed from the SI, generally contributes the largest uncertainty in the GI tract model. This parameter will be discussed in a later section.

5.3.1.6.5 ICRP 30 Biokinetic Models

The ICRP biokinetic models were chosen to represent adult male members of the population. Uncertainties are associated with the approach because they do not account for differences in the metabolic behavior of radionuclides, which vary depending on age, sex, and dietary intakes of an individual at the time of exposure. In addition, many of the models

chosen for dosimetry calculations are based on very limited observational data that cannot be reliably applied across the population.

Below is a list of additional uncertainties associated with the ICRP biokinetic models:

- a) The models have been constructed largely from animal data in such a way that extrapolation to humans has no strong logical or scientific support.
- b) Doses to heterogeneously distributed radiosensitive tissues of an organ (e.g., skeletal and lung tissues) cannot be estimated accurately, since the actual movement of radionuclides in the body is not accurately tracked.
- c) Some radionuclides are assigned the model of an apparently related nuclide (e.g., americium, curium, neptunium are assigned the plutonium model) although differences in metabolism are known.
- d) The growth of radioactive daughters is often not handled realistically, and the format of the models makes it difficult to supply alternative assumptions.
- e) The models often yield inaccurate estimates of excretion even for the average adult.

5.3.1.6.6 ICRP Dose Models

ICRP models estimate doses to organs of the body by considering the distribution of the radioactivity and the interaction of radiation with cells and tissues in these organs. Estimates of the absorbed dose in a region (referred to as the target region) depend upon the spatial relationships of that region to the regions containing the radionuclide (referred to as source regions) and how the activity is distributed in the source region. For organs other than bone, it is assumed that the radionuclides are uniformly distributed in the source regions and that the radiosensitive cells of interest are uniformly distributed in the target region. However, this assumption may bias the dose estimates because of the nonuniformity of the activity that is normally found in human organs.

5.3.1.6.7 Uncertainties Due to Parameter Value Variability

Most discussions concerning the uncertainties in dose estimates focus on the uncertainty associated with model parameter values. These discussions assume that the ICRP metabolic and dose models are correct. The most important parameters of concern for dose assessment calculations are: radionuclide intake rates, organ masses, blood transfer factors, organ uptake rates, and biological half-times of radionuclides. Although parameter value variability can be attributed to measurement and sampling errors and natural biological variation, in many cases, age is the largest source of variability.

Depending on the type of radionuclide ingested, the age and element dependency in the metabolic and physiological processes determines how the dose to target organs varies with age. For example, strontium tends to follow the calcium pathways in the body and

deposits to a large extent in the skeleton. In fact, the fraction of ingested strontium eventually reaching the skeleton at a given age depends largely on the skeletal needs for calcium at that age, even though the body is able to discriminate somewhat against strontium in favor of calcium after the first few weeks of life.

Given the importance of age as a contributor to parameter variability in dose estimates, the possible age dependence in thyroid dose for chronic ingestion of a fixed iodine-131 concentration in milk is examined in more detail below. Some other examples of parameter variability will also be noted.

A simple model that can be used to relate the absorbed dose rate to a target organ due to radioactivity located in that organ can be expressed as follows :

$$\dot{D}(t) = c I f_1 f_2 E [1 - \exp(-\lambda t)] / m \lambda \quad (5-21)$$

where:

$\dot{D}(t)$ = absorbed dose rate (rad/day);

I = radionuclide intake rate (Ci/day);

f_1 = fraction of ingested activity transferred to the blood;

f_2 = fraction of blood activity transferred to the organ;

m = target organ mass (g);

λ = elimination constant (day^{-1}) = $0.693/T_{1/2}$, where $T_{1/2}$ is the effective half-time, including the effects of both biological removal and radioactive decay.

E = energy absorbed by the target organ for each radioactive transformation.

c = proportionality constant ($51.2 \times 10^6 \text{ g rad Ci}^{-1} \text{ MeV}^{-1} \text{ d}^{-1}$).

For simplicity, we will consider the case where t is very large compared to the biological half-life of the incorporated radionuclide, so that the term in the bracket is approximately 1:

$$\dot{D}(t) = c I f_1 f_2 E / m \lambda \quad (5-22)$$

In addition, it is assumed that the parameters remain constant throughout the period of investigation and are independent of each other.

Equation 5-22 is a simplified form of the model used by EPA to estimate the absorbed dose rates to target organs resulting from the ingestion of radioactive material. It represents the absorbed dose rate to a target organ from particulate radiation due to radioactivity that is uniformly distributed in that organ.

For this illustration, the chronic intake of iodine-131 is considered assuming that it behaves metabolically the same as stable iodine. It is further assumed that iodine is rapidly and almost completely absorbed into the bloodstream following inhalation or ingestion. From the blood, iodine enters the extracellular fluid and quickly becomes concentrated in the salivary, gastric, and thyroid glands. It is rapidly secreted from the salivary and gastric glands but is retained in the thyroid for relatively long periods.

The intake and metabolism of iodine have been reviewed extensively in the literature. Two papers have used published data to model the absorbed dose from radioiodine. In the first (Du81), the authors compiled and evaluated the variability in three of the principal biological parameters contained in Equation 5-22: m , λ , and f_2 . In the second (Br69), the author provided age-specific values for most of the same model parameters. Differences in these data illustrate how parameter variability, when used in the same model, can affect absorbed dose rate estimates for members of the general population.

Intake Rate, I

The amount of radioactive material taken into the body over a specified period of time by ingestion or inhalation is expected to be proportional to the rate of intake of food, water, or air containing such material, which, in turn, would depend on such factors as age, sex, diet, and geographical location. Therefore, understanding the patterns of food intake for individuals in the population is important in assessing the possible range of intake rates for radionuclides.

Recent EPA analyses were done to assess the daily intake rates of food and water for individuals in the general population. These studies showed that age and sex played an important role (Ne84). Age significantly affects food intake rates for all of the major food classes and, with one exception, subclasses. The relationships between food intake and age are, in most cases, similar to growth curves; there is a rapid increase in intake at an early stage of physical development, then a plateau is reached in adulthood, followed by an occasional decrease after age 60.

When sex differences were significant, males, without exception, consumed more than females. The study also showed that relative consumption rates for children and adults depend on the type of food consumed. The amount of radioactivity taken into the body per unit intake of food, air, and water depends on its relative density (amount of radioactivity contained in the material per unit volume). The most likely pathway to organs in the body for the ingestion of radioactive iodine comes from drinking milk. According to the above analysis, the daily intake rate of milk by children (under 1 yr) was twice that for an adult (25 to 29 yr) male. The intake rates for milk used in the models are 0.7 L/day and 0.5 L/day for the child and adult, respectively.

Transfer Fraction, f_1

While uncertainty in f_1 is not an important consideration for iodine, it can be very significant for other elements. Experimental studies suggest that the f_1 value for some radionuclides may be orders of magnitude higher in newborns than in adult mammals, with the largest relative changes with age occurring for those nuclides with small adult f_1 values (Cr83). For some radionuclides, the f_1 value appears to decrease rapidly in the first year of life. This can be related to the change in diet during this time period, which could affect both the removal rate from the small intestine to the upper large intestine and the absorption rate from the small intestine to the bloodstream. Studies have indicated that the wall of the small intestine is a selective tissue and that absorption of nutrients is to a large extent controlled by the body's needs (Cr83). In particular, the fraction of calcium or iron absorbed depends on the body's needs for these elements, so the f_1 value for these elements and for related elements such as strontium, radium, and barium (in the case of calcium) and plutonium (in the case of iron) may change as the need for calcium or iron changes during various stages of life.

For some essential elements, such as potassium and chemically similar radioelements, such as rubidium and cesium, absorption into the bloodstream is nearly complete at all ages, so that changes with age and possible homeostatic adaptations in absorption are not discernible. The fraction of a radioelement that is transferred to the blood depends on its chemical form, and wide ranges of values are found in the literature for individuals who ingest the material under different conditions. For example, f_1 values for uranium were found to range from 0.005 to 0.05 for industrial workers, but a higher average value of 0.2 (0.12 to 0.31) is indicated by dietary data from persons not occupationally exposed (ICRP79). EPA has used the 0.2 value for uranium ingestion by the general population.

It appears that all iodine entering the small intestine is absorbed into the blood; hence the f_1 value is taken as 1 for all ages, which is the value used in this analysis.

Organ Masses, m

To a large extent, the variability in organ masses among individuals in the general population is related to age. For most of the target organs listed in Table 5-2, the mass increases during childhood and continues to increase until adulthood, at which time the net growth of the organ ceases; there may be a gradual decrease in mass (for some organs) in later years.

Based on data reviewed by Dunning and Schwarz (Du81), the mass of an adult thyroid ranges from 2 to 62 g. It is expected that this parameter variability would be reflected in large dosimetric variability among adults. Children in the age group from .5 to 2 yr were found to have a mean thyroid mass of 2.1 g, while the adult group had a mean mass of 18.3 g. For this illustration, the same values are used as employed by the ICRP (20 g for the adult thyroid mass and 1.8 g for that of a 6-month-old child), which are also consistent with the recommendation of Bryant (Br69).

Organ Uptake Fraction, f_2

The fraction of a radionuclide taken up from the blood in an organ is strongly correlated with the size of the organ, its metabolic activity, and the amount of material ingested. Iodine introduced into the bloodstream is rapidly deposited in the thyroid, usually reaching a peak slightly after 24 hours. The uptake of iodine-131 by the thyroid is similar to that of stable iodine in the diet and can be influenced by sex and dietary differences. There can be considerable variation among populations.

Dunning and Schwarz (Du81) found a mean f_2 value of 0.47 for newborns, 0.39 for infants, 0.47 for adolescents, and 0.19 for adults. This analysis uses f_2 values of .35 and .15 for a child and adult, respectively.

Effective Half-Life, $T_{1/2}$

Some data suggest a strong correlation between biological half-lives of radionuclides in organs in the body and the age of the individual. Children are expected to exhibit faster elimination rates and greater uptakes (Ro58). For iodine, a range of biological half-lives of 21 to 200 days for adults has been observed, and a similarly wide range would be expected for other age groups (Du81). Rosenberg (Ro58) found a significant correlation between the biological half-life and the age of the individual and an inverse relationship between uptake and age in subjects from 22 to 50 yr of age. Dunning and Schwarz (Du81) concluded that for adults the observed range was from 21 to 372 days; for children in the age group from .5 to 2 yr, the range was 4 to 39 days.

In light of the possible inverse relation between the biological half-life and the f_2 value, this analysis uses biological half-lives of 24 and 129 days, respectively, for children and adults, based on the paper by Bryant (Br69). Including the effect of radioactive decay, these values imply an effective half-life of 6 days in adults and 8 days in children.

Effective Energy per Disintegration, E

The effective energy per disintegration (MeV/dis) of a radionuclide within an organ depends on the decay energy of the radionuclide and the effective radius of the organ containing the radionuclide (ICRP59). It is expected, therefore, that E is an age-dependent parameter which could vary as the size of the organ changes. While very little work has been done in determining E for most radionuclides, some information has been published for iodine-131 and cesium-137. Considering the differences between the child and the adult thyroid, Bryant (Br69) estimates E to be 0.18 MeV/dis for the child and 0.19 MeV/dis for the adult. The above values correspond to a 6-month-old child with a mass of 1.8 g and an f_2 value of 0.35. The corresponding E value for the adult was calculated for a 20-g thyroid with an f_2 value of 0.3.

Taking into account all the age-dependent factors discussed above, this analysis indicates that, for a given concentration of I-131 in milk, the estimated dose rate to the thyroid of a 6-month-old child would be approximately 13 times that to an adult thyroid. In other words, use of adult parameters would underestimate the thyroid dose to the child by about a factor of 13.

5.3.1.6.8 Significance of Parameter Variability to EPA Dose and Risk Assessments

In its radiological risk assessments, EPA is generally interested in estimating the risk to an average individual due to chronic lifetime exposures. Variation in dosimetric parameter values among people and age groups is of reduced importance in this context because such variation gets averaged over a population and/or over a lifetime. Nevertheless, it should be kept in mind that some individuals in a population are going to be at higher risk from a given exposure. Furthermore, despite such averaging, parameter value variability can contribute substantially to the uncertainty in the dose and risk estimates.

Parameter value variation among individuals contributes uncertainty to the models by causing random errors in any measured human data upon which the dosimetric models are based. To the extent that the subjects from whom such data are collected are atypical of the U.S. population (e.g., with respect to health status), parameter variation may also be a source of bias. In this respect, since the parameters contained in the dosimetric models were estimated for adult males, primarily, they may not provide an adequate basis for calculating the average dose or risk in cases where age- and sex-related variations in these parameters are large. This problem becomes more significant in light of the generally higher risks associated with a given dose for childhood exposures (see Chapter 6); if doses are also higher in childhood, the enhanced effect on risk will be compounded.

5.3.1.6.9 Past Approaches Used in Estimating Uncertainties in Calculated Organ Dose

As in any predictive exercise, it is useful to question the reliability of the predictions. Variations in environmental levels, dietary and life style preferences, and the variability of controlling physiological and metabolic processes contribute to the distribution of dose among members of the exposed population. Superimposed on this variability is a component of uncertainty arising from limitations in the predictive ability of the dosimetric models themselves. Various approaches have been taken to understand and quantify these uncertainties.

It has recently become popular to estimate the uncertainty by computing the distribution of dose among exposed individuals. This approach consists of repeated solution of the dosimetric model using parameter values selected at random from a frequency distribution of potential values suggested in the literature. It is assumed that the dosimetric model has been properly formulated, although these models were developed to yield point estimates. Despite these and other difficulties, propagation of parameter uncertainty through the dosimetric equation can provide a measure of the model uncertainty. Application of these methods to the estimation of dose from iodine-131 and cesium-137 ingestion can be found in the literature (Du81, Sc82).

An alternative approach to assessing the potential variability is to consider that the observed frequency distribution of a measurable quantity is closely related to dose. Cuddihy and co-workers (Cu79) have investigated the variability of selected target organ deposition among test animals and some individuals exposed. However, they did not address differences in age, gender, magnitude or duration of exposure.

5.3.1.6.10 Uncertainty Classification of Radionuclides

In this section, radionuclides of interest are classified in terms of the uncertainties in estimated dose per unit intake. Nuclides are placed in broad groups, largely reflecting the general status of information on their biokinetic behavior in the body. It is assumed that the uncertainty associated with the calculation of the energy deposition in the target tissues is a minor contributor to the overall uncertainty.

Classification of Uncertainty in Radionuclide Dose

Establishing numerical values of uncertainty for model dose estimates of each of the many radionuclides, for each route of exposure, is a formidable task. Even if there is agreement on the definition of uncertainty, any quantification will be arbitrary to a degree. No model has been verified in man for any long-term exposure scenario; some of the models may be fundamentally wrong in their formulation. In addition, the data selected to establish the parameters used in the model may not be representative of the population being evaluated. Most risk assessors use some informed scientific judgment in estimating the level of uncertainty in a dose model.

A broad categorization of radionuclides reflecting the estimated magnitude of the dosimetric uncertainties is presented. Because of the problems cited above with respect to the development of models and model parameters, it is quite possible that the error in model estimates may be larger than indicated in some cases. Nevertheless, this exercise is useful since it provides some perspective on the magnitude of the uncertainties in light of current evidence and focuses attention on the largest gaps in knowledge. Ultimately, however, better quantification of dose estimates and their associated uncertainties can be obtained only through the development and verification of improved dosimetric models.

Radioisotopes behave biologically like their stable elements. The elements, in turn, can be broadly grouped as: (1) essential elements and their analogs, (2) inert gases, (3) well-studied toxic metals and (4) others. Uncertainties for each of these categories will be expressed as multiplicative factors, which roughly estimate the 95% upper and lower confidence interval limits. [Since the interval is based on judgment, a preferable term would be "credibility interval" (NIH85).]

Group I - Essential Elements and Their Analogs

Essential elements are controlled by homeostatic mechanisms to within narrow tolerances. Usually, analogs of essential elements have distribution and deposition patterns similar to those of the essential element. The uncertainty expected in calculated dose for essential elements is a factor of two or less in major critical organs, perhaps 3 or less in

other significant tissues and organs. The expected dose uncertainty for analogs of essential elements is perhaps a little greater, a factor of 3 or less in major organs and up to 5 or more in less significant tissues. Important radionuclides of essential elements include hydrogen-3, carbon-14, phosphorus-32, potassium-40, calcium-45, cobalt-60, iodine-129, and iodine-131; important analogs include strontium-89, strontium-90, cesium-134, cesium-137, radium-226, and radium-228.

Group II - Inert Gases

Uptake and retention of inhaled inert gases has been fairly well studied. The uncertainty in dose, particularly average whole body dose, is not expected to be large. However, the gases do not distribute uniformly in body tissues, and the effect of distribution on organ dose estimates has not been carefully addressed. The uncertainty in the calculated dose is expected to be about a factor of 2. This group includes, but is not limited to argon-41, krypton-85, xenon-133, and radon-222.

Group III - Well-Studied Toxic Metals

A number of elements have been extensively studied in animals with limited information available for man. Examples here include toxic elements encountered in industrial activities, e.g., mercury, cadmium, lead, and uranium, for which studies were carried out to help establish safe working conditions. Often the available information is not sufficiently complete to identify the dominant processes governing the biokinetic behavior or is simply fragmentary. For example, while much information exists on the biokinetics of uranium, considerable uncertainty remains associated with the absorption to blood from the small intestine. Uncertainties for dose estimates in this group of elements would be variable, ranging from 2 or less for lead up to about 5 or more for polonium, thorium, uranium, and the transuranics. Nuclides in this group include, but are not limited to lead-210, polonium-210, uranium-235, uranium-238, thorium-230, thorium-232, plutonium-239, plutonium-241, and americium-241.

Group IV - Other Elements

For a number of radionuclides information is largely limited to data from animal studies. While animal studies often are the major source of detailed information on the processes governing the biokinetics, the lack of a general framework for extrapolations to man and the limited information upon which to judge the reasonableness of the extrapolations suggest that the estimates must be considered to be potentially in error by at least an order of magnitude. Nuclides in this group include, but are not limited to cerium-144 and other rare earth elements, technetium-99, curium-244, californium-252, etc.

The groupings listed above represent the Agency's best judgment on the uncertainty of internal radionuclide dose estimates. The primary source of uncertainty is in the biokinetic modeling with little uncertainty in the physics. The magnitudes of the uncertainties posited for each group of radionuclides should be regarded as only rough estimates; however, the qualitative breakdown between groups is fairly reliable.

Specific Problems

Certain radioisotopes and aspects of dosimetry pose unique problems. While the effect of these problems may be to increase the uncertainty in dose estimates, the extent of such an increase has yet to be evaluated.

Long-Lived Bone Seekers

Radioisotopes with effective half-lives that are short compared to the average life span are expected to be in dynamic equilibrium. However, some bone seekers have long effective half-lives; therefore, they do not reach dynamic equilibrium during a life span. Since the relevant human biokinetic data are quite limited, dose estimates for such radionuclides are more uncertain.

Nonuniformity of Distribution

The distribution of an element within an organ may not be uniform; in particular, the distribution may be nonuniform with respect to biological targets of interest. This can be a serious problem with respect to the estimation of relevant doses from internally deposited alpha emitters, given the short range of alpha particles in matter. For example, where an alpha emitter is distributed nonuniformly in bone, the calculation of doses to sensitive cells in the bone and the bone marrow will be difficult. Another example is the uncertainty in estimating doses to cells lining the GI tract from ingested alpha emitters passing through the tract. In some cases, the mucus lining may effectively shield the target cells from irradiation.

5.3.2 External Dose Models

This section is concerned with the calculation of dose rates for external exposure to photons from radionuclides dispersed in the environment. Two exposure models are discussed: (1) immersion in contaminated air and (2) irradiation from material deposited on the ground surface. The immersion source is considered to be a uniform semi-infinite radionuclide concentration in air, while the ground surface irradiation source is viewed as a uniform radionuclide concentration on an infinite plane. In both exposure modes, the dose rates to organs are calculated from the dose rate in air.

Dose rates are calculated as the product of a dose rate factor, which is specific for each radionuclide, tissue, and exposure mode, and the corresponding air or surface concentration. The dose rate factors used were calculated with the DOSFACTOR code (Ko81a,b). Note that the dose rate factors for each radionuclide do not include any contribution for decay products. For example, the ground surface dose factors for cesium-137 are all zero, since no photons are emitted in its decay. To assess surface deposition of cesium-137, the ingrowth of its decay product, metastable barium-137, which is a photon emitter, must first be calculated.

5.3.2.1 Immersion

For immersion exposure to the photons from radionuclides in air, EPA assumes that an individual is standing at the base of a semi-infinite cloud of uniform radionuclide concentration. First, the dose rate factor (the dose rate for a unit concentration) in air is calculated for a source of photons with energy E_γ . At all points in an infinite uniform source, conservation of energy considerations require that the rates of absorbed and emitted energy per unit mass be equal. The absorbed energy rate per unit mass at the boundary of a semi-infinite cloud is just half that value. Hence

$$DRF_\gamma^a (E_\gamma) = 1/2k E_\gamma / \rho \quad (5-23)$$

where:

DRF_γ^a = the immersion dose rate per unit air concentration (rad m³/Ci s);

E_γ = emitted photon energy (MeV);

k = units conversion factor

$$= 1.62\text{E-}13 \text{ (J/MeV)} \times 3.7\text{E+}10 \text{ (dis/s-Ci)} \times 1.0\text{E+}3 \text{ (g/kg)} \times 100 \text{ (rad kg/J)}$$

$$= 5.93\text{E+}2 \text{ (g rad/MeV Ci s); and}$$

ρ = density of air (g/m³).

The above equation presumes that for each nuclide transformation, one photon with energy E_γ is emitted. The dose rate factor for a nuclide is obtained by adding together the contributions from each photon associated with the transformation process for that radionuclide.

5.3.2.2 Ground Surface Irradiation

In the case of air immersion, the radiation field was the same throughout the source region. This allows the dose rate factor to be calculated on the basis of energy conservation without having to consider explicitly the scattering processes taking place. For ground surface irradiation, the radiation field depends on the height of the receptor above the surface, and the dose rate factor calculation is more complicated. The radiation flux per unit solid angle is strongly dependent on the angle of incidence. It increases from the value for photons incident from immediately below the receptor to a maximum close to the horizon. Attenuation and buildup due to scattering must be considered to calculate the dose rate factor. Secondary scattering provides a distribution of photon energies at the receptor, which increases the radiation flux above that calculated on the basis of attenuation. Trubey (Tr66) has provided a useful and reasonably accurate expression to approximate this buildup:

$$B_{en}^a(\mu, r) = 1 + C_a \mu_a r \exp(D_a \mu_a r) \quad (5-24)$$

where B_{en}^a is the buildup factor (i.e., the quotient of the total energy flux and that calculated for attenuation) only for energy in air; μ_a is the attenuation coefficient at the energy of the released photon (m^{-1}); r is the distance between the photon source and the receptor; and the Berger buildup coefficients C_a and D_a are dependent on energy and the scattering medium. The buildup factor is dimensionless and always has a value greater than unity. The resulting expression for the dose rate factor at a height z (m) above a uniform plane is

$$DRF_{\gamma}^a(z, E_{\gamma}) = 1/2k(E_{\gamma}/\rho)(\mu_{en}/\rho)_a \{E_1(\mu_a z) + C_a/(1-D_a)\exp[-(1-D_a)\mu_a z]\} \quad (5-25)$$

where $(\mu_{en}/\rho)_a$ is the mass energy-absorption coefficient (m^2/g) for air at photon energy E_{γ} (MeV); E_1 is the first order exponential integral function, i.e.,

$$E_1(x) = \int_x^{\infty} \frac{\exp(-u)}{u} du \quad (5-26)$$

C_a and D_a are the buildup coefficients in air at energy E_{γ} ; and $k=5.93 \times 10^2$ (g rad/MeV Ci s) as for the immersion calculation.

As for immersion, the dose rate factor for a nuclide combines the contribution from each photon energy released in the transformation process.

5.3.2.3 Organ Doses

The dose rate factors in the preceding two sections are for the absorbed dose in air. For a radiological assessment, the absorbed doses in specific tissues and organs are needed. For this purpose, Kerr and Eckerman (Ke80, Ke80a) have calculated organ dose factors for immersion in contaminated air. Their calculations are based on Monte Carlo simulations of the absorbed dose in each tissue or organ for the spectrum of scattered photons in air resulting from a uniform concentration of monoenergetic photon sources. Kocher (Ko81) has used these data to calculate values of the ratio of the organ dose factor to the air dose factor, $G^k(E_{\gamma})$, for 24 organs and tissues at 15 values of E_{γ} , ranging from 0.01 to 10.0 MeV.

The resulting organ-specific dose rate factor for immersion is

$$DRF_{\gamma}^k(E_{\gamma}) = G^k(E_{\gamma}) DRF_{\gamma}^a(E_{\gamma}) \quad (5-27)$$

For a specific nuclide, the dose rate factor is obtained by taking the sum of the contributions from each photon energy associated with the radionuclide decay.

Ideally, a separate set of $G^k(E_\gamma)$ values would be used for the angular and spectral distributions of incident photons from a uniform plane source. Since these data are not available, Kocher has used the same set of $G^k(E_\gamma)$ values for calculating organ dose rate factors for both types of exposure (Ko81).

5.3.2.4 Uncertainty Considerations in External Dose Rate Factors

In computing the immersion dose rate factor in air, the factor of 1/2 in Equation 5-27, which accounts for the semi-infinite geometry of the source region, does not provide a rigorously correct representation of the air/ground interface. However, Dillman (Di74) has concluded that this result is within the accuracy of available calculations. The radiation field between the feet and the head of a person standing on contaminated ground is not uniform, but for source photon energies greater than about 10 keV, the variation about the value at 1 meter becomes minimal. A more significant source of error is the assumption of a uniform concentration. Kocher (Ko81) has shown that sources would have to be approximately uniform over distances of as much as a few hundred meters from the receptor for the dose rate factors to be accurate for either ground surface or immersion exposures. Penetration of deposited materials into the ground surface, surface roughness, and terrain irregularities, as well as the shielding provided by buildings to their inhabitants, all serve to reduce doses.

The effect of using the same factors to relate organ doses to the dose in air for ground surface as for immersion photon sources has not been studied. The assumptions that the radiation field for the ground surface source is isotropic and has the same energy distribution as for immersion clearly do not hold true, but more precise estimates of these distributions are not likely to change the organ dose rate factors substantially.

Kocher (Ko81) has noted that the idealized photon dose rate factors are "likely to be used quite extensively even for exposure conditions for which they are not strictly applicable... because more realistic estimates are considerably more difficult and expensive [to make]."

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Chapter 7: MOVEMENT AND HEALTH RISKS OF RADIONUCLIDE RELEASES TO THE ACCESSIBLE ENVIRONMENT

7.1 INTRODUCTION

This chapter describes analyses used to assess the health risks caused by the environmental transport of radionuclides once they are released from a repository to the accessible environment. As part of its program to develop 40 CFR Part 191, the Agency estimated population health risks for a 10,000-year period following disposal in mined geologic repositories (see Chapter 8). The level of population health risk was calculated for 10,000 years after disposal of all the existing high-level waste and most of the wastes that are yet to be produced by currently operating U.S. nuclear reactors. This overall level of residual risk to future generations is found to be comparable to the estimated lower limit of risk that those generations would face from the uranium ore used to create the wastes if the ore had never been mined (Wi80). These estimates were used in selecting the containment requirements in the disposal standards. The objective of this chapter is: 1) to describe the environmental pathways that were considered when calculating the fatal cancers committed as a result of releases of radionuclides from disposal systems; and 2) to summarize the results of these calculations. A complete description of the analysis method is described in Sm85. This chapter also describes how the release limits of the containment requirements were derived from the results of these calculations.

In performing these long-term assessments of population health effects, the Agency recognizes that it is pointless to try to make precise projections of the actual risks due to radionuclide releases from repositories. Population distributions, food chains, living habits, and technological capabilities will undoubtedly change over 10,000 years. Unlike geological processes, they can be realistically predicted only for relatively short times. Accordingly, very general models of environmental pathways were formulated as opposed to the detailed analytical techniques that would be appropriate for near-term environmental assessments of specific facilities. Population characteristics similar to those of today were assumed.

The models discussed in this chapter consider risks to populations, as opposed to risks to individuals. Therefore, individual risks caused by potential releases from a repository cannot be determined from these analyses. Analyses that assess individual risks are described in Chapter 8.

7.2 METHODOLOGY

The basic analysis assumed radionuclides can be released from geologic repositories and move through the environment by four pathways: 1) to surface water (e.g., a river) through ground water, 2) to an ocean through surface water, 3) to a land surface directly, or 4) to multiple pathways after the very unlikely possibility of disruption by a volcano or a meteorite. For each of these four release modes, radionuclide movement through the geosphere and the biosphere to the population was modeled, and an estimate was made of

the intake by or exposure to the population through each of these environmental pathways. The environmental pathways included for each of the release modes are described in Table 7.2-1.

Risk conversion factors per unit intake or per unit external exposure were applied to the radionuclide concentrations output by the model to estimate fatal cancers and serious genetic effects to all generations per curie of each different radionuclide release to the accessible environment. The results were used as one of the basis to specify the release limits in Table 1 of 40 CFR Part 191, based on a consideration of only excess fatal cancers. The genetic effects to all generations were lower than the estimated fatal cancers by a factor of two or more for all radionuclides and were not used to establish the release limits. The risk conversion factors used to estimate fatal cancers are listed in Table 7.2-2 (Sm85). These risk conversion factors have been updated (Pa92).

Health effects were calculated for the entire population exposed to the releases from a repository and calculations were not terminated at a specific distance from the repository. Time integrations were performed to obtain the total radioactivity intakes or exposures from the time the repository is sealed ("disposal") until a specified time in the future (usually 10,000 years after disposal). These intakes and exposures were converted to population fatal cancers by multiplying by the appropriate risk conversion factors. The following sections summarize the factors considered in the calculation of the population intake of radioactivity for the internal pathways—or the integrated population exposure for the external pathways—for each of the four release modes.

7.3 RELEASES TO SURFACE WATER

In the surface water release pathway, the repository containment is assumed to be breached—after some initial period—and ground water percolates through the repository into the surrounding geologic media and eventually to an aquifer. The aquifer then flows underground until it intersects a river. To determine the total release to the surface water (river), the release rate was integrated over the time period of interest. The integrated release rate, in equation form, was then used to compute surface water concentrations for use with several environmental pathways. These are discussed in the following subsections.

7.3.1 Drinking Water

It was assumed that the population receives 65 percent of its drinking water from surface waters with no reduction in radionuclide concentrations due to water treatment (Mu77). The annual intake rate of drinking water and the water-based drinks by an individual is 600 liters (ICRP75); thus, 390 liters were assumed to be supplied by surface water. This intake was multiplied by the integrated concentration in the river, the fatal cancer conversion factor per unit curie for ingestion, and the number of people drinking the water to arrive at the number of fatal cancers caused by the release of each radionuclide to the accessible environment. The calculation can be rearranged into an equation that expresses the fatal cancers normalized to a unit curie (Sm85):

Table 7.2-1. Release modes and environmental pathways

<u>Release mode</u>	<u>Pathways included in this release mode</u>
Release to river	Drinking water ingestion Freshwater fish ingestion Food crops ingestion Milk ingestion Beef ingestion Inhalation of resuspended material External dose, ground contamination External dose, air submersion
Releases to ocean	Ocean fish ingestion Ocean shellfish ingestion
Releases directly to land surface	Food crops ingestion Milk ingestion Beef ingestion Inhalation of resuspended material External dose, ground contamination External dose, air submersion
Releases due to volcano/meteorite interaction	
Releases directly to land	Food crops ingestion Milk ingestion Beef ingestion Inhalation of resuspended material External dose, ground contamination External dose, air submersion
Releases to air over land	Food crops ingestion Milk ingestion Beef ingestion Inhalation of dispersed and resuspended material External dose, ground contamination External dose, air submersion
Releases to air over ocean	Ocean fish ingestion Ocean shellfish ingestion

Table 7.2-2 Fatal cancer risk conversion factors^(a)

Radionuclide	Fatal cancers per Ci intake				Fatal cancers from external doses	
	Inhalation ^(b)		Ingestion ^(b)		Air	Ground
	1	2	1	2	Submersion per Ci-y/m ³	Contamination per Ci-y/m ³
C-14	4.15X10 ⁻³	4.15X10 ⁻³	5.91X10 ⁻¹	5.91X10 ⁻¹	0	0
Ni-59	5.87X10 ⁻¹	5.87X10 ⁻¹	5.41X10 ⁻²	5.41X10 ⁻²	6.18X10 ⁻²	1.34X10 ⁻²
Sr-90	5.44X10 ²	5.83X10 ¹	3.89X10 ⁰	3.28X10 ¹	0.00X10 ⁰	0.00X10 ⁰
Zr-93	3.29X10 ¹	8.53X10 ⁰	2.15X10 ⁻¹	2.15X10 ⁻¹	1.78X10 ⁻¹	2.73X10 ⁻²
Tc-99	7.42X10 ⁰	7.42X10 ⁰	7.42X10 ⁻¹	7.42X10 ⁻¹	8.04X10 ⁻⁴	1.90X10 ⁻⁵
Sn-126	7.01X10 ¹	7.01X10 ¹	3.56X10 ⁰	3.56X10 ⁰	3.44X10 ³	6.93X10 ¹
I-129	1.25X10 ¹	1.25X10 ¹	1.88X10 ¹	1.88X10 ¹	1.09X10 ¹	5.71X10 ⁻¹
Cs-135	1.71X10 ⁰	1.71X10 ⁰	2.47X10 ⁰	2.47X10 ⁰	0.00X10 ⁰	0.00X10 ⁰
Cs-137	1.16X10 ¹	1.16X10 ¹	1.70X10 ¹	1.70X10 ¹	9.72X10 ²	1.94X10 ¹
Sm-151	8.62X10 ⁰	8.62X10 ⁰	6.87X10 ⁻²	6.87X10 ⁻²	1.15X10 ⁻³	1.34X10 ⁻⁴
Pb-210	2.74X10 ⁴	3.83X10 ³	6.84X10 ²	6.84X10 ²	1.80X10 ⁰	8.46X10 ⁻²
Ra-226	5.27X10 ⁴	6.64X10 ³	7.78X10 ²	7.78X10 ²	3.18X10 ³	5.69X10 ¹
Ra-228	9.58X10 ⁴	1.84X10 ⁴	1.23X10 ²	1.23X10 ²	4.64X10 ³	7.98X10 ¹
Ac-227	8.64X10 ⁴	5.35X10 ⁴	4.12X10 ²	4.12X10 ²	6.46X10 ²	1.41X10 ¹
Th-229	7.64X10 ⁴	3.07X10 ⁴	1.02X10 ²	1.02X10 ²	4.46X10 ²	1.01X10 ¹
Th-230	8.21X10 ⁴	2.27X10 ⁴	8.01X10 ²	8.01X10 ²	3.17X10 ³	5.67X10 ¹
Th-232	1.25X10 ⁵	3.28X10 ⁴	1.44X10 ²	1.44X10 ²	4.64X10 ³	7.99X10 ¹
Pa-231	1.25X10 ⁵	7.91X10 ⁴	6.01X10 ²	6.01X10 ²	6.94X10 ²	1.53X10 ¹
U-233	2.88X10 ⁴	4.31X10 ³	6.86X10 ⁰	8.01X10 ¹	2.27X10 ¹	5.20X10 ⁻¹
U-234	2.49X10 ⁴	2.74X10 ³	1.74X10 ⁰	7.48X10 ¹	2.27X10 ⁻¹	2.39X10 ⁻²
U-235	2.43X10 ⁴	3.34X10 ³	8.25X10 ⁰	7.92X10 ¹	2.65X10 ²	6.13X10 ⁰
U-236	2.36X10 ⁴	2.60X10 ³	1.65X10 ⁰	7.08X10 ¹	1.78X10 ⁻¹	2.17X10 ⁻²
U-238	2.23X10 ⁴	2.46X10 ³	3.88X10 ⁰	7.62X10 ¹	3.19X10 ¹	7.01X10 ⁻¹

Table 7.2-2 Fatal cancer risk conversion factors^(a) (continued)

Radionuclide	Fatal cancers per Ci intake				Fatal cancers from external doses	
	Inhalation ^(b)		Ingestion ^(b)		Air Submersion per Ci-y/m ³	Ground Contamination per Ci-y/m ³
	1	2	1	2		
Np-237	3.66X10 ⁴	3.51X10 ⁴	2.69X10 ²	2.69X10 ²	3.83X10 ²	8.66X10 ⁰
Pu-238	3.96X10 ⁴	3.56X10 ⁴	2.71X10 ²	2.71X10 ²	1.27X10 ¹	2.48X10 ⁻²
Pu-239	3.92X10 ⁴	3.77X10 ⁴	2.99X10 ¹	2.89X10 ²	1.25X10 ¹	1.11X10 ⁻²
Pu-240	3.91X10 ⁴	3.76X10 ⁴	2.98X10 ¹	2.89X10 ²	1.25X10 ¹	2.38X10 ⁻²
Pu-241	1.52X10 ³	1.74X10 ³	1.37X10 ¹	1.37X10 ¹	8.04X10 ¹	2.45X10 ²
Pu-242	3.72X10 ⁴	3.58X10 ⁴	2.84X10 ¹	2.74X10 ²	1.06X10 ¹	1.98X10 ⁻²
Am-241	4.14X10 ⁴	3.92X10 ⁴	3.00X10 ²	3.00X10 ²	2.68X10 ¹	8.45X10 ¹
Am-243	4.05X10 ⁴	3.88X10 ⁴	2.98X10 ²	2.98X10 ²	3.43X10 ²	8.04X10 ⁰
Cm-245	8.25X10 ⁴	7.95X10 ⁴	6.10X10 ²	6.10X10 ²	1.38X10 ²	3.50X10 ⁰
Cm-246	4.13X10 ⁴	3.96X10 ⁴	3.04X10 ²	3.04X10 ²	9.95X10 ⁻²	2.08X10 ⁻²

- (a) The fatal cancer risk conversion factors in this table are the sum of the risk factors for the listed radionuclide plus any significant daughter products which can develop during the residence time of the radionuclide in the accessible environment.
- (b) When a radionuclide can exist having more than one solubility class, factor 1 refers to the form with a lower solubility class, and factor 2 refers to the form with a higher solubility class (Sm85)

$$\frac{FHE_{np}}{Q_{cp}} = \frac{(P_R)(I_w)(f_{rw})(FCF_{cp})(f_{wt})}{R}$$

FHE_{np} = time integrated population risk commitment for radionuclide, n, in the drinking water pathway.

Q_{cp} = integrated release to the river (curies).

P_R = the number of persons drinking the water.

I_w = per capita consumption of drinking water and water based drinks.

f_{rw} = fraction of intake the from river = 0.65.

FCF_{cp} = fatal cancers per curie intake for ingestion.

f_{wt} = loss from water treatment (assumed 1.0).

R = annual river flow (liters per year).

The ratio P_R/R can be determined for the purpose of this generic evaluation without obtaining site specific data. Using information from Annex D of the 1977 UNSCEAR Report (UNSC77), the annual flow rate of the world's rivers is 3×10^{16} liters. Assuming an average world population of 10^{10} persons over the next 10,000 years, P_R/R is 3.3×10^{-7} person years/liter. This ratio is within the range of values found for various river basins in the United States.

7.3.2 Ingestion of Fish

Fish caught in the river are assumed to contain radionuclides due to uptake from the water. The amount of radionuclides accumulated in the fish (in terms of curies per kg of fish body weight) is a direct function of the radionuclide concentration (curies per liter) in the surface water. The fraction of the radionuclides released to the surface water that is ingested by the population through fish consumption is obtained by calculating the quantity of radionuclides in the fish by the use of bioaccumulation factors, and by determining an average ratio of the population's fish ingestion rate to the river flow rate (3.3×10^{-7} man-kilogram/liter*). Multiplying the bioaccumulation factors for fish, which are given in Table 7.3-1, by the average ratio and the fatal cancer conversion factor for ingestion, calculates the normalized fatal cancers for each radionuclide.

*This ratio is determined by multiplying the person-year/liter (discussed in Section 7.3) by the assumed annual individual fish consumption of 1.0 kg/year (UNSC77).

7.3.3 Ingestion of Food Raised on Irrigated Land

Surface water containing radionuclides released from the repository may be used to spray or irrigate farm land, leading to direct deposition of radionuclides onto the crops and the surface of the soil in which the crops grow. The average fraction of the river flow used for irrigation was assumed to be 0.1, based on the United States average of 0.07 (Sm85, Mu77). Irrigated plants that have incorporated radionuclides through their leaves and root systems are consumed by humans as food, or are consumed by either dairy or beef cattle that transfer radionuclides to milk and meat. The normalized fatal cancers for these pathways were determined by multiplying the following parameters together: a radionuclide-specific intake factor for each pathway (food crops, milk, and beef) as given in Table 7.3-2, the fraction of the river flow used for irrigation, the fraction of irrigated land used for the food source, the average number of people that can be fed per unit area of land by each of the pathways as given in Table 7.3-3 (Sm85), and the fatal cancer conversion factor for ingestion.

7.3.4 Inhalation of Resuspended Material

Some of the radionuclides deposited on the soil by irrigation are resuspended into the air. The subsequent air concentrations were estimated by use of a resuspension factor of $10^{-3}/\text{m}$ multiplied by the integrated soil concentration (Be76). The calculation of the soil concentrations consists of multiplying the fraction of river water used for irrigation (0.1) by the integral expressing the buildup of a radionuclide in soil due to irrigation and the depletion from the root zone as a function of time, and dividing by 15 centimeters to account for the dilution of the concentration due to plowing to the depth of the root zone. The air concentrations are multiplied by an annual inhalation rate of 8400 cubic meters and an average population density of 6.7×10^{-5} persons per square meter (ICRP75, UNSC77, Wo79) to obtain the population intakes. Multiplying by the fatal cancer conversion factors and dividing by the integrated environmental release of the radionuclides produces the normalized fatal cancers.

Table 7.3-1. Bioaccumulation factors for freshwater fish

Radionuclide	Bioaccumulation factor (Ci/kg per Ci/liter)
C-14	NA ^(a)
Ni-59	1.00X10 ²
Sr-90	1.10X10 ¹ (Ho79)
Zr-93	3.33X10 ⁰
Tc-99	4.30X10 ¹ (B182)
Sn-126	3.00X10 ³
I-129	3.30X10 ¹ (Ho79)
Cs-135	1.30X10 ³ (Ho79)
Cs-137	1.30X10 ³ (Ho79)
Sm-151	2.50X10 ¹
Pb-210	1.00X10 ²
Ra-226	5.00X10 ¹
Ra-228	5.00X10 ¹
Ac-227	2.50X10 ¹
Th-229	3.00X10 ¹
Th-230	3.00X10 ¹
Th-232	3.00X10 ¹
Pa-231	1.10X10 ¹
U-233	1.00X10 ¹
U-234	1.00X10 ¹
U-235	1.00X10 ¹
U-236	1.00X10 ¹
U-238	1.00X10 ¹
Np-237	5.00X10 ² (Sc83)
Pu-238	8.00X10 ⁰ (Ri83)
Pu-239	8.00X10 ⁰ (Ri83)
Pu-240	8.00X10 ⁰ (Ri83)
Pu-241	8.00X10 ⁰ (Ri83)
Pu-242	8.00X10 ⁰ (Ri83)
Am-241	8.10X10 ¹ (Ri83)
Am-243	8.10X10 ¹ (Ri83)
Cm-245	2.50X10 ¹
Cm-246	2.50X10 ¹

^(a)NA - Not Applicable

Source: Th72 (unless otherwise noted)

Table 7.3-2 Radionuclide intake factors for farm products raised in areas using contaminated irrigation water

Radionuclide	Radionuclide Intake Factor (Ci intake per Ci/m ² deposited)		
	Food Crops	Milk	Meat
C-14	NA ^(a)	NA	NA
Ni-59	4.38X10 ⁰	3.22X10 ⁻¹	2.48X10 ⁻¹
Sr-90	2.57X10 ⁰	1.07X10 ⁰	8.20X10 ⁻²
Zr-93	4.21X10 ⁰	8.15X10 ⁻²	2.10X10 ⁻¹
Tc-99	1.57X10 ⁰	4.00X10 ⁰	1.31X10 ⁰
Sn-126	1.10X10 ⁰	3.04X10 ⁻¹	9.36X10 ⁰
I-129	1.17X10 ¹	1.03X10 ¹	2.78X10 ⁰
Cs-135	1.40X10 ¹	8.04X10 ⁰	8.84X10 ⁰
Cs-137	8.51X10 ⁻¹	1.74X10 ⁰	1.91X10 ⁰
Sm-151	5.47X10 ⁻¹	4.54X10 ⁻³	4.37X10 ⁻¹
Pb-210	4.98X10 ⁻¹	5.75X10 ⁻²	2.66X10 ⁻²
Ra-226	6.62X10 ⁻¹	1.26X10 ⁻¹	6.26X10 ⁻²
Ra-228	3.95X10 ⁻¹	9.81X10 ⁻²	4.53X10 ⁻²
Ac-227	3.95X10 ⁻¹	4.36X10 ⁻³	2.10X10 ⁻³
Th-229	7.33X10 ⁻¹	1.49X10 ⁻³	6.78X10 ⁻⁴
Th-230	2.77X10 ⁰	3.87X10 ⁻³	1.79X10 ⁻³
Th-232	6.73X10 ⁰	8.51X10 ⁻³	3.93X10 ⁻³
Pa-231	6.92X10 ⁻¹	1.43X10 ⁻³	1.10X10 ⁻³
U-233	1.19X10 ⁰	1.57X10 ⁻¹	2.01X10 ⁻²
U-234	1.19X10 ⁰	1.57X10 ⁻¹	2.01X10 ⁻²
U-235	1.19X10 ⁰	1.57X10 ⁻¹	2.01X10 ⁻²
U-236	1.19X10 ⁰	1.57X10 ⁻¹	2.01X10 ⁻²
U-238	1.19X10 ⁰	1.57X10 ⁻¹	2.01X10 ⁻²
Np-237	5.42X10 ⁻¹	2.52X10 ⁻³	1.94X10 ⁻²
Pu-238	3.92X10 ⁻¹	2.17X10 ⁻⁵	1.67X10 ⁻⁴
Pu-239	4.77X10 ⁻¹	2.37X10 ⁻⁵	1.83X10 ⁻⁴
Pu-240	4.53X10 ⁻¹	2.32X10 ⁻⁵	1.79X10 ⁻⁴
Pu-241	3.90X10 ⁻¹	2.17X10 ⁻⁵	1.67X10 ⁻⁴
Pu-242	4.89X10 ⁻¹	2.40X10 ⁻⁵	1.85X10 ⁻⁴
Am-241	4.35X10 ⁻¹	9.45X10 ⁻⁵	3.18X10 ⁻⁴
Am-243	4.87X10 ⁻¹	1.03X10 ⁻⁴	3.48X10 ⁻⁴
Cm-245	4.10X10 ⁻¹	4.67X10 ⁻³	3.14X10 ⁻⁴
Cm-246	4.08X10 ⁻¹	4.63X10 ⁻³	3.12X10 ⁻⁴

NA^(a) - Not Applicable
Source: Sm85

Table 7.3-3. Values for persons fed per unit area of land

Food	Person fed/m ²
Vegetable Food Crops	4.79 X10 ⁻³
Milk	1.56 X10 ⁻³
Meat	7.85 X10 ⁻⁵

Source: Sm85

7.3.5 External Exposure from Air Submersion

The radionuclides resuspended into the air as described in Paragraph 7.3.4 can also cause immersion exposures to the population. These exposures are calculated from the integrated air concentration, the average population density, and a shielding and occupancy factor of 0.33 (UNSC77, Wo79). The results are multiplied by the fatal cancer conversion factors found in Table 7.2-2 and divided by the integrated environmental releases to obtain the normalized fatal cancers.

7.3.6 External Exposure from Ground Concentration

Finally, the radionuclides deposited on the ground during irrigation can also cause external exposures to persons in the area. Throughout the irrigation period, radionuclides continue to build up on the ground until either irrigation stops or equilibrium is reached with losses through the soil. The methods for estimating these exposures are similar to those applied for air submersion. The integrated soil concentration is calculated in the same way and multiplied by the external risk conversion factor found in Table 7.2-2, by a shielding factor that accounts for the reduction of external dose due to house shielding and occupancy (0.33), and by a correction factor (calculated for each radionuclide) that accounts for the reduction in the external risk commitment due to soil shielding.

7.4 Releases to an Ocean

Releases to a surface water system are assumed to discharge subsequently into an ocean. Since radionuclide decay during travel in the river or depletion of the radionuclide inventory due to river water use and sedimentation is not considered, the radionuclide releases to an ocean are equal to the releases to a surface water. The ocean pathway model has two compartments consisting of a shallow upper layer in which it is assumed that all edible seafood is grown, and a lower layer that includes the remainder of the ocean. Differential equations were developed whose solutions describe the quantities of radionuclides in these two compartments over time. The equation for the upper compartment inventory was divided by the volume of the compartment to determine the time-dependent concentration of radionuclides in the upper layer. This concentration was then used to estimate the fraction

of the radionuclides released to the river that is consumed by the population due to bioaccumulation of radionuclides in ocean fish and shellfish (Sm85). In spite of this conservative approach, the collective dose equivalent is small relative to that from the other components of the release to surface water scenario. For this reason, consideration of more sophisticated ocean models does not seem warranted.

7.5 RELEASES DIRECTLY TO LAND SURFACE

For the land surface pathway models, some of the radioactive waste from the repository is assumed to be brought to the surface after an event such as inadvertent intrusion while drilling for resources. Such releases to the surface are assumed to be over a small area and a short period of time; as such, they can be modeled as instantaneous point sources. The mechanisms distributing the material to humans are resuspension and subsequent dispersion in the atmosphere. After the initial release to the land surface is determined, a time-dependent release rate to the air is estimated using a simple exponential model that depletes the land surface source to account for resuspension and radioactive decay. This release rate is applied in conjunction with an atmospheric dispersion equation to predict air concentrations as a function of time and distance from the source; these air concentrations are then used to estimate ground surface concentrations as a function of time and distance. Once ground surface concentrations are determined, the techniques used to calculate population intake are similar to those described for the surface water release mode. The pathways considered for releases to land surface are ingestion of food raised on land contaminated with radionuclides, including food crops, milk, and meat; inhalation of resuspended radionuclides; external exposure due to air submersion; and external exposure due to ground contamination. The mathematical formulations are the same as the river release scenario except that for the surface release scenario there is no long term replenishment and, therefore, the soil concentration diminishes more rapidly with time (Sm85).

7.6 RELEASES DUE TO A VOLCANIC ERUPTION OR METEORITE IMPACT

Releases to the land surface and directly to the air can be caused by the extremely unlikely events of disruption by volcanoes or meteorites. The methodology described for the land surface release mode is used for the material released to the land surface. For the material released to the air, it is assumed that the radionuclides would be quickly dispersed in such a manner that they would eventually be distributed uniformly within the troposphere. The airborne material is divided into the fraction over land and the fraction over water using the ratio of earth land surface and earth water surface (i.e., 0.29 and 0.71, respectively). Compartment models, with their systems of coupled differential equations, were used to estimate the quantity of radionuclides reaching the land surface or ocean (Sm85). Finally, the amount of radionuclides or radiation exposure reaching people was estimated through the same pathways described for the land surface or the ocean, respectively.

7.7 SPECIAL CONSIDERATIONS FOR CARBON-14 ENVIRONMENTAL RISK COMMITMENT

Unlike the other radionuclides considered in these analyses, stable carbon constitutes a significant fraction of the elemental composition of the human body and man's diet. Thus, transport processes through the different environmental pathways and within plants, animals, and man that apply to trace quantities of other radionuclides do not necessarily apply to radionuclides such as carbon-14 (C-14), where the corresponding stable elements are present in such quantities that saturation effects are significant (Mo79).

Atmospheric releases of C-14 as a carbon dioxide can be evaluated using a diffusion-type model of the carbon cycle developed by Killough (Ki77). It seems clear that this model is the correct calculational procedure to use for releases for the volcano/meteorite release mode where it is assumed that high temperatures would cause carbon releases to be oxidized to carbon dioxide. Models are not available to treat the fatal cancers calculations explicitly for C-14 released to water, land surfaces, or air in a chemical form other than carbon dioxide. A review of the literature indicated that the chemical form of C-14 released in the water and land surface release modes is not well known, but the rate of oxidation to carbon dioxide or other chemical forms of C-14 occurs over a short length of time relative to the extensive integration period for these release modes (NCRP85). Therefore, the most prudent course was concluded to be the use of the Killough carbon dioxide model for all four release modes, realizing that this probably leads to conservative estimates of the fatal cancers for the water and land release modes.

The environmental risk commitment for C-14 is obtained by calculating the total body environmental dose commitment (EDC) per curie released for the 10,000 year assessment period and multiplying by a fatal cancer risk conversion factor. Values of the total body environmental dose commitment per curie of C-14 released to the atmosphere have been calculated by Fowler using the Killough model (Fo79, Ki77). It is estimated that the ingestion pathway contributes 99 percent of the carbon-14 environmental dose commitment (Fo76); however we assumed that the ingestion pathways contribute 100 percent for purposes of computational convenience. For estimating the environmental dose commitment, Fowler's curve of worldwide EDC to the total body per curie release versus time after release was used. For the 10,000 year assessment period a total body EDC of 399 man-rem/curie released was calculated.

The environmental risk commitment is obtained by multiplying the total body environmental dose commitment of 399 man-rem/curie by the fatal cancer risk factor of 1.46×10^{-4} fatal cancers per total body man-rem as given by Fowler* (Fo79), yielding 5.83×10^{-2} fatal cancers/curie released. For C-14, this is the total environmental risk commitment for all the pathways within each release mode; the methodology is not applied separately for each pathway. This does not imply that there is no risk from C-14 for some

*This C-14 fatal cancer risk factor is less than that used for most other radionuclides because a large percentage of the total body dose from C-14 is to adipose tissue and is not effective in producing cancer (Fo79).

pathways; rather it means that all C-14 emitted is assumed to reach equilibrium worldwide regardless of release mode, and to contribute risk solely from ingestion.

7.8 FATAL CANCERS PER CURIE RELEASED TO THE ACCESSIBLE ENVIRONMENT

This section presents the results of all analyses in terms of the premature fatal cancers induced (over 10,000 years) for each curie of the various radionuclides that may be released to the accessible environment. These values were used to calculate the limits in Table 1 of 40 CFR Part 191 of the rule. The fatal cancer estimates for releases to surface water (the sum of releases to a river and releases to the ocean), to land surfaces, and to the atmosphere are tabulated in Table 7.8-1. Table 7.8-2 shows how the various environmental pathways contribute to the fatal cancer per curie estimate for releases to surface water. As can be seen from Table 7.8-2, the dominant pathway for each radionuclide is usually ingestion of surface crops irrigated with contaminated water.

7.8.1 Development of Release Limits for 40 CFR Part 191

The analyses described in this chapter were used to develop radionuclide release limits that correspond to the level of protection chosen for the containment requirements of the final rule (Section 191.13). Since releases to surface water through ground water are usually the most important release mode for mined repositories, and since the health effects per curie released are usually the highest for this release mode, the release limits in 40 CFR Part 191 were based solely on the surface water release mode.

To develop the release limits, the appropriate population risk level must first be chosen. The Agency has chosen to base the containment requirements on a population risk level of no more than 1,000 premature cancer deaths over 10,000 years from disposal of 100,000 metric tons of heavy metal (MTHM) contained in spent fuel (or from disposal of the high-level radioactive wastes produced by this much spent fuel). For convenience, the release limits in 40 CFR Part 191 are stated per 1,000 MTHM and can be adjusted to reflect the actual amount of waste in a disposal system. Therefore, the release limits in 40 CFR Part 191 are to be the amount of each radionuclide that would cause 10 health effects over 10,000 years.

Table 7.8-1 Fatal cancers per curie release to the accessible environment for different release modes

Radionuclide	Releases to surface water	Releases to land surface	Releases due to violent interactions ^(a)
C-14			
Ni-59	6.88X10 ⁻⁵	9.76X10 ⁻⁷	4.17X10 ⁻⁵
Sr-90	2.60X10 ⁻²	4.34X10 ⁻⁵	1.91X10 ⁻³
Zr-93	2.70X10 ⁻⁴	3.81X10 ⁻⁵	2.05X10 ⁻⁴
Tc-99	5.09X10 ⁻⁴	7.80X10 ⁻⁴	2.75X10 ⁻⁴
Sn-126	1.89X10 ⁻²	1.88X10 ⁻³	4.83X10 ⁻²
I-129	6.31X10 ⁻²	3.09X10 ⁻³	4.36X10 ⁻²
Cs-135	1.05X10 ⁻²	7.80X10 ⁻⁴	6.82X10 ⁻³
Cs-137	1.47X10 ⁻²	2.98X10 ⁻⁵	5.25X10 ⁻³
Sm-151	1.94X10 ⁻⁵	1.26X10 ⁻⁷	8.90X10 ⁻⁶
Pb-210	2.07X10 ⁻¹	2.54X10 ⁻⁴	6.87X10 ⁻²
Ra-226	2.64X10 ⁻¹	8.31X10 ⁻³	1.74X10 ⁻¹
Ra-228	3.06X10 ⁻²	1.65X10 ⁻⁵	3.49X10 ⁻²
Ac-227	9.93X10 ⁻²	1.80X10 ⁻⁴	5.17X10 ⁻²
Th-229	7.51X10 ⁻²	2.44X10 ⁻²	1.24X10 ⁻¹
Th-230	1.11X10 ⁰	5.40X10 ⁻¹	2.21X10 ⁰
Th-232	4.89X10 ⁻¹	5.03X10 ⁻¹	1.91X10 ⁰
Pa-231	1.93X10 ⁻¹	3.08X10 ⁻²	1.81X10 ⁻¹
U-233	3.44X10 ⁻²	1.16X10 ⁻³	9.68X10 ⁻³
U-234	3.21X10 ⁻²	1.04X10 ⁻³	7.29X10 ⁻³
U-235	3.45X10 ⁻²	1.28X10 ⁻³	1.12X10 ⁻²
U-236	3.04X10 ⁻²	9.90X10 ⁻³	6.91X10 ⁻³
U-238	3.27X10 ⁻²	1.07X10 ⁻³	7.20X10 ⁻³
Np-237	1.25X10 ⁻¹	1.74X10 ⁻⁴	3.98X10 ⁻²
Pu-238	6.22X10 ⁻²	4.44X10 ⁻⁴	2.86X10 ⁻²
Pu-239	7.52X10 ⁻²	8.95X10 ⁻³	1.59X10 ⁻²
Pu-240	7.30X10 ⁻²	7.52X10 ⁻³	1.52X10 ⁻²
Pu-241	3.12X10 ⁻³	3.89X10 ⁻⁶	1.29X10 ⁻³
Pu-242	7.22X10 ⁻²	9.10X10 ⁻³	1.53X10 ⁻²
Am-241	8.40X10 ⁻²	1.51X10 ⁻³	3.55X10 ⁻³
Am-243	9.85X10 ⁻²	3.50X10 ⁻³	4.87X10 ⁻²
Cm-245	1.79X10 ⁻¹	1.15X10 ⁻²	8.95X10 ⁻²
Cm-246	8.86X10 ⁻²	5.52X10 ⁻³	4.32X10 ⁻²

^(a)Interactions of a meteorite or a volcanic eruption with a repository.

Table 7.8-2 Fatal cancers per curie released to the accessible environment for releases to surface water

Radioisotope	Total	Ingestion						Inhalation		External Dose		
		Drinking Water	Freshwater Fish	Surface Crops	Milk	Beef	Ocean Fish	Ocean Shellfish	Resuspended Materials	Ground Contaminant	Air Submersion	
C-14												
M-89	6.88E-08	7.07E-08	1.80E-08	6.07E-08	6.79E-07	2.83E-08	1.73E-08	7.20E-07	4.01E-10	4.79E-10	1.89E-18	
Sr-90	2.00E-02	4.28E-03	1.20E-04	2.02E-02	1.37E-03	8.28E-08	2.68E-08	4.43E-08	4.88E-09	0.00E+00	0.00E+00	
Zr-93	2.70E-04	2.81E-08	2.38E-07	2.17E-04	6.88E-07	9.80E-08	1.38E-08	9.08E-07	8.80E-08	2.10E-07	7.03E-14	
Ta-99	8.08E-04	9.89E-08	1.08E-08	2.78E-04	1.18E-04	1.81E-08	2.38E-08	1.98E-08	8.88E-11	0.00E+00	2.43E-19	
Sr-126	1.89E-02	4.88E-04	3.88E-03	9.38E-04	4.22E-08	8.84E-08	3.42E-08	1.90E-04	7.83E-09	1.02E-02	1.84E-10	
I-129	6.31E-02	2.48E-03	2.07E-04	6.27E-02	7.88E-03	1.03E-04	8.10E-08	6.08E-08	2.80E-08	7.78E-08	9.88E-13	
Co-138	1.08E-02	3.23E-04	1.07E-03	8.28E-03	7.78E-04	4.28E-08	3.20E-08	3.34E-08	7.34E-09	0.00E+00	0.00E+00	
Co-137	1.47E-02	2.22E-03	7.37E-03	3.47E-03	1.18E-03	6.37E-08	2.84E-08	2.88E-08	1.81E-09	4.32E-04	6.02E-12	
Bm-181	1.84E-08	9.88E-08	6.73E-07	9.00E-08	1.22E-08	8.89E-08	1.04E-07	8.82E-07	3.80E-09	0.00E+00	1.88E-17	
Pb-210	2.07E-01	9.84E-02	2.28E-02	8.10E-02	1.83E-03	3.87E-08	7.42E-03	4.12E-03	4.48E-07	1.38E-07	8.34E-18	
Ra-226	2.84E-01	1.02E-01	1.30E-02	1.23E-01	3.82E-03	9.88E-08	8.41E-03	2.14E-03	1.11E-08	1.38E-02	2.11E-10	
Ra-228	3.08E-02	1.81E-02	2.08E-03	1.18E-02	4.71E-04	1.09E-08	8.03E-08	2.01E-08	8.82E-07	2.81E-04	6.82E-12	
Ac-227	9.83E-02	6.38E-02	3.43E-03	3.80E-02	7.01E-08	1.70E-08	3.87E-04	2.48E-03	6.18E-08	1.48E-04	2.88E-12	
Th-229	7.81E-02	1.33E-02	1.02E-03	1.78E-02	8.93E-08	1.37E-07	2.42E-02	9.07E-03	6.28E-04	1.00E-02	3.04E-10	
Th-230	1.11E+00	1.08E-01	8.01E-03	6.31E-01	1.21E-04	2.81E-08	2.18E-01	7.28E-02	4.78E-04	1.89E-01	2.83E-08	
Th-232	4.89E-01	1.88E-02	1.44E-03	2.32E-01	4.78E-08	1.11E-08	3.88E-02	1.33E-02	7.00E-04	1.80E-01	3.82E-08	
Pa-231	1.83E-01	7.88E-02	2.20E-03	9.88E-02	3.38E-08	1.30E-08	1.88E-03	3.13E-04	6.81E-04	1.02E-02	2.37E-10	
U-233	3.44E-02	1.08E-02	2.87E-04	2.28E-02	4.91E-04	3.18E-08	2.87E-04	4.28E-08	8.83E-08	4.84E-08	1.80E-12	
U-234	3.21E-02	9.77E-03	2.48E-04	2.13E-02	4.88E-04	2.88E-08	2.41E-04	4.01E-08	6.48E-08	6.28E-07	1.80E-14	
U-238	3.48E-02	1.04E-02	2.84E-04	2.28E-02	4.88E-04	3.12E-08	8.88E-04	4.27E-08	6.70E-08	8.33E-04	2.11E-11	
U-239	3.04E-02	9.28E-03	2.38E-04	2.02E-02	4.34E-04	2.78E-08	2.28E-04	3.81E-08	6.21E-08	6.47E-08	1.41E-14	
U-239	3.27E-02	9.80E-03	2.84E-04	2.17E-02	4.87E-04	3.01E-08	2.48E-04	4.11E-08	4.83E-08	3.88E-08	2.84E-12	
Np-237	1.28E-01	3.81E-02	4.48E-02	3.48E-02	2.84E-08	1.02E-08	8.72E-03	1.48E-03	4.88E-08	8.84E-08	2.10E-12	
Pu-238	6.22E-02	3.84E-02	7.23E-04	2.84E-02	2.28E-07	8.88E-08	4.87E-08	6.41E-04	1.82E-08	2.84E-08	2.28E-18	
Pu-239	7.82E-02	3.78E-02	7.71E-04	3.30E-02	2.87E-07	1.04E-07	2.82E-04	2.81E-03	4.47E-04	3.21E-08	6.87E-14	
Pu-240	7.30E-02	3.78E-02	7.71E-04	3.14E-02	2.81E-07	1.01E-07	2.28E-04	2.84E-03	3.80E-04	8.88E-08	6.14E-14	
Pu-241	3.12E-03	1.78E-03	3.88E-08	1.28E-03	1.18E-08	4.48E-08	1.18E-08	1.28E-08	1.38E-07	1.40E-08	2.47E-18	
Pu-242	7.22E-02	3.88E-02	7.71E-04	3.21E-02	2.88E-07	9.88E-08	2.80E-04	2.88E-03	4.44E-04	6.84E-08	6.21E-14	
Am-241	8.40E-02	3.82E-02	8.10E-03	3.13E-02	1.11E-08	1.87E-07	7.08E-04	4.72E-03	6.48E-08	8.42E-08	1.48E-12	
Am-243	8.88E-02	3.88E-02	8.08E-03	3.48E-02	1.20E-08	2.03E-07	2.08E-03	1.37E-02	1.13E-04	8.88E-04	3.84E-11	
Cm-248	1.78E-01	7.87E-02	8.08E-03	8.88E-02	1.11E-04	3.78E-07	4.27E-03	2.84E-02	6.44E-04	7.70E-04	3.74E-11	
Cm-248	8.88E-02	3.87E-02	2.83E-03	2.87E-02	8.48E-08	1.88E-07	2.13E-03	1.42E-02	2.71E-04	3.38E-08	2.70E-14	

Table 7.8-3 summarizes the procedure used to arrive at the release limits in 40 CFR Part 191, Table 1 of the final rule. First, the number of fatal cancers caused per curie released to surface water for each radionuclide (the first column of Tables 7.8-1, 7.8-2 and 7.8-3) was inverted and multiplied by 10 to determine the number of curies of that radionuclide that would cause 10 health effects (shown in the second column of Table 7.8-3). Then, these estimates were rounded to the nearest order of magnitude to reflect the approximate nature of all of these calculations. (For example, if a number was between 100 and 1,000, it was generally rounded to 100 if it was less than 316, the logarithmic midpoint, and to 1,000 if it was more than 316).

Certain judgments were made in determining whether to round values up or down, taking into account uncertainties in the long-term risk estimates for each radionuclide. For example, the projected curies of strontium-90 and the various isotopes of uranium needed to cause 10 health effects are all about the same and are all near the midpoint of the rounding range. However, the release limit for the relatively short-lived strontium-90 was rounded up to 1,000 curies, while the release limits for the very long-lived uranium isotopes (for which ultimate environmental pathways will be more uncertain) were conservatively rounded down to 100 curies.

As indicated in the third column of Table 7.8-3, this procedure and the judgments that were made resulted in a total of four different release limits being determined for the various radionuclides listed: 10, 100, 1,000 and 10,000 curies per 1,000 MTHM. Most of the alpha-emitting radionuclides are subject to the same release limit (100 curies/1,000 MTHM), while most of the beta- or gamma-emitting radionuclides are subject to the same release limit (1,000 curies/1,000 MTHM), with a few exceptions that are subject to higher or lower release limits. Thus, all of the radionuclides may be grouped into four different categories, with the radionuclides in each category being subject to the corresponding release limit, as indicated in Table 7.8-4:*

The release limits listed in Table 1 are the estimated quantities of each radionuclide that would cause 10 premature cancer deaths if released to the accessible environment, which is the maximum number of premature cancer deaths allowed per 1,000 MTHM under the standard. It should be emphasized that a waste package containing multiple radionuclides is not allowed to release up to the above-listed release limits for each radionuclide but rather is subject to a sum-of-the-fractions limit per radionuclide, as described in Note 6 to Table 1.

*It should be noted that the regulatory effect of grouping radionuclides in this way is exactly the same as if the radionuclides were listed individually, as was done in the version of 40 CFR Part 191 promulgated in 1985 and in the February 3, 1992 draft of the standard. The grouping is done only for the purpose of simplifying the structure of Table 1.

Table 7.8-3 Development of Release Limits presented in Table 1 of 40 CFR Part 191

Radionuclide	Fatal cancers per curie released to surface water ^(a)	Curies required to cause 10 fatal cancers ^(b)	Release limit per 1000 MTHM or other waste ^(c) (curies)
C-14	5.83×10^{-2}	172	
Ni-59 ^(d)	6.88×10^{-5}	145349	100
Sr-90	2.60×10^{-2}	385	1000 ^(d)
Zr-93 ^(d)	2.70×10^{-4}	37037	1000
Tc-99	5.09×10^{-4}	19646	1000 ^(d)
Sn-126	1.89×10^{-2}	529	10000
I-129	6.31×10^{-2}	158	1000
Cs-135	1.05×10^{-2}	952	100
Cs-137	1.47×10^{-2}	680	1000
Sm-151 ^(d)	1.94×10^{-2}	515464	1000
Pb-210 ^(e)	2.07×10^{-2}	48	1000 ^(d)
Ra-226	2.64×10^{-2}	38	100 ^(e)
Ra-228 ^(e)	3.06×10^{-2}	327	100
Ac-227 ^(e)	9.93×10^{-2}	101	100 ^(e)
Th-229 ^(e)	7.51×10^{-2}	133	100 ^(e)
Th-230	1.11×10^0	9	100 ^(e)
Th-232	4.89×10^{-1}	20	10
Pa-231 ^(e)	1.93×10^{-1}	52	10
U-233	3.44×10^{-2}	291	100 ^(e)
U-234	3.21×10^{-2}	312	100
U-235	3.45×10^{-2}	290	100
U-236	3.04×10^{-2}	329	100
U-238	3.27×10^{-2}	306	100
Np-237	1.25×10^{-1}	80	100
Pu-238	6.22×10^{-2}	161	100
Pu-239	7.52×10^{-2}	133	100
Pu-240	7.30×10^{-2}	137	100
Pu-241 ^(d)	3.12×10^{-3}	3205	100
Pu-242	7.22×10^{-2}	139	1000 ^(d)
Am-241	8.40×10^{-2}	119	100
Am-243	9.85×10^{-2}	102	100
Cm-245 ^(e)	1.79×10^{-1}	56	100
Cm-246 ^(e)	8.86×10^{-2}	113	100 ^(e)

^(a) From Table 7.8-1.

^(b) Equal to 10 fatal cancers/per curie released to surface water (column 2).

^(c) Cumulative releases to the accessible environment for 10,000 years after disposal.

^(d) Included in 40 CFR Part 191 as any other radionuclide with half-life greater than 20 years that does not emit alpha particles.

^(e) Included in 40 CFR Part 191 as any other alpha-emitting radionuclide with half-life greater than 20 years.

Table 7.8-4 Release Limits for Containment Requirements Specified in Table 1 of 40 CFR Part 191

Radionuclide	Release Limit in Ci per 1,000 MTHM or other unit of waste
Th-230 or -232	10
C-14, I-129, or any alpha-emitting radionuclide other than Th-230 or -232	100
Any beta- or gamma-emitting radionuclide except C-14, I-129, or Tc-99	1,000
Tc-99	10,000

7.8.2 Waste Quantities Equivalent to 1,000 MTHM of Spent Fuel

As noted above, the release limits that are specified in Table 1 of 40 CFR Part 191 apply to every 1,000 MTHM of spent fuel emplaced in the repository. Since 40 CFR Part 191 release limits apply to other categories of waste besides spent fuel, it is necessary to identify the quantities of various non-spent fuel wastes to which the release limits are also to be applied.*

In the version of 40 CFR 191 promulgated in 1985, Note 1(c) specified that the Table 1 release limits applied to each 100 million curies of beta/gamma-emitters with half-lives between 20 and 100 years; Note 1(d) stated that the limits applied to each 1 million curies of beta/gamma-emitters with half-lives greater than 100 years; and Note 1(e) stated that they applied to each 1 million curies of transuranic wastes containing alpha-emitters with half-lives over 20 years. EPA did not provide a detailed explanation of the basis for these quantities. The preamble to the proposed 40 CFR 191 published in 1982 stated that the 1 million Curie quantity specified in Note 1(e) for TRU waste was chosen "so that the standards would require radioactivity from either high-level or transuranic wastes to be isolated with about the same degree of effectiveness." (EPA82a) Furthermore, the Draft EIS stated that the reference values of 1 million curies of TRU waste and 1,000 MTHM of spent fuel "were selected so that about the same fraction of transuranic radionuclides would be retained for either high-level or transuranic wastes." (EPA82b, p. 116) These estimates had

*In the case of high-level waste from reprocessing, the original MTHM of the spent fuel that was reprocessed and its burnup level are often known, and the release limits may be specified on the same basis as for spent fuel. However, in the event that high-level waste is being disposed and these parameters are not known, or another category of waste is being disposed such as defense TRU waste or Greater-Than-Class C waste, the release limits must be determined on a different basis.

been made based on looking at the long-term residual of TRU curies in spent fuel and rounding to the nearest arithmetic order of magnitude.

Further analysis shows that other methods that could be used to arrive at different bases for other wastes. A detailed analysis for one approach is provided in Nu92. Since the numerators in such fractions – the Table 1 release limits – are fixed values, the denominators can be made equivalent as an approach to equivalent release fractions. The denominators are the quantities of each category of waste to which the Table 1 release limits should apply. The denominator for spent fuel is 1,000 MTHM; therefore, the denominator for another waste should be the quantity of such waste that is equivalent to 1,000 MTHM of spent fuel.

Using the just described basis, one method for determining the quantity of a waste that is equivalent to 1,000 MTHM of spent fuel is to identify the number of curies initially present in 1,000 MTHM of spent fuel. This method was used by EPA in developing the 1985 rule. EPA's reference repository inventory, identified in Table 2-1 of its 1982 "Population Risks" report (EPA82c, p. 194), indicates that a repository containing 100,000 MTHM of spent fuel would contain an initial transuranic inventory of 470 million curies. (This value is based on 10-year-old spent fuel; 100,000 MTHM of new spent fuel would have a transuranic inventory of 336 million curies.) Thus, each 1,000 MTHM of spent fuel could be said to equal between 3 and 5 million curies of TRU waste, based on initial inventories; therefore, a quantity somewhere in this range would require about the same fraction of transuranic radionuclides to be retained in HLW and TRU waste, based on a comparison of initial inventories.

For Notes 1(c) and (d), EPA's reference repository inventory gives initial totals of 146 million curies of beta/gamma-emitters with half-lives between 20 and 100 years and 17,000 curies of beta/gamma-emitters with half-lives greater than 100 years. Thus, the quantity given in Note 1(c) of the 1985 rule would permit about the same fraction of this category of radionuclides to be retained in various wastes, if an allowance is made for rounding. However, the quantity in Note 1(d) of the 1985 rule requires a much smaller fractional release of this category of radionuclides from non-spent fuel wastes as from spent fuel, since it is also meant to include unknown quantities of alpha-emitters with half-lives greater than 20 years and was therefore made consistent with the approach in 1(e). This approach was especially deemed appropriate since this category is meant to serve in situations on source term uncertainty in conjunction with Note 5.

Another option is to determine the Curie inventory of 1,000 MTHM of spent fuel integrated over the entire 10,000-year isolation period, rather than just the initial inventory. This method may be preferable since it would require equivalence in the cumulative release fractions for spent fuel and other wastes, rather than just equivalent release fractions initially, when releases are actually the least likely to occur. Since spent fuel and other wastes have different radionuclide compositions with different half-lives, their inventories decay away at different rates. If the Table 1 release limits are applied to equal initial inventories of different waste streams, they will apply to different amounts of different waste streams at future points in the waste isolation period, allowing different cumulative release fractions.

Using integrated curies over 10,000 years, rather than just initial curies, would correct this deficiency.

This method of comparing integrated curies would determine the quantity of a waste that has the same number of radioactive disintegrations over 10,000 years as 1,000 MTHM of spent fuel. A further possible improvement would be to weight the number of integrated curies of each radionuclide by the risk factor for each radionuclide (using the values in Table 7.8-3). The resulting method would determine the quantity of a waste that has the same risk potential over 10,000 years as 1,000 MTHM of spent fuel, and apply the Table 1 release limits to that quantity.

If this method is used, and the Table 1 release limits are applied to the quantity of a waste that has the same risk potential over 10,000 years as 1,000 MTHM of spent fuel, a different quantity would be determined for each waste stream depending on the mixture of radionuclides present. As a result, it would not be possible to specify values in Notes 1(c), (d) and (e), as in the 1985 version of the rule. Instead, Note 1 would specify a formula for determining the quantity of any waste that is equivalent to 1,000 MTHM of spent fuel in terms of total risk potential over 10,000 years. The risk potential of 1,000 MTHM of spent fuel over 10,000 years is:

$$\frac{\sum I_{LSF} R_i}{1000 \text{ MTHM}}$$

where I_{LSF} is the number of integrated curies of each radionuclide i present in 1,000 MTHM of spent fuel from 100 to 10,000 years after disposal, and R_i is the risk factor for each radionuclide. To determine the equivalent quantity Q_j of a waste j , this expression may be set equal to the risk potential over 10,000 years of Q_j ,

$$\frac{\sum I_{ij} R_i}{Q_j} = \frac{\sum I_{LSF} R_i}{1000 \text{ MTHM}}$$

and the equation may be solved for Q_j :

$$Q_j = \frac{\sum I_{ij} R_i}{\sum I_{LSF} R_i / 1000 \text{ MTHM}}$$

This is the quantity of any waste j that is equivalent to 1,000 MTHM of spent fuel in terms of risk potential, and to which the Table 1 release limits would be applied under the formula method. Under this option, Notes 1(c), (d) and (e) would be replaced with a single Note 1(c) stating that the Table 1 release limits apply to the quantity determined by this formula. Thus, under the formula method, EPA would not prescribe in Note 1 any

numerical quantities to which the Table 1 release limits apply for non-spent fuel wastes. The quantities would instead be determined in the future, when the compositions of these other wastes are better known, based on the formula. This single formula could be used for any waste, including defense transuranic or high-level waste and various categories of Greater-Than-Class C waste.

EPA is currently considering the above discussed options for determining the quantities of non-spent fuel wastes to which to apply the Table 1 release limits, and requests public comment on these options. EPA also welcomes public comments on other alternatives, such as:

- basing the release standards for non-spent fuel wastes on technical achievability for each waste stream;
- using radionuclide mass, rather than the Curie inventory or the risk potential, to compare quantities of spent fuel and other wastes (this is equivalent to the integrated Curie method, but integrated until complete decay rather than to 10,000 years);
- limiting each reference repository to a certain level of total radionuclide release;
- allowing a risk to all future generations from waste disposal equivalent to the risk to present populations from the fuel cycle operations that produced the waste; and
- making risk based a policy determination of the maximum risk that should be allowed as a result of the geologic disposal of other categories of waste (e.g., whereas the rule would allow 1,000 premature cancer deaths over 10,000 years as a result of the disposal of 100,000 MTHM of commercial spent fuel, how many premature cancer deaths should it allow as a result of the disposal of various categories of defense waste?).

7.8.3 Burnup Level

The release limits in Table 1 apply to spent fuel of medium burnup, i.e., 25,000 to 40,000 MWd/MTHM. For spent fuel of higher or lower average burnup, adjustments need to be made to the unit of waste to which the release limits apply.

In general, as the burnup level of fuel increases, the concentration of various long-lived fission products and actinides in the fuel also increases. Therefore, if no adjustment were made to the units of waste to which the release limits apply, then the same quantity of radionuclides would be authorized to be released regardless of the total inventory of radionuclides; i.e., different release fractions would apply to fuels of different burnup. To provide equal release fractions for all spent fuel regardless of the burnup level, an adjustment is necessary to reflect the different radionuclide concentrations.

Thus, Note 3 to Table 1 states that for fuels with burnups outside of the medium range of 25,000 to 40,000 Mwd/MTHM, the unit of waste shall be multiplied by "the ratio of 30,000 Mwd/MTHM divided by the fuel's actual average burnup..." For example, in the case of 1,000 MTHM of fuel with an average burnup of 50,000 Mwd/MTHM, the unit of waste to which the release limits of Table 1 would apply would be $1,000 \text{ MTHM} \times 0.6 = 600 \text{ MTHM}$.

7.8.4 Wastes Arising From Fuel with Poorly Known Burnup and/or Original MTHM

As noted above, for wastes which are neither spent fuel nor high-level waste derived from spent fuel, the 1985 rule stated that Notes 1(c), (d) and (e) are to be used to determine the quantity of waste that is equivalent to 1,000 MTHM of spent fuel and for which the release levels in Table 1 are authorized.

In addition, as provided by Note 5 to Table 1, Notes 1(c), (d) and (e) may also be used for spent fuel and high-level waste in situations where either the burnup level and/or the number of metric tons from which a waste stream arose is not known or there is a range of values for these parameters because of uncertainties in the history of the fuel. Note 5 states that when the original MTHM and average burnup cannot be quantified, the units of waste derived from Notes 1(a) and (b), which require knowledge of the original MTHM and average burnup levels, shall not be used. Instead, Notes 1(c), (d) and (e) are to be used. The note also states that if uncertainties regarding the original MTHM and average burnup result in a range of values being considered for these variables, then

the calculations described in previous Notes will be conducted using the values that result in the smallest Release Limits, except that the Release Limits need not be smaller than those that would be calculated using the units of waste defined in (c), (d) and (e) of Note 1.

This means simply that from the range of original MTHM and average burnup figures that a waste stream is believed to possibly have, the smallest MTHM and average burnup figures should be assumed (so that a correspondingly small release amount is allowed) to calculate the release limit multiplier. However, if the resulting release amounts are smaller than would be calculated under Notes 1(c), (d) and (e), the values obtained using the method of Notes 1(c), (d) and (e) may be substituted.

7.9 UNCERTAINTY ANALYSIS

Environmental pathway dosimetry and risk models generally employ an environmental transport methodology consisting of multiplicative chain algorithms incorporating several variables. When regulatory analyses are performed, the tendency is to choose conservative values for these variables due to the inherent uncertainty in the parameters. The multiplicative nature of the models means that conservatism in choosing values for individual parameters can lead to larger conservatism in the result. The problem with this approach is that widespread conservatism can lead to extremely conservative and sometimes unrealistic results.

The consideration of uncertainty in individual parameter values used in environmental pathway models has been a subject of discussion in the technical community for more than a decade (Ba79a,b, Ho79, Ru79, Sh79, Ri83, Ru83, An84, Ho82).

When considering the uncertainty in the input parameters associated with environmental pathway calculations, the most common procedure has been to consider the range of reported parameter values qualitatively and to use judgement to select the "best" value to use for a particular application. More recently, attempts have been made to analyze the distribution of data for individual parameters statistically and to choose a mean or median as the "best" value for regulatory purposes.

It appears that the most systematic mechanism for considering uncertainty in multiplicative chain models would be to include a probability distribution representing current uncertainty about parameter values in the input data and to run a sufficient number of cases (with parameter values for each case chosen by a suitable sampling procedure) such that the distribution of results can be evaluated. The results of this type of analysis could be considered in choosing an appropriate set of single-valued parameters to apply for regulatory calculations. Alternately, a decision might be made to perform the regulatory calculations probabilistically, then choose limits for a standard (or to perform calculations to see if a limit is met) at a specified confidence level. We believe the subject needs additional study to determine the most appropriate use of uncertainty analysis for standards setting applications; however, it is clear to us that a quantitative analysis of the uncertainties is very useful for identifying the important parameters for more intensive consideration.

Most of the technical analyses discussed in this chapter were performed prior to the increase in emphasis on uncertainty in risk assessment calculations. In most cases, point values for each parameter were chosen after review of the range of values reported in the literature and were chosen to be near the mean or median value to avoid obtaining unrealistically conservative results.

In response to one of the recommendations of the SAR Subcommittee that reviewed the technical basis for 40 CFR Part 191, the Agency commissioned an uncertainty analysis of the calculations that produced the estimates for fatal cancers per curie of radioactivity released to surface water (ENV85, SCA91). The surface water pathway models were reviewed and a sensitivity analysis was performed to identify the key uncertainty parameters. The uncertainties in these parameters were characterized by probability distributions that were propagated through the models using a simulation technique, which produced uncertainty distributions in the estimates of fatal cancers per unit of radionuclide release to surface water.

An example of the results of the uncertainty analysis is presented in Figure 7.9-1, which shows uncertainty distributions for the fatal cancers risk for Am-243 releases to surface water. Figure 7.9-1 shows a probability plot of the fatal cancers per 10,000 year per curie of Am-243 released to surface water. This figure indicates a 60 percent probability, considering parameter uncertainties, that a release limit of 10 curies for Am-243 will result in 10 or less deaths over 10,000 years per 1000 MTHM. The level of confidence indicated in the example is not necessarily representative of all radionuclides analyzed (ENV85, SCA91).

Figure 7.9-1 Probability Distribution of population risks per curie of Am-243 released to surface water.

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Chapter 11: PERSPECTIVES ON RISKS FROM THE DISPOSAL OF SPENT NUCLEAR FUEL, HIGH-LEVEL AND TRANSURANIC RADIOACTIVE WASTES

As a basis for comparison with the risks associated with spent nuclear fuel, high-level and transuranic radioactive waste (hereafter referred to as radioactive waste), estimates have been made of the risks from five radiation sources. The risks from three of these sources, natural background radiation, natural radionuclide concentrations in ground water, and uranium ore bodies are relatively constant and will continue far into the future. The risks from the other two sources, generation of nuclear power and nuclear weapons fallout, are more near-term and will decrease with time. The estimates of population risks from natural background radiation, generation of nuclear power, and nuclear weapons fallout are higher than those estimated from radioactive waste after disposal. The estimates of risks from uranium ore bodies vary over a large range. Some of the estimates of the risks from the disposal of radioactive waste are above the lower bound of this range and are less than the higher part of the range.

Risks allowed under other regulatory standards, both U.S. and international, are also discussed. Because of the unique nature of 10 CFR 191 and the need to consider intergenerational risks, considerable care must be exercised in comparing risks under other standards, but they can provide a sense of perspective.

11.1 RADIATION RISKS FROM NATURAL SOURCES

11.1.1 Risks Associated with Natural Background Radiation

In 1987, the National Council on Radiation Protection and Measurements (NCRP) issued NCRP Report 94, Exposure of the Population in the United States and Canada from Natural Background Radiation (NCRP87b), as an update of its previous Report No. 45, Natural Background Radiation in the United States (NCRP75). NCRP87b reflects the fact that since publication of NCRP75, it has become apparent that the inhalation of the short-lived decay products of radon-222 indoors is the most significant source of natural exposure to radiation. In addition, the update reflects other refinements, among them the increase in quality factor for alpha radiation from 10 (used in NCRP75) to 20 and the use of the effective dose equivalent concept. Except for those places where a separate reference is noted, the information in this section is taken from NCRP87b.

Natural radiation has two components: external (from outside the body), and internal where radioactive materials enter the body through inhalation or ingestion. NCRP87b first considers the sources separately and then combines their effects to provide the total dose equivalent to man, based on best estimates of the dose equivalents to selected important body organs. Humans are exposed to natural background radiation from primordial radionuclides, cosmogenic radionuclides, and cosmic radiation.

Primordial radionuclides are mostly isotopes of the heavy elements and belong to the three radioactive series headed by uranium-238 (uranium series), uranium-235 (actinium series), and thorium-232 (thorium series). The major primordial radionuclides that decay directly to a stable state are potassium-40 and rubidium-87. Although the isotopic abundance of potassium-40 is small (0.0118 percent), potassium is so widespread that potassium-40 contributes about one-third of both the external terrestrial and the internal whole-body dose from natural background radiation.

Cosmogenic radionuclides mostly have low or intermediate atomic numbers and are produced by interactions of cosmic nucleons with target atoms in the atmosphere or in the earth. The four major contributors to human exposure are carbon-14, tritium, beryllium-7, and sodium-22.

Cosmic radiation originates when charged particles, primarily protons, from extra-terrestrial sources that are incident upon the Earth's atmosphere, have sufficiently high energies to generate secondary particles that penetrate to ground level. The secondary particles (mostly high-energy muons and electrons) produce an absorbed dose in ground-level air at a rate comparable to that from the beta and gamma radiation generated by terrestrially distributed radionuclides.

Table 11.1-1, adapted from NCRP87b, provides the estimated total effective dose rate for a member of the population in the United States from natural background radiation. The rounded total effective dose rate was found to be 300 mrem/yr, or 3.0 mSv/yr.

Table 11.1-1 Estimated total effective dose rate from various sources of natural background radiation for a member of the population in the United States^a

Source	Total effective dose rate (mrem/yr)
Cosmic	27
Cosmogenic	1
Terrestrial	28
In the Body	200
Rounded Totals	300

^a Adapted from NCRP87b, Table 9.7.

The UNSC88 report also provided estimates of annual effective dose equivalent rates from natural sources in areas of normal background. UNSC88 reports a total of 240

mrem/yr, 80 mrem/yr from external sources and 160 mrem/yr from internal radiation. The differences between the UNSCEAR and NCRP estimates are contained in the estimates of effective dose equivalent rate from inhalation of radon decay products.

NCRP87b does not provide quantitative information on the variability of exposure in the United States. The report notes that there are many locations in the United States where exposures differ markedly from the mean. Earlier information on the variability of the dose equivalent received by people in the United States has been published by Oakley (Oa72) and revised by Bogen (Bo80). NCRP75 noted that cosmic radiation annual dose equivalent in the United States varies from 29 mrem at sea level to about 125 mrem at 3200 meters (10,500 feet), the altitude of Leadville, Colorado. The terrestrial component dose equivalent from radionuclides in the Earth has been characterized as low (annual average of 23 mrem) for the Atlantic and Gulf coastal plain, moderate (annual average of 46 mrem) for most of the rest of the United States, and high (annual average of 90 mrem) for a small region in the Colorado Front Range. While these earlier numbers do not reflect the current state of knowledge about the contribution from radon-222, they provide a perspective on the degree of variability present.

EPA85 estimated that people in the United States annually receive a total of about 50 to 195 mrem from natural background radiation. The highest annual dose equivalents, received in large urban areas, were estimated to be about 130 mrem in Denver, Colorado and about 115 mrem in Albuquerque, New Mexico, with the highest annual average being about 115 mrem, in Colorado. Table 11.1-2 gives the EPA85 distribution of the dose received by the population. It shows that about 60 percent of the United States population receives an annual dose equivalent ranging from 65 to 90 mrem; 20 percent more than 90 mrem; and 5 percent more than 105 mrem.

EPA estimates that an excess of about 400 fatal cancers per million person-rem can be expected (EPA89). This means that an increase of one mrem/yr in natural background radiation (about one percent) to everyone in the present U.S. population would result in about 85 excess fatal cancers per year. A one mrem/yr increase in radiation exposure to five million people (a population that might be exposed to releases from a large high-level radioactive waste repository) would result in about two additional fatal cancers per year.

Risks associated with natural background radiation are similar to those from radioactive wastes; both sources cause health effects and persist for long periods of time. The risk from natural background radiation is much larger than that projected for releases from a radioactive waste repository. However, in general, the variability of the risk from natural background radiation does not seem to greatly influence the choices people make in selecting a place to live.

**Table 11.1-2 Distribution of natural radiation annual dose equivalents
(Terrestrial, cosmic, and internal)^a**

Dose equivalent^b (mrem)	Percentage of U.S. population receiving more than stated dose
50	100
65	80
75	50
90	20
105	5
120	1

^aFrom EPA85.

^bAnnual dose equivalent includes 25 millirems from internal radiation.
Values have been rounded to the nearest 5 mrem.

11.1.2 Risks Associated with Natural Radionuclide Concentrations in Ground Waters

One part of human exposure to background radiation comes from naturally occurring radionuclides in the Earth that, in turn, are taken up by ground water. Radium is the most important of these naturally occurring materials likely to occur in public water supply systems. Uranium is also found in ground waters due to its natural occurrence. Thus, it is expected that some drinking water sources will contain concentrations of uranium (Co83, 84).

Surveys of radionuclides in ground-water systems indicate a United States range of: 0.1 to 50 picocuries/liter (pCi/l) for radium-226 (with some isolated sources exceeding 100 or more pCi/l); 0 to 74 pCi/l for gross alpha concentrations (although most of the United States gross alpha concentrations are below 3 pCi/l); and 0 to 650 pCi/l for total uranium. Elevated radium-226 concentrations are seen along the Atlantic coastal region (North and South Carolina, Georgia, and Florida) and the Midwest (Illinois, Wisconsin, Minnesota, and Kansas). Low levels, or almost none, are found in treated drinking waters in the western states. In general, only those locations with elevated uranium concentrations show elevated alpha concentration levels, and these are primarily in the Rocky Mountain region (New Mexico, Colorado, Wyoming, and Montana) and in Maine and Pennsylvania in the east (Co83, Dr81).

Of the nearly 60,000 community water supplies in the United States, about 80 percent use ground-water sources and over 90 percent of the ground-water supplies serve less than

3,300 people. In general, radium in drinking water is a problem for small water supply systems, with an estimated 600 supplies, half of which serve 500 or fewer people, exceeding 5 pCi/l, and 70 supplies, serving approximately 890,000 people, exceeding 20 pCi/l of radium-226. It is estimated that 500 supplies exceed 5 pCi/l of radium-228, while 40 systems exceed 20 pCi/l and 15-20 exceed 30 pCi/l. The corresponding exposure estimates are that 1.3 million people using ground water supplies receive water with radium-228 levels above 5 pCi/l, 164,000 are exposed to water exceeding 20 pCi/l, and about 82,000 are exposed to water exceeding 30 pCi/l. It is estimated that approximately 1500 community water supplies, serving approximately 875,000 people, exceed 20 pCi/l of uranium (EPA91).

11.1.3 Risks from Uranium Ore Bodies

A comparison of the risks from a high-level waste repository with those from an undisturbed uranium ore body offers another useful perspective. In this assessment, it is assumed that uranium dissolved by the ground water eventually reaches a stream (Wi80). As the uranium moves along the aquifer, radium-226 and other daughters are formed. An ore body containing 620,000 tons of uranium oxide would have to be mined to produce the high-level radioactive wastes contained in the model repository. An estimate of the risks presented by a model ore body containing estimated uranium reserves of 10,000 metric tons of uranium oxide was performed using the same generic environmental model described in Chapter 7 of this report (Sm82).

The magnitude of risks from uranium ore bodies cover a wide range, depending on the ability of uranium and its decay products (particularly radium-226) to leave the ore body and reach people. To estimate a minimum impact, a uranium concentration in ground water at the low end of the reported range was selected (Fi55). The yearly release of radium-226 to a stream from the model ore body was projected to be 120 microcuries. Based on a risk conversion factor of 2.64×10^1 fatal cancers per curie of radium-226 released to surface water (see Chapter 7), this would produce 3.2×10^5 fatal cancers per year. Applying the same risk conversion factor to estimated releases from three site specific case studies would result in estimated fatal cancers per year from released radium-226 ranging from 5.0×10^2 to 2.9×10^1 .

The minimum impact is estimated to be about 10 excess fatal cancers over 10,000 years for the model generic ore body. Based on pre-operational data for three actual uranium mines reported in Wi80, the maximum impact from the generic ore body is estimated to be about 100,000 excess cancers over 10,000 years. These estimates, based on data from uranium mines, are likely to be high because the reported concentrations in the ground water include some measurements from the oxidizing uranium-rich ground water. Therefore, the risk to future generations from an amount of uranium ore equivalent to that which would be used to produce 100,000 metric tons of spent fuel appears to range from 10 to 100,000 fatal cancers over 10,000 years.

11.2 RADIATION RISKS FROM MAN-MADE SOURCES

11.2.1 Risks from Nuclear Power Generation

Ellett and Richardson estimated that exposure to the radiation incident to the generation of 1 gigawatt-year of electrical energy could result in an average of about 1.2 fatal cancers in the first 100 years (the estimated number of fatal cancers represents a probability of incidence in the entire exposed population) (EI77). This is broken down into elements of the fuel cycle in Table 11.2-1 (EI77).

Table 11.2-1 Cancer Risks for Elements of the Nuclear Fuel Cycle

Element of Nuclear Fuel Cycle	Fatal Cancers / GW-yr
Extraction and Fabrication	0.5
Reprocessing	0.3
Nuclear Reactors	0.1
Occupational	0.2
Accidents	0.1
Total	1.2

More recent information reduces this estimate to 0.8 fatal cancers. Operation of the 50 gigawatts of nuclear power generation in 1977 at 60 percent of capacity could cause about 24 fatal cancers per year (ADL79). To the extent that radionuclides such as carbon-14, thorium-230, and radium-226 and its daughters persist beyond 100 years, this estimate may be low. Table 11.2-2 gives the fuel cycle doses as they are presented by UNSC88.

Table 11.2-2 Collective dose per unit practice of nuclear power generation (man-Sv per GW)

Element of the fuel cycle	Over the next 100 years	Over the next 10,000 years
Mill Tailings (radon), long term	1.5	150.0
Globally dispersed nuclides and waste	6.0	60.0
Local and regional exposures	4.0	4.0
Occupational Exposures	12.0	12.0
Total	24	230

UNSC88 provides information on the release of radioactive gases from operating reactors and fuel processing facilities and the resulting global collective dose commitments. This information can be used to provide a perspective for evaluating the risk posed by a repository. UNSC88 states that the "nuclides giving rise to a global collective dose commitment are sufficiently long-lived and migrate through the environment, thus achieving widespread distribution." Examples of these include carbon-14, tritium, krypton-85, and iodine-129. UNSC88 gives normalized collective effective dose equivalent commitments for each of the globally dispersed nuclides in terms of man-Sv per GW of electricity generated.

An estimate of the collective effective dose equivalent commitment over a 10,000-year period due to globally dispersed radionuclides from civilian reactor operation and fuel reprocessing can be estimated by multiplying the UNSC88 numbers by the amount of electricity generated by nuclear reactors. The International Atomic Energy Agency report, "Nuclear Power Reactors in the World," gives the cumulative net generation by nuclear reactors to 1990 as 18,090.2 TW(e)-hours, or approximately 2,065 GW-years (GW-yr) of generation (IAEA91). For example, the estimated collective effective dose equivalent commitment over 10,000 years from global dispersion of carbon-14 released by the generation of electricity from nuclear reactors (including fuel reprocessing) through 1990 thus would be:

$$(63 \text{ man-Sv/GW-yr})(2,065 \text{ GW-yr})(100 \text{ person-rem/man-Sv}) = 1.3 \times 10^7 \text{ person-rem}$$

If similar collective dose equivalent commitments based on the UNSC88 data are calculated for tritium, krypton-85, and iodine-129, and subsequently multiplied by the Agency's dose-to-risk conversion factor of 4×10^4 fatal health effects per rem, the resulting health effects over time are as shown in Table 11.2-1.

Table 11.2-3 Fatal health effects, truncated to different times for globally dispersed radionuclides, weighted for the fraction of fuel reprocessed.

Radio-nuclide	Years				
	10	100	1,000	10,000	1,000,000
Kr-85	5.8	10	10	10	10
H-3	0.25	0.33	0.33	0.33	0.33
C-14	140	520	990	5200	5200
I-129	-	0.07	0.13	0.77	120

(Based on normalized collective dose equivalent commitment numbers from UNSC88 and 2,065 Gw-yr of reactor operation through 1990, per IAEA91. UNSC88 notes that the environmental transfer of H-3, C-14, and Kr-85 are fairly well established and that reliable estimates of collective dose commitments can be made. For I-129, which has a half-life of 16 million years, UNSC88 cautions that there are substantial uncertainties in predicting population size, dietary habits, and environmental pathways.)

A different perspective can be gained by looking at operating reactors and fuel reprocessing in terms of activity released. The National Council on Radiation Protection and Measurements (NCRP) estimated that world-wide civilian reactor operations and fuel reprocessing have resulted in the release of approximately 235,000 Ci of carbon-14 through 1990 (NCRP85). The NCRP notes that this estimate was made from limited experimental data and does not include any releases from reactors operated for weapons production or military purposes, some of which started operation in the 1940s. Applying the Agency's risk conversion factor for carbon-14 of 5.83×10^{-2} fatal cancers per curie released would result in an estimate of more than 13,000 health effects. By contrast, the 40 CFR 191 release limit of 100 Ci per MTHM allows a release of 10,000 Ci of carbon-14 from a 100,000 MTHM repository which results in about 600 health effects.

11.2.2 Risks from Nuclear Weapons Fallout

Fallout from atmospheric testing of nuclear weapons has added to the radioactivity in air, water, food, and of residual radionuclides in soil (primarily tritium, strontium-90, cesium-137, carbon-14, and plutonium) and is a small but continual source of radiation exposure.

The UNSCEAR 1982 report calculated the effective dose equivalent commitment for the population of the northern hemisphere of 4.5 mSv (450 mrem), with 2.1 mSv (210 mrem) delivered from the start of testing up to the year 2000. NCRP87b estimated that the annual effective dose equivalent to an individual is less than 0.01 mSv (1 mrem) and will continue to decline assuming no resumption of testing which releases radioactivity to the atmosphere.

The total effect of fallout is perhaps more effectively viewed in terms of the global effect of the long-lived nuclides, particularly carbon-14. NCRP85 notes that nuclear weapons testing from 1952 to 1963 is estimated to have injected 9.6 MCi of carbon-14 into the atmosphere, while as noted above in Section 11.1.4, nuclear power operations have released about 235 KCi of carbon-14 through 1990. These provided a peak specific activity for carbon-14 in the atmosphere of 10.8 pCi/gC, which is now estimated to have diminished to 7.5 pCi/gC (NCRP85). Multiplying the EPA's value of 5.83×10^{-2} fatal cancers per curie released by the total release from weapons testing results in about 560,000 potential fatal cancers.

11.3 RISKS UNDER OTHER REGULATORY STANDARDS

11.3.1 Regulatory Standards for Radiation

11.3.1.1 Other EPA Regulatory Standards

In addition to 40 CFR 191, the EPA has promulgated other regulations and standards governing releases of radionuclides. These include 40 CFR 190, *Environmental Radiation Protection Standards for Nuclear Power Operations*, 40 CFR 61, the *National Emission Standards for Hazardous Air Pollutants; Radionuclides*, or NESHAPs; and 40 CFR Parts 141 and 142, the proposed *National Primary Drinking Water Regulations*.

In 40 CFR 190, the EPA established environmental standards for the uranium fuel cycle, excluding mining and radioactive waste disposal. The standards limit the annual dose equivalent to any member of the public from all phases of the fuel cycle (not including radon and its daughters) to 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ. In addition, to protect against buildup in the environment, the standard sets limits for long-lived radionuclides, including I-129. That standard states that for normal operation of fuel cycle facilities, operations shall be conducted in such a manner as to provide reasonable assurance that the total release of I-129 entering the general environment from the entire uranium fuel cycle, per GW-yr of electrical energy produced by the fuel cycle, must be less than 5 millicuries. If it is assumed that 5 millicuries had been released for each of the 2,065 GW-yr of electricity generated to date (IAEA91), this would result in a total release of 10.3 Ci to date. Multiplying this by the EPA's calculated number for I-129, 6.3×10^{-2} fatal cancers per curie released to surface water, results in about 0.7 fatal cancer over 10,000 years.

In its December 1989 promulgation of 40 CFR 61, the EPA issued standards for emissions of radionuclides other than radon from DOE facilities, facilities licensed by the Nuclear Regulatory Commission (NRC) and non-DOE facilities. These standards state that emissions of radionuclides to the ambient air from the regulated facilities shall not exceed those amounts that would cause any member of the public to receive in any year an effective dose equivalent of 10 mrem. The EPA's risk assessment for the DOE facilities showed that this level corresponded to an incidence, within 80 km of a facility, of 0.28 cancer deaths per year, and a maximum individual risk of 2.0×10^{-4} . For fuel cycle facilities, the EPA estimated an incidence within 80 Kilometers (km), of 0.1 cancer death per year or a maximum lifetime risk for an individual of 1.5×10^{-4} (EPA89a).

The Agency's proposed National Primary Drinking Water Regulations (40 CFR Parts 141 and 142) for radionuclides provide the following maximum contamination levels: radium-226, 20pCi/l; radium-228, 20 pCi/l; radon-222, 300 pCi/l; uranium, 20 micro g/l; adjusted gross alpha, 15 pCi/l; and beta and photon emitters, 4 mrem ede/yr. In general, these limits yield doses of between 4 mrem/yr and 20 mrem/yr to individuals drinking the contaminated water. The estimated lifetime cancer risk to an individual consuming 2 liters per day of water is between 10^{-4} and 10^{-6} , based on these limits (EPA91).

11.3.1.2 Nuclear Regulatory Commission Regulations

Two regulations promulgated by the NRC provide additional perspective. In Appendix I to 10 CFR 50, the NRC provides numerical guides for design objectives and limiting conditions for operation to meet the "As Low As Is Reasonably Achievable" (ALARA) criterion for releases of radioactive material from light-water-cooled nuclear power reactors. In 10 CFR 20, the NRC establishes standards for protection against radiation hazards arising out of activities licensed by the NRC.

11.3.1.2.1 10 CFR 50, Appendix I

In establishing its ALARA principle in Appendix I, the NRC provides a two-tiered approach. Appendix I first provides specific numerical design objectives. The applicant for a nuclear power reactor license must provide "reasonable assurance" that these design objectives are met. In addition, Appendix I requires that,

"the applicant shall include in the radwaste system all items of reasonably demonstrated technology that, when added to the system sequentially and in order of diminishing cost-benefit return, can for a favorable cost-benefit ratio effect reductions in dose to the population reasonably expected to be within 50 miles of the reactor."

Also Appendix I, the NRC also introduced, as an interim measure, the value of \$1,000 per total body man-rem to be used in cost-benefit analyses to satisfy the ALARA principle (i.e., measures which would cost \$1,000 or less per avoided man-rem would be included). Later, in its 1991 promulgation of revisions to 10 CFR 20, the NRC noted that it has not adopted a requirement that a numerical cost-benefit analysis be used to demonstrate ALARA.

As a specific limit, Appendix I recommends a design limit of 10 mrem/yr for the maximum calculated radiation dose to any one organ in an individual at the site boundary of a commercial light water nuclear power plant. This limit would result in an individual lifetime fatal cancer risk of 1×10^{-4} (NCRP89).

11.3.1.2.2 10 CFR 20

The standards for protection against radiation established in 10 CFR 20 apply to "all persons who receive, possess, use or transfer" material licensed by the NRC, including power reactor licensees. Part 20 sets out permissible doses, levels, and concentrations for restricted and unrestricted areas. The 1960 version of Part 20 noted that the NRC would approve proposed limits on radiation levels if "the applicant demonstrates that the proposed limits are not likely to cause any individual to receive a dose to the whole body in any period of one calendar year in excess of 0.5 rem."

The 1960 regulation further required that radiation levels not exceed that which would result in an individual receiving a dose in excess of 2 mrem/hr in any one hour or a dose in excess of 100 mrem in any seven consecutive days. For effluents to unrestricted areas, the regulation set out specific limits by individual radionuclide.

In the May 1991 revision to Part 20, the limit for protection of the general public was changed to an explicit limit of 0.1 rem per year. The revision retained the limit of 2 mrem/hr for any one hour but did not retain the limit of 100 mrem in any seven consecutive days. Part 20 notes that, in addition to its other requirements, licensees engaged in uranium fuel cycle operations subject to the provisions of 40 CFR 190 (described in section 11.3.3) must also comply with that part. Part 20 also incorporates the ALARA principle.

Applying the Agency's risk conversion factor of 4×10^{-4} fatal cancers per rem to the Part 20 limit for the general public, 0.1 rem per year, would result in an individual risk of 4×10^{-5} per year.

11.3.2 Regulatory Standards for Chemicals

Comparisons can be made with risks allowed under regulatory standards governing chemicals. Two specific EPA standards have been reviewed: the Underground Injection Control (UIC) Program, and the Hazardous Waste Management System Land Disposal Restrictions. These programs are discussed in the following sections in terms of their similarities and differences to 40 CFR 191. Chemicals may pose risks from either toxicity or carcinogenicity. Before describing these standards in particular, it may be useful to provide a general perspective on the comparison of risks from radiation and chemicals.

The NCRP has published a report, "Comparative Carcinogenicity of Ionizing Radiation and Chemicals," (NCRP89) which provides some information on the usefulness of applying methods used in expressing the risk of radiation exposure to the expression of risks from chemical exposure. NCRP89 notes that evaluation of the carcinogenic risks of chemicals differs from evaluation of the carcinogenic risks of radiation in several important respects, listed below:

- o Chemicals vary widely in their capacity to cause cancer ("carcinogenic potency").
- o The amount of data on individual chemicals is far smaller than the amount of data on radiation. For most chemicals, human data are lacking, and for many chemicals there are no available data on carcinogenicity.
- o External penetrating radiation affects cells directly, whereas chemicals and "internally deposited" radionuclides must be transported through the body and, in many cases, metabolized before reaching or acting on affected cells.

- o Chemicals and many radionuclides (e.g., iodine-131) often affect primarily only one organ, which may vary among species, while external penetrating radiation can affect virtually any organ, depending on the source and energy of the emissions.
- o Finally, chemicals often undergo metabolism to yield carcinogenic derivatives, the concentration of which is the relevant dose parameter.

NCRP89 states that because of these factors, analyses of the risks of chemicals are generally less precise than those of radiation. The report does provide some estimated occupational risks for various chemicals, as calculated by Albert (Alb83), in comparison with estimated risks for ionizing radiation (Table 11.3-1). NCRP cautions, however, that the calculations for radiation represent "best" estimates, while the calculations for the chemicals are deliberately more "conservative."

Table 11.3-1 Estimated Lifetime Cancer Risks Associated with Occupational Exposure Limits for Various Chemical and Physical Agents^a

Agent	Lifetime Risk of Cancer ^b
Radon (4 WLM/yr)	0.03
Radiation (50 mSv/yr)	0.02
Vinyl Chloride	0.0007
Acrylonitrile	0.03
Arsenic	0.003
Coal Tar	0.01
Benzene	0.01

^a From Alb83

^b Values tabulated denote calculated probabilities of an individual's developing cancer at some time during life following continuous annual occupational exposure for an entire working lifetime at the maximum permissible level.

In the preamble to the National Primary Drinking Water Regulations, discussed earlier in Section 11.3.1.1, the Agency discusses the adverse health effects of uranium from both its radioactive carcinogenicity and its chemical toxicity. The regulations set the Maximum Contaminant Level Goal (MCLG) for uranium at zero based on its carcinogenicity. The Maximum Contaminant Level (MCL), however, is set at 20 micro-grams per liter based on its chemical toxicity to the kidneys.

Risk has not been the single determining factor in establishing standards. NCRP89 states that "When exposures to chemical carcinogens have been high enough to cause noticeably large risks, there has usually been little doubt about the need to reduce the risks." In the case of vinyl chloride, it turned out to be cost-effective to reduce workplace exposure a thousand fold, to 1 ppm. This level was chosen because it was a feasible objective and could be reasonable attained, rather than by a specific calculation of risk.

11.3.2.1 Underground Injection Control of Hazardous Wastes

There are many similarities regarding the analysis of the disposal of hazardous wastes which relate to the potential health risks from disposal of radioactive wastes, e.g., the use of man-made and geologic media to physically separate them from the pathways of conventional human activity, the use of the 10,000-year regulatory time frame, the use of predictive models to demonstrate compliance over this period, and the application of the Safe Drinking Water Act (SDWA) requirements to restrict contamination of ground water outside the disposal area.

However, differences also exist. These differences arise from the unique physical and chemical attributes of the wastes being disposed of which affect the number of sites and the period over which the material remains hazardous. Other key differences include the "physical" properties of the two materials (e.g., liquid vs. solid), the process for emplacing them, and the engineered and geologic barriers used to segregate the wastes. These differences are critical in examining the wastes' respective ability and likelihood to create human health risks.

Background - Hazardous waste injection wells are regulated under the authority of both the SDWA and RCRA. The Hazardous and Solid Waste Amendments of 1984 (HSWA) imposed substantial new responsibilities on those who handle hazardous waste. These amendments prohibit the continued land disposal of untreated hazardous waste beyond certain dates, unless the Administrator of the EPA determines that prohibition is not required in order to protect human health and the environment for as long as the wastes remain hazardous.

The EPA's final rule on the underground injection of hazardous chemicals was published in the July 26, 1988 *Federal Register*. The final rule essentially prohibits the underground injection of hazardous wastes unless:

- (1) the waste has been treated in accordance with the requirements of Part 226 pursuant to section 3004(m) of RCRA; or
- (2) the applicant satisfactorily demonstrates to the EPA that there will be no migration of hazardous constituents from the injection zone for as long as the waste remains hazardous.

The EPA's Underground Injection Control (UIC) Branch estimated in 1991 that there are about 180 permitted Hazardous Class I injection wells spread over 80 facilities which require no-migration petitions in order to continue to operate. Of this number, 65 have submitted no-migration petitions. At that time, 36 petitions had been granted, 10 had been withdrawn (many of these facilities have decided to discontinue hazardous waste injection activities), with the remainder in various stages of action.

Regulatory Time Frame - Both the UIC program and 40 CFR 191 rely on the same regulatory time frame, i.e., 10,000 years. However, the reasons for the selection of this time period differ, basically because of the qualities of the wastes involved.

In regulating the underground injection of hazardous chemicals into Class I wells, the EPA specified a 10,000-year time frame because it believed such a time period would provide containment for a substantially long time frame and would allow time for geochemical transformations which would render the waste either nonhazardous or immobile. The following reasons were used in selecting the 10,000-year regulatory period for disposal of HLW: 1) it is long enough for releases through ground water from poorly selected and designed facilities to reach the accessible environment; 2) major geologic changes take much longer than 10,000 years to occur, so that the likelihood and characteristics of geologic events which might disrupt the disposal system are reasonably predictable over this period; and 3) compliance with quantitative standards for a substantially longer period would introduce much more uncertainty.

Predictive Models and Compliance Requirements - To demonstrate no-migration in a UIC petition, applicants must submit models demonstrating that the waste would not reach a point of discharge, either vertically or horizontally, for a period of 10,000 years. Petitioners must supply computer models and supporting information which includes site specific data regarding geology and hydrology as well as data relating to the injected constituents.

Class I injection wells are subject to continuous and extensive monitoring and reporting requirements. The EPA can require information on the formations found in the well bore, including logging, coring, testing and formation fluid sampling, other than in the injection and confining zones. For wells injecting corrosive waste, the Agency requires continuous corrosion monitoring of well construction materials and may require monitoring for other wastes. The EPA requires operators to test the well's mechanical integrity by looking for fluid movement behind the casing and for leaks in the tubing, at least every five years. Class I wells are also required to develop an ambient monitoring system which is consistent with Section 1426 of the SDWA. Each Class I hazardous waste injection well must also have a Waste Analysis Plan which further describes testing and monitoring requirements.

The Part 191 requirements call for a "reasonable expectation" that their various quantitative tests for disposal system performance will be met. Unequivocal numerical proof of compliance is neither necessary nor likely to be obtained. 40 CFR 191 states that

containment requirements can be implemented only through analytical projections of disposal performance. Compliance is generally to be demonstrated through long-term modeling projections of disposal system performance. The implementing agencies are required to perform analyses before the waste is emplaced in the system.

Unlike injection wells, high-level waste disposal systems will not rely on active institutional controls to isolate the wastes beyond 100 years after disposal. The EPA believes that conventional regulatory concepts involving monitoring of emissions or ambient levels of pollution would not be sufficient because such surveillance cannot be relied upon for the long period involved. However, disposal systems will be monitored to detect substantial changes from their expected performance until the implementing agencies determine that there are no significant concerns to be addressed by further monitoring. In addition, disposal sites are required to be identified by permanent markers, widespread records, and other passive controls to warn future generations of the dangers and locations of the wastes.

Statutory Definition of Injection Zone - The regulations relating to UIC and nuclear waste disposal systems are similar in that they do not seek to regulate or control waste within the disposal area; but instead seek to contain it there. The regulatory concern is for preventing the migration of dangerous waste to the accessible environment, not in controlling the hazardous material contained in the disposal area.

Essentially, the UIC program permits the use of certain geologic formations or parts of formations in the inaccessible subterranean environment for waste disposal so long as this disposal is sufficiently removed from ground-water or surface-water resources. Some amount of penetration into confining material within the injection zone may occur but should not be considered migration because the geologic material is considered part of the injection zone.

Part 191 does not seek to regulate water quality within the controlled area because geologic media within the controlled area are an integral part of the disposal system's capability to provide long-term isolation. Thus a certain area of the natural environment is envisioned to be dedicated to keeping these materials away from future generations and may not be suitable for other uses. 40 CFR 191 limits this "controlled area" to no larger than 100 square kilometers or five kilometers in any direction from the original emplacement of the wastes in the disposal system. The proposed rule allows the implementing agencies to choose a smaller area if appropriate.

Geologic and Man-Made Barriers - Both 40 CFR 191 and the UIC regulations rely on the natural attributes of the geologic site to act as barriers to release of the waste. Nuclear waste repositories would be constructed in suitable host media at depths greater than 300 meters by conventional mining techniques. Disposal of hazardous wastes through underground injection must be sited beneath the lowermost USDW and are restricted to areas that are geologically suitable.

In addition to the use of the geologic disposal zone described above, the disposal of radioactive waste relies on engineered and natural barriers to protect against the release of waste. Nuclear waste disposal employs a "redundant" system which anticipates that the unexpected failure of one or more barriers will be compensated for by other barriers. Different engineered barriers may be appropriate depending on the type of waste involved. They could include canisters, the physical and chemical forms of the waste itself, waste package overpacks, or other structures within the disposal system which will prevent or substantially delay the release of waste to the environment. In addition, recovery of the wastes is not to be precluded, in order to allow for advances in future scientific understanding.

Risks - In reviewing petitions for variances, the EPA applies health-based criteria which have been calculated by assuming chronic (lifetime) exposure by ingestion or inhalation of contaminated media. For carcinogens, the maximum residual risk level is 1×10^{-6} for Class A and B carcinogenic constituents, and 1×10^{-5} for Class C carcinogens. The Agency will also consider "additivity" for constituents in the same environmental medium e.g., air.

11.3.2.2 Hazardous Waste Land Disposal Restrictions

The EPA's land disposal restrictions seek to protect the environment from adverse consequences of land disposal of hazardous waste. In many ways, the regulatory paradigm restricting land disposal of hazardous waste is very similar to that governing the disposal of radioactive waste. Important similarities between the two include: allowing land disposal only under the assumption that none or very little of the material will escape to the accessible environment; the use of models to demonstrate compliance, particularly regarding air and water pathways; the use of man-made and natural barriers to contain the wastes; and the use of health-related standards to set allowable limits.

However, there are important differences between the EPA's land disposal regulations and Part 191. These differences stem from the chemical attributes and characteristics of the waste being disposed. For example, the nation's overall hazardous waste stream is comprised of thousands of compounds with varying health and environmental risks. Further, Part 191 and the land ban restrictions differ with regard to the potential number of sites regulated, the application of vastly different regulatory time periods, and the application of different standards and assurances to measure compliance.

Background - The Hazardous and Solid Waste Amendments (HSWA) of 1984 mandated specific restrictions relating to the land disposal of hazardous waste. HSWA requires that all hazardous wastes must meet treatment standards based on the performance of best demonstrated available technology (BDAT) before emplacement into the land. Owners/operators of such facilities who do not meet these treatment standards are prohibited from land disposal of hazardous waste.

The EPA promulgated a final rulemaking establishing an overall framework for land disposal restriction in 51 FR 40572. The final rule, as codified in 40 CFR 268.6, allows facility owners/operators who do not meet BDAT treatment standards an opportunity to petition for a variance from land disposal prohibitions. In order to receive a variance, a petitioner must "demonstrate to a reasonable degree of certainty, that there will be no migration of hazardous constituents from the disposal unit ... for as long as the wastes remain hazardous."

Variations can be issued by either the EPA or by states which have EPA-approved land disposal restrictions and "no-migration" programs. Variations can only be obtained for specific sites and specific wastes and may be valid for up to ten years, but not longer than the term of the facility's RCRA permit.

If after a variance has been granted, the EPA determines that migration is occurring or has occurred from the unit (i.e., that a release from the unit exceeds Agency-approved health-based or environmental-based exposure levels), the Agency must revoke the variance. This approach is based on the belief that, with the exception of the air medium, once even a single incidence of migration has occurred, the "no-migration" demonstration has failed.

Regulatory Time Frame - Determining how long "the wastes remain hazardous" is a waste- and site-specific decision. Because some wastes degrade naturally to health-based standards in a short time period, the petitioner may only have to demonstrate this and show "no-migration" from the unit during this period. For other materials, such as metals, the petitioner may have to demonstrate that "no-migration" will occur for many thousands of years.

Predictive Models and Compliance Requirements - The petitioner must provide long-term assurance that the "no-migration" criteria will be met. Petitioners must rely on modeling and theoretical long-term projections to demonstrate that any migration from the disposal site would be at concentrations that did not pose a threat to human health and the environment. Depending on the unit and waste, models may have to account for waste constituent mobility which takes into consideration site geology, soils, and climatology as well as the physical and chemical characteristics of the waste and leachate.

Restrictions on wastes destined for land disposal focus primarily on the relationship between the land disposal of hazardous waste and ground-water quality. However, the potential for air emission of hazardous constituents from the waste into the atmosphere must also be addressed.

In addition, persons seeking exemption from land ban disposal restrictions must submit a monitoring plan that describes how compliance will be verified. The plan must describe the media to be monitored, monitoring frequency, and the equipment and analytical techniques used for sampling. Before granting a final variance, the EPA may require a site to implement a trial monitoring program.

Statutory Definition of Disposal Area - Petitions for variances must demonstrate that hazardous constituents will not exceed Agency-approved, human health-based levels (or environmentally protective levels, if appropriate) beyond the boundary of the disposal unit. Agency-approved health-based levels or environmentally based levels have been determined for ground water, surface water, soil, and air. Generally the disposal unit boundary is defined as the outermost limit of the engineered components.

Geologic and Man-Made Barriers - Man-made barriers or engineered systems alone cannot meet the "no-migration" standard. Like nuclear waste disposal systems, "no-migration" variances require multiple barriers and/or assurances. However, man-made barriers may be given favorable consideration in combination with partial waste treatment or other barriers that are expected to last substantially longer than the hazardous life of the waste.

Individual Annual Dose Limits - In reviewing petitions for variances, the EPA will compare the calculated concentration of hazardous constituents to Agency-approved levels. For example, the Agency would compare the constituent concentrations in leachate to the Maximum Contaminant Levels (MCLs) or Ambient Water Quality Criteria (AWQC). If an MCL or AWQC is not available for a constituent, the appropriate health-based levels would be the Reference Dose (RfD) for non-carcinogens and the Risk Specific Dose (RSD) for carcinogenic compounds. If a health-based level has not been determined for a constituent, that constituent must not exceed analytical detection limits.

Total Release Limits - Unlike nuclear waste disposal regulations, land disposal restrictions do not require standards to measure the total expected releases of hazardous chemical constituents over time.

Risks - As noted in the discussion of the UIC program, in reviewing petitions for variances, the EPA applies health-based criteria which have been calculated by assuming chronic (lifetime) exposure by ingestion or inhalation of contaminated media. The maximum residual risk levels of 1×10^{-6} for Class A and B carcinogenic constituents, and 1×10^{-5} for Class C carcinogens apply. The Agency will consider "additivity" for constituents in the same environmental medium, *e.g.*, air.

11.3.3 International Standards

In 1985, the International Commission on Radiological Protection (ICRP) published ICRP 46, "Radiation Principles for the Disposal of Solid Radioactive Wastes." (ICRP85) That report set out the basic measure of long-term performance of a repository, a limit on the average dose within the critical group, with the critical group defined as being composed of representatives of the public that are expected to receive the highest effective dose. The proposed dose limit to this member of the public from all sources of radiation, excluding medical and natural sources, was set at 1 mSv/yr (0.1 rem/yr). The report recommended an apportionment of the proposed dose limit among various sources and nations, with an

example individual dose limit for a given repository of 0.1 mSv/yr (0.01 rem/yr). Multiplied by the Agency's 4×10^4 fatal cancers per rem, this would equate to a risk of 4×10^6 per year.

In 1989, the International Atomic Energy Agency (IAEA) issued reports 96 and 99 in its Safety Series (IAEA89a, IAEA89b). These documents presented criteria and guidance for the underground disposal of nuclear wastes. Along with ICRP46, they have been used by various individual governments in the development of standards and regulations. Table 11.3-2 presents a compilation of nuclear waste regulations by country (CAR91, IEAL87). In Table 11.3-2, the term "protection limit" means the maximum allowable exposure limit for individual members of the public during the operational phase of repository activities. The "system performance goal" is the maximum dose to a member of the public after closure of the repository.

The range of risk for an individual posed under the repository system performance goals is from 2×10^6 to 1.2×10^5 fatal cancers per year.

11.4 CONTAINMENT REQUIREMENTS AND INTERGENERATIONAL RISKS

Past radiation standards have been focused on activities where releases to the environment, or direct exposures, were controlled to limit the exposure of individuals in the current generation. The subject activities were those where there were typically annual releases and related control options to mitigate them. The radionuclides of concern typically decayed in a relatively short time. In these instances, a focus on limiting annual individual exposures to individuals is appropriate. The need to limit annual individual exposure is also a concern in high-level and transuranic waste disposal, which is why the proposed 40 CFR 191 has specific provisions for dealing with this issue. This is not, however, the intent and purpose of the containment requirements, and therefore comparison with such standards is inappropriate.

The containment requirements are intended to serve two basic purposes: 1) provide a measure of waste repository integrity, and 2) assure that the collective exposure of future populations is limited. This collective exposure is the summed product of all the individual exposures and is directly proportional to the estimate of health effects. Controlling the annual individual dose cannot accomplish either of these purposes for a situation where releases are highly improbable for hundreds of years and the materials remain hazardous for thousands of years. Even the approach suggested by the International Commission on Radiological Protection in their Report 46 does not accomplish these objectives because it focuses on individual dose and risk.

Some reviewers have attempted to compare the containment requirements with current annual dose limits by assuming that the allowed release was evenly emitted over the 10,000 years. In these instances, they have argued that the resultant annual "allowed" doses were far smaller than were other radiation standards. Although this is not an impossible scenario,

it is not the more likely one for which the containment requirements are directed. As a measure of waste repository integrity, the containment requirements focus on requiring an analysis of the events that could disrupt the engineered and natural barriers. If these events occur they are most likely to be at some time in the relatively distant future. Therefore, the releases are highly unlikely to be evenly spread out over the 10,000 years of the required analysis. In fact some of the possible releases, for which the assessment is intended, e.g., volcanic eruption, could release materials over a relatively short time.

Table 11.3-2 Comparison of National Nuclear Waste Regulations*

Country	Protection Limit (preclosure) (mSv/yr)	Repository System Performance Goal (mSv/yr)
Argentina	0.3	0.3
Belgium	5	none set
Canada	5	geologic criteria set
FRG	0.3	0.3
France	none	none set
Japan	0.5	0.05
Nordic	0.1	0.1 plus activity inflow (proposed)
Switzerland	5	0.1
United Kingdom	0.1	0.1
United States	0.25	0.25 plus curie release

* From CAR91 and IEAL87.

In terms of intergenerational risk, the containment requirements provide a level of protection commensurate with the protection provided today. According to data in UNSC88, the collective risk being allowed over the next 10,000 years because of a repository is comparable to that of a 100-year committed dose assessment from the rest of the fuel cycle activities and a factor of ten less than the impact of those activities over all time.

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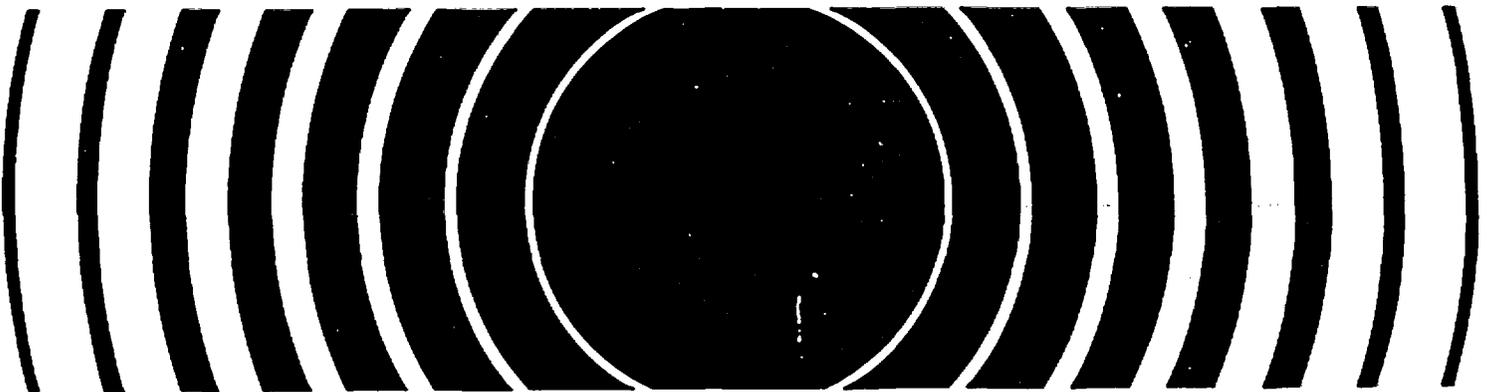
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 EPA

Regulatory Impact Analysis For EPA's High-Level Waste Standard (40 CFR Part 191)

Draft



**Regulatory Impact Analysis
For EPA's High-Level Waste
Standard (40 CFR Part 191)**

DRAFT

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EXECUTIVE SUMMARY

REGULATORY BACKGROUND

The Environmental Protection Agency (EPA) is responsible for developing generally applicable environmental standards for the disposal of spent nuclear fuel and high-level wastes (HLW) and transuranic (TRU) radioactive waste. EPA promulgated standards on August 15, 1985 (40 CFR Part 191). However, those parts of the standards dealing with disposal (Subpart B) were vacated by a U.S. Court of Appeals on July 17, 1987, and were remanded to the Agency for further consideration. Since then, the Agency has been in the process of reproposing these disposal standards.

The focus for this Regulatory Impact Analysis (RIA) is primarily with Subpart B, "Disposal", of the proposed 40 CFR 191, working draft 3 and Subpart C, which was newly created in the process of reproposing the rule. Subpart B has three major components. The first component, a containment requirement, includes numeric limits on the cumulative releases of radionuclides to the environment for the first 10,000 years after disposal (intended to produce less than 1,000 health effects per 100,000 MTHM over a 10,000-year period) taking into account the probability of all possible events. The second component consists of assurance requirements that, among other things, require that both natural and engineered barriers be used. The third part consists of a standard for protection of the individual. Working draft 3 of the rule proposed options of 10 and 25 millirem annual committed effective dose equivalent to the individual for the time period options of 1000 and 10,000 years, for all exposure pathways.

Subpart C contains the fourth component of the rule that has been studied for its impact. It contains, in working draft 3, groundwater protection requirements, which are to be consistent with EPA's drinking water standards, 40 CFR 141. Compliance with the individual and ground water requirements assumes only that "expected" processes occur, such as the normal flow of ground water, and that intrusion, seismic, and volcanic events do not occur.

The Environmental Protection Agency's (EPA) standards embodied in 40 CFR 191 are the primary standards regulating the environmental impacts of nuclear waste management. The regulatory history leading to 40 CFR 191 places primary responsibility for the environmental consequences of nuclear waste management with the EPA. Congress passed the Nuclear Waste Policy Act of 1982 (Public Law 97-425), whereby the EPA was to "...promulgate generally applicable standards for the protection of the general environment from off-site releases from radioactive material in repositories..." not later than January 1984.

The Nuclear Regulatory Commission (NRC) and the Department of Energy (DOE) also regulate nuclear waste management. The NRC regulates the means to the goal of environmental protection by promulgating technical requirements for licensing repositories and sites for storage of nuclear waste from non-defense activities. In carrying out these responsibilities it has promulgated 10 CFR 60, also referred to as the "NRC rule". The Department of Energy (DOE) is charged with building and operating nuclear waste sites. The DOE also has sole responsibility for nuclear waste from DOE defense sites, which are not governed by the NRC. However, DOE defense sites must conform to EPA standards.

PROFILES OF NUCLEAR WASTE GENERATORS

Nuclear waste generators include producers of both HLW (including spent fuel) and TRU waste. Generators of HLW include commercial nuclear power plants, from which spent fuel and, eventually, scrap from decommissioning must be disposed of, and DOE non-commercial generators of HLW.

Commercial nuclear power plants produce the largest amount of wastes in terms of radioactivity. These facilities fund the bulk of the expenditures for waste disposal through fees paid to the Nuclear Waste Fund. DOE has prepared forecasts of the level of operation of commercial nuclear plants and their generation of HLW under a scenario in which there will be no new plants and under an alternative scenario in which there are some new plants. Out to the year 2010, the mass, radioactivity, and thermal power produced depends very little on the choice of scenario. This is due to the long lead times required for construction and licensing of nuclear power plants. By the end of this period, however, significant differences in waste quantities, radioactivities, and thermal power begin to appear.

Levels of radioactivity and thermal power will grow less rapidly than the quantities of spent-fuel. This is due to the natural process of radioactive decay. Thus, by 2020 the mass of spent-fuel will be 4.3 larger for the first scenario and 4.7 times larger for the second than in 1988, while the total radioactivity will only be 1.9 to 2.6 times higher. The accumulation of thermal power will grow to 1.8 to 2.5 times the 1988 level.

DOE operates the four non-commercial sites which generate HLW. Two, the Savannah River Site and the Hanford Reservation, generate nuclear materials for weapons, and two others, the Idaho National Engineering Laboratory and the West Valley Site, are involved with nuclear fuel reprocessing and research. By 2020 there will be a decline in the volume of non-commercial HLW of 19 per cent accompanied by an increase in their radioactivity and thermal power of 16 and 18 per cent respectively. This is due to the conversion of the waste form of the wastes to glass.

Two groups of sites handle transuranic (TRU) waste: sites of operations that continue to generate TRU waste but are not intended to store it, and sites at which a major continuing activity is the storage of TRU waste. The first category includes the Rocky Flats Plant (RFP), Argonne National Laboratory East (ANL-E), the Lawrence Livermore National Laboratory (LLNL), and the Mound Plant (MP). In the second category are the Hanford Reservation (HANF), the Idaho National Engineering Laboratory (INEL), the Los Alamos National Laboratory (LANL), Oak Ridge National Laboratory (ORNL), Sandia National Laboratory (SNL), the Savannah River Site (SRS), and the Nevada Test Site (NTS).

Since 1970, it has been required that TRU wastes be stored in an easily retrievable manner, but in a way that provides greater confinement than on-site storage. TRU wastes will eventually be disposed of in the Waste Isolation Pilot Plant (WIPP) outside of Carlsbad, New Mexico, once it becomes operational, or at some other plant for disposal if the WIPP is determined to be unsuitable.

The categories of stored TRU waste are retrievable-stored, contact handled waste; retrievable-stored, remote handled waste; and buried waste. With respect to contact-handled TRU waste the greatest volume is stored at INEL where there are 36,640 cubic meters while the greatest radioactive content, 1,026,800 Curies, is stored at the Savannah River Site. The total volume of TRU waste stored as of the end of 1988 was 251,000 cubic meters. This TRU waste contains 4,500,000 Curies of radioactivity.

Most TRU wastes, by volume, are generated by DOE defense-related activities at the Rocky Flats Plant, the Hanford Facility, and the Los Alamos National Laboratory. Nearly one-half of all TRU waste comes from weapons components manufactured at the Rocky Flats Plant and subsequent plutonium recovery at all three sites. The second largest source of TRU wastes is decontamination and decommissioning projects, which account for one-fourth of the total. About one-fifth of TRU wastes come from laboratory activities, which can produce exotic TRU isotopes. There is currently a moratorium on reprocessing and plutonium recycling, so the amounts of TRU wastes from fuel cycle activities are minimal. A small amount of TRU waste is also being generated in industrial and government-sponsored fuel fabrication and research.

By volume, the Rocky Flats Plant is expected to generate 47 percent of the total in future years through 2013, all of it contact-handled. The site accumulating the most waste, measured by radioactivity, is the Savannah River Site which contains 83 percent of the total radioactivity. The 732.5 cubic meters of TRU waste accumulated there annually contains 127,850 Curies. By 2013, the expected volume of contact handled waste will be 120,243 cubic meters with a radioactive content of 12,238,000 Curies, an increase of nearly 10,000,000 Curies over the 1988 level.

REFERENCE CASE STUDIES FOR A HLW FACILITY

For the purpose of this RIA, the selected reference case consists of DOE's "no new orders, end-of-reactor life" spent fuel projection (86,757 MTHM of spent fuel), two repositories (the first disposing of 70,000 MTHM-equivalent of spent-fuel, defense HLW, and West Valley HLW, and the second disposing of the remainder of the waste), and the intact spent-fuel assumption. Regarding the spent-fuel projections assumption, the DOE has stated that the "no-new orders, end-of-reactor life" projection is currently the assumption used in their planning. In addition, the DOE has also stated their intention to use the intact spent-fuel assumption as a basis for planning. The first repository is assumed to be located in the tuff formation at the Yucca Mountain site in Nevada. Detailed cost studies have been conducted for this site. They indicate that, in undiscounted 1988 dollars, the cost of developing the first repository is \$7 billion. Discounted over the life of the project, using a discount rate of 2%, these costs equal \$3.6 billion.

A few parameters of nuclear waste management design are of special importance because they can be thought of as policy variables. To some extent, these properties can be manipulated by EPA or program managers. These parameters are the geologic media in which the facility will be located, the leach rate associated with the waste form, and the life of the canister containing the waste form. For the reference case these parameters are:

Geologic Media - Tuff
Waste Form Leach Rate - 10^{-5} parts per year
Canister Life - 300 years

Because the repository is assumed to hold 70,000 MTHM-equivalent of spent-fuel, rather than the 100,000 MTHM for which the implicit limit of 1,000 statistical health effects was derived, the implicit limit for the reference case is 700 health effects in 10,000 years. Throughout this discussion, health effects are not discounted, even though costs sometimes are. Because health effects will occur so far in the future, any discount rate would reduce total health effects to zero.

Given these assumptions, the expected total number of statistical health effects, taking all possible events such as volcanos, earth quakes, and intrusions into account, was 4. This is substantially below the 700 statistical health effects per 10,000 years implicit limit of 40 CFR 191. The release fractions from the facility are below the limits set by the containment requirement. The parameter that most influences the results is the probability of volcanic activity. The probability of volcanic activity is increased to over six times the level in the base case, but the probability that there will be over 700 health effects is still below 0.1 and the

probability that there will be 7,000 or more health effects is still below 0.001.

The reference case analysis also shows zero discharge of radiation to ground water in 10,000 years. This clearly meets the requirement that management and storage of radioactive waste shall not cause any increase in the levels of radioactivity in any underground source of drinking water outside the controlled area which may cause a violation of any primary drinking water regulation under 40 CFR Part 141. The individual protection requirement requires that exposure to individuals through all pathways not exceed 10 millirems. Studies of the Yucca Mountain site infer that the only pathway of concern is ground-water. As stated above, ground-water was determined to have zero discharge.

REFERENCE CASE STUDIES FOR A TRU WASTE FACILITY

The DOE plans to dispose of TRU waste from national defense programs in a geologic repository located in bedded salt in Eddy County, New Mexico. The TRU waste repository is designated as the Waste Isolation Pilot Plant (WIPP). The construction of the WIPP was authorized by the Department of Energy National Security and Military Applications of Nuclear Authorization Act of 1980 (Public Law 96-164) and is classified as a defense activity of the DOE, exempt from Nuclear Regulatory Commission regulations.

A significant portion of the TRU waste management system currently exists as facilities and equipment at the WIPP site. Consequently, sunk costs for these existing facilities and equipment were used as reference costs. In addition to costs for currently existing facilities and equipment, estimates of future capital and operating expenditures were needed to identify the life-cycle costs for the reference TRU waste management system.

The figures given are considered to be conservative: when multiplied by the contingency rates of 0 percent in the Pilot Plant Phase, 10 percent in the Full Operation and Operation Phase-Out periods, and 15 percent in the Decontamination and Decommissioning, the grand total rises from \$3.3 billion to \$3.6 billion in undiscounted 1988 dollars. Discounted over the life of the project, using a 2% discount rate, these costs equal \$2.3 billion.

Design parameters affected by 40 CFR 191 are the geologic media in which the facility will be located, the leach rate associated with the waste form, and the life of the canister containing the waste form. For the TRU reference case these parameters are:

Geologic Media - Salt
Waste Form Leach Rate - unlimited
Canister Life - 0 years

The quantity of waste assumed to be stored in the TRU reference facility is 9.7 million Curies of which 5.6 million curies have half lives greater than 20 years. This is equivalent to 5,600 MTMH of waste. The implied limit on the number of statistical health effects that could be allowed to result from storage of these wastes in WIPP is 56.

The WIPP, as currently planned and constructed, will meet the containment criteria of 40 CFR 191 as well as its individual and groundwater protection criteria. With respect to the containment criteria, a release fraction of .09 is predicted which corresponds to 5 expected health effects in 10,000 years. The limit, as stated above, is 56 health effects for a facility with the waste inventory expected at WIPP. With respect to the individual and groundwater protection criteria, it is likely that "no release of radionuclides to the accessible environment will occur from the salt repository under expected or degraded conditions by way of normal groundwater flow within 10,000 years."

COST-EFFECTIVENESS OF ALTERNATIVES FOR HLW DISPOSAL

Thirty-six options were studied for the HLW repository and one, the reference case, for TRU waste. Cost-effectiveness analysis is performed for HLW and the value of 40 CFR 191 is discussed based on those results. The conclusion applies to TRU waste as well. The value of 40 CFR 191 is that, regardless of how inexpensively 40 CFR 191 can be complied with and how few expected health effects are avoided relative to the reference cases, the regulation prevents other geologic media, other sites, or any engineered barriers from being employed such that the resulting system does not meet the criteria. The regulation protects the population from the universe of possibilities that have not yet been studied. The fact that 40 CFR 191 does not add to the implementation costs of proposed efforts to dispose of HLW or transuranic waste should not detract from its intrinsic value.

Due to the history of HLW disposal, costs and benefits that accrue from 40 CFR 191 are impossible to discern from those that accrue to NRC actions, such as 10 CFR 60, and DOE actions. NRC and DOE actions on HLW disposal anticipated those of EPA. In this respect it may be argued that there either are or are not cost and health impacts from EPA actions. However, the function of this RIA is not to attribute these impacts, but to examine their magnitude and the cost effectiveness of various options.

The conformance of the various alternatives for a HLW management system to the containment standard of 40 CFR 191 was discussed in the Background Information Document. The analysis considered canister lifetimes of zero and 1,000 years and the waste form release rate was varied from one part in 1,000 (10^{-3}) per year to one part in 1,000,000 (10^{-6}) per year.

The reference case assumed there would be a first repository holding 70,000 MTHM for the HWL repository and a second repository for the remaining waste. The cost-effectiveness analysis deals only with the first repository. Under the assumption that the first repository will contain 70,000 MTHM of waste, the limit implied by 40 CFR 191 is 700 statistical health effects in 10,000 years. Both bedded salt and tuff meet that requirement even with zero canister life and a waste form leach rate of 10^{-3} parts per year. In fact, the highest number of health effects for those two media, just over 70 in 10,000 years, is for salt with a canister life of zero years and a waste form leach rate of 10^{-3} parts per year. In other words, both bedded salt and tuff will meet the limit with no special engineering barriers in place. However, basalt requires a limit on the waste form leach rate of 10^{-4} to meet the limit. Conformance with the requirement of 10 CFR 60 that waste form leach rates not exceed 10^{-5} ensures compliance with the containment standard of 40 CFR 191 in all three media.

A large number of options exist for meeting the containment requirements for the HLW repository. Health effects for these options range from about 1 in the most protective case to approximately 1610 for the worst case during the 10,000 year period of the analysis. Basalt has the worst performance under all options with a health effects range of from about 14 to the worst case of approximately 1610 mentioned above. Of the three media, basalt relies most heavily on low waste form leach rates to keep health effects to acceptable levels. The smallest number of health effects, approximately 1, is associated with tuff, a 1,000 year canister and a waste form leach rate of 10^{-6} .

NRC's rule, 10 CFR 60, requires that minimum canister life be somewhere between 300 and 1,000 years and waste form leach rates be less than 10^{-5} . If the NRC requirements are met, repositories in all three media studied comply with 40 CFR 191. The highest number of health effects expected from HLW in any media and complying with 10 CFR 60 is approximately 97. This result is for basalt with a leach rate of 10^{-5} and a 300 year canister. The highest number of health effects for a salt media would be 6 and for tuff, 4. In the absence of NRC's rule, tuff and salt would continue to meet the EPA standard with expected health effects a little below 80 for salt and below 40 for tuff. For basalt, EPA's standard could just barely be met with zero year canisters and leach rates between 10^{-4} and 10^{-5} . These parameters would not satisfy the NRC rule.

Reducing the waste form leach rate is more cost effective than increasing canister life. The most cost-effective options are those that include a canister life of zero. Increasing the canister life increases costs by a proportionally larger amount than the avoided statistical health effects are increased.

A leach rate of 10^{-5} and a canister life of 300 years is the most cost-effective option that meets 10 CFR 60. Another option, zero canister life and a leach rate of 10^{-6} is even more cost effective, but it is not allowed by 10 CFR 60 because of the zero year canister life. Moving from the first of these options to the second would save \$170 million while simultaneously eliminating 3 statistical health effects, implying a savings of \$66 million per statistical health effect averted. The move from the first option to a three hundred year canister and a leach rate of 10^{-6} is the cheapest way to purchase additional protection, beyond the minimum provided by 10 CFR 60, the move would produce no net benefits unless the reduction of an undiscounted health effect was considered to be worth \$95 million or more in discounted dollars.

The results quoted from the BID did not include consideration of gaseous C-14 releases. In the event that the repositories are not able to contain gaseous C-14 releases that will reach the air pathway, special canisters may be necessary to achieve compliance. A quick analysis has been performed to examine the costs of such mitigation and potential benefits. A large amount of uncertainty is associated with both the cost and health effects estimates. In the next iteration of this report we will try both to improve these estimates and to provide estimates of the uncertainty.

These estimates suggest a range for the cost-benefit ratio of C-14 reduction. Combining high cost assumptions (\$2.1 billion, in present value) with poorest performance of the barrier (reduction of 1 health effect) leadership to a cost-benefit ratio of \$2.1 billion per statistical health effect averted. Combining low cost assumptions (\$2.1 billion, in present value) with best performance (3,000 statistical health effects averted) for the other end of the range gives a bound of \$473,000 per statistical health effect averted. A midpoint between the two estimates gives a value of \$1.2 million per health effect averted.

This analysis is based on discounting of the costs and not the health effects that are prevented due to these expenditures. It should be noted that given the time frame over which the health effect would occur (10,000 years) if any positive discount rate were to be applied they would become insignificant.

COST OF DEMONSTRATING COMPLIANCE FOR A HLW FACILITY

EPA recognizes that its requirements that the DOE demonstrate compliance with 40 CFR Part 191 for HLW disposal will impose costs on DOE and add to the cost of HLW disposal. To a certain extent it is believed that these costs will be very close to those that would have been necessary to demonstrate compliance with 10 CFR Part 60. However, in order to have a clear understanding of those costs imposed on DOE by either regulation, an attempt will be made in the next iteration of this RIA to estimate these costs. It is likely

that most of these costs will have been undertaken in the development of the facility itself and that those that remain will be small compared to the overall costs of the development of the facility.

COST-EFFECTIVENESS OF ALTERNATIVES FOR TRU WASTE DISPOSAL

The reference case for TRU waste is the WIPP facility that has already been constructed. It is constructed in salt, and canister lives of zero years and unlimited waste form leach rates are assumed. Modelling results shown in the BID inferred that the facility with these parameters would meet the containment requirement. The studies show that if the equivalent of 5,600 MTHM of waste were stored in a salt repository, about 5 health effects would occur in 10,000 years. This compares to a hypothetical implicit limit of 56 health effects calculated for 40 CFR 191 for this amount of waste storage. The BID also shows that the individual protection requirement, and the ground-water protection requirement would also be met. No alternative media, canister lives, or waste form leach rates were modelled or costed.

The cost of the reference case for TRU is \$3.6 billion in undiscounted 1988 dollars and \$2.3 billion, discounted at 2%. Because no reduced waste forms, leach rates, or special canisters were utilized, it is apparent that there are measures that could reduce the number of health effects over 10,000 years to below the 5 estimated for the reference case. The data in the cost-effectiveness study for HLW implies that the number of health effects can be reduced by 95 to 99 per cent by use of reduced waste form leach rates and increased canister lives. If the cost of canister lives for TRU waste were similar to that for HLW, then the cost of reducing nearly 5 statistical health effects would be over \$1 billion or \$200 million per statistical health effects in undiscounted dollars. The cost-effectiveness ratio for the similar jump from the least stringent to the most stringent option for HLW is only \$28 million per statistical health effects. Thus it is likely to not be as cost-effective to reduce statistical health effects by adding more stringent engineered barriers for TRU waste as it is for HLW.

The reference case results from actions taken in the absence of any specific requirements regarding the release of radionuclides to the environment. The geologic media is sufficient, according to the BID, to contain the radionuclide release to the environment sufficiently to meet all requirements of 40 CFR 191 in the absence of engineered controls. The NRC requirements do not apply to WIPP. Were they to apply, the costs for such a facility will increase.

Just as is the case with HLW, these results indicate that a transuranic waste facility is capable of meeting 40 CFR 191 with little or no additional effort beyond that of the reference case.

Due to the history of TRU disposal, costs and benefits that accrue from 40 CFR 191 are impossible to discern from those that accrue to DOE actions. DOE actions on TRU waste disposal anticipated those of EPA. In this respect it may be argued that there either are or are not cost and health impacts from EPA actions. However, the function of this RIA is not to attribute these impacts, but to examine their magnitude and the cost effectiveness of various options. In this sense the regulation is shown to be a reasonable one in that it provides the desired level of protection without imposing a cost burden for implementing the design.

COST OF DEMONSTRATING COMPLIANCE

EPA recognizes that its requirements that the DOE demonstrate compliance with 40 CFR Part 191 for TRU waste disposal will impose costs on DOE and add to the cost of HLW disposal. The TRU waste facility is not subject to 10 CFR Part 60. In order to have a clear understanding of those costs imposed on DOE by either regulation, an attempt will be made in the next iteration of this RIA to estimate these costs. It is likely that most of these costs will have been undertaken in the development of the facility itself and that those that remain will be small compared to the overall costs of the development of the facility.

ECONOMIC IMPACTS

Two separate economic impact analyses are developed; one for the commercial generators of nuclear power; and the second for government programs. This dichotomy is necessitated by the dissimilar nature by which costs of a regulation on industry versus those on government are translated into impacts. The analysis is done with respect to HLW only. No attempt to divide the costs between commercial and defence HLW was made. In the analysis total incremental costs were applied to commercial waste. The discussion of charges for defence high level waste is qualitative in nature.

Three configurations of a HLW storage facility are analyzed. In the first configuration HLW is buried in tuff with no restrictions regarding leach rate or canister life. The closest observation to this for which there is data on the expected number of statistical lives lost uses canisters with zero life expectancy and a waste form leach rate of 10^{-3} . This configuration is associated with 39 statistical health effects and costs \$4.1 billion in present value terms. A discount rate of 2 percent was used in cost calculations.

The second configuration is the minimum for meeting 10 CFR 60. It is associated with 4 statistical health effects and costs \$4.7 billion in present value terms. The third configuration is the maximum protection. It would be associated with approximately 1 statistical health effects in 10,000 years, and cost \$5.1 billion

in present value terms. All three configurations meet the requirements of 40 CFR 191.

Reductions in statistical health effects which result from moving from the least costly configuration in tuff to the minimum cost for meeting 10 CFR 60 total 35, a decrease of 89.8 percent. The next step, to the 1,000 year canister and leach rate of 10^{-6} will reduce an additional 3 statistical health effects in 10,000 years, or 73 percent more.

Total revenues of the electric utility industry in the U.S. were approximately \$164 billion in 1988. The \$99 million annualized cost of the least stringent option in tuff is 0.06 percent of those total revenues. The more stringent configurations are, respectively \$113.8 million, or 0.07 percent, and \$125.2 million, or 0.08 percent of total revenue. Incremental increases in annualized costs are, from the first configuration to the second, \$14.8 million or 0.009 percent, and from the second to the third, \$11.4 million or 0.007 percent of the total revenue.

Costs to nuclear power generators are computed under the assumption that they alone bear all costs of HLW disposal. The incremental impact on rates charged to users of nuclear power of going from configuration 1 to 2 is .00000281 cent/kwh while proceeding to the most stringent option adds another .00000216 cent/kwh. The impact on rates is smaller if spread over all electric power regardless of its source. For the first increment, the electric rate would increase by .000000574 cent/kwh and for the second by .000000442 cent/kwh.

The effects on sales and average revenues are likewise very small. The most stringent option would cost the 107 million consumers (including industrial, residential, commercial, and other consumers) each only \$1.17 per year on average. The annual incremental cost per customer of advancing from the least stringent to the most stringent of these options is about \$0.25. It can be concluded that the customer is unlikely to notice costs of these magnitudes.

Regarding the welfare measures, the annual "deadweight loss", which measures inefficiencies to the economy, is \$50 for the most stringent option. Other welfare measures show that consumers are assumed to shoulder the entire burden. The loss in consumer surplus is just equal to the cost of each configuration. Because producer profits are unchanged, the changes in producer revenues just equal the changes in producer costs.

The impacts of expenditures to implement 40 CFR 191 are not uniform across the various regions of the U.S. Since the costs are borne only by utilities that have nuclear capacity, 18 states will not be directly affected because they have no nuclear capacity. The

region with the highest percentage of its electric generation provided by nuclear power is New England, with 35 percent. The region of the contiguous states with the lowest percent of generation provided by nuclear power is the west south central region with 7 percent.

The annual direct cost of the regulation in states with no nuclear power is zero. There will be no direct economic impact on those states. Annual impacts to other states will vary depending on the proportion of electricity they produce using nuclear power. For the second increment of increased stringency, the highest cost, \$1.5 million, is borne by Illinois and the lowest for states with nuclear generation, \$14 thousand, by Colorado. When costs are spread over all consumers of electricity generated by all fuels in a state, the highest cost in cents per kilowatt hour (c/kwh) is .000002. This rate increase would apply to Connecticut, Vermont, and South Carolina.

The second analysis of the economic impacts of 40 CFR 191 on defense HLW generators showed that impacts on industry identified.

1. INTRODUCTION AND SUMMARY

The Environmental Protection Agency (EPA) is responsible for developing generally applicable environmental standards for the disposal of spent nuclear fuel and high-level wastes (HLW) and transuranic radioactive waste (TRU). EPA promulgated standards on August 15, 1985 (40 CFR Part 191). However, those parts of the standards dealing with disposal (Subpart B) were vacated by a U.S. Appeals Court on July 17, 1987, and were remanded to the Agency for further consideration. Since then the Agency has been in the process of reproposing these disposal standards.

The focus for this Economic Impact Analysis (EIA) has been primarily with Subpart B, "Disposal", of the proposed 40 CFR 191, working draft 3 (Subpart B of the regulation originally promulgated was specifically remanded to EPA) and Subpart C, which was newly created in the process of reproposing the rule. Basically, Subpart B has three major components. The first component, a containment requirement, includes numeric limits on the cumulative releases of radionuclides to the environment for the first 10,000 years after disposal (intended to produce less than 1,000 premature health effects per 100,000 MTHM over a 10,000-year period). The second component consists of assurance requirements that, among other things, require that both natural and engineered barriers be used. The third part consists of a standard for protection of the individual. Working draft 3 of the rule may propose options of 10 and 25 millirem annual committed effective dose equivalent to the individual for the time period options of 1,000 and 10,000 years, for all exposure pathways.

Subpart C contains the fourth component of the rule that has been studied for its impact. It contains, again in the latest working draft, groundwater protection requirements, which are to be consistent with EPA's drinking water standards, 40 CFR 141. Compliance with the individual and ground water requirements assumes only that "expected" processes occur, such as the normal flow of ground water, and that intrusion, seismic, and volcanic events do not occur.

This RIA develops and assesses information relevant to economic concerns related to disposal of high-level and transuranic nuclear waste. After describing the proposed rule in Chapter 1, it profiles the electric utility industry and non-commercial facilities that have generated and will continue to generate high-level and transuranic wastes in Chapter 2.

Chapter 3 provides a description of and costs of the various activities involved in the disposal of both HLW and TRU waste. These detailed summaries constitute the "reference case", the best documented case against which other alternatives to nuclear waste disposal are compared.

Chapter 4 introduces other cost data and estimates of the environmental releases and their consequences in terms of the number of statistical health effects. Using this data, it develops a cost and benefit analysis for alternative configurations of a HLW facility. Similar analysis was not possible for TRU wastes, since cost and benefit data for developing such a facility in other media and using different canisters and waste forms were not available.

Chapter 5 provides a quantitative study of the cost impact of three configurations of the HLW facility on the electric utility industry and a qualitative discussion of the distribution of HLW defence costs.

1.1 REGULATORY HISTORY OF 40 CFR PART 191

The Environmental Protection Agency's (EPA) standards embodied in 40 CFR 191 are the primary standards regulating the environmental impacts of nuclear waste management. The regulatory history leading to 40 CFR 191 places primary responsibility for nuclear waste management with the EPA. The EPA is in charge of protecting the environment through regulation of emission levels from nuclear waste storage sites.

The Nuclear Regulatory Commission (NRC) and the Department of Energy (DOE) also regulate nuclear waste management. The NRC regulates the means to the goal of environmental protection by promulgating technical requirements for licensing repositories and sites for storage of nuclear waste from non-defense activities. The NRC has been the main source of actual regulations protecting the environment from nuclear waste, since the EPA has not issued standards as specific as the NRC standards. This situation will continue until 40 CFR 191 is promulgated. Ultimately however, NRC standards must be in conformance with EPA's standards. In the last decade, the NRC has occasionally amended their regulations to conform with altered proposed versions of 40 CFR 191.

The Department of Energy (DOE) is charged with building and operating nuclear waste sites. The DOE also has sole responsibility for nuclear waste from DOE defense sites, which are not governed by the NRC. DOE defense sites must also conform to EPA standards.

A brief history of 40 CFR 191 and related regulatory activities follows:

In 1955, the Atomic Energy Commission (AEC) requested that a National Academy of Sciences/National Research Council (NAS-NRC) Advisory Committee be established to consider the disposal of high-level radioactive wastes within the United States. This committee's report, issued in 1957, recommended that:

- 1) the AEC continue to develop processes for the solidification of high-level radioactive liquid wastes, and
- 2) naturally occurring salt formations are the most promising medium for the long-term isolation of these solidified wastes.

Project Salt Vault, conducted by the AEC from 1965 to 1967 in an abandoned salt mine near Lyons, Kansas, demonstrated the safety and feasibility of handling and storing solid wastes in salt formations.

In 1968, the AEC again requested the NAS-NRC to establish a Committee on Radioactive Waste Management (CRWM) to advise the AEC concerning its long-range radioactive waste management plans and to evaluate the feasibility of disposing of solidified radioactive wastes in bedded salt. The CRWM concluded that the use of bedded salt is satisfactory for the disposal of radioactive waste.

In March 1976, the Office of Management and Budget (OMB) established an interagency task force on permanent disposal methods for high-level radioactive commercial wastes. This task force defined the scope of the responsibility of each Federal agency for high-level waste management, including the preparation of environmental standards for high-level waste by the EPA.

A status report on the management of commercial radioactive nuclear wastes, published in May 1976 by the President's Federal Energy Resources Council (FERC), also called for an accelerated comprehensive government radioactive waste program and gave the EPA the responsibility of establishing general environmental standards governing waste disposal activities, including high-level radioactive wastes that must be delivered to Federal repositories for long-term management.

Based on the findings of the OMB interagency task force formed in March 1976, President Ford announced that the experts had concluded that the most practical method for long-term disposal of high-level radioactive wastes from commercial power plants is geologic storage in repositories located in stable formations deep underground. Among the EPA's responsibilities, the Agency was to issue general environmental standards governing nuclear facility releases to the biosphere above the natural background radiation level. These standards were to place a numerical limit on the long-term radiation releases outside the boundary of the repository. In December 1976, the EPA announced its intent to develop environmental radiation protection standards for high-level radioactive wastes to assure the protection of public health and the general environment.

In 1978, President Carter established the Interagency Review Group (IRG) to develop recommendations for the establishment of an administrative policy and supporting implemental programs to address the long-term management of nuclear wastes. The IRG report re-emphasized EPA's role in developing generally applicable standards for the disposal of high-level wastes, spent nuclear fuel, and transuranic wastes (DOE79). In a message to Congress on February 12th, 1980, the President outlined a comprehensive national radioactive waste management program, based on IRG recommendations, which called for an interim strategy that would rely on mined-out geologic repositories for disposal of high-level and transuranic wastes.

In November 1978, the EPA published proposed "Criteria for Radioactive Wastes", which were intended as Federal Guidance for storage and disposal of all forms of radioactive wastes. However, in March 1981 the EPA withdrew the proposed criteria because the many different types of radioactive wastes made the issuance of generic disposal guidance too difficult. On December 29, 1982, the EPA published a proposed rule on "Environmental Radiation Protection Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Wastes".

Shortly after this was published, Congress passed the Nuclear Waste Policy Act of 1982 (Public Law 97-425), whereby the EPA was to "...promulgate generally applicable standards for the protection of the general environment from off-site releases from radioactive material in repositories..." not later than January 1984 (NWPA83a).

In parallel with public review and comment efforts in 1983, the Agency conducted an independent scientific review of the technical basis for the proposed 40 CFR 191 standards through a special Subcommittee for the Agency's Science Advisory Board (SAB), which released a final report on February 17, 1984. Although the SAB review found that the Agency's analyses in support of the proposed standards were comprehensive and scientifically competent, the report contained several findings and recommendations for improvement. The report was publicly released on May 8, 1984. Public response to the SAB report was released in August 1985 (EPA85a), and responses to general comments received from the public on the proposed rule also were published in August 1985 (EPA85b).

On February 8, 1985, the Natural Resources Defense Council, Inc., the Environmental Defense Fund, the Environmental Policy Institute, the Sierra Club, and the Snake River Alliance brought suit against the Agency and the Administrator because they had failed to comply with the January 7, 1984 deadline mandated by the NWPA for promulgation of the standards. A consent order was negotiated with the plaintiffs that required the standards to be promulgated on or before August 15, 1985. At that time, the EPA

issued the final rule and environmental standards under 40 CFR 191 (EPA85a).

In 1986, five environmental groups led by the Natural Resources Defense Council and four states filed petitions for a review of 40 CFR 191. These suits were consolidated and argued in the U.S. Court of Appeals for the First Circuit in Boston. In July 1987, the Court rendered its opinion and noted three findings against the Agency and two favorable judgements. This resulted in the remand of Subpart B.

The Court found (1) that the high-level wastes standards should have been consistent with the Safe Drinking Water's Act (SDWA) underground injection section or the Agency should have explained that a different standard was adopted and why this was justified. The Court noted that the EPA is not necessarily incorrect in promulgating the proposed standards, but that the EPA never acknowledged the interrelationship of the SDWA and HLW rules, nor did it present a reasonable explanation for the divergence between them. (2) The court also supported the petitioner's argument that the EPA arbitrarily selected the 1,000 year limit for individual protection requirements (section 191.15) under undisturbed performance. Although the choice of this limit was judged to be inadequately supported, the Court indicated that the 1,000 year criterion is not inherently flawed. The criterion was remanded either for reconsideration or, at least, for a more thorough EPA explanation for its basis. (3) Finally, the Court found that the Agency did not provide sufficient opportunity for notice and comments on Section 191.16 (Ground Water Protection Requirements) since the section was added to Subpart B after the standards were proposed. This section was remanded for a second round of notice and comments. There were, however, no rulings issued on technical ground.

In August 1987, the Justice Department asked the First Circuit Court to reinstate all of 40 CFR 191 except for the Sections 191.15 and 191.16 which were found defective. The Natural Defense Council filed an opposing opinion. The Court then issued an Amended Decree that reinstated Subpart A, but continued the remand of Subpart B.

1.2 The Rule's Requirements

40 CFR Part 191 Subpart B has three major components: the containment requirement discussed in section 1.2.1, the assurance requirements discussed briefly in section 1.2.2, and the individual and ground water protection requirements discussed in Section 1.2.3. Compliance with the containment requirement is modeled based on all possible events, including low probability events such as the repository being disturbed by major volcanic and seismic disturbances or intrusion by man. Compliance with the individual and ground water requirements assumes only that

"expected" process occur, such as the normal flow of ground water, and that intrusion, seismic, and volcanic events do not occur. These less likely events are referred to as "unexpected occurrences."

1.2.1 Containment Requirement and Its Relation to Population Health Risks

The containment requirement states that "the cumulative releases of radionuclides to the accessible environment for 10,000 years after disposal from all significant processes and events that may affect the disposal system shall: (1) have a likelihood of less than one chance in 10 of exceeding the quantities calculated according to Table 1 (Appendix B); and (2) have a likelihood of less than one chance in 1,000 of exceeding ten times" those quantities. The release limits are in terms of curies "per 1,000 MTHM¹ or other unit of waste."² These units are defined in NOTE 1 to the table. There is a separate limit for each of fourteen radionuclides. Table 1 of Appendix B is reproduced in Table 1-1.

The containment requirement is further explained in notes to Table 1 of Appendix B; Appendix C, non-binding guidance for implementation of Subpart B; and a discussion for inclusion in the Federal Register describing 40 CFR Part 191.

The notes cover a number of technical matters, some of which are discussed here. Note 7 to Table 1 of Appendix B describes how to interpret the containment requirement in cases where more than one radionuclide is released. The note states that a fractional number is to be determined for each radionuclide. This fraction is to be calculated by dividing the amount of the radionuclide released by that radionuclide's individual limit. Each radionuclide's fraction is to be summed. The sum is then compared to the containment requirements which are interpreted as follows: events and processes affecting the disposal system shall (1) have a likelihood of less than one chance in 10 of exceeding one and (2) have a likelihood of less than one chance in 1,000 of exceeding 10.

A graph depicting the rule is shown in Figure 1-1. The horizontal axis, labeled "consequence," is the sum of the fractions and the vertical axis is the cumulative probability. The graph is termed a "decumulative ogive" (PAR78) and represents the containment requirement. It is referred to in the BID as a

¹ The equivalent release of one metric ton of heavy metal.

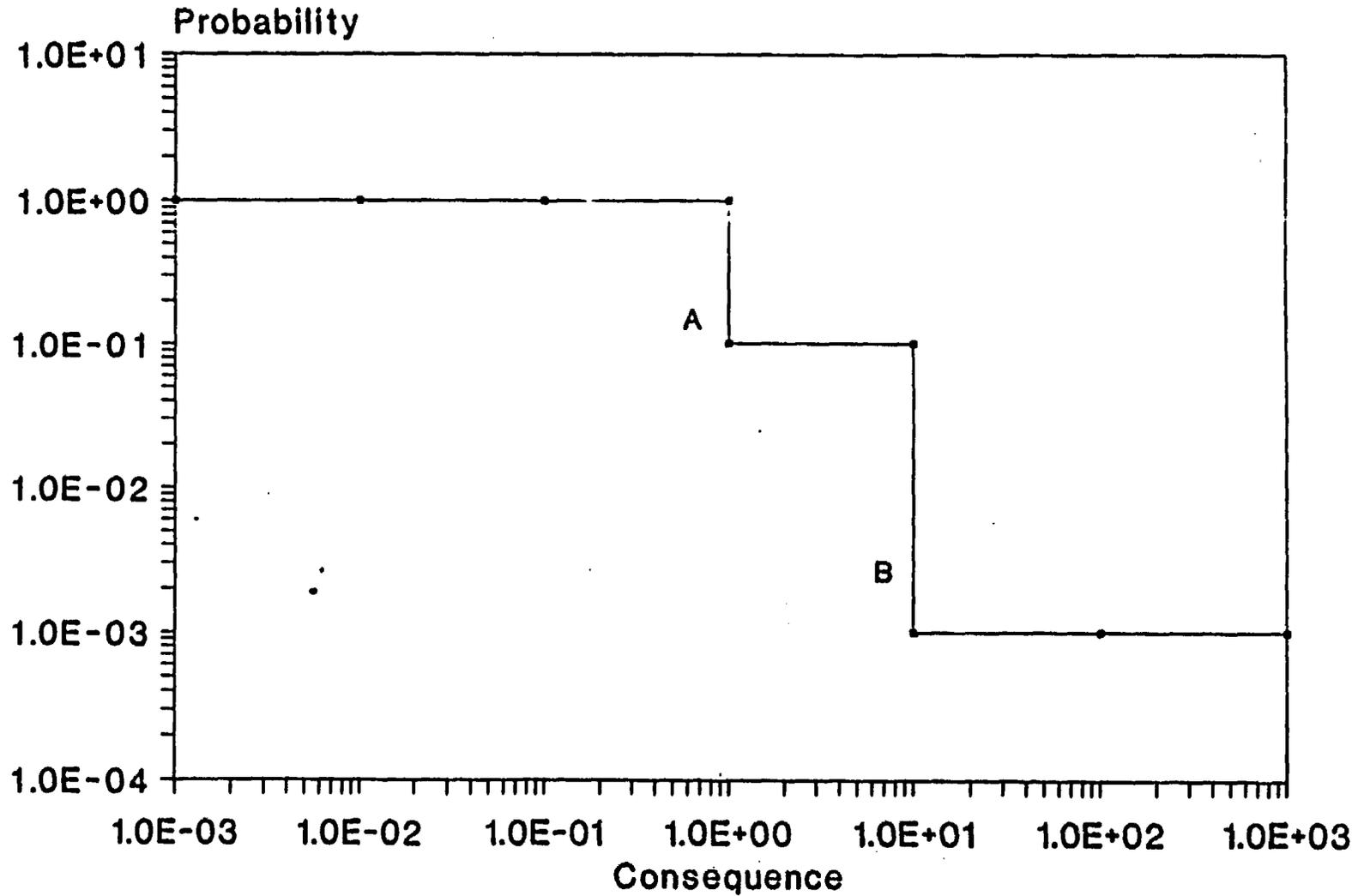
² With respect to TRU waste, it is an amount of TRU waste containing 1,000,000 curies of alpha-emitting transuranic radionuclides with half lives greater than 20 years.

Table 1-1. Cumulative Releases to the Accessible Environment for 10,000 years After Disposal

Radionuclide	Release Limit per 1,000 MTHM or other unit of waste (curies)
Americum-241 or -243	100
Carbon-14	100
Cesium-135 or -137	1000
Iodine-129	100
Neptunium-237	100
Plutonium-238, -239, -240, or -242	100
Radium-226	100
Strontium-90	1000
Technetium-99	10000
Thorium-230 or -232	10
Tin-126	1000
Uranium-233, -234, -235, -236, or -238	100
Any other alpha-emitting radionuclide with a half-life greater than 20 years	100
Any other radionuclide with a half-life greater than 20 years that does not emit alpha particles	100

Source: Appendix B, Table for Subpart B of 40 CFR 191

Figure 1-1. Limits Due to Containment Requirements of 40 CFR Part 191



8

Source: ADL90b

"complementary cumulative distribution function" (CCDF). The guidance document states that the EPA assumes "... the implementing agency will assemble all of the results of the performance assessments to determine compliance with 191.13 into a "complementary cumulative distribution function that indicates the probability of exceeding various levels of cumulative release." The facility complies with the containment requirement if the graph of the CCDF is contained by the graph in Figure 1-1. Points A and B are the only points given by the rule but the rest of the limit is implied by those two points. A decumulative ogive for the probability of releases from the disposal should be situated below these limits.

In the section containing guidance for implementation of Subpart B, EPA states the assumptions it used in applying the rule in a specific case. While these assumptions are non-binding, and not formally part of the rule, EPA expects that these same assumptions will be used by the people who will be in charge of implementing the rule.

Health effects attributable to 40 CFR Part 191 are discussed in the Federal Register notice that is published along with the rule. There it is assumed that the equivalent of 100,000 metric tons of spent reactor fuel is to be stored in the repository. The release of each radionuclide expected to be in the fuel was then "back calculated" such that 1,000 or less premature health effects attributable to the facility would occur in the 10,000 year period commencing with its opening. The values were standardized by dividing by one hundred to give the limit that would apply to each 1,000 metric tons of spent reactor fuel. These calculations are the basis of Table 1 of Appendix B. For each 1,000 metric tons of spent reactor fuel stored, compliance with the limits in the table would lead to no more than ten health effects (limited to health effects induced by radiation because other health effects such as genetic alterations were not significant in number relative to health effects) in 10,000 years due to all expected and unexpected occurrences. Assuming that the relationship between the quantity stored in the system and the number of health effects is linear, the most health effects expected from a system in compliance with the rule and storing, for example, 70,000 metric tons of spent reactor fuel is 700 or less in 10,000 years. Thus the discussion in the federal record provides the link between the rule and the expected health effects if a facility complies with the rule. The rule could be stated in terms of health effects, assuming the facility is storing the equivalent of 100,000 metric tons of spent reactor fuel, as: events and processes affecting the disposal system shall (1) have a likelihood of less than one chance in 10 of exceeding 1,000 health effects and (2) have a likelihood of less than one chance in 1,000 of exceeding 10,000 health effects in 10,000 years. The discussion did not mention the assumptions EPA made regarding the size and distribution of the populations

surrounding the facilities for which these calculations were carried out.

1.2.2 Assurance Requirements

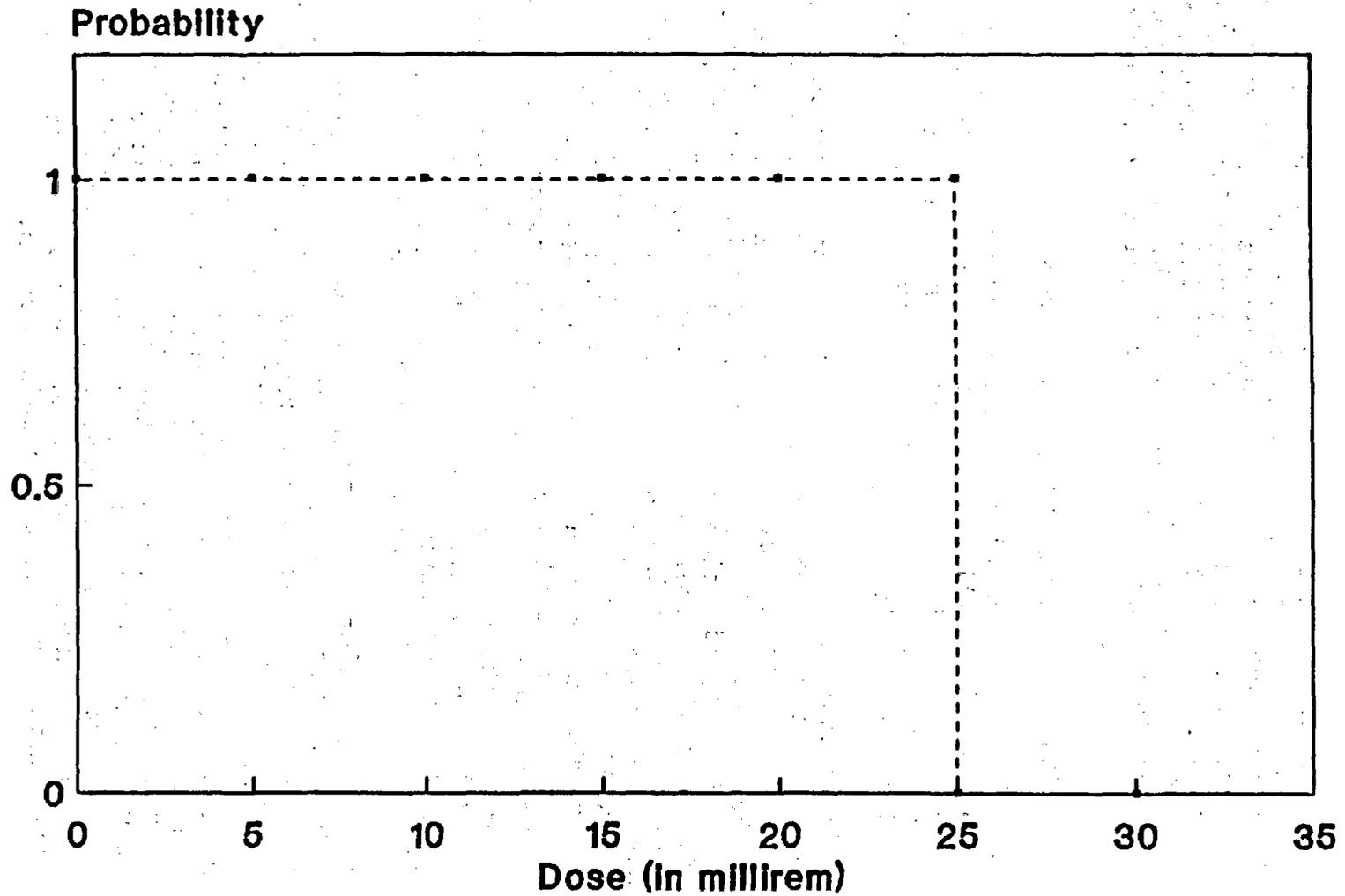
These requirements are general in nature and describe methods for assuring that the containment requirements are met. They seek to reduce the likelihood of intrusion, forbid reliance on institutional and physical mechanisms that require active intervention by human beings for long run security, require that a combination of geologic and engineered barriers are used to isolate the waste, require that the site be monitored until it is clear that it is performing as predicted, that truly long term considerations are used in selecting the site, that recovery of the waste from the sites is feasible, and that the systems be designed to reduce releases into the environment using the maximum achievable control technology taking into account technical, economic, public health and environmental considerations. These requirements are not analyzed in this document.

1.2.3 Individual and Ground Water Protection Requirements

The EPA considered that the annual dose limits for individual protection from radionuclides from all pathways could be 10 millirem or 25 millirem. Of this, 4 millirem is from drinking water, in conformance with the Drinking Water Standard, 40 CFR 141. Whether this is a whole body or other dose is not yet certain as 40 CFR 141 has not yet been completed.

These options contrast with the containment requirement in their treatment of probability. The containment requirement allows the possibility of large releases of radioactivity over 10,000 years so long as the chance of the large release is less than one in 1,000. In other words, the containment requirement includes consideration of low probability events such as volcanic eruptions that have high potential releases of radioactivity. The individual and ground water requirements ignore low probability events and call for a 100 percent chance that the limit will not be exceeded in their absence for either 1,000 years or 10,000 years. This is illustrated in Figure 1-2, for a 25 millirem standard. The graph shows the limit for a decumulative ogive. The ogive is for the subset of events considered to be normal. It must reach zero for a dose of 25 millirems.

Figure 1-2. Probability Treatment
of Individual Protection Requirement



1.3 Regulatory History of High-Level and Transuranic Waste Management Systems

Although 40 CFR 191 will regulate any long term underground storage of nuclear waste, in the near future it will be applied to two different waste systems: the High-Level Waste (HLW) Management System, sometimes called the commercial repository because it is managed by the Office of Civilian Radioactive Waste Management (OCRWM), and the Transuranic (TRU) Waste Management System. In the following, the separate regulatory histories of the two systems are discussed.

1.3.1 High-Level Waste Management System

In the DOE, the Office of Civilian Radioactive Waste Management, as established by Sec. 304 of the Nuclear Waste Policy Act of 1982 (Public Law 97-425), will manage any high-level waste management facility that will be built. This office annually reports to Congress on its activities.

The NRC will oversee any high-level waste depository under Sec. 121b of the 1982 act, which authorizes it to "promulgate technical requirements and criteria that it will apply, under the Atomic Energy Act of 1954 (specifically, Sections 103 and 104) and the Energy Reorganization Act of 1974" and to review applications for high-level and spent nuclear waste repositories. The NRC regulation governing this is 10 CFR 60 (Disposal of High-Level Radioactive Wastes in Geologic Repositories), which the chosen site will be required to meet. 10 CFR 60 prescribes rules governing the DOE licensing of geologic repositories to receive and possess source, special nuclear, and byproduct materials. Regulation 10 CFR 60 was promulgated in 1983, under the authority of the 1982 Act (Section 121B). However, 10 CFR 60 was developed in parallel with the 1982 Act; as such, the Act anticipates and includes many of the regulatory requirements as the lawmakers had seen the NRC regulations before the Act was passed.

Prior to this regulation, the NRC had no specific regulations governing nuclear waste management. Thus, Section 217e of the 1982 Act requires only that "The system of engineered barriers and selected geology used in a test and evaluation facility shall have a design life at least as long as that which the Commission requires by regulations issued under this Act, or under the Atomic Energy Act of 1954 for repositories". (Stat 2251) Section 217f requires that within 1 year after the enactment of the Act, the Secretary and the NRC "shall reach a written understanding establishing the procedures for review, consultation, and coordination in the planning, construction and operation of the test and evaluation facility under this section."

All such applications for site licenses must include a Safety Analysis Report, which shows the conditions which may contribute

to or detract from isolation, and "evaluate the repository for the period after permanent closure, assuming anticipated processes and events, giving the rates and quantities of releases of radionuclides to the accessible environment as a function of time; and a similar evaluation which assumes the occurrence of unanticipated processes and events." This report should also include "a description and assessment of the site at which the proposed geologic repository operations area is to be located with appropriate attention to those features of the site that might affect geologic repository operations, area design, and performance."

Section 60.31 provides for construction authorization standards for the NRC, and Section 60.41, general standards for issuing a license, which must show the application is in accordance with the rules and regulations of the Commission and with the Atomic Energy Act and the Energy Reorganization Act. The performance of the repository must show releases of radioactive materials conform with Part 20 of the act and with "such generally applicable environmental standards for radioactivity as may have been established by the EPA."

Section 60.113 of 10 CFR 60 provides that the release rate of the nuclides from the engineered barrier "shall not exceed one part in 100,000 per year of the inventory of that radionuclide calculated to be present at 1,000 years following permanent closure, or such other fraction as may be approved or specified by the Commission; provided that this requirement does not apply to any radionuclide which is released at a rate less than 0.1 percent of the calculated total release rate limit." This "shall be taken to be one part in 100,000 per year of the inventory of radioactive waste, originally emplaced in the underground facility, that remains after 1,000 years of radioactive decay." The same section states that "The geological repository shall be located so that pre-waste-emplacment ground water travel time along the fastest path of likely radionuclide travel from the disturbed zone to the accessible environment shall be at least 1,000 years or such other travel time as may be approved or specified by the Commission". It also says that "(c)ontainment of HLW within the waste packages will be substantially complete for a period to be determined by the Commission...provided that such period shall be not less than 300 years nor more than 1,000 years after permanent closure of the geologic repository."

Section 60.113 also states that "On a case-by-case basis, the Commission may approve or specify some other radionuclide release rates, designed containment period or pre-waste-emplacment ground water travel time, provided that the overall system performance objective, as it relates to anticipated processes and events, is satisfied. Among the factors that the Commission may take into account are: (1) Any generally applicable environmental

standard for radioactivity established by the Environmental Protection Agency...."

However, the NRC must comply with the Federal guidance on radiation protection, formally established under the Federal Radiation Council, which limits all operations of NRC licensees' exposure to citizens to no more than 0.5 rem/year from all pathways. EPA 40 CFR 190 standards for uranium fuel-cycle facilities also apply to fuel-cycle facilities licensed by the NRC. These standards prohibit releasing radioactive effluents which would result in doses greater than 25 mrems/year.

The EPA's radiation standards (40 CFR 191) apply to waste from civilian reactors as well. Nuclear programs are those that are covered by subchapter F 90-192 of the EPA's CFR Title 40. Under the Nuclear Waste Policy Act of 1982, section 121a directs that "Not later than 1 year after the date of the enactment of this act (January 7, 1983), the Administrator, pursuant to authority under other provisions of law, shall, by rule, promulgate generally applicable standards for protection of the general environment from offsite releases from radioactive material in repositories". Section 113 of the 1982 Act states that "Any recommendation made by the Secretary under this section shall be considered a major Federal action significantly affecting the quality of the human environment for purposes of the National Environmental Policy Act of 1969."

The Nuclear Waste Policy Act of 1982 asks that sites for nuclear waste storage be recommended, and reports on the recommended sites be issued. Section 112a authorized the Secretary of Energy to issue, by July 1983, general guidelines for the recommendations of sites for repositories "following consultation with the Administrator of the EPA," among others, and "the concurrence of the NRC". The Secretary of Energy is to use these guidelines to consider candidate sites for recommendation under Section 112b, and is to issue environmental assessments of the sites, showing how they accord with these guidelines. Section 113c says the Secretary of Energy may conduct site characterization activities only as considered necessary to prepare an "application to be submitted to the NRC for a construction authorization for a repository at such candidate site", and for "Compliance with the National Environmental Policy Act of 1969 (42 USC 4321 et seq).".

The 1982 act thus concurrently gives authority for regulations governing nuclear waste management to the DOE, the NRC, and the EPA. The act does not specifically mention technical standards within it. The EPA's standards are to protect the environment, and the DOE Office of Civilian Radioactive Waste Management is to carry out the functions of the Secretary under this act, which include siting and evaluating potential facilities and gaining title to high-level nuclear waste for its disposal under Section

302. The NRC is to develop guidelines to judge "(i) application for authorization to construct repositories; (ii) applications for licenses to receive and possess spent nuclear fuel and high-level radioactive waste in such repositories; and (iii) applications for authorization for closure and decommissioning of such repositories."

The Nuclear Waste Policy Amendments Act of 1987 specifies the Yucca Mountain site to be the single site to be evaluated for suitability as a repository for nuclear waste from civilian reactors. The 1987 Amendments act also authorizes the DOE to locate a monitored retrievable storage facility to provide temporary storage of spent nuclear fuel. It also establishes a Nuclear Waste Negotiator within the Executive Office of the President to find a state or Indian tribe willing to host a repository or MRS facility. This was made into a separate Office of the Nuclear Waste Negotiator in Public Law 100-507 in 1988. The process of filling this office has only recently begun.

1.3.2 Transuranic (TRU) Waste Management System

For the past decade, the effort to dispose of transuranic (TRU) waste has centered on the design and construction of the Waste Isolation Pilot Plant (WIPP). WIPP was established under Section 213 of the DOE's National Security and Military Applications of the Nuclear Energy Authorization Act of 1980 (PL96-164). WIPP is authorized by the Department of Energy, and is to be administered by the Assistant Secretary of Energy for Defense Programs. It is a research and development facility designed to demonstrate the safe disposal of radioactive wastes which result from the defense activities and programs of the U.S. These nuclear defense programs are exempted from regulation by the Nuclear Regulatory Commission (NRC). Construction of the WIPP began in 1981.

In January 1990, the DOE issued a supplement to their 1980 Environmental Impact Statement. This Supplemental Environmental Impact Statement (SEIS) analyzes risks associated with the operation, closure, and long-term stability of the facility. The WIPP must comply with environmental standards, and costs are not a consideration in the SEIS. The SEIS does, however, consider socioeconomic costs.

The WIPP is not regulated by the Nuclear Regulatory Commission. No DOE facility is regulated by the NRC except for the high-level waste depository from civilian reactors. However, transportation of nuclear waste to WIPP is governed by the NRC. Also, NRC's 10 CFR 60 and 71 are seen as being implemented in EPA's 40 CFR 191, subparts A and B. The WIPP facility must adhere to EPA's disposal standards, but the DOE alone is responsible for deciding whether WIPP is complying with the EPA standards.

In preparing their environmental impact statement, the DOE assumed compliance with 40 CFR 191, parts A and B for the WIPP facility. With regard to section B, WIPP has assumed compliance with the original standard before remanding until the new version is issued, as specified in their agreement with the state of New Mexico. NRC's 10 CFR 60 implements many portions of 40 CFR 191, parts A and B, although WIPP is not subject to the NRC. However, the WIPP was always in conformance with NRC 10 CFR 60's containment and groundwater standards: this is partially because the TRU waste WIPP deals with is not as radioactive as waste from nuclear reactors about which 40 CFR 191 is primarily concerned, and partially because the salt caverns which contain the WIPP facility have proved to be such an effective barrier against waste leakage. WIPP will demonstrate compliance with the assurance requirements of Section B prior to proceeding with the Disposal Phase.

The SEIS for WIPP describes long term emissions from nuclear waste in compliance with DOE order 5820.2A. This order dictates the plans which must be developed by a nuclear waste management facility in order to comply with DOE standards. DOE order 5480.1B of May 23, 1986 also establishes an overall framework of program requirements for safety, environmental and health protection, including criteria for radiation exposure and radioactive effluent releases for operating facilities and sites. The Final Safety Analysis Report by WIPP is being prepared in compliance with DOE order 5480.1B and with commitments made in the Working Agreement for Consultation and Cooperation between the State of New Mexico and the DOE. The working agreement is a systematic analysis of the potential hazards associated with WIPP operations, and the information it contains allows for an informed choice among the alternatives addressed in the current WIPP SEIS.

WIPP is governed by EPA regulations as well. Even though PL96-164, which established the WIPP, does not specifically mention the EPA, as a federal facility, the WIPP is required to comply with all existing EPA standards under the National Environmental Policy Act (NEPA) of 1969 (PL 91-190). NEPA requires a statement of environmental impacts and resource commitments and alternatives to any major Federal action which significantly affects the quality of the environment. The WIPP will comply with the NEPA and the Council on Environmental Quality (CEQ) regulations to issue the required environmental impact statements. The WIPP will also issue another SEIS at the conclusion of the Test Phase to support a decision to proceed to the Disposal Phase.

The WIPP is also governed by the Federal Land Policy and Management Act (FLPMA), since the WIPP site is now a public land, and under the jurisdiction of the U.S. Bureau of Land Management. The DOE has submitted an application for an administrative

modification for the WIPP site, although the DOE would prefer a legislative withdrawal on the needed land. The proposed legislation was introduced in both houses during the first session of the 100th Congress, which would permanently transfer the WIPP site from the Department of the Interior to the DOE, but Congress was adjourned before the bill could be enacted. Legislative withdrawal was pursued in the 101st Congress. Such a legislative withdrawal would be permanent. The FLPMA authorizes (by administrative modification) the Secretary of the Interior to withdraw 5,000 acres or more for an initial period that is not to exceed 20 years and is subject to renewal.

The other major EPA regulation which governed WIPP during the 1980s was the Resource Conservation and Regulation Act (RCRA) which governs mixed waste (waste of which some portion is hazardous and some is radioactive). This has been so since July 3, 1986 (51 FR 24504), when the EPA published a notice of its determination that wastes containing both hazardous and radioactive contaminants were subject to regulation by the RCRA, and that states must obtain authority from the EPA to regulate the hazardous constituents of "radioactive mixed waste" in order to obtain or retain authorization to administer or enforce a RCRA Subtitle C hazardous waste program. The hazardous chemical components of DOE-generated mixed waste are subject to regulation under the RCRA. EPA regulations allow the disposal of hazardous waste that does not meet the treatment standards if a petitioner can demonstrate to the EPA "to a reasonable degree of certainty, that there will be no migration of hazardous constituents from the disposal unit...for as long as the wastes remain hazardous." A RCRA "no migration" variance petition, as provided by 40 CFR 268.6, was submitted by the DOE to the EPA in March 1989. Additional information supporting the no-mitigation petition was provided in October 1989. The EPA is currently evaluating this petition. Under the RCRA, the EPA or the State of New Mexico will issue the DOE a permit to operate the WIPP as a hazardous waste management facility.

The only possible exception to WIPP's compliance with EPA regulations was the 1954 Atomic Energy Act (AEA), which had within it the flexibility to override all other regulations on radiation and which governed all atomic defense program activities. The exception no longer applies, because in 1986 the DOE signed an agreement with the EPA that all DOE radioactive waste which is hazardous under the RCRA will be subject to regulations under both the RCRA and the AEA. This agreement is 10 CFR 962 dated May 1, 1987.

The Secretary of Energy is interpreted by PL96-164 to now have the authority to see if WIPP is complying with EPA regulations under 40 CFR 191. EPA has the option of stating its opinion if WIPP is in compliance. The EPA can concur, but it does not

technically approve WIPP's compliance with EPA standards. The DOE alone approves WIPP compliance with EPA standards.

In June of 1989, Secretary of Energy James D. Watkins announced a 10 point program to demonstrate the DOE's commitment to comply with environmental requirements and protection of the public. These initiatives included the decision that the WIPP would not open until the Secretary and a non-DOE review board consider it safe. A panel of experts from industry, academia, and government has been formed to review current plans for demonstrating WIPP's technical and operational adequacy. They submitted their first findings in October 1989.

In addition, the Advisory Committee on Nuclear Facility Safety was formed under the provisions of the Federal Advisory Committee Act (Public Law 92-463). The Committee gives advice directly to the Secretary of Energy on the safety of the DOE's production and utilization facilities. The Committee reviews technical information regarding the WIPP facility and serves as a focal point for feedback from other oversight groups and from the public.

2. PROFILES OF WASTE GENERATORS

Nuclear waste generators include producers of both HLW (including spent fuel) and TRU waste. Profiles of the individual facilities that produce these wastes are provided in the Appendix. The profiles describe the activities that generate the wastes, the level and nature of employment at the facilities, their locations, the quantities of wastes generated and expected to be generated by 2020, and the degree of urgency for finding disposal for the wastes of each facility. This chapter provides an overview.

2.1 Generators of High-Level Waste

Generators of HLW include commercial nuclear power plants, from which spent fuel and, eventually, scrap from decommissioning must be disposed of, and DOE non-commercial generators of HLW. These two categories of waste producers are profiled in sections 2.1.1 and 2.1.2., respectively.

2.1.1 Commercial Nuclear Power Generators

Commercial nuclear power plants produce the largest amount of wastes in terms of radioactivity. These facilities will be funding the bulk of the expenditures for waste disposal. The discussion in the Appendix focuses on several aspects of this industry including a description of the wastes generated, industry operating and financial characteristics, and short and long term prospects. The purpose of this discussion is to highlight background information that is useful for evaluating future waste disposal requirements, the financial capability of the electric utility sector to fund waste disposal, and the economic impact of waste disposal costs upon various regions and industries.

There are several ways by which the quantities of wastes generated at commercial nuclear power plants can be measured. Several measures of volume are available including metric tons of initial heavy metal (MTHM), cubic meters, and number of fuel rods. The MTHM measure is the most commonly employed measure of spent fuel volumes, although the cubic meters measure is useful in comparing volumes of spent fuel with volumes of other types of radioactive waste. Volumes of waste are important because any containment strategy must be capable of accommodating the anticipated volumes. As of December 31, 1988 there were 7966 cubic meters of commercial spent-fuel assemblies. This total includes the spacing between the fuel rods.

The quantities of wastes may also be measured in terms of their radioactivity. The importance of the radioactivity is obvious, as this is the source of the potential health effects. While spent-fuel accounts for only a small share of the total volume of

radioactive wastes, it makes up most of the radioactivity. For example, in terms of high-level waste, spent-fuel made up 1.24 percent of volume and 94.0 percent of radioactivity (ORLN89). Note that since volume comparisons are made in cubic meters, which for spent-fuel includes the spacing between the fuel assembly rods, the volume is arguably somewhat inflated. One reason that spent-fuel accounts for so much of the radioactivity is the relatively older ages of the defense wastes.

The quantities of waste may also be measured in terms of thermal power. Thermal power is a measure of the rate of heat energy emission that results from the radioactive decay of a material. It is an important parameter in building a nuclear waste storage facility, because facilities must be designed to manage the amount of thermal power contained by wastes stored at facilities. Table 2-1 provides information on the current and projected mass, radioactivity, and thermal power of spent-fuels. Data are provided for two of DOE's three scenarios, which are discussed in the Appendix: the no new orders scenario and the lower reference case scenario. Information is provided both in terms of annual discharges and cumulative inventories.

Out to the year 2010, the mass, radioactivity, and thermal power that will be produced depends very little on the choice of scenario. This is due to the long lead times required for construction and licensing of nuclear power plants. By the end of the forecast period, however, significant differences in waste quantities, radioactivities, and thermal power begin to appear.

One important observation is that the levels of radioactivity and thermal power will grow less rapidly than the quantities of spent-fuel. This is due to the natural process of radioactive decay. Thus, by 2020 the mass of spent-fuel will be 4.3 to 4.7 times larger than in 1988, while the total radioactivity will only be 1.9 to 2.6 times higher. The accumulation of thermal power grows to 1.8 to 2.5 times the 1988 level.

2.1.2 DOE Non-Commercial Generators of High-Level Waste

DOE operates four sites which generate high-level waste. Two, the Savannah River Site and the Hanford Reservation, generate nuclear materials for weapons, and two others, the Idaho National Engineering Laboratory and the West Valley Site, are involved with nuclear fuel reprocessing and research. The descriptions of each site found in the Appendix are intended to profile each site, its financial resources, and its future waste disposal needs in order to provide a reference baseline for the WIPP facility. The profiles include descriptions of current waste storage capabilities, the amount of current nuclear waste, and future waste generation projections.

Table 2-2 summarizes the quantities of High-Level Wastes generated by the three defense sites and the West Valley Demonstration Project. In the table, the quantities of HLW are given in units of volume (thousands of cubic meters), radioactivity (millions of curies), and thermal power (thousands of Watts). Overall, there is a decline in the volume of High-Level Wastes of 19 per cent accompanied by an increase in their radioactivity and thermal power of 16 and 18 per cent respectively. This is due to the conversion of the waste form of the wastes to glass.

2.2 Generators of Transuranic Waste

Two groups of sites are included in the discussion of transuranic (TRU) waste: sites of operations that continue to generate TRU waste but are not intended to store it, and sites at which a major continuing activity is the storage of TRU waste. The first category includes the Rocky Flats Plant (RFP), Argonne National Laboratory East (ANL-E), the Lawrence Livermore National Laboratory (LLNL), and the Mound Plant (MP). In the second category are the Hanford Reservation (HANF), the Idaho National Engineering Laboratory (INEL), the Los Alamos National Laboratory (LANL), Oak Ridge National Laboratory (ORNL), Sandia National Laboratory (SNL)³, the Savannah River Site (SRS), and the Nevada Test Site (NTS).

Transuranic wastes (TRU) are defined as wastes emitting more than 100 nanocuries per gram of alpha-emitting transuranic (uranium and heavier elements) isotopes, with half-lives greater than 20 years. Prior to 1982, TRU waste was defined as having a concentration of alpha-emitting radionuclides greater than 10 nCi/g TRU. In 1982, the definition was changed to include only those wastes with TRU concentrations greater than 100 nCi/g. As a result, about a half of the 2.3 million cubic feet of waste stored at the Radioactive Waste Management Complex (RWMC) at the INEL, is expected to be reclassified as low-level waste, and is not proposed to be shipped to WIPP.

Since 1970, it has been required that TRU wastes be stored in an easily retrievable manner, but in a way that provides greater confinement than on-site storage. TRU wastes will eventually be disposed of in the Waste Isolation Pilot Plant (WIPP) outside of Carlsbad, New Mexico, once it becomes operational, or at some other plant for long-term storage if the WIPP is determined to be unsuitable. DOE estimates of the amount of TRU at each site

³ Because Sandia National Laboratory has an accumulation of only 3 cubic meters of TRU waste and is not generating more, it is not discussed further in the text. There are, however, references to it in tables 2-9 and 2-10.

Table 2-1 Projections of Mass and Radioactivity of Permanently Discharged Spent Fuel, 1988-2020

DOE/EIA No New Orders Case						
	Volume, m ³	Radioactivity, Mil. Ci		Thermal Power, 10 ⁶ W		
Year	Annual Rate	Accumulation	Annual Rate	Accumulation	Annual Rate	Accumulation
	¹					
1988	769	7,966	9,493	18,654	39	70.5
1989	946	8,864	11,300	21,300	46.4	80.8
1990	815	9,722	10,200	21,500	42.2	80.9
1991	807	10,486	10,300	22,200	42.6	83.5
1992	942	11,475	12,400	25,100	51.6	95
1993	811	12,239	10,400	24,500	43.2	91.7
1994	811	13,094	10,800	25,400	45	94.9
1995	859	13,908	10,700	26,000	44.8	97.1
1996	946	14,855	11,900	28,000	50.1	104.8
1997	720	15,574	9,900	27,000	41.7	100.5
1998	815	16,433	11,000	28,400	46.3	105.9
1999	763	17,196	10,200	28,500	43.4	106
2000	954	18,150	12,300	31,100	52	116.4
2001	668	18,774	9,300	29,400	39.6	108.8
2002	819	19,637	10,600	30,700	44.8	113.7
2003	720	20,356	9,900	30,700	41.9	113.3
2004	855	21,211	11,800	33,100	50.3	123
2005	720	21,887	9,400	31,800	39.7	116.9

Table 2-1. (cont.) Projections of Mass and Radioactivity of Permanently Discharged Spent Fuel, 1988-2020

DOE/EIA No New Orders Case (cont.)

Year	Volume, m ³		Radioactivity, Mil. Ci		Thermal Power, 10 ⁶ W	
	Annual Rate	Accumulation	Annual Rate	Accumulation	Annual Rate	Accumulation
2006	906	22,745	11,700	34,100	49.5	126.5
2007	851	23,640	11,700	35,200	49.7	130.7
2008	771	24,411	10,000	34,300	42.6	126.6
2009	946	25,357	12,700	37,200	53.7	138
2010	862	26,220	11,400	37,000	47.8	136.6
2011	946	27,166	12,500	38,600	52.5	142.7
2012	950	28,116	12,000	38,900	50.4	143.5
2013	1,078	29,146	13,800	41,300	57.5	152.4
2014	1,097	30,242	13,300	41,800	55.2	153.7
2015	624	30,867	8,800	37,900	36.8	137.2
2016	763	31,678	10,400	38,800	43.3	140.2
2017	668	32,346	9,300	37,900	39	136.7
2018	497	32,843	6,600	35,100	27.8	125.2
2019	358	33,201	5,200	33,000	22	116.3
2020	672	33,873	8,800	35,800	37.1	128.2

1 Reported Historical Data for 1988.

Source: ORNL89, Tables 1.3 and 1.4

Table 2-1. (cont.) Projections of Mass and Radioactivity of Permanently Discharged Spent Fuel, 1988-2020

DOE/EIA Lower Reference Case						
	Volume, m ³		Radioactivity, Mil. Ci		Thermal Power, 10 ⁶ W	
Year	Annual Rate	Accumulation	Annual Rate	Accumulation	Annual Rate	Accumulation
1988	764	7,966	9,493	18,654	39	70.5
1989	895	8,864	1,000	21,000	45.3	79.6
1990	808	9,678	10,200	21,400	42.2	80.6
1991	802	10,486	10,500	22,400	43.6	84.4
1992	936	11,475	12,400	25,200	51.6	95.2
1993	805	11,239	10,400	24,500	43.2	91.8
1994	849	13,094	11,000	25,600	45.8	95.8
1995	805	13,905	10,200	25,600	42.9	95.5
1996	939	14,087	12,300	28,200	51.6	105.8
1997	715	15,570	9,900	27,100	41.6	100.7
1998	808	16,385	11,000	28,500	46.4	106.1
1999	758	17,149	10,600	28,900	44.8	107.5
2000	945	18,102	12,300	31,200	52	116.9
2001	665	18,771	9,500	29,700	40.4	109.9
2002	811	19,680	10,700	31,000	45.4	114.8
2003	715	20,400	9,900	30,800	41.8	113.8
2004	895	21,255	12,200	33,500	51.7	124.8
2005	715	21,974	9,400	31,900	39.8	117.7

Table 2-1. (cont.) Projections of Mass and Radioactivity of Permanently Discharged Spent Fuel, 1988-2020

DOE/EIA Lower Reference Case (cont.)

Year	Volume, m ³		Radioactivity, Mil. Ci		Thermal Power, 10 ⁶ W	
	Annual Rate	Accumulation	Annual Rate	Accumulation	Annual Rate	Accumulation
2006	898	22,881	11,900	34,500	50.3	127.9
2007	846	23,732	11,900	35,600	50.3	132
2008	764	24,455	10,100	34,500	42.6	127.3
2009	986	25,492	13,300	38,000	56.2	141.1
2010	854	26,402	11,300	37,300	47.7	137.9
2011	983	27,392	13,000	39,400	54.4	145.5
2012	986	28,342	12,700	39,900	52.9	147.2
2013	1,114	29,507	15,000	42,900	62.2	158.6
2014	1,306	30,735	15,700	44,900	65	165.8
2015	934	31,674	12,100	42,400	50.5	155.1
2016	1,161	32,890	15,300	45,500	63.5	166.9
2017	1,158	34,055	15,200	46,400	63.5	170.8
2018	942	35,005	12,500	44,500	52	162.6
2019	936	35,948	12,300	44,300	51.7	161.5
2020	1,204	37,160	16,200	48,400	68	178.5

1 Reported Historical Data.

Source: ORNL89, Tables 1.3 and 1.4

Table 2-2. Projected HLW Inventories

Year	Volume (10 ³ m ³)			Radioactivity (10 ⁶ Ci)			Thermal Power (10 ³ W)	
	Class	Other	Total	Class	Other	Total	Glass	Total
Savannah River Site								
1988		128.3	128.3		661	661		1830
1990		123.6	123.6		664	664		1884
1995	1.0	93.1	94.1	88	761	849	246	2498
2000	2.0	71.5	73.5	188	649	837	539	2422
2005	2.6	59.5	62.1	275	588	863	772	2502
2010	3.3	54.8	58.1	317	541	858	903	2477
2015	3.4	48.6	52.0	340	548	888	951	2579
2020	3.5	45.8	49.3	350	534	884	970	2550
Hanford Reservation								
1988		243.5	243.5		446	446		1307
1990		245.6	245.6		415	415		1196
1995		242.5	242.5		392	392		1138
2000		238.0	238.0		342	342		997
2005		240.0	240.0		303	303		886
2010		241.6	241.6		269	269		788
2015		243.3	243.3		239	239		701
2020		243.3	243.3		213	213		624
Idaho Chemical Processing Plant								
1988		11.0	11.0		67	67		196
1990		11.5	11.5		77	77		227
1995		11.5	11.5		85	85		246
2000		13.5	13.5		128	128		373
2005		15.0	15.0		172	172		502
2010		13.3	13.3		182	182		532
2015		14.9	14.9		251	251		749
2020		17.2	17.2		284	284		828

Table 2-2. (cont.) Projected HLW Inventories

Year	Volume (10 ³ m ³)			Radioactivity (10 ⁶ Ci)			Thermal Power (10 ³ W)	
	Glass	Other	Total	Glass	Other	Total	Glass	Total
Total Defense								
1988		382.8	382.8		1174	1174		3333
1990		380.7	380.7		1156	1156		3307
1995	1.0	347.1	348.1	88	1238	1326	246	3882
2000	2.0	323.0	325.0	188	1119	1307	539	3792
2005	2.6	314.5	317.1	275	1063	1338	777	3890
2010	3.3	309.7	313.0	317	992	1309	903	3797
2015	3.4	306.8	319.2	340	1038	1378	951	4020
2020	3.5	306.3	309.8	350	1031	1381	970	4002
Projected Commercial HLW Inventories (West Valley Demonstration Project)								
1988		2.1	2.1		29	29		87
1990		1.6	1.6		28	28		83
1995	0.2	0	0.2	25	0	25	74	74
2000	0.2	0	0.2	22	0	22	66	66
2005	0.2	0	0.2	20	0	20	59	59
2010	0.2	0	0.2	18	0	18	52	52
2015	0.2	0	0.2	16	0	16	47	47
2020	0.2	0	0.2	14	0	14	42	42
Total Projected HLW Inventories								
1988		384.9	384.9		1203.0	1203.0		3420.0
1990		382.3	382.3		1184.0	1184.0		3390.0
1995	1.2	347.1	348.3	113.0	1238.0	1351.0	320.0	3956.0
2000	2.2	323.0	325.2	210.0	1119.0	1329.0	605.0	3858.0
2005	2.8	314.5	317.3	295.0	1063.0	1358.0	836.0	3949.0
2010	3.5	309.7	313.2	335.0	992.0	1327.0	955.0	3849.0
2015	3.6	306.8	319.4	356.0	1038.0	1394.0	998.0	4067.0
2020	3.7	306.3	310.0	364.0	1031.0	1395.0	1012.0	4044.0

Source: ORNL89, Table 2.

site includes all stored retrievable waste. Most TRU wastes are classified as "contact-handled", which can be handled directly as the shielding provided by the waste package is enough to prevent exposure. These wastes have a surface dose rate of less than 200 milliRoentgen per hour (mR/h). "Remote-handled" wastes are those that have enough contamination from beta, gamma, or neutron activity to require remote handling: remote handled wastes have a surface dose rate of greater than 200 mR/h.

The quantities of TRU waste generated by and accumulated at designated storage facilities, expressed in volume, mass, radioactivity, and thermal power for each type of stored waste, are shown in Table 2-3. The categories of stored waste are retrievable-stored, contact handled waste; retrievable-stored, remote handled waste; and buried waste. With respect to contact-handled TRU waste the greatest volume is stored at INEL where there is 36,640 cubic meters while the greatest radioactive content, 1,026,800 Curies, is stored at the Savannah River Site. The total volume of TRU waste stored as of the end of 1988 is 251,000 cubic meters. This TRU waste contains 4,500,000 Curies of radioactivity.

Most TRU wastes, by volume, are generated by DOE defense-related activities at the Rocky Flats Plant, the Hanford Facility, and the Los Alamos National Laboratory. Nearly one-half of all TRU waste comes from weapons components manufactured at the Rocky Flats Plant and subsequent plutonium recovery at all three sites. The second largest source of TRU wastes is decontamination and decommissioning projects, which account for one-fourth of the total. About one-fifth of TRU wastes come from laboratory activities, which can produce exotic TRU isotopes. There is currently a moratorium on reprocessing and plutonium recycling, so the amounts of TRU wastes from fuel cycle activities are minimal. A small amount of TRU waste is also being generated in industrial and government-sponsored fuel fabrication and research.

Table 2-4 projects the annual accumulations of contact and remote-handled TRU waste at each facility. By volume, the Rocky Flats Plant is expected to generate 47 percent of the total, all of it contact-handled. The site accumulating the most waste, measured by radioactivity, is the Savannah River Site which contains 83 percent of the total radioactivity. The 732.5 cubic meters of TRU waste accumulated there annually contains 127,850 Curies.

Table 2-5 shows both the annual accumulations and total amounts of waste. The projections show the accumulations at the rate indicated in table 2-4. By 2013, the expected volume of contact handled waste will be 120,243 cubic meters with a radioactive content of 12,238,000 Curies, an increase of nearly 10,000,000 Curies over the 1988 level.

Table 2-3. TRU Waste Inventories as of December 31, 1988

Site	Volume (m ³)		Radioactivity (10 ³ Ci)		Thermal Power (10 ³ W)	
	Annual	Total	Annual	Total	Annual	Total
Retrievably stored, contact-handled						
HANF	96.0	9,876.0	0.1	0.7	0.0	0.0
INEL	1,065.0	36,640.0	125.6	876.8	3.1	19.8
LANL	179.7	7,179.0	52.6	367.4	1.9	12.1
ORNL	32.7	607.5	9.7	667.7	0.1	1.2
SNL	0.0	0.0	0.0	0.0	0.0	0.0
SRS	275.0	3,297.0	147.0	1,026.8	4.9	31.4
NTS	10.2	596.0	0.4	2.8	0.0	0.0
Subtotal	1,658.6	58,195.5	335.4	2,942.2	10.0	64.5
Retrievably stored, remote-handled						
HANF	0.20	137.0	0.0	0.4	0.0	0.0
INEL	4.77	32.8	0.0	0.0	0.0	0.0
LANL	0.00	11.1	0.0	0.4	0.0	0.0
ORNL	0.00	1,304.0	0.0	1,537.0	0.0	2.0
SNL	0.00	0.0	0.0	0.0	0.0	0.0
SRS	0.00	0.0	0.0	0.0	0.0	0.0
NTS	0.00	0.0	0.0	0.0	0.0	0.0
Subtotal	4.97	1,484.9	0.0	1,537.8	0.0	2.0
Buried						
HANF	0.0	109,000.0	0.0	0.6	0.0	0.0
INEL	0.0	57,100.0	0.0	0.0	0.0	0.0
LANL	0.0	14,000.0	0.0	9.2	0.0	0.3
ORNL	0.0	6,200.0	0.0	14.1	0.0	0.3
SNL	0.0	3.0	0.0	0.0	0.0	0.0
SRS	0.0	4,534.0	0.0	38.4	0.0	1.2
NTS	0.0	0.0	0.0	0.0	0.0	0.0
Subtotal	0.0	190,837.0	0.0	62.3	0.0	1.8
TOTAL	1,663.57	250,517.40	335.40	4,542.30	10.00	68.30

Source: ORNL89, Table 3.9.

Table 2-4. Projected Annual TRU Waste Generation, 1989-2013

Site	Average Annual Volume (m ³)	Average Annual Alpha Radioactivity (Ci)
Contact-handled		
Storage Sites		
HANF	143.5	451.4
INEL	88.3	16.3
LANL	200.0	19,900.0
ORNL	25.0	580.0
SNL	0.0	0.0
SRS	732.5	127,850.0
NTS	0.0	0.0
Generators		
ANL-E	7.4	29.8
LLNL	59.0	175.0
MP	38.2	85.2
RFP	1,188.0	5,240.0
Subtotal	2,481.9	154,327.7
Remote-handled		
Storage Sites		
HANF	41.4	unknown
INEL	1.9	33.4
LANL	0.2	<0.01
ORNL	6.0	8.5
SNL	0.0	0.0
SRS	0.0	0.0
NTS	0.0	0.0
Generators		
ANL-E	3.6	2.5
LLNL	0.0	0.0
MP	0.0	0.0
RFP	0.0	0.0
Subtotal	53.1	44.5
TOTAL	2,535.0	154,372.2

Source: ORNL89, Table 3.10.

Table 2-5. Projected TRU Inventories

Year	Volume (m ³)		Radioactivity (10 ³ Ci)		Thermal Power (10 ³ W)	
	Annual	Total	Annual	Total	Annual	Total
Contact-Handled						
1988	1659	58196	335.4	2342	10.0	64.6
1990	2482	63160	395.8	3134	11.7	88.0
1995	2482	75569	395.8	5113	11.7	146.7
2000	2482	87979	395.8	7092	11.7	205.3
2010	2482	112798	395.8	11050	11.7	322.6
2013	2482	120243	395.8	12238	11.7	357.8
Remote-Handled						
1988	5	1485	0	1537	0	0
1990	53	1591	0.2	1538	0	0
1995	53	1897	0.2	1539	0	0
2000	53	2122	0.2	1540	0	0
2010	53	2653	0.2	1541	0	0
2013	53	2812	0.2	1542	0	0

Source: ORNL89, Table 3.1

The sections A.4.1.1 - A.4.1.10 describe all the DOE facilities which store or generate TRU waste that is projected to eventually be transported to the WIPP facility or another transuranic waste management system. Also included in the discussion are descriptions of the current TRU waste storage capacity at DOE interim storage sites, and DOE's alternative plans for stored wastes if delays in the opening of the WIPP facility are extensive or if the facility does not open at all. Until WIPP is operational, TRU waste will continue to be stored at the interim storage sites. Each interim site is required to develop site-specific waste management plans describing how the projected newly generated TRU waste will be managed until a final disposal site is operational.

3. REFERENCE WASTE MANAGEMENT SYSTEM

To perform an impact analysis of regulatory alternatives for 40 CFR 191, a reference waste management system, representing the planned system to be in place in the absence of further Environmental Protection Agency (EPA) action regarding 40 CFR 191, is assumed. The estimated costs for this system are derived from the most recent information available from the Department of Energy (DOE). This cost information will form the basis for the economic analyses performed in this RIA. In addition to reporting the costs for the reference system, the system description also presents design assumptions and key system-parameters in order to evaluate compliance strategies with alternative standards.

The reference system, and its associated costs, are described separately for:

1. the program to dispose of spent nuclear fuel and high-level radioactive wastes administered by DOE's Office of Civilian Radioactive Waste Management (OCRWM), and
2. the transuranic (TRU) waste disposal program at the Waste Isolation Pilot Plant (WIPP).

The following sections contain a summary description of the key elements of the system, a discussion of the selection basis for the reference system costs assumed for this RIA, a presentation of the projected cost estimates, and a discussion of the repository and waste package design -- the primary elements of the waste management system potentially impacted by 40 CFR 191.

3.1 High-Level Waste Management System

3.1.1 Summary System Description

The high-level waste management system authorized by the Nuclear Waste Policy Act (NWPA), as amended, (NWPA87) is composed of three major system-elements: the transportation system, the monitored retrievable storage (MRS) facility, and the mined geologic disposal system (MGDS). The transportation system will accept spent nuclear fuel from commercial nuclear power reactors or from other storage locations. In general, this waste will be transported to an MRS for temporary storage and staging. When scheduled for disposal, the waste will be transported from the MRS facility to the site of the geologic repository. The transportation system will also accept defense high-level waste (HLW) produced in atomic energy defense programs, commercial HLW from the West Valley Demonstration Project, and other radioactive materials as designated by the DOE and the NRC for disposal in the MGDS. These wastes will be shipped directly to the repository for final preparation and emplacement. The

transportation system will include a cask system, transportation system support facilities, and transportation system support equipment.

The MRS facility will receive, store, and stage shipments of spent nuclear fuel to the repository. Shipments will be received from reactor sites via truck or rail. Facilities will be provided for the handling and temporary storage of spent nuclear fuel. When scheduled for final disposal, the waste will be retrieved from storage and shipped to the repository by trains. During steady state operation, when the receipt rate at the MRS is equal to the shipping rate, the MRS facility would serve as a staging facility for transportation: spent fuel received in truck and rail casks would be transferred to large capacity rail casks for shipment to the repository.

The MGDS, which includes the repository waste handling facilities, will receive spent-nuclear fuel from the MRS facility and HLW from the defense sites and the West Valley Demonstration Project primarily via rail. Some spent nuclear fuel (e.g., from reactors located close to the repository) and/or other HLW may possibly be sent directly to the MGDS via truck and/or rail. The received waste will be prepared for disposal, transferred underground, and emplaced in the host rock. After all the waste has been emplaced, a "caretaker" period will begin which complies with Nuclear Regulatory Commission's (NRC) fifty-year retrievability requirements under 10 CFR 60. (GAO89) If, at the end of this period, the NRC is satisfied that the repository is performing as expected, the repository will be prepared for permanent closure by backfilling the underground repository, sealing shafts and ramps, and decommissioning the surface facilities.

3.1.2 Selection of Reference System Costs

The NWPA requires the DOE to submit a report to Congress each year which assesses the adequacy of the fee charged to utilities to cover the entire costs of the waste management program.

(GAO89) In support of this effort, the DOE publishes a detailed set of life-cycle cost estimates for an assumed set of reference systems. This analysis, referred to as the "Total System Life-Cycle Cost (TSLCC) Analysis" represents the official long-term financial plan for the program. The analysis reflects the up-to-date program plans, policies, designs and data that are available. The most recent TSLCC analysis was completed in May 1989 (GAO89) and is the source of the reference costs for the high-level waste management system in this RIA.

The May 1989 TSLCC analysis presents costs for five cases whose total costs range from \$23.8 to \$32.7 billion, expressed in undiscounted, constant 1988 dollars. The five cases are distinguished by:

(1) spent-fuel generation projection; (2) number of repositories needed to dispose of all the waste (one or two); and (3) whether or not the spent-fuel is first consolidated at the MRS, placed in canisters and then packaged in disposal containers at the repository and emplaced, or left intact, packaged in disposal containers at the repository and then emplaced. All of the cases assumed that an MRS facility is an integral part of the waste management system and that the first repository (in the cases with two repositories assumed) is located in the tuff formation at the Yucca Mountain site in Nevada. For the purpose of this RIA, the selected case consists of the "no new orders, end-of-reactor life" spent fuel projection (86,757 MTHM of spent fuel (GAO89)), two repositories (the first disposing of 70,000 MTHM-equivalent of spent-fuel, defense HLW, and West Valley HLW, and the second disposing of the remainder of the waste), and the intact spent-fuel assumption. Regarding the spent-fuel projections assumption, the DOE has stated (DOE89a) that the "no-new orders, end-of-reactor life" projection is currently the assumption used in their planning. In addition, the DOE has also stated their intention to use the intact spent-fuel assumption as a basis for planning. (DOE89b) Regarding one versus two repositories, the DOE has not yet taken a position on the need for a second repository. Two repositories are assumed because of the NWPA-prescribed 70,000 MTHM disposal limit placed on the first repository until a second repository begins acceptance of waste. The total system life-cycle cost for this case is estimated by DOE to be \$31.2 billion in undiscounted, constant 1988 dollars. Of this, \$7.0 billion is for the first repository. Discounted at 2 percent, the first repository will cost \$3.6 billion. The next section will present the details of this reference system cost.

One point to be mentioned before presenting the cost estimates is that the assumed facility start-up dates in the May 1989 TSLCC analysis are the year 2000 for the MRS facility and 2003 for the first repository. In November 1989, the DOE announced in their report to Congress restructuring the high-level waste program, (DOE89d) that the reference start-up dates have now become 1998 for the MRS facility and 2010 for the repository. DOE is currently assessing the life-cycle costs for the restructured program and, as of this time, these new estimates are not yet available. Therefore, this analysis assumes the costs given in the May 1989 analysis. It is likely that the total costs (expressed in constant dollars) for the repository facility (including waste package costs)--the element most likely to be potentially affected by the EPA standard--will be relatively unchanged by the delay, even though the years in which the costs will be incurred would change.

3.1.3 Reference Costs

The major cost categories during the life-cycle of the high-level waste management system are development and evaluation, transportation, first repository, second repository, MRS facility, and benefits payments. The life-cycle will extend until decommissioning and closure of second repository is completed (estimated to be 2087 for the case selected). The total estimated costs of the major categories are summarized below in Table 3-1. The cost estimates for each major category are expressed on an annual basis in Table 3-2 and the cost estimates for first repository and second repository costs are shown in Figures 3-1 and 3-2.

Table 3-1. Summary of Reference Costs

Major Cost Category	1988 Dollars, Billions
Development and Evaluation	13.1
Transportation	2.3
First Repository	7.0
Second Repository	6.6
MRS Facility	1.4
Benefit Payments	0.9
Total System	*\$31.2

* Column may not add to total due to independent rounding.

These costs are based on the receipt, management and disposal of 96,727 MTHM of waste of which the predominant type is spent-fuel from commercial reactors. The total consists of 86,757 MTHM of spent fuel, 8,875 MTHM-equivalent of defense HLW, and 640 MTHM of commercial HLW from the West Valley Demonstration Project.

The remainder of this section explains the basis of each cost category and provides a more detailed breakdown of the cost estimates.

Table 3-2. Annual Total-System Costs for the Two-Repository System No-New Orders, End-Of-Reactor-Life Case with Intact Disposal (Millions of 1988 \$)

Year	Development & Evaluation	Benefits	Transportation	First Repository	Second Repository	MRS Facility	Total
1983	207	0	0	0	0	0	207
1984	310	0	0	0	0	0	310
1985	346	0	0	0	0	0	346
1986	428	0	0	0	0	0	428
1987	488	0	0	0	0	0	488
1988	382	0	0	0	0	0	382
1989	424	0	0	0	0	0	424
1990	406	14	0	0	0	0	420
1991	622	14	0	0	0	0	636
1992	535	13	0	28	0	0	576
1993	458	13	0	44	0	0	515
1994	420	12	0	42	0	14	488
1995	321	12	0	18	0	18	369
1996	310	12	6	24	0	14	366
1997	309	12	18	21	0	61	421
1998	317	11	24	111	0	150	613
1999	321	11	12	188	0	86	618
2000	313	14	51	225	0	43	646
2001	266	14	23	225	0	43	571
2002	262	14	36	192	0	57	561
2003	131	19	49	123	0	51	373
2004	50	18	51	128	0	64	311
2005	48	18	49	138	0	69	322
2006	46	18	65	151	0	61	341
2007	46	17	67	178	0	47	355
2008	46	17	135	226	0	29	453
2009	46	16	75	219	0	40	396
2010	46	15	75	221	0	29	386
2011	56	15	74	221	0	29	395
2012	71	15	71	223	0	29	409
2013	96	15	71	224	0	29	435
2014	121	14	75	225	0	29	464
2015	146	14	74	225	0	29	488
2016	146	13	68	220	0	29	476
2017	177	12	67	224	0	29	509
2018	253	12	65	216	0	29	575
2019	317	16	68	216	0	53	670
2020	367	16	88	227	21	29	748

Table 3-2. (cont.) Annual Total-System Costs for the Two-Repository System
 No-New Orders, End-Of-Reactor-Life Case with Intact Disposal (Millions of 1988 \$)

Year	Develop- ment & Eval- uation	Benefits	Trans- portation	First Repository	Second Repository	MRS Facility	Total
2021	294	16	65	219	82	29	705
2022	250	15	79	219	82	29	674
2023	223	15	58	214	43	23	576
2024	202	13	37	207	50	23	532
2025	194	13	35	180	50	23	495
2026	195	13	51	172	174	30	635
2027	187	12	27	145	258	28	657
2028	187	12	4	20	288	1	512
2029	187	12	2	20	372	9	602
2030	178	12	2	20	403	2	617
2031	179	9	2	20	332	0	542
2032	91	10	17	20	293	0	431
2033	29	10	36	20	249	0	344
2034	29	10	111	20	348	0	518
2035	29	10	53	20	350	0	462
2036	29	10	66	20	347	0	472
2037	29	10	43	20	346	0	448
2038	29	10	54	20	339	0	452
2039	29	8	50	20	332	0	439
2040	29	8	26	20	146	0	229
2041	29	8	22	20	136	0	215
2042	29	8	22	20	136	0	215
2043	17	8	2	20	27	0	74
2044	17	8	2	20	27	0	74
2045	17	8	2	20	27	0	74
2046	17	8	0	20	27	0	72
2047	17	6	0	20	27	0	70
2048	17	6	0	20	27	0	70
2049	17	6	0	20	27	0	70
2050	17	6	0	20	27	0	70
2051	17	6	0	20	27	0	70
2052	17	6	0	20	27	0	70
2053	17	6	0	36	27	0	86
2054	17	6	0	37	27	0	87
2055	17	6	0	37	27	0	87
2056	17	6	0	37	27	0	87
2057	17	6	0	37	27	0	87
2058	17	6	0	37	27	0	87

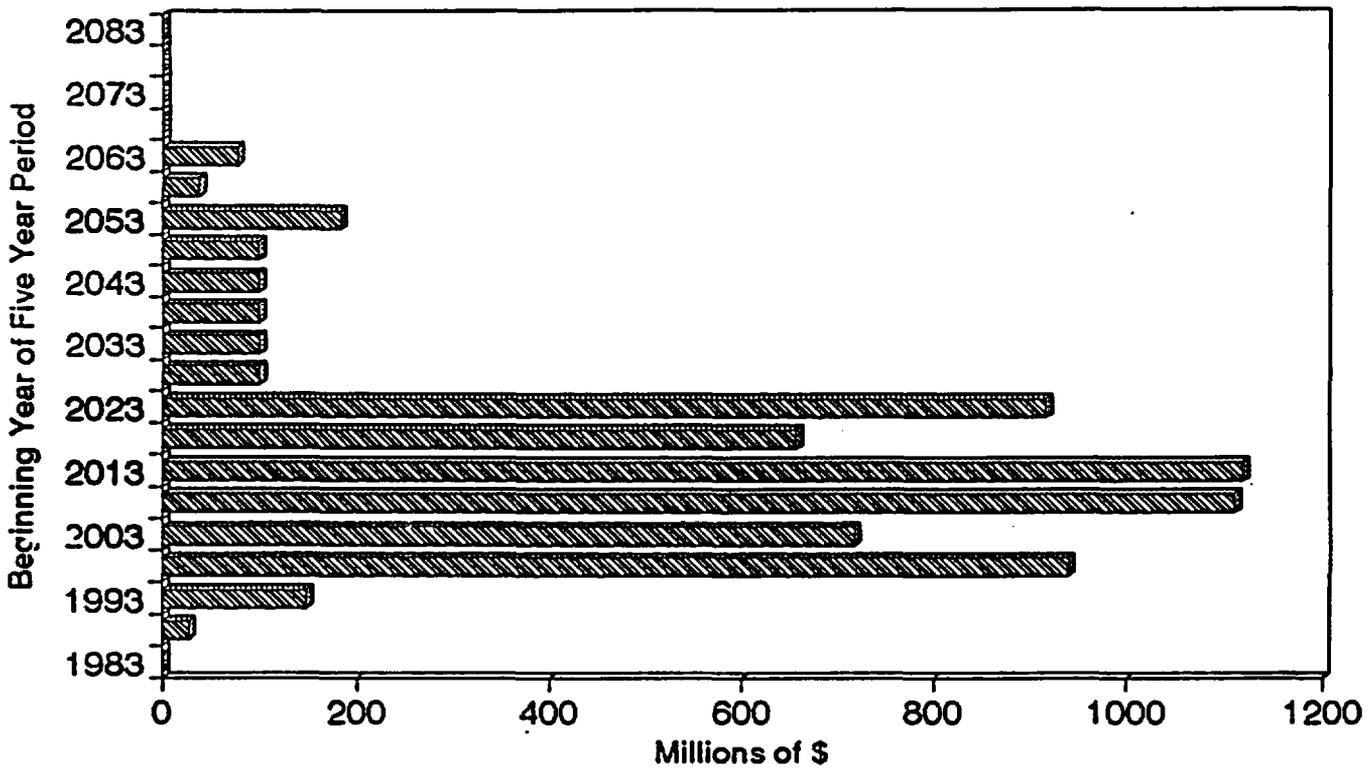
Table 3-2. (cont.) Annual Total-System Costs for the Two-Repository System
 No-New Orders, End-Of-Reactor-Life Case with Intact Disposal (Millions of 1988 \$)

Year	Develop- ment & Eval- uation	Benefits	Trans- portation	First Repository	Second Repository	MRS Facility	Total
2059	17	4	0	37	27	0	85
2060	17	4	0	37	27	0	85
2061	17	4	0	37	27	0	85
2062	17	4	0	19	27	0	67
2063	17	4	0	19	27	0	67
2064	17	4	0	19	27	0	67
2065	17	4	0	19	27	0	67
2066	17	4	0	19	27	0	67
2067	16	2	0	0	27	0	45
2068	16	2	0	0	27	0	45
2069	16	2	0	0	27	0	45
2070	16	2	0	0	27	0	45
2071	16	2	0	0	27	0	45
2072	16	2	0	0	27	0	45
2073	16	2	0	0	27	0	45
2074	16	2	0	0	27	0	45
2075	16	2	0	0	27	0	45
2076	16	1	0	0	27	0	44
2077	16	1	0	0	27	0	44
2078	16	1	0	0	27	0	44
2079	16	1	0	0	27	0	44
2080	16	1	0	0	27	0	44
2081	16	1	0	0	27	0	44
2082	16	1	0	0	45	0	62
2083	16	1	0	0	76	0	93
2084	16	1	0	0	76	0	93
2085	16	1	0	0	62	0	79
2086	16	1	0	0	62	0	79
2087	16	1	0	0	31	0	48
Total	13055	856	2325	7006	6582	1387	31211

Source: DOE89a

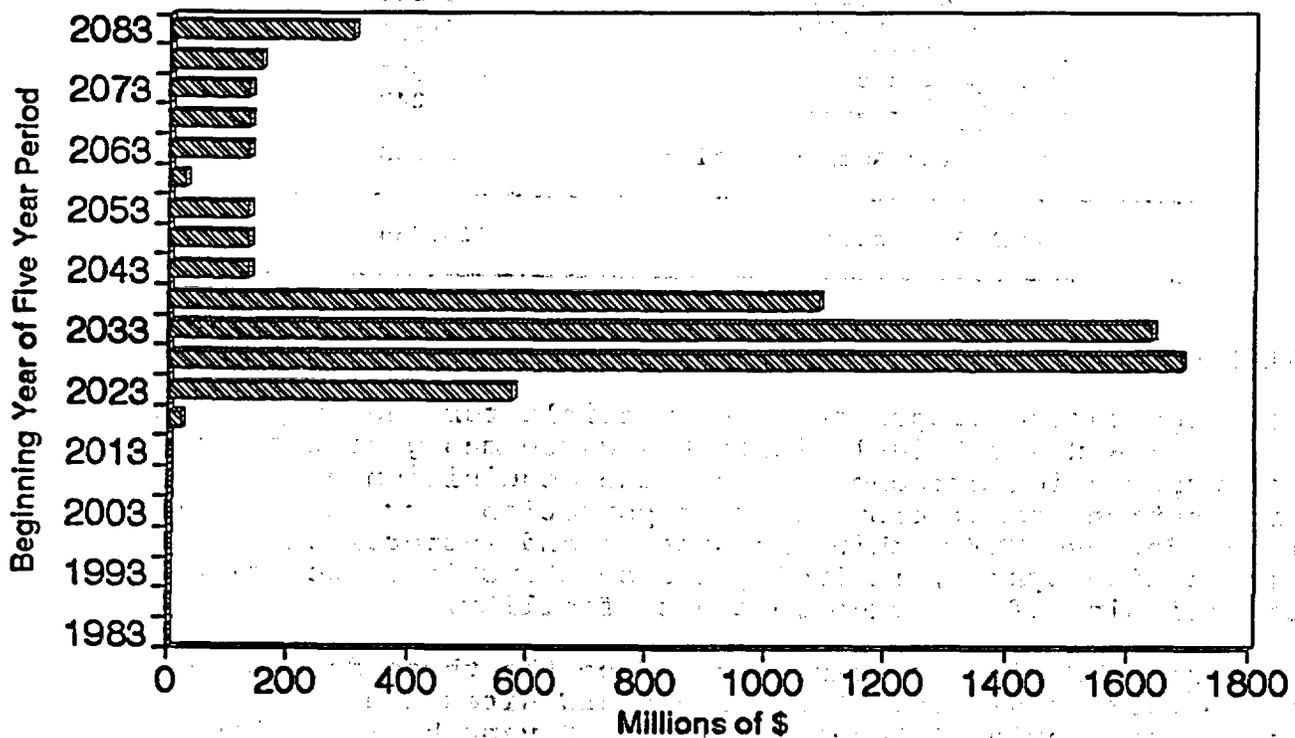
Figure 3-1

Sums of First Repository Life-Cycle Costs for Five Year Periods



Source: DOE 89a

Figure 3-2
Sums of Second Repository Life Cycle
Costs for Five Year Periods



Source: DOE 89a

Development and Evaluation

The development and evaluation (D&E) category covers all the siting, preliminary design development, testing, regulatory and institutional activities associated with the repository, the facility for MRS, and the transportation system. The D&E category also includes administration activities by the Federal Government and NRC activities relating to licensing the waste management facilities and certifying the transportation casks. Table 3-3 provides a breakdown of the D&E costs.

Table 3-3. Reference D&E Cost Estimates

Cost Category	Millions of 1988 Dollars
First Repository	5,206
Second Repository	3,051
MRS Facility	292
Transportation	976
Systems Integration	249
NRC Fees	718
Government Administration	2,563
Total D&E Costs	\$13,055

Transportation

The transportation component is responsible for the shipment of defense HLW and spent-fuel within the waste management system. The transportation component covers the acquisition of transportation casks; provides transportation services for spent-fuel and defense HLW within the system; and supports the transportation system, including the construction, operation, and decommissioning of a cask-maintenance facility.

The defense HLW will be transported from the three defense HLW sites to the repository. The defense HLW sites are Idaho National Engineering Laboratory, Idaho; Savannah River Plant, South Carolina; and Hanford Reservation, Washington. All of the defense HLW will be shipped to the repository via rail. Canisters containing the solidified HLW are placed in shipping casks for transport to the repository. Costs for the shipment of commercial HLW from the West Valley Demonstration Project have been excluded from these transportation costs.

All reactor sites that have the capability to ship spent-fuel by rail to the MRS are assumed to do so by dedicated trains. All other spent fuel shipments to the MRS are assumed to be by truck.

The spent fuel is shipped from the MRS to the repository by dedicated trains. The intact spent fuel assemblies are placed in shipping casks for both rail and truck transport. Table 3-4 provides a breakdown of the transportation costs. The from-MRS transportation costs assume that the routing distance between the MRS facility and the repository is about 2,900 miles. (DOE89e)

Table 3-4. Reference Transportation Cost Estimates

Cost Category	Millions of 1988 Dollars
<u>Spent-fuel transportation</u>	
From Reactors	901
From MRS Facility	577
Cask-Maintenance Facility	484
Total Spent-Fuel Costs	1,962
<u>Defense HLW transportation</u>	
Defense Waste	307
Cask-Maintenance Facility	56
Total Defense HLW Costs	363
Total Transportation Costs	\$2,325

Repository (First and Second)

The repository will consist of surface and underground facilities at the mined geologic disposal site. The principal function of the surface facilities is to receive, package, and transfer defense HLW, West Valley HLW, and spent-fuel to the underground facilities. Surface facilities include excavated material handling facilities, the waste handling building, support buildings, etc. The packaged waste is transported through the underground facilities to the waste emplacement tunnels where waste packages will be emplaced in a borehole for permanent disposal. The underground facilities consist of shafts, ramps, support facilities, and access and waste emplacement tunnels. Repository costs cover the engineering, construction, operation, closure and decommissioning for the repositories, and waste package costs.

Tables 3-5 and 3-6 provide a breakdown of the repository costs. The first repository costs are based on an assumed repository in tuff at the Yucca Mountain candidate site in Nevada, while the

second repository costs are for a generic rock type with average mining conditions and operating productivities. Section 3.2.4 provides a more detailed description of the tuff repository and waste package design.

Monitored Retrievable Storage (MRS) Facility

The MRS facility is assumed to receive intact spent-fuel assemblies from reactors. The intact assemblies are removed from the shipping casks at the MRS and placed in dry cask storage until the repository is ready to accept waste. After the repository is open, the intact spent-fuel assemblies are placed in shipping casks and transported by rail to the repository. The MRS facilities include a receiving-and-handling building, maintenance shops, administration building, support areas, dry storage facilities, etc. The MRS costs cover the engineering, construction, operation, and decommissioning costs of the MRS facility. Table 3-7 provides a breakdown of the MRS costs.

Benefits Payments

The Nuclear Waste Policy Amendments Act of 1987 (NWPAA87) allows the Secretary of Energy to enter into benefits agreements with States or Indian Tribes who volunteer to site a repository or MRS facility. In return for waiving its rights to disapprove a site recommendation, the State or Indian Tribe will receive annual payments for the life of the facility.

The DOE's May 1989 TSLCC analysis estimates of the benefits payments and are presented in Table 3-8. These estimates were based on the yearly estimates contained in the NWPAA (i.e., \$5 to \$20 million per year depending on the facility and status of operation).

Table 3-6. Reference Second Repository Cost Estimates

Cost Category	Millions of 1988 Dollars
<u>Second Repository</u>	
Construction	2,245
Operation	3,985
Closure and decommissioning	352
Total Second Repository Costs	\$6,582

Table 3-7. Reference MRS Facility Cost Estimates

Cost Category	Millions of 1988 Dollars
Construction	343
Operation	1,020
Decommissioning	24
Total MRS Facility Costs	\$1,387

Table 3-8. Reference Total Benefits Payments

Cost Category	Millions of 1988 Dollars
First Repository	476
Second Repository	193
MRS Facility	187
Total Benefits Payments	\$856

3.1.4 Design of the Repository and Waste Package for Tuff

The elements of the waste management system most likely to be impacted by the EPA standards (40 CFR 191) are the repository facility and the waste package. This section presents a detailed description of the repository and waste package designs that were used to develop the reference repository costs for the tuff repository. Also noted, are the regulatory requirements upon which the designs are based.

3.1.4.1 Repository Design

The site of the candidate Yucca Mountain repository is about 85 miles northwest of Las Vegas, Nevada, on Federal land controlled by the DOE, the U.S. Air Force, and the Bureau of Land Management. The repository will consist of the surface and underground facilities. The surface facilities would be situated on the sloping terrain at the eastern base of the mountain, and the underground waste disposal area would be under the ridgeline of the mountain. The surface and underground areas are assumed to be connected by four shafts and two ramps, one ramp used exclusively for transporting waste and the other used for moving mined tuff to and from the surface.

The life cycle of the repository consists of phases that include engineering and construction, operation, as well as closure and decommissioning. Engineering and construction would include the license application design, final procurement and construction design, Title III design, and construction of the repository to a level for initial waste emplacement. The operations phase of the repository begins with initial waste emplacement and ends fifty years later (meeting the 10 CFR 60 fifty-year retrievability requirement). The reference repository design is for the time required for emplacement of 70,000 MTHM, based on the annual waste acceptance rates assumed, while the emplacement period is estimated to last 25 years, followed by 25 years of caretaker operations. Repository closure and decommissioning follow the operational phase. The underground facilities will be backfilled and the openings to the surface sealed. Surface facilities will be dismantled and the site will be decontaminated and decommissioned.

The design requirements (SNL87) of a repository evolve from federal laws and regulations. Program requirements, which are generic and apply to the MGDS regardless of the host medium, have been derived from statutory and regulatory requirements--the NWPA (as amended), 40 CFR 191 (promulgated in 1985), 10 CFR 60, 10 CFR 960, and 10 CFR 20. In addition to these regulatory requirements, a series of DOE directives provide additional design requirements for the MGDS.

Repository Design Assumptions

The following design assumptions are used for the reference repository design:

- The spent-fuel mix was approximately 2/3 by weight (MTHM) from pressurized-water reactors (PWRs) and 1/3 from boiling-water reactors (BWRs).
- The initial power density may not exceed 55.6 kW/acre for 10-year-old PWR fuel or 50.3 kW/acre for 10-year-old BWR fuel.
- The temperature of the borehole wall may not exceed 235°C.
- Fifty years after waste emplacement, the temperature in the panel-access drifts may not significantly exceed 50°C.
- The depth of overburden must be greater than 200 meters.
- The percentage of tuff that can be excavated and still maintain stability in the underground drifts is 30 percent.

Surface Facilities

The principal surface facilities area occupies about 82 acres. A spoils pile, for the tuff excavated from the underground disposal area, is located about one mile to the north. The principal surface facilities area is divided by security fences into three functional areas dedicated to waste receiving and inspection, general support facilities, and the waste processing operations. Casks containing defense HLW, West Valley HLW, or spent-fuel would initially enter the waste receiving and inspection area. Incoming casks are inspected and cleaned, and moved into the waste operations area. Rail sidings are provided for the temporary storage of incoming casks and empty casks awaiting return to the shippers.

The general support area is comprised of the administration building; medical, health physics, fire, and plant security facilities; the food service area; the central warehouse; the motor pool; and shops. Personnel and non-radioactive materials enter the central facilities through the general support area.

The waste operations area encompasses all the buildings associated with the processing of radioactive waste, including site-generated waste. The waste handling building (WHB) is the center of high-level waste processing operations. Spent-fuel assemblies and HLW canisters containing solidified high-level waste from the defense sites and the West Valley Demonstration

Project are encapsulated into disposal containers at the WHB before being transported to the underground facilities. The WHB consists of five receiving bays; cask unloading cells for defense or West Valley HLW and spent-fuel, welding and inspection stations, and decontamination chambers. The waste packages containing the defense HLW, West Valley HLW, or spent-fuel are placed on a waste package transporter and shipped out of the WHB.

Underground Facilities

The underground facilities of the repository will be constructed at a depth of about 1,000 feet in tuff. The facilities will include three parallel main entry drifts and a number of waste-emplacements panels--areas in which the waste will be emplaced. Each emplacement panel will be 1,400 feet wide and roughly rectangular in shape. Spaced within each emplacement panel will be a number of emplacement drifts, in which boreholes would be drilled for the emplacement of waste containers. A perimeter drift will define the boundary of the facility. Also included in the underground facilities will be areas for waste emplacement support shops, drift development support shops, performance confirmation and inspection, and a dedicated testing area. The entire underground area will cover about 1,400 acres.

As soon as two of the waste-emplacements panels have been completely developed, waste emplacement in the first panel would begin. This approach will allow the underground-development and waste-emplacements operations to proceed essentially in parallel. The underground-development and waste-emplacements operations will use different access drifts and ventilation systems to provide sufficient separation to protect the workers from radiation.

A transporter brings the waste package from the surface facilities into the underground facilities. The transporter would travel through the underground facilities until it reaches a specified borehole in an emplacement drift. The reference waste-emplacements mode is vertical emplacement. Boreholes will be drilled vertically into the floor of the emplacement drifts, and a single waste package would be emplaced in each borehole. Boreholes for defense and West Valley HLW are 20 feet deep and about 30 inches in diameter. Spent-fuel boreholes are 25 feet deep and about 30 inches in diameter. The HLW and spent-fuel waste packages will be commingled, i.e., waste packages of defense HLW or West Valley HLW are alternated with spent-fuel waste packages in the same emplacement drift. The reference borehole spacing in commingled emplacement drifts is a 7.5 foot distance between the center of each package. The number of spent fuel waste packages is greater than the number of HLW packages. Consequently, not all emplacement drifts will have commingled waste packages. The emplacement queue for the waste packages will depend on the calculated heat output of each waste package and the design constraints mentioned above.

3.1.4.2 Waste Package Design

There will be two different waste packages used for disposal. One is designed for defense and West Valley HLW, the other for spent-fuel. The reference waste package is designed as a thin-walled (.375 inches) right circular cylindrical shell with a separate top and a bottom made of stainless steel. The top will include a pintle for container handling purposes and an identification system. The reference material for both packages is stainless steel. It is assumed that a rolling process will be used to form the cylindrical shell. The seam of the shell will be welded. The bottom will be welded to the shell.

During waste package closure at the repository WHB, the waste will be loaded in the package and the package will be decontaminated as necessary. Plasma arc welding is assumed for welding the top to the container. This weld would receive certification before the waste package would be allowed to leave the WHB.

Canisters containing either solidified defense HLW or West Valley HLW will be shipped to the repository from the HLW sites. At the WHB, the canister is installed in a HLW waste package that is fitted with minimal stabilizers to provide mechanical restraint during handling, transporting, and emplacement operations.

The spent-fuel waste package will have an internal structure in which the PWR and/or BWR assemblies would be placed. The optimal design for a spent-fuel waste package will contain both PWR and BWR assemblies. However, it is expected that situations would occur in which only PWR or BWR assemblies are available for packaging at the WHB. An internal structure that optimize either PWR or BWR assemblies packaging would then be used. The general specifications of the HLW and spent-fuel waste packages are given in Table 3-9.

3.1.5 Health Consequences of the Reference Case

In this section the health consequences of 40 CFR 191, as applied to the reference case for a high-level waste repository, are discussed. Because the Nuclear Waste Policy Amendments Act of 1987 requires that continuing feasibility studies be focused exclusively on the Yucca Mountain, Nevada, site, an updated performance assessment is available for this one site only. Hence the reference case considered here has all the properties possessed by the model of Yucca Mountain constructed by A. D. Little in the course of their most recent performance assessment.

Numerous parameters must be set in developing a model of a nuclear waste storage facility that combines the geologic and engineered properties of the facility as well as the expected properties of the substances to be reposed therein. However, a

few parameters are of special importance because they can be thought of as policy variables. To some extent, these properties can be manipulated by EPA or program managers.

Table 3-9.
Waste Package Characteristics-Tuff Repository

Package	Assemblies or MTHM	Dimensions (inches)	Loaded Weight (lbs.)
PWR and BWR Mixed	3 PWR + 4 BWR*	28 x 187.5	9,700
PWR Only	10 PWR	28 x 187.5	8,300
BWR Only	4 BWR	28 x 187.5	8,800
Defense HLW	0.5 MTHM equivalent	26 x 126	5,500
Commercial HLW	2.13 MTHM equivalent	26 x 126	5,300

* Average MTHM for a PWR assembly is .43 and for a BWR assembly is .18.

These parameters are the geologic media in which the facility will be located, the leach rate associated with the waste form, and the life of the canister containing the waste form. For the reference case these parameters are:

Geologic Media - Tuff
Waste Form Leach Rate - 10^{-5} parts per year (ADL90b)
Canister Life - 300 years (ADL90c)

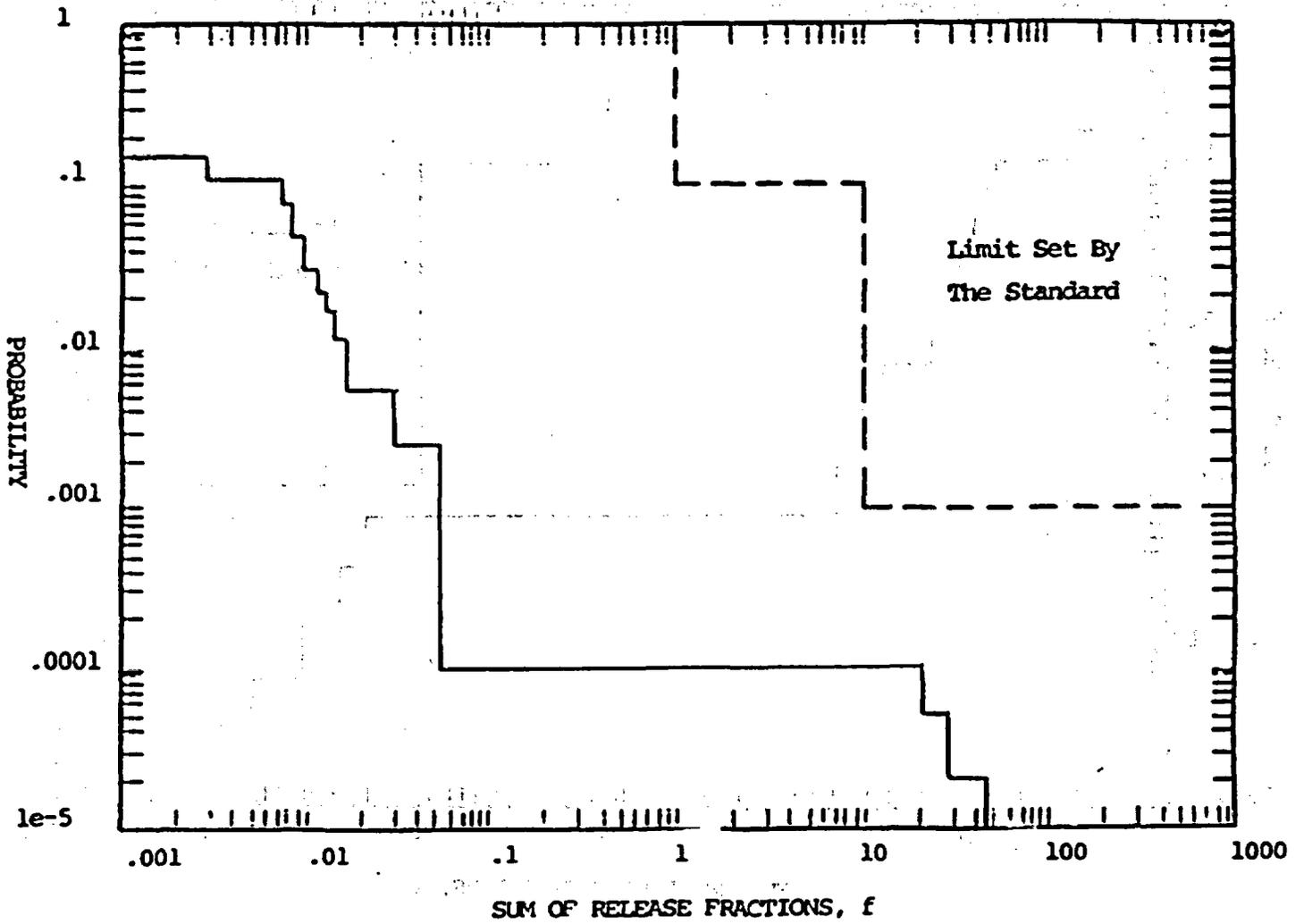
In addition, it is assumed that the repository will contain a maximum of 70,000 MTHM. (ADL90a) Former studies have based their results on 100,000 MTHM of waste.⁴ In order to correspond to 70,000 MTHM of waste, results from former studies will be reduced by 30 percent in presentation of the results. Likewise, the limit on the number of health effects in 10,000 years implied by 40 CFR 191 will be reduced to 700 from 1,000. This can be done

⁴ ADL90b states that the report uses 100,000 MTHM even though the Repository is limited to 70,000 MTHM. This was done to be consistent with previous EPA documents.

because the limit on the number of health effects is modelled as a linear function of the volume of waste.

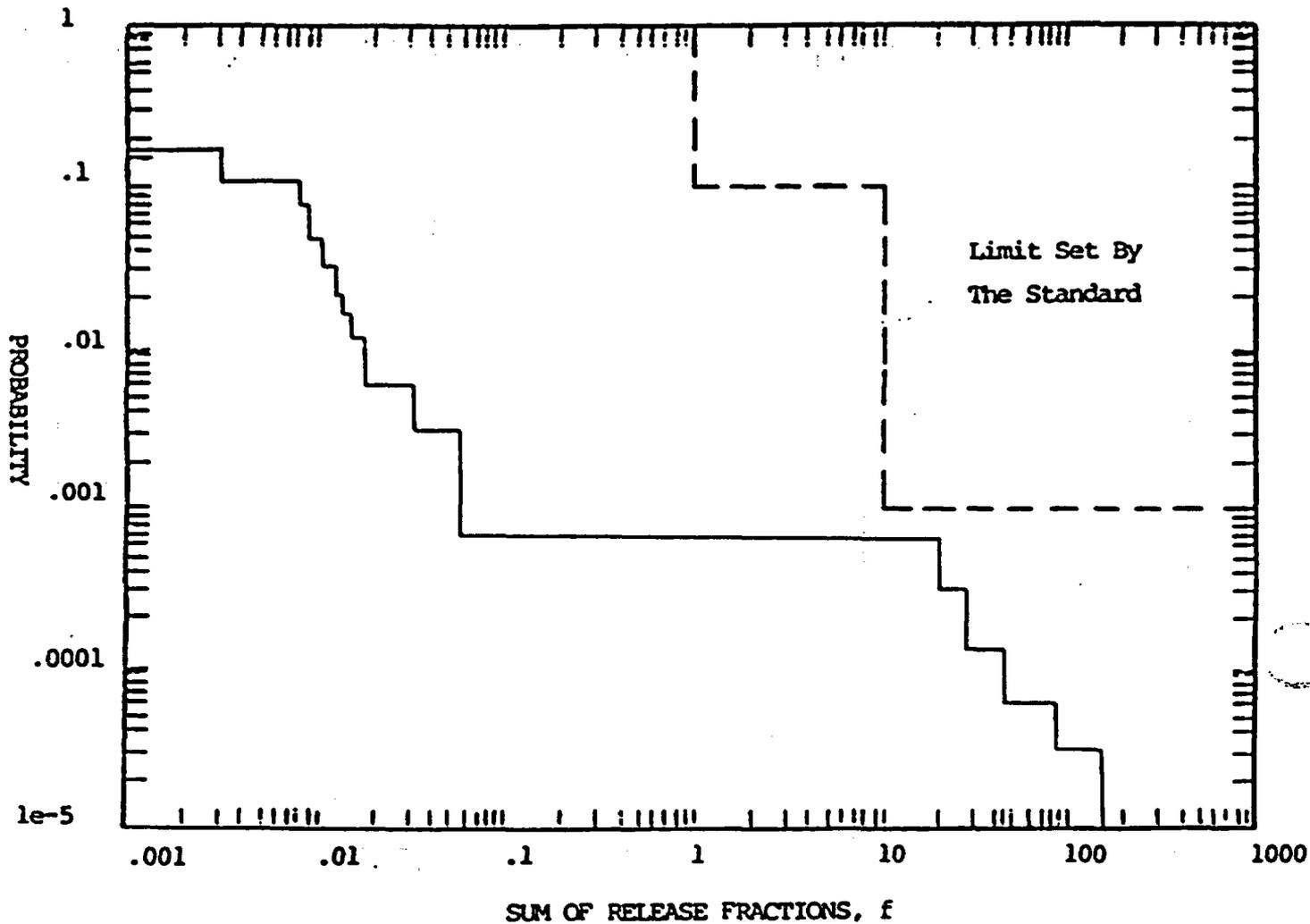
Given these assumptions, the expected total number of health effects, taking all scenarios into account, was 4. (ADL90a) This is substantially below the 700 health effects per 10,000 years implied by 40 CFR 191. However, to judge if the containment requirement of 40 CFR 191 is satisfied, the probabilistic behavior of the radioactive emissions must also be considered, as discussed in Section 1.2.1. Figure 3-3 shows that the probability that health effects due to emissions from the reference case facility equals or exceeds 700 is less than 0.1 while the probability that it equals or exceeds 7,000 is less than 0.001. In this figure a release fraction of 1 would cause 700 health effects and a release fraction of 10 would cause 7,000 health effects. As can be seen the solid line depicting the release fractions from the facility is below the broken line depicting the limits set by the containment requirement. This indicates compliance with the containment requirement. Sensitivity tests were also conducted to determine how dependent the results are on the assumed parameters of the model. (ADL90a) The parameter that most influences the results is the probability of volcanic activity. Figure 3-4 shows the results for the highest probability of volcanic activity modelled. The probability of volcanic activity is increased to over six times the level in the base case, but the probability that there will be over 700 health effects due to health effects is still below 0.1 and the probability that there will be 7,000 or more health effects is still below 0.001.

Figure 3-3. CCDF for the Reference Case



Source: ADL90b

Figure 3-4. CCDF for Modified Volcano Probability



Source: ADL90b

The reference case analysis for Yucca Mountain, shows zero discharge of radiation to ground water in 10,000 years (ADL90a). This clearly meets the requirement that management and storage of radioactive waste shall not cause any increase in the levels of radioactivity in any underground source of drinking water outside the controlled area which may cause a violation of any primary drinking water regulation under 40 CFR Part 141. The individual protection requirement requires that exposure to individuals through all pathways not exceed 10 millirems. Studies of the Yucca Mountain site infer that the only pathway of concern is ground-water. As stated above, ground-water was determined to have zero discharge.

3.2 Transuranic Waste Management System

3.2.1 Summary System Description

The DOE plans to dispose of TRU waste from national defense programs in a geologic repository located in bedded salt in Eddy County, New Mexico. The TRU waste repository is designated as the Waste Isolation Pilot Plant (WIPP). The construction of the WIPP was authorized by the Department of Energy National Security and Military Applications of Nuclear Authorization Act of 1980 (Public Law 96-164) (DOE79) and is classified as a defense activity of the DOE, exempt from Nuclear Regulatory Commission regulations.

The life-cycle of the TRU waste management system began in 1976 when conceptual design work on the WIPP was initiated. The period from Fiscal Year 1976 (FY 1976) until FY 1989 can be generally characterized as a period of research and development (R&D) and construction of WIPP facilities. The TRU waste management system is currently entering a six year transition and pilot plant phase that extends from FY 1990 to FY 1995. The full operations phase is expected to last for seventeen years (FY 1996 to FY 2012). Full operations will then be phased out over a three year period (2013 to 2015). The TRU waste management system life-cycle will end in 2019 after a four-year decontamination and decommissioning phase.

The major elements of the TRU waste management system are the transportation system and the WIPP. It is expected that TRU waste from ten sites will be shipped to the WIPP for disposal. The estimated volume of TRU waste to be disposed of is 6.2 million cubic feet of contact-handled (CH) waste and 250,000 cubic feet of remote handled (RH) waste. (FEIS90)

3.2.2 Selection of Reference System Costs

A significant portion of the TRU waste management system currently exists as facilities and equipment at the WIPP site. Consequently, sunk costs for these existing facilities and

equipment were used as reference costs. In addition to costs for currently existing facilities and equipment, estimates of future capital and operating expenditures were needed to identify the life-cycle costs for the reference TRU waste management system. The reference WIPP sunk costs and estimates of future expenditures were provided for this RIA by the DOE. (WIPPa, WIPpb)

Transportation reference costs are based on current WIPP life cycle projections which were provided by the DOE as well. Currently, figures are only available for contact-handled waste, which comprises 97 percent of the waste going into WIPP. These figures cover the amount WIPP will pay to its trucking contractor to haul the waste, the WIPP costs to handle waste packages, fees paid to states in transit, maintenance costs, and the cost of purchasing the waste packages themselves.

It should be noted here, that the costs for Type A TRU waste disposal inner packages (55-gallon drums, 30 gallon drums, standard waste boxes, etc.) containing the contact-handled and remote-handled TRU waste at the sites are not currently part of the cost of the WIPP or transportation elements of the TRU waste management system. The cost of the waste disposal packages will be absorbed by the individual TRU waste storage and/or generator site that ships TRU waste to the WIPP. The cost for the outer Type B packages (TRUPACT-II and the remote-handled package) and a Type B package maintenance facility are included in the WIPP repository costs. Thus, the cost for Type A waste packages are additional to the estimates provided below for the TRU waste management system.

3.2.3 Reference Costs

The major cost categories during the life-cycle of the TRU waste management system are transportation and WIPP repository. The summary of reference costs for the TRU waste management system are contained in Table 3-10. These costs are undiscounted. The total system life cycle cost, discounted at 2 percent, is \$2.3 billion. The costs, in constant 1983 dollars, for the TRU waste management system are expressed on an annual basis in Table 3-11.

Table 3-10. Summary of TRU Waste Management System Costs

Cost Category	Millions of 1988 Dollars
Repository (WIPP)	3,484.7
Transportation	111.8
Total System	3,596.5

Transportation

The WIPP is designed to dispose of a waste inventory consisting of 6.2 million cubic feet of contact-handled (CH) TRU waste and 250,000 cubic feet of remote-handled (RH) TRU waste that is currently in retrievable storage or projected to be generated through the year 2013. The sites that are expected to send contact-handled TRU waste and/or remote-handled TRU waste to the WIPP are Idaho National Engineering Laboratory, Idaho; Rocky Flats Plant, Colorado; Hanford Reservation, Washington; Savannah River Plant, South Carolina; Los Alamos National Laboratory, New Mexico; Nevada Test Site, Nevada; Oak Ridge National Laboratory, Tennessee; Argonne National Laboratory-East, Illinois; Lawrence Livermore National Laboratory, California; and Mound Plant, Ohio.

Table 3-11
Annual TRU Waste Management System Costs
(in 1,000's of 1988 \$)

Fiscal Year	WIPP Phase	Transportation (See note 1.)	Repository (See note 2.)	Total System (in \$1,000's)
1976	construction		6,700	6,700
1977	construction		11,547	11,547
1978	construction		29,431	29,431
1979	construction		31,461	31,461
1980	construction		24,986	24,986
1981	construction		16,928	16,928
1982	construction		38,845	38,845
1983	construction		110,296	110,296
1984	construction		121,000	121,000
1985	construction		67,945	67,945
1986	construction		13,773	13,773
1987	construction		35,566	35,566
1988	construction		115,302	115,302
1989	construction		127,424	127,424
1990	transition		106,504	106,504
1991	pilot plant	2,484	111,460	113,944
1992	pilot plant	2,484	107,461	109,945
1993	pilot plant	2,484	101,369	103,853
1994	pilot plant	2,484	95,188	97,672
1995	pilot plant	2,484	89,981	92,465
1996	full operation	4,969	98,333	103,301
1997	full operation	4,969	98,801	103,769
1998	full operation	4,969	98,801	103,769
1999	full operation	4,969	98,801	103,769
2000	full operation	4,969	98,801	103,769
2001	full operation	4,969	98,801	103,769
2002	full operation	4,969	98,801	103,769
2003	full operation	4,969	98,801	103,769
2004	full operation	4,969	98,801	103,769
2005	full operation	4,969	98,801	103,769
2006	full operation	4,969	98,801	103,769
2007	full operation	4,969	98,801	103,769
2008	full operation	4,969	98,608	103,576
2009	full operation	4,969	98,124	103,093
2010	full operation	4,969	97,641	102,609
2011	full operation	4,969	97,157	102,126
2012	full operation	4,969	96,481	101,449

Table 3-11 (cont.)
 Annual TRU Waste Management System Costs
 (in 1,000's of 1988 \$)

Fiscal Year	WIPP Phase	Transportation (See note 1.)	Repository (See note 2.)	Total System (in \$1,000's)
2013	phase out	4,969	84,590	89,558
2014	phase out	4,969	76,566	81,534
2015	phase out	4,969	64,868	69,837
2016	decommission		65,694	65,694
2017	decommission		60,641	60,641
2018	decommission		53,566	53,566
2019	decommission		42,449	42,449
Total		\$111,792	\$3,484,698	\$3,596,490

Note 1: Transportation costs computed based on WIPPC.
 They are for CH waste only.

Note 2: Repository costs from WIPPd. They include a contingency factor.

The current plan is that all shipments of contact-handled and remote-handled TRU waste to the WIPP will be by truck. Trucks will pick up the TRU waste at the respective sites around the country and transport the waste to the WIPP via pre-approved routes.

The contact-handled TRU waste will be placed in Type A packages at the waste sites. The contact-handled TRU packages are expected to be Type A 55-gallon drums or standard waste boxes for transport to the WIPP. Type A and B packing specifications are contained in NRC regulations, under 10 CFR 71. The proposed design of the Type B package for the contact-handled TRU waste is the TRUPACT-II. The TRUPACT-II has a capacity of fourteen 55-gallon drums or two standard waste boxes. A "low-boy" trailer carrying three TRUPACT-II packages per trip will transport the contact-handled TRU waste to the WIPP.

The package in which the remote-handled TRU waste will be transported to the WIPP will also be able to meet TYPE B design criteria. The DOE is currently developing the Type B package that will be used for the remote-handled TRU waste. The remote-handled TRU waste will be contained at the sites in 55-gallon drums, 30-gallon drums, or similar containers that will be loaded into the Type B packages designed for transportation of the remote-handled TRU waste to the WIPP.

Transportation costs for WIPP are projected over a 25 year life cycle, and broken down into four components:

- 1) **Trucking Contract Cost.** This is the cost that the trucking company charges to haul the waste, and includes the cost of drivers, driver training, fuel, and the cost to lease the tractors which haul the trailers on which the Type B waste packages are placed. A total of 40,905,000 miles will be covered hauling all the waste from the sites to WIPP, at a shipping cost of \$1.71 per mile.
- 2) **Fleet Purchase.** This is the cost of the TRUPACT-II waste containers and the cost of the trailers on which they are placed. The cost for one TRUPACT-II is around \$288,500 in 1988 dollars. The current plan is to purchase 17 of them.
- 3) **Operational Cost.** This is estimated to cost \$1,167 per shipment in 1988 dollars, and WIPP is using a figure of 16,763 shipments to move all the wastes from the sites to WIPP. Operational costs cover state fees incurred in transit and the cost of loading and unloading the Type B packages from the trailers at the origin sites and at the destination.

- 4) **Maintenance Cost.** This covers spare parts and other miscellaneous costs.

The total life-cycle cost for transportation to and from WIPP is an estimated \$111,792,000 or \$2.73 per mile in 1988 dollars over a total distance of 40,905,000 miles. Note again that this figure only covers contact-handled waste.

Table 3-12 contains a summary of the transportation costs of the TRU waste management system.

Table 3-12. Reference TRU Waste Transportation Cost Estimates

Cost Category	Thousands of 1988 Dollars
Trucking Contract Cost	67,382
Operational Cost	19,719
Fleet Purchase	16,473
Maintenance	8,218
Total Transportation Costs	111,792

Repository

The WIPP is designed to receive, inspect, and dispose of contact-handled and remote-handled TRU waste in a repository mined in the bedded salt underlying the WIPP site. The surface facilities at the WIPP include the waste handling building, shaft filter building, warehouses, etc. The underground facilities include the shafts that connect the surface to the underground repository horizon, the waste-disposal area, an experimental area, and a equipment and maintenance support area.

All TRU waste is received at the waste handling building of the WIPP where it is inspected, inventoried, and prepared for disposal. From the waste handling building, the TRU waste is transported via the waste shaft to the underground facilities for disposal.

The figures given are considered to be conservative: when multiplied by the contingency rates of 0 percent in the Pilot Plant Phase, 10 percent in the Full Operation and Operation Phase-Out periods, and 15 percent in the Decontamination and Decommissioning, the grand total rises from \$3,283,042,800 to \$3,596,040,000.

Table 3-13 contains a summary of the capital, waste operating, waste capital equipment, general plant projects, and contingency costs for the WIPP. Also included in the summary WIPP costs are expenditures related to the development, testing, manufacture, and maintenance of TRUPACT-II and remote-handled TRU waste Type B packages.

Table 3-13. Reference Repository Cost Estimates

Cost Category	Thousands of 1988 Dollars
Capital Costs	408,932
Waste Operating	2,682,913
Waste Capital Equipment	147,600
General Plant Project	43,597
Contingency	201,655
Total Repository Costs	3,484,698

Waste Disposal Packages

As discussed above, the cost of the disposal waste packages containing the TRU are not included within the reference costs of the transportation and repository elements of the TRU waste management system. Waste package costs are absorbed by the TRU waste generators and/or storage facilities. Unit costs for TRU disposal waste packages are contained in Table 3-14.

Table 3-14: Reference Unit Cost for TRU Waste Disposal Packages

Waste Disposal Package	1988 Dollars
CH TRU 55 Gallon Drum	\$73
CH TRU Standard Waste Box	\$1,357
RH Disposal Canister	n/a

3.2.4 Design of Repository and Waste Packages for TRU Waste

The elements of the TRU waste management system most likely to be impacted by 40 CFR 191 are the repository and waste packages. This section discusses in detail the repository facility and the waste packages that will be used for TRU disposal.

3.2.4.1 . Repository Design

The WIPP site is located in Eddy County in southeastern New Mexico. The site about 25 miles east of Carlsbad, New Mexico and about 45 miles southwest of Hobbs, New Mexico. The site is in an area known as Los Medanos which is relatively flat and sparsely populated. The land the WIPP site occupies is owned by the Bureau of Land Management.

The WIPP site boundary is defined by a 16 square mile area. There are two control zones within this boundary. Control Zone I contains most of the surface facilities and occupies approximately 100 acres. Control Zone II covers about 1,800 acres that overlies the maximum limit of the underground facilities.

Surface Facilities

The principal surface structure is the waste handling building. All TRU waste is routed through this building. The building has separate areas for the receipt, inventory, and transfer of the contact-handled and the remote-handled TRU waste to the waste shaft.

The areas for the contact-handled waste include a shipping-and-receiving, receiving-and-inspection, inventory-and-preparation, and a overpack-and-repair room for damaged waste packages. The separate facilities for the remote-handled waste include shipping-and-receiving areas, shipping-cask preparation and decontamination, an area for unloading casks, and a hot cell. The contact-handled and remote-handled TRU waste is transferred to the waste shaft through separate air-locks. In addition to the TRU waste facilities, the waste handling building also contains change rooms, a health physics laboratory, equipment for ventilation and filtration, and an attached facility for the maintenance of Type B packages. Some of the other surface facilities at the WIPP site include a shaft filter building, warehouses, a construction management and maintenance complex, a safety and emergency services building, a security building, etc.

Underground Facilities

The underground facilities of the WIPP are constructed beneath the surface facilities in the bedded salt of the Salado Formation (the underground repository horizon), approximately 2,150 feet beneath the surface. The underground facilities consist of the shafts that connect the surface facilities to the underground

repository horizon and the facilities excavated in the underground repository horizon.

There are four shafts at the WIPP. These shafts are used for air intake, salt-handling, TRU waste handling, and air exhaust, respectively.

The facilities in the underground repository horizon include the waste disposal area, an experimental area, an equipment and maintenance area, and connecting drifts. The waste disposal area will consist of emplacement and access drifts laid out in a "room-and-pillar" configuration. The total waste disposal area needed to emplace the 6.2 million cubic feet of contact-handled TRU waste and the 250,000 cubic feet of remote-handled TRU waste is estimated to be about 100 acres. To date, only about 15 acres of this area is completed. The experimental area and the equipment and maintenance areas are located to the north of the waste disposal area and are largely completed.

Contact-handled TRU waste will be transported to the underground repository horizon via the waste shaft. It is expected that the contact-handled TRU waste arriving in the repository horizon will be contained in 55-gallon drums or standard waste boxes placed on pallets. The pallets on which the TRU waste has been placed will be transported to an emplacement drift where the pallet will be placed on the floor of the drift or stacked on top of previously emplaced contact-handled TRU waste packages. Emplacement drifts are to be backfilled with salt.

Remote-handled TRU waste will be transported to the underground facilities in a facility cask holding one canister of remote-handled TRU waste via the waste shaft. The canister is horizontally emplaced in steel-lined holes in the salt pillars. The holes will then be capped with a shielded steel plug. These drifts will not be backfilled until a decision on retrieval of the remote-handled TRU waste has been made.

3.2.4.2 . Waste Packaging Design

TRUPACT-II

The TRUPACT-II container will be used for shipping contact-handled (CH) TRU waste. It has been designed and constructed to meet the regulations issued by the NRC for "Type B packaging" in 10 CFR Part 71. A Type B packaging with double containment is the type of container that must be used for the transport of TRU waste containing more than 20 curies of plutonium per package. A certificate stating that the TRUPACT-II complies with the NRC regulations was issued by the NRC on August 30, 1989.

The TRUPACT-II container is a cylinder with a flat bottom and a domed top; it is transported in an upright position. The overall

dimensions of the TRUPACT-II are approximately 9 ft in diameter by 10 ft in height; the inner containment vessel is approximately 6 ft in diameter by 8 ft in height. The inner and outer containment vessels have removable lids that are held in place by banded lockrings and retaining tabs. The containment vessels are nonvented and are designed for a maximum normal operating pressure of 50 pounds per square inch.

The inner containment vessel is a stainless-steel pressure vessel that contains the waste payload. The payload is protected by spacers that are made of aluminum honeycomb and are located in each of the two domed heads of the inner vessel. Compression of double O-rings between the lid and the body form a bore-type seal. As the lid is lowered onto the body, retaining tabs on a lockring slide through recesses in the mating tabs on the body. When the lid is fully engaged, the lockring can be rotated to the closed position; the lockring cannot be rotated unless the lid is correctly mated to the body.

The outer containment assembly is made of stainless steel and polyurethane foam. It consists of an exterior stainless-steel shell and a stainless-steel pressure vessel, the outer containment vessel. Between these steel shells there is a layer of fire-retardant polyurethane foam to limit fire damage. There is a lockring which secures the lower body of the outer containment vessel to the upper lid of the vessel by double O-ring seals which maintains, as it does with the inner vessel, leaktight seals under both normal and accident conditions.
(FEIS80)

NUPAC 72B CASK

The NuPac 72B shipping cask will be used to transport remotely handled (RH) TRU waste. The 72B cask is a scaled-down version of the NuPac 125B cask, which is certified as NRC Type B packaging, and which is being used to transport debris from the core of the damaged Three Mile Island reactor.

The NuPac 72B is a cylindrical cask that has a separate inner vessel within an outer cask protected by impact limiters at each end. Neither containment vessel is vented; each is capable of withstanding an internal pressure of 150 lb per square inch (gauge), and each has a capacity of 8,000 lb of payload.

The inner containment vessel is made of stainless steel and has a cavity for the payload canister approximately 26.5 inches in diameter and 123 inches long. The lid is secured to the body of the vessel by means of eight closure bolts. Internal spacers are provided at the top, bottom, and at two locations near the middle of the inner vessel to center the canister and facilitate the insertion and removal of the canister.

The outer cask is a stainless-steel vessel constructed to two concentric shells enclosing a cast-lead shield approximately 1.9 inches thick, which is for gamma radiation. The outer cask is approximately 142 inches long and has an outer diameter of 42 inches. It is protected at each end by stainless-steel shells filled with polyurethane foam. These act as impact limiters as well as thermal insulators to protect the seal areas from fire during an accident.

3.2.5 Health Consequences of the Reference Case

In this section the health consequences of 40 CFR 191, as applied to the reference case for a transuranic waste repository, are discussed. There is only one site under consideration because PL96-164, the National Security and Military Applications of Nuclear Energy Authorization Act of 1980, requires DOE to investigate and build the repository at the one specific site. The one site, the Waste Isolation Pilot Project (WIPP), is now partially completed and in a multi-year test phase. Because WIPP is not subject to NRC regulation, there are no requirements for the longevity of waste disposal canisters or maximum leach rates due to the waste forms. Hence the reference case for a transuranic waste repository is the model that has been developed for the WIPP.

While there are numerous parameters that must be set in developing a model of a nuclear waste storage facility that combines the geologic and engineered properties of the facility as well as the physical and chemical properties of the substances to be stored therein, a few parameters are of special importance because they can be thought of as policy variables. To some extent, these properties can be manipulated by EPA or program managers.

These parameters are the geologic media in which the facility will be located, the leach rate associated with the waste form, and the life of the canister containing the waste form. For the reference case these parameters are:

Geologic Media - Salt
Waste Form Leach Rate - unlimited
Canister Life - 0 years

The quantity of waste assumed to be stored in the reference facility is 9.7 million Curies of which 5.6 million curies have half lives greater than 20 years. Thus the implied limit on the number of health effects that could be allowed to result from storage of these wastes in WIPP is 56.

Analysis demonstrates that the WIPP, as currently planned and constructed, will meet the containment criteria of 40 CFR 191 as well as its individual and groundwater protection criteria. With

respect to the containment criteria, the background information document predicts a release fraction of .09 (ADL90a) which corresponds to 5 health effects in 10,000 years. The limit, as stated above, is 56 health effects for a facility with the waste inventory expected at WIPP. With respect to the individual and groundwater protection criteria, it has been determined that "no release of radionuclides to the accessible environment will occur from the salt repository under expected or degraded conditions by way of normal groundwater flow within 10,000 years." (ADL90a)



4. ALTERNATIVES TO THE REFERENCE CASES: THEIR COSTS, BENEFITS AND EFFECTS ON REQUIREMENTS OF 40 CFR PART 191

This chapter describes the costs and benefits of providing an underground waste disposal for HLW and TRU waste in terms of reductions in the number of statistical health effects under those options for which studies conducted for EPA have provided the necessary information. Thirty-six options were studied for the HLW repository and one, the reference case, for TRU waste. Cost-effectiveness analysis is performed for HLW and the value of 40 CFR 191 is discussed based on those results. The conclusion applies to TRU waste as well. The value is that, regardless of how inexpensively 40 CFR 191 can be complied with and how few health effects are reduced relative to the reference cases, the regulation prevents other geologic media, other sites, or any engineered barriers from being employed that do not meet the criteria. The regulation protects the population from the universe of possibilities that have not yet been studied. The fact that 40 CFR 191 does not add to the implementation costs of proposed efforts to dispose of HLW or transuranic waste should not detract from its intrinsic value.

4.1 Costs and Benefits of High Level Waste Facility

The base case identified in Chapter 1 is the HLW management system that can be constructed at lowest cost in the tuff formation at Yucca Mountain, Nevada to meet basic design requirements other than those of 40 CFR 191. This is a change in the base case relative to that used in previous economic and regulatory impact analyses of 40 CFR 191 (EPA80, EPA85a). In those analyses the facility had not yet been designed, its cost had not been estimated, and the particular design had not been analyzed with respect to its release of radionuclides. In those generic analyses, only the least costly materials and processes were assumed to be used unless there were reason to use a more costly approach. If more costly materials or processes were required in order to meet the requirements of 40 CFR 191, then the benefits would have been the reduction in health effects over 10,000 years, presumably from a number greater than 700 to a number less than 700 (i.e., the limit implied by the containment requirement for a repository containing 70,000 MTHM of waste).

At the time of the 1980 study very little was known regarding the projected releases from the repositories, so no benefit analysis was performed. The analysis provided a range of costs and economic impacts that would occur if additional emission reductions were required. The costs of 40 CFR 191 were considered to be the additional costs of HLW management required to meet the standard.

The 1985 RIA also relied on generic data in estimating costs of various options. In addition, it provided graphical data

depicting the number of health effects for those options. A discussion of the ground-water and individual protection requirements was included.

In contrast to previous analyses, the current study takes note of existing designs which incorporate the requirements of 10 CFR 60. Models of radionuclide release rates based on these designs have been designed and constitute the reference case with respect to radioactive releases and their effects on the containment requirement, the individual protection requirement, and the ground-water protection requirement. Detailed costing of the reference case as discussed in Chapter 3 has also become available due to the annual life cycle cost estimates performed by the Office of Civilian Radioactive Waste Management of the Department of Energy. However, the most recent available data on the cost of alternative waste form leach rates and canister lives is still that from the 1985 RIA (EPA85a).

Modelling of the various options conducted for the BID indicates, in a preliminary way, that the standards set by 40 CFR 191 are met for all wastes expected to be stored in a HLW management facility under a variety of options. In some cases this is due to the properties of the geologic facility alone and, in some, due to the application of 10 CFR 60, the NRC rule. One of the options that meets the standard of 40 CFR 191 is the reference case, tuff with a leach rate of 10^{-5} and a canister life of 300 years. From these observations the conclusion might be drawn that with respect to the geologic media analyzed by EPA, 40 CFR 191 imposes no costs beyond those already incurred by the act of burying the waste with no special engineered barriers or at most burying the waste in conformance with 10 CFR 60.

Sections 4.1.1 through 4.1.3 describe the alternatives to the reference case for which information is available; develop their costs; and depict the number of health effects lost for each option for which information is available. Section 4.1.4 discusses the satisfaction of the containment requirement, the individual protection requirement, and the ground-water protection requirement. Section 4.1.5 then uses the data developed in sections 4.1.2 and 4.1.3 to analyze the benefits and costs of the alternatives.

4.1.1 Alternatives to the Reference Case for a High Level Waste Management System

Section 3.1 describes the reference case for the High Level Waste Management System. With regard to what might be regarded as "the policy variables," the reference case facility is assumed to meet the requirements of 10 CFR 60--that the leach rate of the waste forms not exceed 10^{-5} and that the lives of the canisters used

exceed 300 years. The geologic media used in the reference case is tuff.

Alternatives to the reference case include facilities constructed in other geologic media, specifically salt and basalt; longer canister lives, specifically 1,000 years; and alternative waste forms with lower leach rates, specifically 10^{-6} . The costs and effects of each of these alternatives is discussed in the following sections.

4.1.2 Costs of the Alternatives to the Reference Case for a High Level Waste Management System

The cost of the alternatives are composed of the cost of constructing a repository for 70,000 MTHM of waste in a particular medium (referred to here as Repository Costs), the cost of canisters of the specified durability (Canister Costs), and the costs of waste forms with the given leach rates (Waste Form Costs). Information regarding costs comes from several sources; no single source contains all the costs needed for the comparison. As a result, some assumptions are required to make the estimates.

Comparable repository costs are provided for tuff, salt, and basalt for 1986 by the Office of Civilian Radioactive Waste Management of the U.S. Department of Energy. Changes in repository design assumptions and accounting practices, the decision by the U.S. Congress to limit analysis to Yucca Mountain only, and the shift in concept from consolidating fuel assemblies to storing fuel assemblies intact complicate comparison. Table 4-1 depicts the present value of repository costs for tuff, salt, and basalt in 1988 dollars under the 1986 assumptions and the 1988 assumptions. The 1988 values for salt and basalt are estimates obtained by multiplying their 1986 values by the ratio of 1988 tuff repository costs to 1986 tuff repository costs. It can be observed that repository costs are lower for tuff than for salt or basalt.

Information regarding waste form costs and canister costs were obtained from the 1985 RIA (EPA85a). They are assumed to be the same regardless of the choice of geologic media even though, for example, a stainless steel container will not last as long in salt as in basalt. The present values of the costs do differ, however, over the life cycle of a repository as explained below.

Table 4-1 shows the present value of the total costs of choosing various waste form leach rates. A waste form leach rate of no more than 10^{-5} is required under 10 CFR 60. This rate is the lower limit of the "very good" category listed in the 1985 RIA while 10^{-6} is the upper end. These are assumed to cost \$14 and \$20 per kg of heavy metal in the original substance. In

addition, cost of leach rates of 10^{-3} and 10^{-4} were assumed to be \$10 and \$12 per kg of heavy metal. In Table 4-1 shows the present value of the total costs of choosing various canister lives. These were calculated by observing the percentage of

Table 4-1 Present Value of Cost Factors by Media
(in millions of FY 1988 Dollars)

Cost Factor	GEOLOGIC MEDIA		
	Basalt	Salt	Tuff
REPOSITORY TYPE			
First Repository, 1988 Assumptions	7,402	5,492	3,634
First Repository, 1986 Assumptions	6,158	4,569	3,023
WASTE FORM LEACH RATE			
10 ⁻³	447	436	418
10 ⁻⁴	536	524	502
10 ⁻⁵	636	621	595
10 ⁻⁶	908	887	849
CANISTER LIFE			
0	0	0	0
300	454	443	425
1000	681	665	637

Calculated using 2% discount rate over lifecycle of facility.
Sources: EPA85a, DOE87a, DOE89a

total deposits emplaced in each year of the life cycle of a repository. The volume of heavy metal was then estimated and the cost of the waste form or canisters then derived for that year. Present values over the life cycle of the repository were then calculated using a 2 per cent discount rate. Present values vary across media because the emplacements of spent fuel are assumed to proceed at different rates in the different media. The 300 year canister corresponds to the "minimum" category described in the 1985 RIA and is assumed to cost \$10 per kg of heavy metal in the original substance. The 1,000 year canister corresponds to the 1985 RIA's "good" category and is assumed to cost \$15 per kg of heavy metal in the original substance. The 10,000 year canister corresponds to the 1985 RIA's "very good" canister and is assumed to cost \$40 per kg of heavy metal. It was assumed that the cost of a zero year canister is \$0 per kg of heavy metal.

Table 4-2 sums the repository, canister, and waste form costs of each alternative studied for High Level Waste disposal, enumerated in present value terms and in constant 1988 dollars. The costs were also annualized using the life cycle of the repository for the number of years and 2 per cent for the discount rate.

4.1.3 Impacts of the Alternatives to the Reference Case for a High Level Waste Management System with Respect to Containment Requirements

The conformance of the various alternatives for a HLW management system to the containment standard of 40 CFR 191 was discussed in the Background Information Document (ADL90a). The analysis considered canister lifetimes of zero and 1,000 years and the waste form release rate was varied from one part in 1,000 (10^{-3}) per year to one part in 1,000,000 (10^{-6}) per year.

Figure 4-1 displays the results of the analysis for the three different geologic media (salt, basalt, and tuff) that have been considered for the first repository. The reference case in Chapter 3 assumed there would be a first repository holding 70,000 MTHM and a second repository for the remaining waste. This analysis deals only with the first. Under the assumption that the first repository will contain 70,000 MTHM of waste, the limit implied by 40 CFR 191 is 700 health effects in 10,000 years. As Figure 4-1 shows, both bedded salt and tuff meet that requirement even with zero canister life and a waste form leach rate of 10^{-3} parts per year.

The first part of the report deals with the general situation in the country. It is noted that the economy is still in a state of depression, and that the government has taken various measures to stimulate it. The second part of the report deals with the financial situation, and the third part deals with the social situation.

The financial situation is particularly serious, and it is noted that the government has had to resort to various measures to raise revenue. The social situation is also a cause for concern, and it is noted that the government has taken various measures to improve it.

The report concludes with a number of recommendations for the government. It is suggested that the government should continue to take measures to stimulate the economy, and that it should also take measures to improve the social situation.

Table 4-2. Costs by Medium, Leach Rate, and Canister Life (Millions of FY 1988 Dollars)

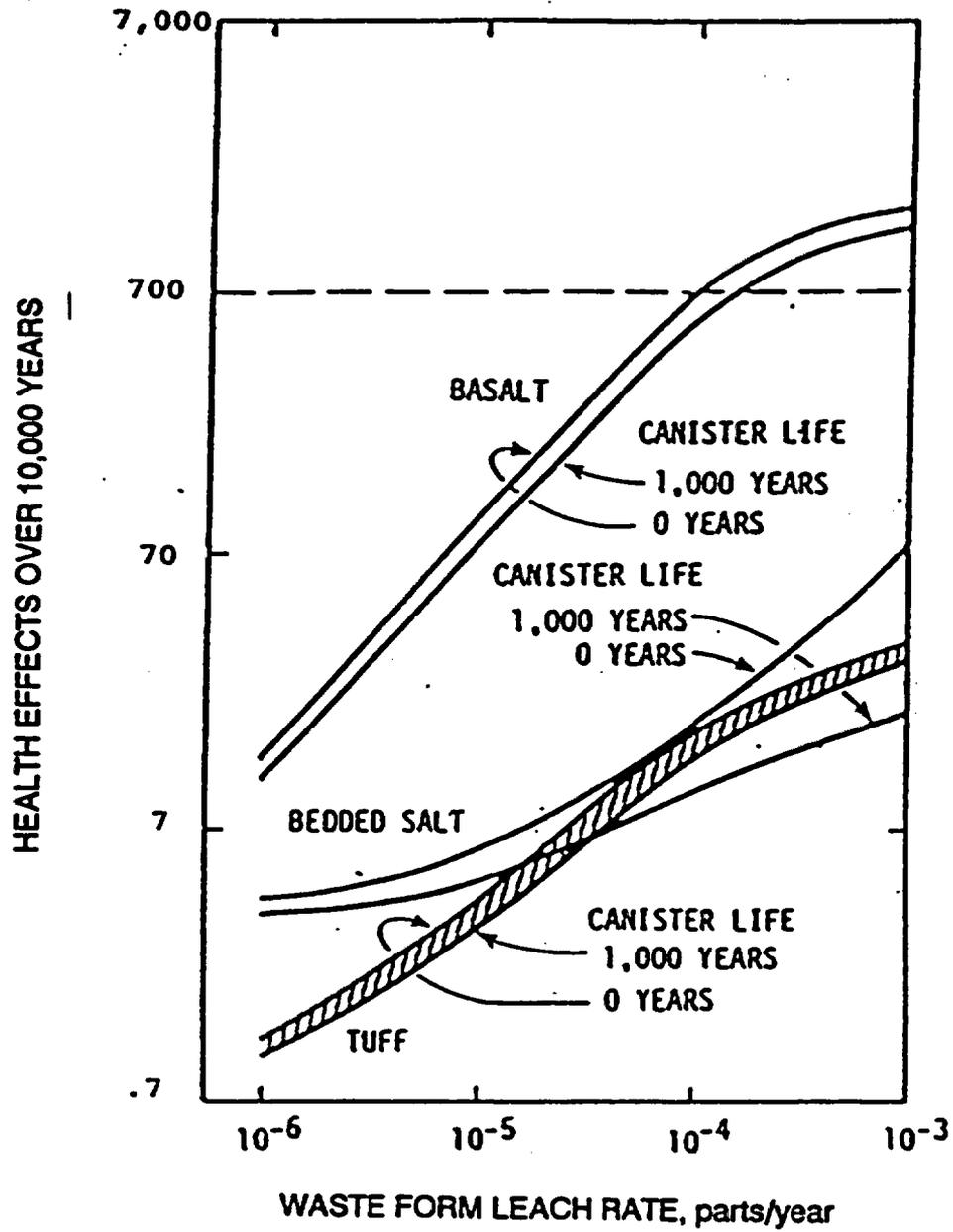
Geologic Medium	Repository Cost, 1988	Canister Life (Years)	Canister Cost	Waste	Waste	Total Cost (Present Value)	Total Cost (Annualized)
				Form	Form		
				Leach Rate	Cost		
BASALT	7,402	0	0	10 ⁻³	447	7,849	192
BASALT	7,402	300	454	10 ⁻³	447	8,303	203
BASALT	7,402	1000	681	10 ⁻³	447	8,530	209
BASALT	7,402	0	0	10 ⁻⁴	536	7,938	194
BASALT	7,402	300	454	10 ⁻⁴	536	8,392	205
BASALT	7,402	1000	681	10 ⁻⁴	536	8,619	211
BASALT	7,402	0	0	10 ⁻⁵	636	8,038	197
BASALT	7,402	300	454	10 ⁻⁵	636	8,491	208
BASALT	7,402	1000	681	10 ⁻⁵	636	8,718	213
BASALT	7,402	0	0	10 ⁻⁶	908	8,310	203
BASALT	7,402	300	454	10 ⁻⁶	908	8,764	214
BASALT	7,402	1000	681	10 ⁻⁶	908	8,991	220
SALT	5,492	0	0	10 ⁻³	436	5,928	145
SALT	5,492	300	443	10 ⁻³	436	6,372	156
SALT	5,492	1000	665	10 ⁻³	436	6,593	161
SALT	5,492	0	0	10 ⁻⁴	524	6,015	147
SALT	5,492	300	443	10 ⁻⁴	524	6,459	158
SALT	5,492	1000	665	10 ⁻⁴	524	6,681	163
SALT	5,492	0	0	10 ⁻⁵	621	6,113	149
SALT	5,492	300	443	10 ⁻⁵	621	6,556	160
SALT	5,492	1000	665	10 ⁻⁵	621	6,778	166
SALT	5,492	0	0	10 ⁻⁶	887	6,379	156
SALT	5,492	300	443	10 ⁻⁶	887	6,822	167
SALT	5,492	1000	665	10 ⁻⁶	887	7,044	172

Table 4-2. (cont.) Costs by Medium, Leach Rate, and Canister Life (Millions of FY 1988 Dollars)

Geologic Medium	Repository Cost, 1988	Canister Life (Years)	Canister Cost	Waste Form Leach Rate	Waste Form Cost	Total Cost (Present Value)	Total Cost (Annualized)
TUFF	3,634	0	0	10 ⁻³	418	4,052	99
TUFF	3,634	300	425	10 ⁻³	418	4,476	109
TUFF	3,634	1000	637	10 ⁻³	418	4,689	115
TUFF	3,634	0	0	10 ⁻⁴	502	4,135	101
TUFF	3,634	300	425	10 ⁻⁴	502	4,560	112
TUFF	3,634	1000	637	10 ⁻⁴	502	4,772	117
TUFF	3,634	0	0	10 ⁻⁵	595	4,228	103
TUFF	3,634	300	425	10 ⁻⁵	595	4,653	114
TUFF	3,634	1000	637	10 ⁻⁵	595	4,865	119
TUFF	3,634	0	0	10 ⁻⁶	849	4,483	110
TUFF	3,634	300	425	10 ⁻⁶	849	4,908	120
TUFF	3,634	1000	637	10 ⁻⁶	849	5,120	125

Source: Table 4-1.

Figure 4-1. The Effect of Canister Life and Waste Form Leach Rate on Population Risks for Three Potentially Suitable Repository Media



Source: ADL90a

In fact, the highest number of health effects for those two media, just over 70 in 10,000 years, is for salt with a canister life of zero years and a waste form leach rate of 10^3 parts per year. In other words, both bedded salt and tuff will meet the limit with no special engineering barriers in place. However, basalt requires a limit on the waste form leach rate of 10^4 to meet the limit. Conformance with the requirement of 10 CFR 60 that waste form leach rates not exceed 10^5 ensures compliance with the containment standard of 40 CFR 191 in all three media.

4.1.4 Impacts of the Alternatives to the Reference Case for a High Level Waste Management System with Respect to Individual and Ground-Water Protection Requirements

The background information document states that "...undisturbed ground-water flow does not result in a release to the accessible environment over 10,000 years..." (ADL90a) from tuff or salt. Therefore, alternative canisters and waste forms will have no effect on the meeting of individual and ground-water protection requirements in these media. With no release to ground-water, the ground-water protection criterion is clearly satisfied. Also, in tuff, the background information document states that there is no normal release path other than ground-water. Hence, the proposed facilities also satisfy the individual protection criterion in the medium studied. Although these results are specific to the site studied, they demonstrate that there is at least one feasible site and support the possibility that a number of feasible sites exist.

4.1.5 Cost-Effectiveness of 40 CFR 191

In this section, the costs and benefits of HLW disposal options are considered. At the same time the interrelationship of 40 CFR 191 and 10 CFR 60 is explored. Due to limitations in benefit data, cost effectiveness analysis is performed rather than cost-benefit analysis. Comparing the cost per averted health effect of the possible options to a base case or to each other allows for a discussion of the cost-effectiveness of the options. This is the basic approach economists use: to compare the incremental cost of moving to the next most stringent option to the incremental benefits realized from of such a move.

The cost-effectiveness comparison is used here rather than the more familiar cost-benefit analysis. The reason for this is several-fold. It allows a comparison of several different disposal options so that the cost-effectiveness of the preferred standard may be placed in context. Cost-effectiveness analysis is also appropriate when the costs and benefits are measured in different ways. In the current analysis two factors prohibit the direct comparison of costs and benefits in a formal cost-benefit framework. The first is that the costs are discounted but the benefits are not. Discounting of benefits over the long time-frames involved (10,000 years for the health effects in this case) or even over a relatively short period such as several hundred years, will give a present value of zero, for any reasonable discount rate. The second is that the costs and benefits are expressed in different terms, costs in dollars, benefits in averted health effects. There is no monetization of the increase or decrease in the number of statistical health effects that coincides with the choice of options. Finally the costs and benefits of the least costly option, tuff with no canisters and no restriction on waste form leach rates, are not known. What is known are the absolute costs of such a repository, but not the costs incremental to what may have occurred in the absence of 10 CFR 60 or 40 CFR 191. The same applies for the health effects. To somewhat surmount this problem, the various options are compared to suggest what the most cost-effective level of stringency might be.

In the analysis that follows the costs and benefits of the standard were examined for a number of combinations of leach rates and canister lives that might meet it. These options were discussed in section 4.1.2 and are shown in Table 4-2. These options are the result of the choices of studies performed by EPA to evaluate 40 CFR 191. However, when the health effects data are considered, only two canister lives, zero and 1,000 years are available for the analysis. Health effects for a third canister life, 300 years, were estimated by interpolation.

For HLW, Figure 4-1 shows the large number of options that exist for meeting the containment requirements for the HLW repository

and the number of health effects associated with each option, assuming the emplacement of 70,000 MTHM of HLW. These health effects range from about 1 in the most protective case to approximately 1,610 for the worst case for the 10,000 year period of the analysis. Basalt has the worst performance under all options with a health effects range of from about 14 to the worst case of approximately 1,610 mentioned above. Of the three media, basalt relies most on low waste form leach rates to keep health effects to acceptable levels. The smallest number of health effects, approximately 1, is associated with tuff, a 1000 year canister and a waste form leach rate of 10^{-6} .

In EPA's modeling results, the choice of geologic media has a much larger impact on expected health effects than do engineering features. In basalt, a waste form with a leach rate of less than 10^{-4} would meet the rule with less than 700 health effects, but a faster leach rate would be allowed if some other option, such as a hypothetical super canister, were available (health effects start up dramatically at the end of the life of the 1000 year canister so that if a 10,000 year canister were to be used basalt and a fast leach rate would meet 40 CFR 191.) Any of the points shown on Figure 1 for either salt or saturated tuff keep health effects below approximately 77.

Figure 4-1 also illustrates the effect of 10 CFR 60. It requires that minimum canister life be somewhere between 300 and 1,000 years and waste form leach rates be less than 10^{-5} . If the NRC requirements are met, repositories in all three media studied comply with 40 CFR 191. Figure 4-1 also shows that the highest number of health effects expected from HLW in any media and complying with 10 CFR 60 is approximately 97. This result is for basalt with a leach rate of 10^{-5} and a 300 year canister. The highest number of health effects for a salt media would be 6 and for tuff, 4. In the absence of NRC's rule, tuff and salt would continue to meet the EPA standard with health effects a little below 80 for salt and below 40 for tuff. For basalt, EPA's standard could be just barely met with zero year canisters and leach rates between 10^{-4} and 10^{-5} . These parameters would not satisfy the NRC rule.

Table 4-3 shows both the total costs and the total statistical health effects associated with thirty-six options for the disposal of 70,000 MTHM of HLW. It combines the information from Table 4-2 and Figure 4-1. Figure 4-2 illustrates Table 4-3. The following statements rely on Table 4-2: The largest cost is associated with the preparation of the geologic media. Of the three media, basalt is most expensive with a present value of \$7.4 billion (discounted at 2 percent to attain a present value). This is followed by salt, with a present value cost of \$5.5 billion, and tuff, with a present value cost of \$3.6 billion. It

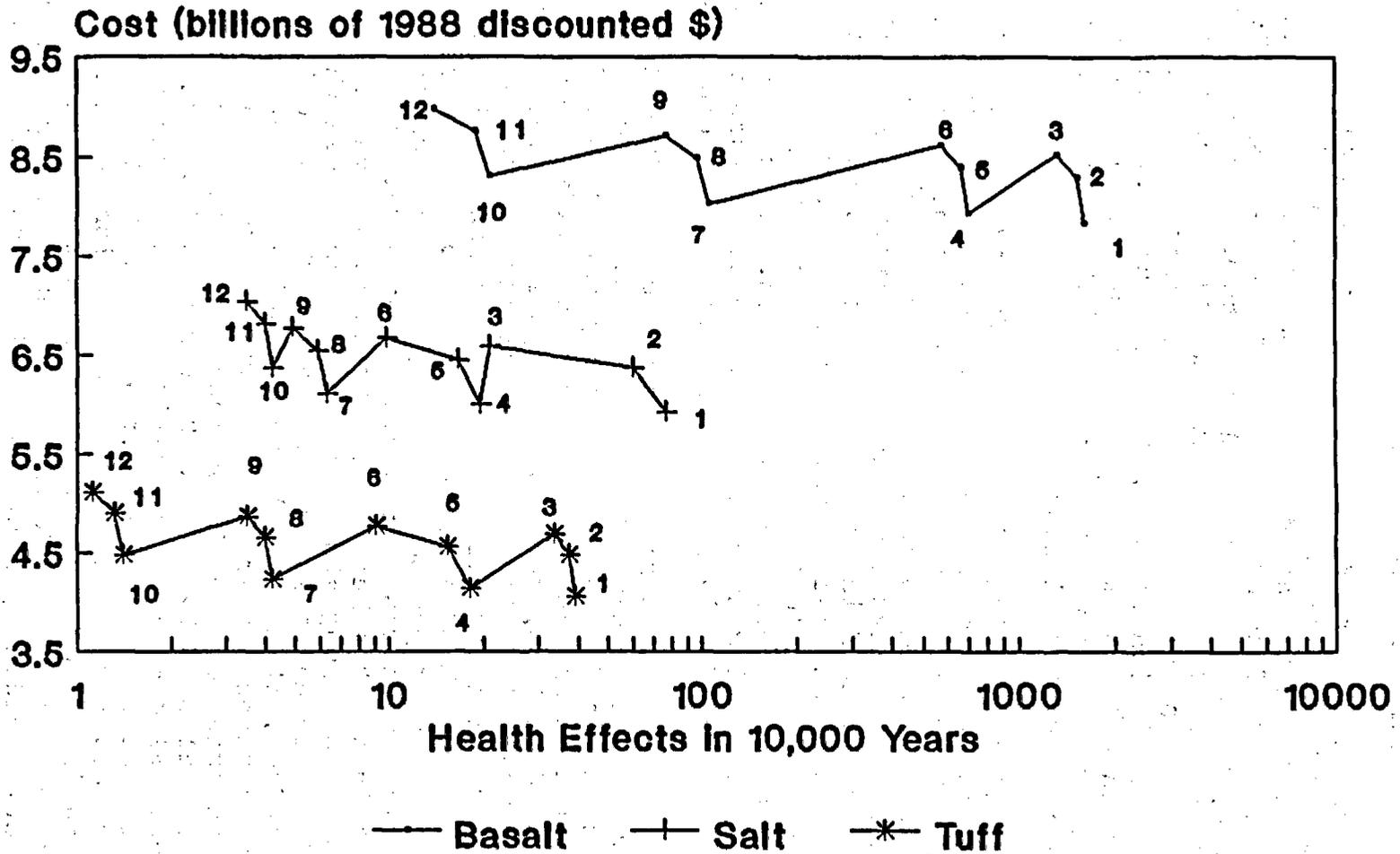
Table 4-3. Costs and Risks for Alternative Configurations of HLW Management System (Costs in millions of discounted 1988 \$)

Geologic Media	Option	Leach Rate	Longevity of Canister	Number of Health Effects in 10,000 yr	Present Value of Cost for Option
BASALT	1	10 ⁻³	0	1,610	7,849
BASALT	2	10 ⁻³	300	1,526	8,303
BASALT	3	10 ⁻³	1000	1,330	8,530
BASALT	4	10 ⁻⁴	0	700	7,938
BASALT	5	10 ⁻⁴	300	662	8,392
BASALT	6	10 ⁻⁴	1000	574	8,619
BASALT	7	10 ⁻⁵	0	105	8,038
BASALT	8	10 ⁻⁵	300	97	8,491
BASALT	9	10 ⁻⁵	1000	77	8,718
BASALT	10	10 ⁻⁶	0	21	8,310
BASALT	11	10 ⁻⁶	300	19	8,764
BASALT	12	10 ⁻⁶	1000	14	8,991
SALT	1	10 ⁻³	0	77	5,928
SALT	2	10 ⁻³	300	60	6,372
SALT	3	10 ⁻³	1000	21	6,593
SALT	4	10 ⁻⁴	0	20	6,015
SALT	5	10 ⁻⁴	300	17	6,459
SALT	6	10 ⁻⁴	1000	10	6,681
SALT	7	10 ⁻⁵	0	6	6,113
SALT	8	10 ⁻⁵	300	6	6,556
SALT	9	10 ⁻⁵	1000	5	6,778
SALT	10	10 ⁻⁶	0	4	6,379
SALT	11	10 ⁻⁶	300	4	6,822
SALT	12	10 ⁻⁶	1000	4	7,044
TUFF	1	10 ⁻³	0	39	4,052
TUFF	2	10 ⁻³	300	38	4,476
TUFF	3	10 ⁻³	1000	34	4,689
TUFF	4	10 ⁻⁴	0	18	4,135
TUFF	5	10 ⁻⁴	300	15	4,560
TUFF	6	10 ⁻⁴	1000	9	4,772
TUFF	7	10 ⁻⁵	0	4	4,228
TUFF	8	10 ⁻⁵	300	4	4,653
TUFF	9	10 ⁻⁵	1000	4	4,865
TUFF	10	10 ⁻⁶	0	1	4,483
TUFF	11	10 ⁻⁶	300	1	4,908
TUFF	12	10 ⁻⁶	1000	1	5,120

Source: Table 4-1, ADL90

Figure 4-2. Costs of Alternative HLW Configurations

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Options numbered on chart are defined in Table 4-3.

is also noted that a tuff repository meeting only 40 CFR 191 has a present value cost of \$3.6 billion.

Costs for different waste forms and canisters are also incorporated in Table 4-2 and are part of the total in Table 4-3. Due to the different timing of activities in the different media, the present values of using a particular canister type or waste form varies by a small amount, although the undiscounted values are not different. The cost of using a 300 year canister is around \$400 to \$500 million. The cost increases to around \$600 million to \$700 million for a 1,000 year canister. The cost of a waste form with a leach rate of 10^{-3} is around \$600 million, while switching to a waste form with a release rate of 10^{-6} makes the cost approximately \$800 million.

A comparison of the options for which modeling results are available allows for a comparison of the cost-effectiveness of the options. Of course, to make a comparison of stringency an option must be compared to other options preferably both less and more stringent. An analytic problem that exists is in evaluating the simplest case, tuff in combination with a zero canister life and no restriction on waste form, which is the lowest cost option that would comply with 40 CFR 191. There is nothing less stringent insofar as costs to compare it to. It can, however, be compared to more stringent options and the comparison can be used to judge the cost-effectiveness of becoming more stringent.

The move from the least stringent option to meet 40 CFR 191 to one that would also comply with 10 CFR 60 can be looked at. To set up the scenario for 40 CFR 191, a repository in tuff is used and no restriction on waste form or canister life is used. This would cost \$3.6 billion, the cost of repository development for tuff. The number of health effects attributable to the highest leach rate for which data is provided is approximately 39 per 10,000 years. 10 CFR 60 at a minimum cost would require construction of the repository in tuff, using a minimal (300 year) canister, and a waste form with a leach rate of 10^{-3} . This would result in about 4 health effects in 10,000 years and cost \$4.6 billion. This suggests that a reduction in health effects is achieved in moving to 10 CFR 60 from 40 CFR 191 at \$28 million per health effect, at most. The additional health effects of allowing leach rates higher than 10^{-3} are not known, nor is the leach rate associated with the unprocessed waste. The maximum health effects allowed by 40 CFR 191 for a 70,000 MTHM repository would be 700. \$1.4 million per health effect is the lowest cost-effectiveness ratio for the move from 40 CFR 191 to 10 CFR 60. This estimate divides the additional \$1 billion for going from no expense on canisters or improved waste form leach rates to the costs required to meet 10 CFR 60 by the nearly 700 statistical health effects avoided by doing so. This is the most cost-

effective 10 CFR 60 would be compared to the minimum requirements of 40 CFR 191.

Other comparisons are possible and are shown in Table 4-4. It is possible to decrease the number of statistical health effects by using waste forms with reduced leach rates or longer lasting canisters. Table 4-4 shows the cost per health effect averted attributed to moves from the option named in the left column to the option named in the top row. The first six options detailed in Table 4-4 satisfy 40 CFR 191 but not 10 CFR 60. Options 1, 2, and 3 set the leach rate at 10^3 . Option 1 then uses a zero year canister; Option 2, a 300 year canister; and Option 3, a 1,000 year canister. Options 4, 5 and 6 set the leach rate at 10^4 . Option 4 uses a zero year canister; Option 5, a 300 year canister; and Option 6, a 1,000 year canister. Like the first six options, Options 7 and 10 also comply with 40 CFR 191 but not with 10 CFR 60. Both use zero year canisters; Option 7 with a 10^5 leach rate and Option 10 with a 10^6 leach rate. The remaining four options satisfy 10 CFR 60. Option 8 is the least costly option in tuff that meets the NRC rule; it consists of using a 300 year canister and a waste form with a leach rate of 10^5 . It was used as the least costly 10 CFR 60 complying option above. Option 9 consists of using a 1,000 year canister. Option 11 consists of reducing the waste form leach rate by a factor of 10 to 10^6 . Option 12 consists of improving the canister and the leach rate to 1,000 years and 10^6 respectively. Tuff is the media used in all these options due to its dominating performance.

It can be noted from Table 4-3 that as higher numbered options are chosen, the number of health effects always declines, while the cost generally increases but occasionally declines. For example, moving from option 6 to option 7 reduces the number of health effects by about 5 while the cost declines by \$544 million. These results follow from decreasing the leach rate from 10^4 to 10^5 at a cost of \$93 million while decreasing canister life from 1000 years to zero years at a savings of \$637 million. These negative cost-effectiveness ratios indicate options which reduce both costs and the number of statistical health effects. They are superior to options with positive cost-effectiveness ratios.

Table 4-4 makes it clear that reducing the waste form leach rate is more cost effective than increasing canister life. Going across any row, the cost per health effect avoided dips for options 4, 7, and 10. These are the options that include a canister life of zero. Increasing the canister life increases costs by a proportionally larger amount than the health effects are decreased.

Table 4-4 suggests that option 8 is the most cost-effective option that meets 10 CFR 60. Starting from option 1, 2, 3, 4, 5, or 6 on the left of Table 4-4 and moving across the row, the column for option 8 has the lowest cost-effectiveness values including negatives for the NRC approved options. For example,

Table 4-4. Cost-effectiveness of Options for Meeting 40 CFR 191 in Tuff

Cost per Health Effect Avoided (in millions of discounted 1988 \$)

From Option:*		To Option:*										
		2 (10 ⁻³ 300)	3 (10 ⁻³ 1000)	4 (10 ⁻⁴ 0)	5 (10 ⁻⁴ 300)	6 (10 ⁻⁴ 1000)	7 (10 ⁻⁵ 0)	8** (10 ⁻⁵ 300)	9** (10 ⁻⁵ 1000)	10 (10 ⁻⁶ 0)	11** (10 ⁻⁶ 300)	12** (10 ⁻⁶ 1000)
(10 ⁻³ , 0)***	1	253	114	4	21	24	5	17	23	11	23	28
(10 ⁻³ , 300)	2	-	54	-18	4	10	-7	5	11	0	12	18
(10 ⁻³ , 1000)	3	-	-	-36	-7	3	-16	-1	6	-6	7	13
(10 ⁻⁴ , 0)	4	-	-	-	156	70	7	36	50	21	46	58
(10 ⁻⁴ , 300)	5	-	-	-	-	33	-29	8	26	-5	25	39
(10 ⁻⁴ , 1000)	6	-	-	-	-	-	-111	-23	17	-38	17	44
(10 ⁻⁵ , 0)	7	-	-	-	-	-	-	2022	910	91	236	290
(10 ⁻⁵ , 300)	8	-	-	-	-	-	-	-	433	-66	95	163
(10 ⁻⁵ , 1000)	9	-	-	-	-	-	-	-	-	-182	19	107
(10 ⁻⁶ , 0)	10	-	-	-	-	-	-	-	-	-	5055	2275
(10 ⁻⁶ , 300)	11	-	-	-	-	-	-	-	-	-	-	1083

Table 4-4. (cont.) Cost-effectiveness of Options for Meeting 40 CFR 191

Incremental Costs (in millions of discounted 1988 \$)

From Option:*		To Option:*										
		2 (10 ⁻³ 300)	3 (10 ⁻³ 1000)	4 (10 ⁻⁴ 0)	5 (10 ⁻⁴ 300)	6 (10 ⁻⁴ 1000)	7 (10 ⁻⁵ 0)	8** (10 ⁻⁵ 300)	9** (10 ⁻⁵ 1000)	10 (10 ⁻⁶ 0)	11** (10 ⁻⁶ 300)	12** (10 ⁻⁶ 1000)
(10 ⁻³ , 0)***	1	425	637	84	508	721	177	601	814	431	856	1068
(10 ⁻³ , 300)	2	-	212	-341	84	296	-248	177	389	7	431	644
(10 ⁻³ , 1000)	3	-	-	-553	-129	84	-460	-36	177	-206	219	431
(10 ⁻⁴ , 0)	4	-	-	-	425	637	93	518	730	348	772	985
(10 ⁻⁴ , 300)	5	-	-	-	-	212	-332	93	305	-77	348	560
(10 ⁻⁴ , 1000)	6	-	-	-	-	-	-544	-119	93	-289	135	348
(10 ⁻⁵ , 0)	7	-	-	-	-	-	-	425	637	255	679	892
(10 ⁻⁵ , 300)	8	-	-	-	-	-	-	-	212	-170	255	467
(10 ⁻⁵ , 1000)	9	-	-	-	-	-	-	-	-	-382	42	255
(10 ⁻⁶ , 0)	10	-	-	-	-	-	-	-	-	-	425	637
(10 ⁻⁶ , 300)	11	-	-	-	-	-	-	-	-	-	-	212

Table 4-4. (cont.) Cost-effectiveness of Options for meeting 40 CFR 191

Incremental Health Effects Avoided

From Options:		To Options:										
		2 (10 ⁻³ 300)	3 (10 ⁻³ 1000)	4 (10 ⁻⁴ 0)	5 (10 ⁻⁴ 300)	6 (10 ⁻⁴ 1000)	7 (10 ⁻⁵ 0)	8** (10 ⁻⁵ 300)	9** (10 ⁻⁵ 1000)	10 (10 ⁻⁶ 0)	11** (10 ⁻⁶ 300)	12** 10 ⁻⁶ , 1000)
(10 ⁻³ , 0)***	1	2	6	21	24	30	35	35	36	38	38	38
(10 ⁻³ , 300)	2	-	4	19	22	28	33	34	34	36	36	36
(10 ⁻³ , 1000)	3	-	-	15	18	24	29	30	30	32	32	32
(10 ⁻⁴ , 0)	4	-	-	-	3	9	14	14	15	17	17	17
(10 ⁻⁴ , 300)	5	-	-	-	-	6	11	11	12	14	14	14
(10 ⁻⁴ , 1000)	6	-	-	-	-	-	5	5	6	8	8	8
(10 ⁻⁵ , 0)	7	-	-	-	-	-	-	0	1	3	3	3
(10 ⁻⁵ , 300)	8	-	-	-	-	-	-	-	0	3	3	3
(10 ⁻⁵ , 1000)	9	-	-	-	-	-	-	-	-	2	2	2
(10 ⁻⁶ , 0)	10	-	-	-	-	-	-	-	-	-	0	0
(10 ⁻⁶ , 300)	11	-	-	-	-	-	-	-	-	-	-	0

*Note: Options defined in Table 4-3.

**Note: Options that meet 10 CFR 60

***Note: (Leach Rate, Longevity of Canister)

Source: Table 4-3

the lowest of these values in row 3 is \$1 million savings per health effect avoided. Option 8 consists of a leach rate of 10^{-5} and a canister life of 300 years.

If option 8 is the starting point, moving to option 10 is the most cost-effective move. This option consists of reducing the waste form leach rate to 10^{-6} and reducing the canister life to zero. Moving from option 8 to option 10 would save \$170 million while simultaneously avoiding 3 health effects implying a savings of \$66 million per statistical health effect averted. Moving to option 9, 11 or 12 is more costly per statistical health effect averted. However, Option 10 is not allowed by 10 CFR 60 because of the zero year canister life. While the move to option 11 is the cheapest way to purchase additional protection, beyond the minimum provided by 10 CFR 60, the move would produce no net benefits unless the reduction of an undiscounted health effect was considered to be worth \$95 million or more in discounted dollars.

For a tuff repository with characteristics similar to the Yucca Mountain site, the individual and ground-water requirements provide an extra protection to populations living near the repositories, but they do not add to the cost of compliance. EPA generic modeling suggests, zero discharge of radiation to ground-water in 10,000 years. The analysis for salt shows the same, even when no special canisters or waste forms are used. These modeling results support the case that the requirements of 40 CFR 61 for management and storage of radioactive waste shall prevent any increase in the levels of radioactivity in any underground source of drinking water outside the controlled area which may cause a violation of any primary drinking water regulation under 40 CFR Part 141.

The individual protection requirement requires that exposure to individuals through all pathways not exceed 10 millirems. As has been shown, ground-water has been determined to have zero discharge in the modeling results. There is, therefore, no impact of moving the limit down to zero.

4.1.6 Costs and Benefits of C-14 Mitigation

Recent evidence suggests the possibility that gaseous releases of C-14 might occur for certain geologic conditions. While it is most likely to be released from unsaturated geological sites, in some unsaturated sites the air currents inside the repository is such that C-14 would not be expected to be released. Compliance with 10 CFR 60 would prohibit such releases and only a violation of this standard would create a violation of 40 CFR 191. These releases may result in a large number of health effects. In anticipation of such an event the costs and benefits of technological barriers have been briefly examined. A large

amount of uncertainty is associated with both the cost and health effects estimates. In the next iteration of this report we will try both to improve these estimates and to provide estimates of the uncertainty.

Two studies have been done to estimate the cost of engineered barriers to prevent the release of C-14 within the 10,000 year period proscribed by 40 CFR part 191. The DOE, in a June 17, 1992 presentation to the EPA Science Advisory Board Subcommittee on Carbon-14, presented results based on their assumptions of C-14 inventories and release (SAIC 92). Their posited engineered barrier was a ceramic metal canister that remained air-tight for 10,000 years.

An estimated 25,000 of these would be required with considerable uncertainty as to their effectiveness. The total system cost for this fix was estimated to be \$3.25 billion. This cost is incremental to the original cost of a canister to meet 10 CFR 60. However, since most of the cost is spread over the 50 year operational period, the actual discounted cost is \$2.1 billion at a 2 percent discount rate over the 50 year life of the facility.

The EPA, in a similar study using a 10,000 year 'super' canister estimated the incremental total life cycle cost as being \$2.1 billion (SCA92). Discounted over the 50 year operating period of the repository at 2 percent the present value would be \$1.4 billion.

Health effect projections may have a high degree of uncertainty associated with them due to the untested nature of these technologies. To address this a range of assumptions is made from perfect performance for the technology to total ineffectiveness. Following on this, a best case assumption can be made that zero health effects would occur from gaseous releases as a result of the canisters. As a worst case the assumption is approximately 3,000 health effects resulting from a total failure of the canisters to contain the gaseous releases (SCA 92).

These estimates suggest a range for the cost-benefit ratio of C-14 reduction. Combining high cost assumptions (\$2.1 billion) with poorest performance of the barrier (reduction of 1 health effect) leader to a cost-benefit ratio of \$2.1 billion per statistical health effect averted. Combining low cost assumptions (\$1.4 billion) with best performance (3,000 statistical health effects averted) for the other end of the range gives a bound of \$473,000 per statistical health effect averted. A midpoint between the two estimates gives a value of \$1.2 million per health effect averted.

This analysis is based on discounting of the costs and not the health effects that are prevented due to these expenditures. It

should be noted that given the time frame over which the health effect would occur (10,000 years) if any positive discount rate were to be applied they would become insignificant.

4.2 Transuranic Waste Facility

As discussed in section 3.2.5, the reference case is the WIPP facility that has already been constructed. It is constructed in salt, and canister lives of zero years and unlimited waste form leach rates are assumed. Modelling results shown in the BID (ADL90a) inferred that the facility with these parameters would meet the containment requirement. The studies show that if the equivalent of 100,000 MTHM of waste were stored in a salt repository, about 90 health effects would occur in 10,000 years. This compares to a hypothetical implicit limit of 1,000 health effects calculated for 40 CFR 191 for this amount of waste storage. The BID also shows that the individual and the groundwater protection requirements would also be met. No alternative media, canister lives, or waste form leach rates were modelled or costed.

The cost of the reference case, as developed in Chapter 3, is \$3.6 billion in undiscounted 1988 dollars. In the reference case an equivalent of 5,600 MTHM of waste is stored in WIPP. Because no reduced waste forms, leach rates, or special canisters were studied, it is apparent that there are measures that could reduce the number of health effects over 10,000 years to below the 5 estimated for the reference case. The data in Table 4-3 implies that the number of health effects can be reduced by 95 to 99 per cent by use of reduced waste form leach rates and increased canister lives. If the cost of canister lives for TRU waste were similar to that for HLW, then the cost of saving nearly 5 health effects would be over \$1 billion or \$200 million per statistical life saved. The cost-effectiveness ratio for the similar jump from HLW Option 1 to HLW Option 12 is only \$28 million per statistical life saved. Thus it is likely to not be as cost-effective to save health effects by adding more stringent engineered barriers for TRU waste as it is for HLW.

The reference case results from actions taken in the absence of any specific requirements regarding the release of radionuclides to the environment. NRC requirements do not apply to WIPP. The geologic media is sufficient, according to the BID, to contain the radionuclide release to the environment sufficiently to meet all requirements of 40 CFR 191 in the absence of special canisters and waste forms. The costs for such a facility would be expected to increase if the requirements of 10 CFR 60 were applied.

Just as is the case with HLW, these results indicate that a transuranic waste facility is capable of meeting 40 CFR 191 with little or no additional effort. In this sense the regulation is

shown to be a reasonable one in that it provides the desired level of protection without imposing a cost burden for implementing the design. This differs from imposing a cost burden for demonstrating the viability of the design, which 40 CFR 191 unavoidably does do.

Again it should be noted that the value of 40 CFR 191 is in what it prevents. It prevents HLW or TRU waste from being buried in any geologic medium in conjunction with any engineered barrier unless it can be reasonably demonstrated that the environmental releases are within the stated levels. 40 CFR 191 does this without imposing unnecessary costs for measures that are not required to meet these levels.

4.3 Cost of Demonstrating Compliance

EPA recognizes that its requirement that DOE demonstrate compliance with 40 CFR Part 191 for TRU disposal may impose some costs on DOE and add to the cost of HLW disposal. In order to have a clear understanding of those costs imposed on DOE, an attempt will be made in the next iteration of this RIA to estimate these costs.

It is likely that many, if not all, of these cost will have been undertaken in the development of the facility itself and that those that remain will be small compared to the overall costs of the development of the facility. At a TRU site DOE does not incur costs of demonstrating compliance to EPA. If, at some future time, DOE is required to demonstrate compliance to EPA the costs would be expected to be similar to those for HLW.

5. ECONOMIC IMPACT MODELS

This section provides descriptions of the methodologies and models used to evaluate the economic impacts of the incremental costs of alternative disposal requirements. Two separate economic impact analyses are developed; one for the commercial generators of nuclear power; and the second for government programs. This dichotomy is necessitated by the dissimilar processes by which costs of a regulation on industry versus those on government are translated into impacts. The analysis is done with respect to HLW only. In the following total incremental costs were applied to commercial waste. The discussion of charges for defence high level waste is qualitative in nature.

5.1 Impacts of Fees Charged to Commercial Waste Generators

Two methods of measuring the impact of waste disposal fees on the industry and the private sector are examined. Both are relatively rudimentary owing on one hand to the relatively small cost of disposal in comparison with the much larger size of the electric generation industry and on the other hand to the difficulties in modeling the impact of increases in energy costs as they are incorporated as inputs to intermediate and final goods and eventually passed, in varying degrees, along to the consumer.

The first method employed reflects the simplest possible assumptions and is identical to the methods utilized in the 1985 high level waste RIA. In this method it is assumed that impacts can be determined by comparing annual cost of the regulation to total industry revenue to determine the change in average electricity rates. Consumers are assumed not to change their purchasing patterns as a result of the increased price and the impact for a given consumer is merely the product of the change in average rates times the number of units purchased.

The second method makes use of more complex economic concepts to derive a more accurate portrayal of the changes in welfare resulting from the imposition of disposal fees. Specifically, empirical estimates of elasticity of demand are employed to model the impact of the fee on the quantity of electricity demanded. This also allows calculation of the impact of the fee on consumer surplus and thus economic welfare.

Three configurations of a HLW storage facility are discussed in this chapter. In the first configuration HLW is buried in tuff with no restrictions regarding leach rate or canister life. The closest observation to this for which there is data on the expected number of health effects lost uses canisters with zero life expectancy and a waste form leach rate of 10^{-3} . Table 4-3 shows that this configuration is associated with 39 statistical

health effects and costs \$4.1 billion in present value terms. Table 4-2 indicates an annualized cost of \$99 million per year. A discount rate of 2 percent was used in both cost calculations.

The second configuration buries waste with a leach rate of 10^{-5} in a 300 year canister . It is the minimum approach for meeting 10 CFR 60. It is associated with 4 health effects and costs \$4.7 billion in present value terms or \$113.8 million per year annualized. The third configuration is the maximum protection modelled. It involves burying waste with a leach rate of 10^{-6} in a 1,000 year canister. It would be associated with approximately 1 health effects in 10,000 years, and cost \$5.1 billion in present value terms or \$125.2 million per year when the costs are annualized. All three configurations meet the requirements of 40 CFR 191.

Reductions in health effects which result from moving from the least costly configuration in tuff to the minimum cost for meeting 10 CFR 60 total 35, a decrease of 90 percent, at an increase in annualized cost of \$14.7 million per year which is an increase of 15 percent. The next step, to the 1000 year canister and leach rate of 10^{-6} will eliminate an additional 3 health effects in 10,000 years, or 73 percent more, at an annual cost of \$11.4 million, a ten percent increase in cost.

5.1.1 Comparison of the incremental regulatory cost to total revenues of the electric utilities

Total revenues of the electric utility industry in the U.S. were approximately \$164 billion in 1988 (EEI89). The \$99.1 million cost of the least stringent option in tuff is 0.06 percent of those total revenues. The more stringent configurations are, respectively \$113.8 million, or 0.07 percent, and \$125.2 million or 0.08 percent of total revenue. Incremental increases in cost are, from the first configuration to the second, \$14.8 million or 0.009 percent, and from the second to the third, \$11.4 million or 0.007 percent of the total revenue.

5.1.2 Application of a model based on elasticity of demand for electricity

This section discusses the features and results of a model of the electric utility industry that computes consumer surplus, welfare cost, and changes in total revenue. The model uses as inputs estimates of the price elasticity of demand for residential, commercial, industrial, and other consumers of electricity (see Table 5-1); the amounts of electricity generated by nuclear and non-nuclear generators (see Table 5-2); quantities of electricity sold (see Table 5-3); and average revenues (see Table 5-4). These data are shown for each state in the U.S. The model

provides all of its results at the state, regional, and national levels. The primary calculations are at the state level.

Measuring changes in consumer and producer surpluses is a widely accepted operational method of evaluating welfare losses due to changes in the price or availability of a product. There are both theoretical and practical considerations in the application of this technique. One theoretical issue in the measurement of consumer surplus concerns the implicit change of the consumer's income as the price of the commodity is changed. In the case of this regulation, the imposition of the waste disposal fee will raise the price of electricity and thus reduce income of electric power consumers. Although the impact of this income change on consumer surplus is an interesting conceptual issue, its impact on the measurement of consumer surplus is negligible. Therefore, implicit changes in consumer income are ignored in this analysis.

A second theoretical issue that is less easily ignored is the question of whether to approach the measurement of consumer surplus in a partial or general equilibrium framework. The partial equilibrium framework assumes that the prices of goods and services remain constant even though there is a second order shift in the demand for them. General equilibrium analysis is more complex because the effects of the initial shift must be traced through the entire economy. Only a partial equilibrium model is considered here due to the small size of the fees compared to the electric power industry and the difficulties in tracing such a widely used intermediate input through the economy.

A third theoretical issue concerns the shape of the demand curve which raises the issue of the size of a perturbation for which the empirical measures of elasticity can be expected to remain valid. Two specifications of the demand curves that might be derived are constant elasticity demand curves and linear demand curves. Constant elasticity demand curves have the same elasticity at all points on the demand curve, and thus, as the quantity demanded gets smaller, the downward slope of these curves gets steeper. On the other hand, linear demand curves have constant slope, but the elasticity of demand is different at each point on the curve. Regardless of the nature of the curve, it must be assumed that the empirically estimated elasticities hold over the area under consideration. Given the small changes in demand to be caused by the imposition of the waste disposal fees, the assumptions on the shape of the demand curves are not likely to cause inaccuracies in the measurements of welfare loss. Constant elasticity demand curves are used in this study.

Perhaps the most crucial assumption utilized in the economic impact model is that of a horizontal supply curve. This assumption is based on two considerations. The first is simply that the change in the quantity demanded caused by the imposition

of waste disposal fees is so small that it will not cause a significant change in the average costs of producing electricity. The second is that there is a fair amount of evidence in the

Table 5-1. Long Run Price Elasticity of Demand by Sector and Region

	Total	Res.	Comm.	Ind.	Other
Total U.S.	-1.05	-0.69	-1.05	-1.40	-1.05
New England	-0.99	-0.62	-1.05	-1.40	-1.05
Connecticut	-0.97	-0.62	-1.05	-1.40	-1.05
Maine	-1.04	-0.62	-1.05	-1.40	-1.05
Massachusetts	-0.98	-0.62	-1.05	-1.40	-1.05
New Hampshire	-1.01	-0.62	-1.05	-1.40	-1.05
Rhode Island	-0.97	-0.62	-1.05	-1.40	-1.05
Vermont	-1.00	-0.62	-1.05	-1.40	-1.05
Middle Atlantic	-1.02	-0.62	-1.05	-1.40	-1.05
New Jersey	-1.00	-0.62	-1.05	-1.40	-1.05
New York	-1.01	-0.62	-1.05	-1.40	-1.05
Pennsylvania	-1.05	-0.62	-1.05	-1.40	-1.05
East North Central	-1.16	-0.93	-1.05	-1.40	-1.05
Illinois	-1.13	-0.93	-1.05	-1.40	-1.05
Indiana	-1.18	-0.93	-1.05	-1.40	-1.05
Michigan	-1.17	-0.93	-1.05	-1.40	-1.05
Ohio	-1.18	-0.93	-1.05	-1.40	-1.05
Wisconsin	-1.14	-0.93	-1.05	-1.40	-1.05
West North Central	-1.12	-0.93	-1.05	-1.40	-1.05
Iowa	-1.14	-0.93	-1.05	-1.40	-1.05
Kansas	-1.11	-0.93	-1.05	-1.40	-1.05
Minnesota	-1.18	-0.93	-1.05	-1.40	-1.05
Missouri	-1.09	-0.93	-1.05	-1.40	-1.05
Nebraska	-1.09	-0.93	-1.05	-1.40	-1.05
North Dakota	-1.10	-0.93	-1.05	-1.40	-1.05
South Dakota	-1.08	-0.93	-1.05	-1.40	-1.05
South Atlantic	-0.98	-0.62	-1.05	-1.40	-1.05
Delaware	-1.04	-0.62	-1.05	-1.40	-1.05
D.C.	-1.09	-0.62	-1.05	-1.40	-1.05
Florida	-0.88	-0.62	-1.05	-1.40	-1.05
Georgia	-1.01	-0.62	-1.05	-1.40	-1.05
Maryland	-1.01	-0.62	-1.05	-1.40	-1.05
North Carolina	-1.01	-0.62	-1.05	-1.40	-1.05
South Carolina	-1.07	-0.62	-1.05	-1.40	-1.05
Virginia	-0.96	-0.62	-1.05	-1.40	-1.05
West Virginia	-1.06	-0.62	-1.05	-1.40	-1.05
East South Central	-1.06	-0.62	-1.05	-1.40	-1.05
Alabama	-1.07	-0.62	-1.05	-1.40	-1.05
Kentucky	-1.09	-0.62	-1.05	-1.40	-1.05
Mississippi	-1.00	-0.62	-1.05	-1.40	-1.05
Tennessee	-1.05	-0.62	-1.05	-1.40	-1.05

Table 5-1. (cont.) Long Run Price Elasticity of Demand by Sector and Region

	Total	Res.	Comm.	Ind.	Other
West South Central	-1.07	-0.74	-1.05	-1.40	-1.05
Arkansas	-1.05	-0.74	-1.05	-1.40	-1.05
Louisiana	-1.08	-0.74	-1.05	-1.40	-1.05
Oklahoma	-1.03	-0.74	-1.05	-1.40	-1.05
Texas	-1.07	-0.74	-1.05	-1.40	-1.05
Mountain	-0.99	-0.50	-1.05	-1.40	-1.05
Arizona	-0.93	-0.50	-1.05	-1.40	-1.05
Colorado	-0.95	-0.50	-1.05	-1.40	-1.05
Idaho	-1.01	-0.50	-1.05	-1.40	-1.05
Montana	-1.08	-0.50	-1.05	-1.40	-1.05
Nevada	-0.97	-0.50	-1.05	-1.40	-1.05
New Mexico	-1.01	-0.50	-1.05	-1.40	-1.05
Utah	-1.02	-0.50	-1.05	-1.40	-1.05
Wyoming	-1.18	-0.50	-1.05	-1.40	-1.05
Pacific	-0.98	-0.50	-1.05	-1.40	-1.05
California	-0.97	-0.50	-1.05	-1.40	-1.05
Oregon	-0.97	-0.50	-1.05	-1.40	-1.05
Washington	-1.03	-0.50	-1.05	-1.40	-1.05
Pacific Noncontiguous	-0.91	-0.22	-1.05	-1.40	-1.05
Alaska	-1.05	-0.93	-1.05	-1.40	-1.05
Hawaii	-1.06	-0.50	-1.05	-1.40	-1.05

Source: RFF84

Table 5-2. Commercial Generation of Electricity in the U.S.

	Total Generation (GWH)	Nuclear Generation (GWH)	% Nuclear Generation
Total U.S.	2,704,250	526,973	19.5
New England	93,383	32,499	34.8
Connecticut	36,379	22,251	61.2
Maine	9,530	5,017	52.6
Massachusetts	34,664	1,117	3.2
New Hampshire	6,997	0	0.0
Rhode Island	764	0	0.0
Vermont	5,049	4,114	81.5
Middle Atlantic	317,806	85,928	27.0
New Jersey	40,160	23,890	59.5
New York	124,714	24,175	19.4
Pennsylvania	152,932	37,862	24.8
East North Central	465,198	106,893	23.0
Illinois	123,340	69,166	56.1
Indiana	83,954	0	0.0
Michigan	88,868	17,808	20.0
Ohio	124,034	8,455	6.8
Wisconsin	45,003	11,464	25.5
West North Central	214,979	37,865	17.6
Iowa	27,729	3,163	11.4
Kansas	31,385	6,650	21.2
Minnesota	40,252	12,288	30.5
Missouri	59,740	8,935	15.0
Nebraska	20,635	6,828	33.1
North Dakota	27,351	0	0.0
South Dakota	7,886	0	0.0
South Atlantic	526,309	144,010	27.4
Delaware	8,953	0	0.0
D.C.	469	0	0.0
Florida	124,053	26,198	21.1
Georgia	82,375	15,149	18.4
Maryland	40,384	11,734	29.1
North Carolina	78,406	29,146	37.2
South Carolina	65,183	40,746	62.5
Virginia	45,172	21,037	46.6
West Virginia	81,314	0	0.0
East South Central	228,905	26,504	11.6
Alabama	67,518	12,981	19.2
Kentucky	76,436	0	0.0
Mississippi	25,093	9,582	38.2
Tennessee	59,858	3,940	6.6

Table 5-2. (cont.) Commercial Generation of Electricity in the U.S.

	Total Generation (GWH)	Nuclear Generation (GWH)	% Nuclear Generation
West South Central	355,752	26,473	7.4
Arkansas	33,764	8,895	26.3
Louisiana	56,774	13,785	24.3
Oklahoma	44,035	0	0.0
Texas	221,180	3,792	1.7
Mountain	239,474	23,600	9.9
Arizona	61,562	22,940	37.3
Colorado	30,895	660	2.1
Idaho	6,746	0	0.0
Montana	24,821	0	0.0
Nevada	20,296	0	0.0
New Mexico	26,369	0	0.0
Utah	29,637	0	0.0
Wyoming	39,147	0	0.0
Pacific	250,622	43,202	17.2
California	125,981	30,863	24.5
Oregon	41,048	6,339	15.4
Washington	83,593	6,000	7.2
Pacific Noncontiguous	11,822	0	0.0
Alaska	4,195	0	0.0
Hawaii	7,627	0	0.0

Source: EBI89

Table 5-3. Sales of Electricity to Ultimate Customers by Sector and Region

	Total (GWH)	Res. (GWH)	Comm. (GWH)	Ind. (GWH)	Other (GWH)
Total U.S.	2,578,062	892,866	699,100	896,498	89,598
New England	102,398	37,069	36,169	27,209	1,949
Connecticut	26,923	10,300	9,961	6,305	356
Maine	11,264	3,904	2,569	4,616	175
Massachusetts	44,727	15,511	17,886	10,243	1,087
New Hampshire	8,848	3,464	1,947	3,339	99
Rhode Island	6,220	2,319	2,354	1,361	186
Vermont	4,416	1,572	1,453	1,345	46
Middle Atlantic	300,646	95,945	99,075	92,290	13,337
New Jersey	62,112	20,656	25,143	15,844	469
New York	125,643	37,460	46,598	30,155	11,430
Pennsylvania	112,891	37,828	27,334	46,291	1,438
East North Central	446,415	136,879	106,280	188,530	14,726
Illinois	110,041	33,980	30,551	37,942	7,568
Indiana	71,675	22,486	15,058	33,474	657
Michigan	82,517	25,316	19,530	36,324	1,346
Ohio	134,316	38,713	29,004	62,238	4,360
Wisconsin	47,866	16,383	12,137	18,552	794
West North Central	183,221	68,909	48,105	61,151	5,055
Iowa	28,838	10,677	6,320	11,025	816
Kansas	25,829	9,121	8,583	7,708	417
Minnesota	45,728	14,996	7,792	22,131	808
Missouri	52,227	21,339	17,534	12,552	802
Nebraska	17,259	6,813	4,996	4,104	1,346
North Dakota	7,107	3,050	1,521	2,070	466
South Dakota	6,235	2,913	1,359	1,562	400
South Atlantic	499,926	199,187	139,545	145,387	15,806
Delaware	7,543	2,533	2,109	2,854	47
D.C.	9,380	1,465	4,793	2,809	312
Florida	130,241	63,972	45,892	16,356	4,021
Georgia	74,346	27,609	19,850	25,984	903
Maryland	47,564	18,483	11,006	17,446	628
North Carolina	85,540	32,212	21,513	30,211	1,604
South Carolina	52,809	17,172	10,790	24,113	734
Virginia	70,115	28,192	18,768	15,690	7,465
West Virginia	22,388	7,549	4,823	9,925	91
East South Central	211,667	75,826	33,457	97,750	4,634
Alabama	56,638	19,641	9,588	26,758	651
Kentucky	54,078	16,811	8,439	26,446	2,382
Mississippi	28,080	11,415	5,928	10,115	622
Tennessee	72,872	27,960	9,502	34,431	980

Table 5-3. (cont.) Sales of Electricity to Ultimate Customers by Sector and Region

	Total (GWH)	Res. (GWH)	Comm. (GWH)	Ind. (GWH)	Other (GWH)
West South Central	346,955	121,809	88,168	124,788	12,189
Arkansas	25,273	9,946	5,783	8,931	613
Louisiana	60,011	20,134	13,088	23,559	3,230
Oklahoma	37,326	14,475	10,100	10,719	2,032
Texas	224,346	77,255	59,197	81,579	6,315
Mountain	150,269	47,309	46,269	49,820	6,870
Arizona	38,916	14,731	12,346	9,261	2,578
Colorado	29,335	9,551	12,683	6,295	806
Idaho	17,165	5,449	4,562	6,807	346
Montana	12,942	3,301	2,620	6,438	582
Nevada	13,684	4,968	3,489	4,685	543
New Mexico	12,755	3,394	4,264	4,032	1,065
Utah	14,507	4,151	4,244	5,321	791
Wyoming	10,964	1,764	2,062	6,980	158
Pacific	324,782	106,181	98,233	105,531	14,838
California	200,637	64,639	70,706	54,988	10,304
Oregon	39,312	14,338	10,605	13,633	736
Washington	84,832	27,203	16,922	36,909	3,798
Pacific Noncontiguous	11,784	3,751	3,799	4,041	194
Alaska	4,065	1,599	1,785	545	136
Hawaii	7,719	2,151	2,014	3,495	58

Source: EBI89

Table 5-4. Average Revenue by Sector and Region

	Total (c/kwh)	Res. (c/kwh)	Comm. (c/kwh)	Ind. (c/kwh)	Other (c/kwh)
Total U.S.	6.35	7.48	7.04	4.70	6.20
New England	7.89	8.79	7.88	6.53	10.09
Connecticut	8.38	9.22	8.32	6.89	11.98
Maine	6.70	8.25	7.09	5.07	9.41
Massachusetts	7.80	8.49	7.66	6.81	9.71
New Hampshire	8.27	9.46	8.64	6.72	11.81
Rhode Island	7.94	8.63	7.67	7.19	8.20
Vermont	8.10	9.13	8.29	6.60	10.70
Middle Atlantic	8.00	9.62	8.75	5.54	7.89
New Jersey	8.51	9.79	8.42	6.77	16.07
New York	8.54	10.46	9.64	4.94	7.30
Pennsylvania	7.12	8.68	7.54	5.52	9.91
East North Central	6.24	7.90	7.04	4.57	6.45
Illinois	7.31	9.74	7.51	5.17	6.39
Indiana	5.62	7.15	6.32	4.24	8.06
Michigan	6.60	7.40	7.63	5.44	8.18
Ohio	5.74	7.59	6.97	4.02	5.81
Wisconsin	5.49	6.66	6.00	4.10	6.25
West North Central	5.98	7.10	6.30	4.45	6.04
Iowa	6.00	7.58	6.56	4.14	6.05
Kansas	6.55	7.83	6.52	5.01	7.43
Minnesota	5.37	6.76	5.94	4.20	6.25
Missouri	6.46	7.33	6.47	4.93	7.12
Nebraska	5.30	5.95	5.43	3.85	5.98
North Dakota	5.75	6.22	6.49	4.91	4.04
South Dakota	5.99	6.69	6.50	4.20	4.52
South Atlantic	6.19	7.24	6.47	4.49	6.14
Delaware	6.66	8.40	6.96	4.84	9.24
D.C.	6.14	6.28	6.62	5.31	5.61
Florida	7.06	7.82	6.74	5.10	6.76
Georgia	6.23	7.02	7.12	4.67	7.33
Maryland	5.84	6.71	6.51	4.43	7.93
North Carolina	6.14	7.47	6.25	4.64	6.43
South Carolina	5.60	7.13	6.20	4.24	5.50
Virginia	5.67	6.58	5.71	4.08	5.50
West Virginia	4.80	5.93	5.44	3.61	8.19
East South Central	5.51	6.03	6.24	4.85	5.87
Alabama	5.53	6.50	6.64	4.42	5.65
Kentucky	5.14	5.66	5.36	4.77	4.72
Mississippi	6.18	6.54	7.12	5.06	8.79
Tennessee	5.52	5.71	6.07	5.17	6.98

Table 5-4. (cont.) Average Revenue by Sector and Region

	Total (c/kwh)	Res. (c/kwh)	Comm. (c/kwh)	Ind. (c/kwh)	Other (c/kwh)
West South Central	5.70	6.96	6.14	4.10	6.17
Arkansas	6.36	7.61	6.51	4.83	6.73
Louisiana	5.85	7.13	6.79	4.18	6.23
Oklahoma	5.51	6.74	5.80	3.66	5.13
Texas	5.61	6.88	6.02	4.05	6.42
Mountain	5.89	7.23	6.49	4.15	5.21
Arizona	7.42	8.75	8.06	5.25	4.47
Colorado	5.90	6.93	5.70	4.52	7.72
Idaho	3.70	4.74	4.12	2.58	3.90
Montana	4.10	5.43	4.75	3.18	3.88
Nevada	5.40	6.01	6.03	4.43	4.23
New Mexico	7.37	9.01	8.24	5.31	6.44
Utah	6.25	7.80	7.02	4.53	5.55
Wyoming	4.35	5.85	5.40	3.64	5.58
Pacific	6.33	6.93	7.46	4.93	4.52
California	7.96	8.54	8.69	6.87	5.11
Oregon	4.31	4.79	4.86	3.35	4.61
Washington	3.42	4.25	3.95	2.62	2.90
Pacific Noncontiguous	8.08	9.18	8.63	6.37	11.90
Alaska	8.98	9.65	8.49	7.45	13.57
Hawaii	7.61	8.83	8.76	6.20	7.98

Source: EEI89

economic literature that long-run average cost curves have relatively large flat sections on their bottoms. That is, the marginal cost of production is constant and equal to long-run average cost through a relatively wide range.

A graphical representation of the economic impact model is provided in Figure 5-1. S_1S_1 is the original supply curve and D_1D_1 is the original demand curve for electricity. An increase of the fee charged to electric utilities for the storage of HLW causes an upward shift in the horizontal supply curve to S_2S_2 . P_1 is the initial price and Q_1 is the initial quantity. The upward shift of the supply curve from S_1S_1 to S_2S_2 causes the price of electricity to increase to P_2 and the quantity demanded to decrease to Q_2 . The points G, H, I, J, K, L, M, and N will be used below in discussing other concepts related to the model including consumer surplus, producer surplus, change in total revenue, and deadweight welfare loss.

Consumer surplus is defined as the total expenditure that consumers would have been willing and able to make in excess of the total expenditure they did in fact make to obtain a given amount of electricity. Graphically the consumer surplus is the area under the demand curve and above the price paid. This area is undefined given that we are assuming constant elasticity of demand. That is, for the consumer surplus to be well defined we would have to put point N where the demand curve crosses the vertical axis, but with constant elasticity demand curves, the demand curve does not intersect the axes. In Figure 5-1 the magnitude of consumer surplus is initially approximated by area LIN. When price increases to P_2 , consumer surplus becomes MKN. Although consumer surplus is poorly defined, the change in consumer surplus is well defined as LIKM. This area is composed of two other areas: JLMK and IJK. Consumer surplus is decreased by this amount when supply shifts upward from S_1S_1 to S_2S_2 .

Producer surplus is defined as the difference between the lowest total revenue for which the firm would have produced a given quantity of the electricity and the total revenue the firm actually received. Graphically it is represented by the area between the price and the supply curve. Because the supply curve is horizontal there is no producer surplus in this model. However, there is a change of revenues that just equals the change in costs. The change in revenues is due to the decrease of revenue attributed to reduced production (GHIJ), and the increase of revenue due to increased price (JLMK). The amount JLMK is transferred from consumers to producers. However the amount IJK is just lost. It is referred to as the "deadweight welfare loss."

Figure 5-1. The Economic Impact Model

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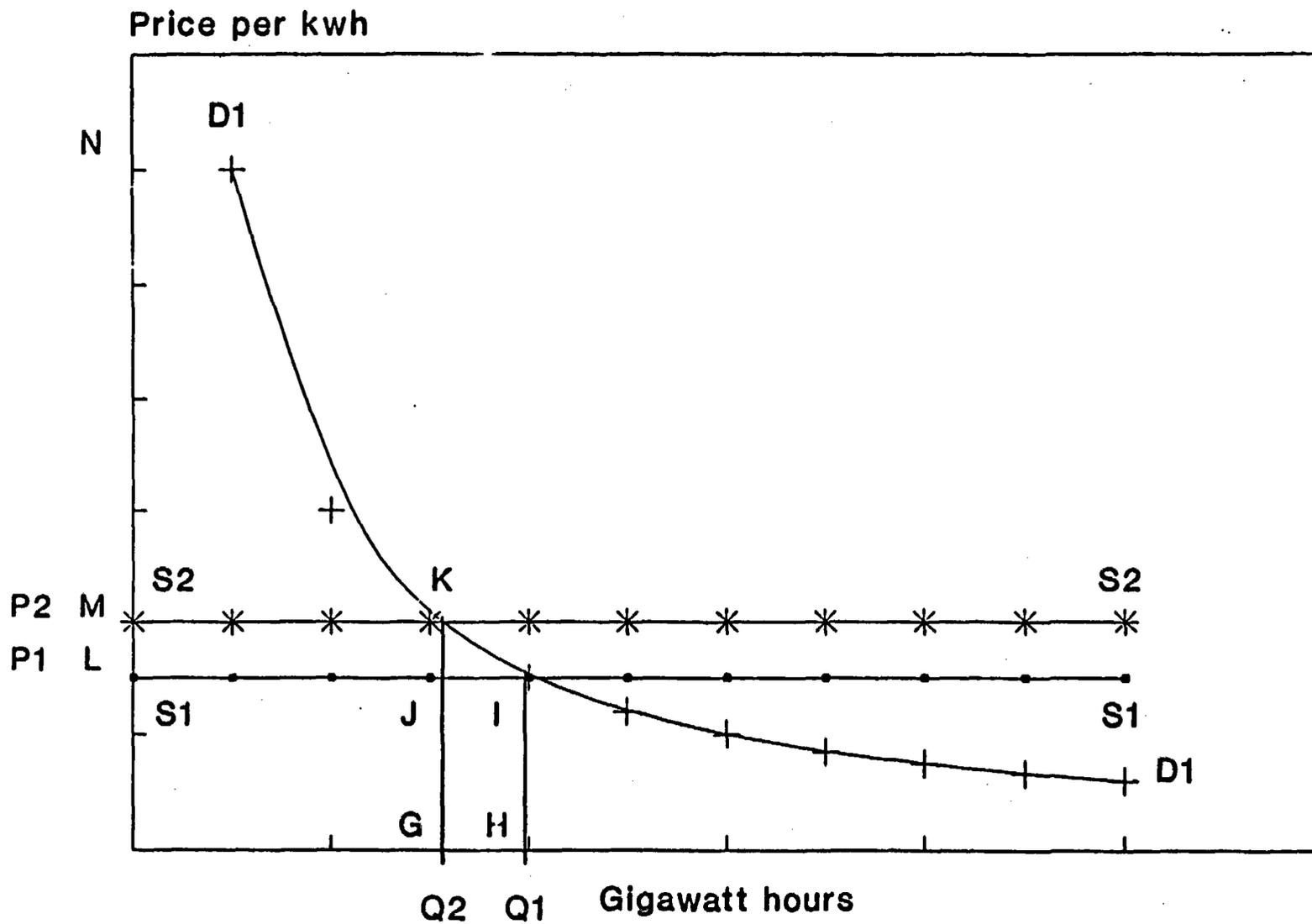


Table 5-5 summarizes the economic impacts of the three configurations and the incremental differences between the first configuration and the second and between the second and the third. There are five categories of impact evaluators. Costs of regulation are presented in millions of dollars and are the same values that were discussed above. These are the initial costs of the regulation that induce the other effects shown in Table 5-5. Costs to nuclear power generators are computed under the assumption that they alone bear all costs of HLW disposal. The incremental impact on rates charged to users of nuclear power of going from configuration 1 to 2 is .00000281 cent/kwh while proceeding to the most stringent option adds another .00000216 cent/kwh. The impact on rates is smaller if spread over all electric power regardless of its source. For the first increment, the electric rate would increase by .000000574 cent/kwh and for the second by .000000442 cent/kwh.

The effects on sales and average revenues are likewise very small. The most stringent option would cost the 107 million consumers (including industrial, residential, commercial, and other consumers) each only \$1.17 per year on average. The annual incremental cost per customer of advancing from the least stringent to the most stringent of these option is about \$0.25. It can be concluded that the customer is unlikely to notice costs of these magnitudes.

Regarding the welfare measures, the annual "deadweight loss", which measures inefficiencies to the economy, is \$50 for the most stringent option. The other welfare measures show that consumers are assumed to shoulder the entire burden. The loss in consumer surplus is just equal to the cost of each configuration. Because producer profits are unchanged, the changes in producer revenues just equal the changes in producer costs.

The impacts of expenditures to implement 40 CFR 191 are not uniform across the various regions of the U.S. Since the costs are borne only by utilities that have nuclear capacity, 18 states will not be directly affected because they have no nuclear capacity as is shown in Table 5-2. The region with the highest percentage of its electric generation provided by nuclear is New England, with 35 percent. The region of the contiguous states with the lowest percent of generation provided by nuclear power is the west south central region with 7.

Table 5-6 shows how these considerations translate into regulatory costs by state. The case shown in Table 5-6 is for an increase from configuration 2 to configuration 3, the total U.S. cost of which is shown to be \$11.4 million in Table 5-5. As is shown in Table 5-6, the annual direct cost of the regulation in states with no nuclear power is zero. There will be no direct economic impact on those states. Annual impacts to other states will vary depending on the proportion of electricity they produce

Table 5-5. Summary of Economic Impacts of Alternative Configurations for HLW Facilities for the Total U.S.

Impact Evaluators	Configuration			Increment	
	1 (See Note 1.)	2 (See Note 2.)	3 (See Note 3.)	2 - 1	3 - 2
COST OF REGULATION					
(\$, MIL)	99.1	113.8	125.2	14.8	11.4
NUCLEAR (CENTS/KWH)	0.00001880	0.00002160	0.00002380	0.00000281	0.00000216
ALL FUELS (CENTS/KWH)	0.00000384	0.00000441	0.00000486	0.00000058	0.00000044
SALES					
BEFORE REGULATION (GWH)	2,578,062	2,578,062	2,578,062	2,578,062	2,578,062
AFTER REGULATION (GWH)	2,578,060	2,578,060	2,578,060	2,578,062	2,578,062
PERCENT CHANGE	-6.33 10 ⁻⁵	-7.27 10 ⁻⁵	-8.00 10 ⁻⁵	-9.46 10 ⁻⁶	-7.29 10 ⁻⁶
AVERAGE REVENUE					
BEFORE REGULATION (CENTS/KWH)	6.35	6.35	6.35	6.35	6.35
AFTER REG. (CENTS/KWH)	6.35	6.35	6.35	6.35	6.35
PERCENT CHANGE	6.05 10 ⁻⁵	6.95 10 ⁻⁵	7.65 10 ⁻⁵	9.04 10 ⁻⁶	6.96 10 ⁻⁶
WELFARE MEASUREMENTS					
DEADWEIGHT LOSS (\$)	-31.38	-41.52	-50.08	-7.00	-4.15
CHANGE IN CONSUMER SURPLUS (\$,MIL.)	-99.10	-113.80	-125.20	-14.80	-11.40
CHANGE IN PRODUCER REVENUE (\$, MIL.)	-4.57	-5.25	-5.78	-0.68	-0.53
OTHER					
NUMBER OF CUSTOMERS	106,827,000	106,827,000	106,827,000	106,827,000	106,827,000
COST OF REGULATION PER CUSTOMER (\$)	0.93	1.07	1.17	0.14	0.11

Note 1: Zero year canister, waste form leach rate of 10⁻³.

Note 2: 300 year canister, waste form leach rate of 10⁻⁵.

Note 3: 1000 year canister, waste form leach rate of 10⁻⁶.

Source: Tables 5-1, 5-2, 5-3, and 5-4

Table 5-6. Incremental Cost of Moving from Configuration 2 to Configuration 3 by Region

	Cost of Regulation (\$)	Cost of Regulation Nuclear (cents/KWH)	Cost of Regulation all Fuels (cents/KWH)
Total U.S.	11,400,000	0.000002	0.000000
New England	703,050	0.000002	0.000001
Connecticut	481,356	0.000002	0.000002
Maine	108,533	0.000002	0.000001
Massachusetts	24,164	0.000002	0.000000
New Hampshire	0	no nuclear	0.000000
Rhode Island	0	no nuclear	0.000000
Vermont	88,998	0.000002	0.000002
Middle Atlantic	1,858,879	0.000002	0.000001
New Jersey	516,812	0.000002	0.000001
New York	522,977	0.000002	0.000000
Pennsylvania	819,068	0.000002	0.000001
East North Central	2,312,415	0.000002	0.000001
Illinois	1,496,267	0.000002	0.000001
Indiana	0	no nuclear	0.000000
Michigan	385,240	0.000002	0.000000
Ohio	182,907	0.000002	0.000000
Wisconsin	248,001	0.000002	0.000001
West North Central	819,133	0.000002	0.000000
Iowa	68,425	0.000002	0.000000
Kansas	143,859	0.000002	0.000001
Minnesota	265,826	0.000002	0.000001
Missouri	193,291	0.000002	0.000000
Nebraska	147,710	0.000002	0.000001
North Dakota	0	no nuclear	0.000000
South Dakota	0	no nuclear	0.000000
South Atlantic	3,115,366	0.000002	0.000001
Delaware	0	no nuclear	0.000000
D.C.	0	no nuclear	0.000000
Florida	566,741	0.000002	0.000000
Georgia	327,718	0.000002	0.000000
Maryland	253,841	0.000002	0.000001
North Carolina	630,515	0.000002	0.000001
South Carolina	881,458	0.000002	0.000002
Virginia	455,093	0.000002	0.000001
West Virginia	0	no nuclear	0.000000

Table 5-6. (cont.) Incremental Cost of Moving from Configuration 2 to Configuration 3 by Region

	Cost of Regulation (\$)	Cost of Regulation Nuclear (cents/KWH)	Cost of Regulation all Fuels (cents/KWH)
East South Central	573,361	0.000002	0.000000
Alabama	280,818	0.000002	0.000000
Kentucky	0	no nuclear	0.000000
Mississippi	207,287	0.000002	0.000001
Tennessee	85,234	0.000002	0.000000
West South Central	572,690	0.000002	0.000000
Arkansas	192,425	0.000002	0.000001
Louisiana	298,211	0.000002	0.000000
Oklahoma	0	no nuclear	0.000000
Texas	82,032	0.000002	0.000000
Mountain	510,538	0.000002	0.000000
Arizona	496,261	0.000002	0.000001
Colorado	14,278	0.000002	0.000000
Idaho	0	no nuclear	0.000000
Montana	0	no nuclear	0.000000
Nevada	0	no nuclear	0.000000
New Mexico	0	no nuclear	0.000000
Utah	0	no nuclear	0.000000
Wyoming	0	no nuclear	0.000000
Pacific	934,588	0.000002	0.000000
California	667,659	0.000002	0.000000
Oregon	137,132	0.000002	0.000000
Washington	129,798	0.000002	0.000000
Pacific Noncontiguous	0	no nuclear	0.000000
Alaska	0	no nuclear	0.000000
Hawaii	0	no nuclear	0.000000

Source: Tables 5-1, 5-2, 5-3, 5-4, 5-5

using nuclear power. The highest cost, \$1.5 million, is borne by Illinois and the lowest for states with nuclear generation, \$14 thousand, by Colorado. When costs are spread over all consumers of electricity generated by all fuels in a state, the highest cost in cents per kilowatt hour (c/kwh) is .000002. This rate increase would apply to Connecticut, Vermont, and South Carolina.

5.2 Impacts of Charges Applied to Defence High Level Waste

The evaluation of the economic impacts of Federal government expenditures to comply with waste disposal are not developed through the use of a quantitative model. Instead a qualitative discussion is provided. A quantitative model is not analytically useful because, given the small size of the waste disposal expenditures relative to the Federal budget, it is not possible to determine the origin of the funds, their impacts, or their opportunity costs.

In terms of the origin of the funds, several broad groups of possibilities exist. One set of possibilities is that the funds are raised by increasing taxation and/or government borrowing. In the absence of a direct tax or bond issuance to fund this specific program, as is the case with the fee charged to utilities to cover their share of waste disposal, it will be impossible to trace the impact in anything more than the most general terms. As a result it will also be impossible to identify the alternative uses to which these funds would have been put in the private sector. Thus the impact as well as the opportunity cost of diverting the funds to public use cannot be accurately quantified. Conversely, there appears to be no reason to assume that the impacts will fall disproportionately on any particular group in society.

A second set of possibilities is that the funds are raised at the expense of other government programs. This could take the form of a slight change in funding over an unspecified group of programs or a larger change in specific programs. It might be argued that DOE nuclear programs would be the most likely victims of such cuts. Again, it will be impossible to determine the exact origin of funds and, thus, the impacts or opportunity costs.

Regardless of the origin of the funds, the most useful way of analyzing the expenditures are as transfer payments, in that the expenditures are generated through taxation or some other means, and spent on the waste disposal program. This transfer of funds by the Federal government is likely to have a positive impact on certain regions and industries. The impacts are the result of fiscal injections and could be measured in terms of the costs for particular goods and services, which are specified in Chapter 3. These costs could also be converted to associated impacts in terms of increased employment and income although this is not

done. The local area chosen for the waste disposal site will most certainly receive increased infusions of Federal dollars. There should be no disproportionate increase, however, in Federal taxes paid by the localities or industries involved.

Beyond the positive benefits accruing to certain localities and industries no discernable patterns of economic impacts should occur. Since expenditures will be derived from general revenues, no negative impacts on particular industry segments and/or small businesses should occur.

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A. APPENDIX

A.1 Commercial Nuclear Power Plants

A.1.1 Types of Wastes Generated at Commercial Nuclear Power Plants

The waste streams resulting from the operations of commercial nuclear power plants can be divided into two broad groups. The first group, spent fuel, refers to the nuclear fuel that has been permanently discharged from a reactor after it has been irradiated. These wastes are generated throughout the operational phase of the power plant. The second group of wastes, which are the result of contamination of equipment and plant during operation, must be managed during the decommissioning of the power plants.

Spent fuels from commercial nuclear power plants originate from either light-water reactors (LWRs) or from one-of-a-kind reactors such as the Fort St. Vrain high-temperature gas cooled reactor (HTGR), now partially shutdown. Most LWR spent fuel assemblies are stored in pools at reactor sites. The remainder were stored at the West Valley Demonstration Project (WVDP) site at West Valley, New York, and at the Midwest Fuel Recovery Plant (MFRP) at Morris, Illinois. However, the WVDP is currently being decommissioned and all of the utility owned spent fuel assemblies have been returned to the utilities. Spent fuels from the one-of-a-kind reactors are currently stored at Hanford and the Idaho National Engineering Laboratory (INEL). Spent fuel from the Fort St. Vrain HTGR is stored at the Idaho Chemical Processing Plant at INEL.

The fuel in reactors is packaged in fuel rods and after 3 or 4 years in use are removed from the reactor. Spent fuel is highly radioactive and generates a lot of heat. Since the assemblies contain unused uranium, fission products, and transuranic elements including plutonium; they are generally considered a form of high level waste (HLW).

It was originally intended that spent fuel would be stored for 6 months in water-filled basins at reactor sites to dissipate thermal heat and allow decay of short-lived fission products. The spent fuel would then be reprocessed and the resultant liquid high-level waste solidified and disposed of in a repository. Since no repository has been developed and no commercial reprocessing is being done, spent fuel has remained in storage at the reactors.

For LWRs the volume of wastes from decommissioning will be significantly less than the volume of wastes produced from normal lifetime operations. The decommissioning of power reactors will result in large volumes of low level wastes but will also produce a small volume of high-activity waste. The timing of this waste generation activity is uncertain. This is due to uncertainty as to when decommissioning operations will be undertaken at each

facility, and as to whether facilities will be either decommissioned upon shutdown or put into safe storage to allow for additional radioactive decay before decommissioning.

The wastes to be managed during LWR decommissioning can be grouped into three major categories. The first, neutron-activated wastes, includes the reactor vessel and its internal components. The second, surface contaminated wastes, includes much of the piping and equipment in the reactor containment and auxiliary control buildings. The third, miscellaneous category, includes a small, but significant, group of materials, such as protective clothing, hand tools, and obsolete equipment.

A.1.2 Industry Operating Characteristics

In 1988 commercial nuclear power generating plants accounted for approximately 95 gigawatts of generating capability, which represents about 14 percent of the total U.S. generating capability of 678 gigawatts. At the same time, these plants provided 527 thousand gigawatts of generation, representing approximately 19 percent of the total generation of 2,704 gigawatt hours (EIA88a). The fact that nuclear plants provide a high percentage of generation, relative to their percentage of capacity, reflects their lower variable operating costs, and their use as baseload generating capacity.

The status of U.S. nuclear generating units as of December 31, 1988 is shown in Table A-1. Of the 123 nuclear generating units, 108 were in commercial operation, 15 were in the construction pipeline and none were on order. Of the 15 plants in the construction pipeline, 3 were in low power testing, 6 were under construction and 6 were indefinitely deferred. Of the six under construction only one was less than 70 percent complete.

As of the end of 1989 the number of operable generating units had increased to 110. However, the total number of plants in all stages of construction and operation had dropped to 121 (MER89).

The overall capacity factor (utilization rate) achieved by U.S. nuclear units in 1989 was 62.3 percent, down slightly from 63.5 percent in 1988. However, the 1988 figure was the highest since 1978, and the capacity factor had not been higher than 58.5 percent since that time (MER89).

Table A-2 presents annual historical nuclear power plant operations data for the years 1973 through 1989. Included is information on the number of units, generating capability, electric generation, percent of electric generation and capacity factors. In terms of units, capability and generation, the history of the nuclear industry can be divided into three periods. The first, which ended in 1978, was typified by a high rate of growth. The second period, which lasted until 1981, was typified by a modest decline that most

Table A-1. Status of U.S. Nuclear Generating Units as of December 31, 1988

Status	Number of Units	Net Summer Capability (GWe)
Operable		
In Commercial Operation	108	95.1
In Power Association	0	0
Total	108	95.1
In Construction Pipeline		
In Low-Power Testing	3	3.2
Under Construction		
More than 90 Percent Complete	3	3.5
70 to 90 Percent Complete	2	2.3
50 to 70 Percent Complete	1	1.1
Indefinitely Deferred	6	7.4
Total	15	17.4
Reactors on Order	0	0
Total	123	112.5

A-3

Source: EIA89

Table A-2.
Nuclear Power Plant Operating Data, 1973-1989

Year	Operable Units (as of December 31)	Net Summer Capability of Operable Units (Million Kilowatts)	Nuclear Electric Generation (kwh, mil.)	Nuclear Portion of Domestic Electric Generation (Percent)	Capacity Factor (Percent)
1973	39	22.6	83.5	4.5	53.7
1974	48	31.8	114.0	6.1	47.9
1975	54	37.2	172.5	9.0	56.0
1976	61	43.7	191.1	9.4	54.9
1977	65	46.2	250.9	11.8	63.4
1978	70	50.7	276.4	12.5	64.7
1979	68	49.6	255.2	11.4	58.5
1980	70	51.7	251.1	11.0	56.4
1981	74	55.9	272.7	11.9	58.4
1982	77	59.9	282.8	12.6	56.7
1983	80	63.0	293.7	12.7	54.4
1984	86	69.7	327.6	13.6	56.3
1985	95	79.4	383.7	15.5	58.0
1986	100	85.2	414.0	16.6	56.9
1987	107	93.6	455.3	17.7	57.4
1988	108	94.7	527.0	19.5	63.5
1989	110	97.9	529.4	19.1	62.3

Source: MBR89

likely resulted from the backlash of the Three-Mile Island accident in 1979. Since 1981 the industry has witnessed slow but steady growth. Projections for the future of the industry are reviewed below. The distribution of nuclear generating capacity is not uniform across the U.S. Figure A-1 presents a map showing the location of commercial nuclear power reactors. The exhibit indicates the concentration of nuclear power plants in the Northeast, the Upper-Midwest, the Mid- and South-Atlantic and the West Coast.

A.1.3 Industry Financial Characteristics

Two aspects of industry financial and ownership characteristics are of interest both in understanding and analyzing impacts on the commercial nuclear power industry. The first concerns the ownership and market shares of the individual utilities. The second concerns the cost of nuclear plants compared to other baseload generating plants and the sensitivity of these comparisons to disposal costs.

One topic of interest is the ownership and market shares in the nuclear power plant industry. The concentration of nuclear power plant ownership is an important policy factor, as the costs of regulatory policy may fall heavily on those utilities and their customers. According to information on principal ownership, provided by the DOE, the 108 commercial nuclear power plants that were operable as of the end of 1988 were held by 56 utilities.

This is only part of the picture, however, as some plants vary in size and some utilities are principal owners of more than one plant. In fact, the ownership of nuclear power industry capacity is relatively concentrated, as approximately 50 percent of the capacity is held by only ten utilities. The nuclear capacities and market shares for these ten utilities are provided in Table A-3. Commonwealth Edison of Illinois is by far the largest utility in terms of nuclear capacity with just over 12 percent of total industry capacity. The next largest utility has only about half as much market share. The Tennessee Valley Authority, which currently has four plants in the construction pipeline, will be almost as large as Commonwealth when, and if, these plants become operable. At that time over 20 percent of the nuclear capacity will be held by two utilities.

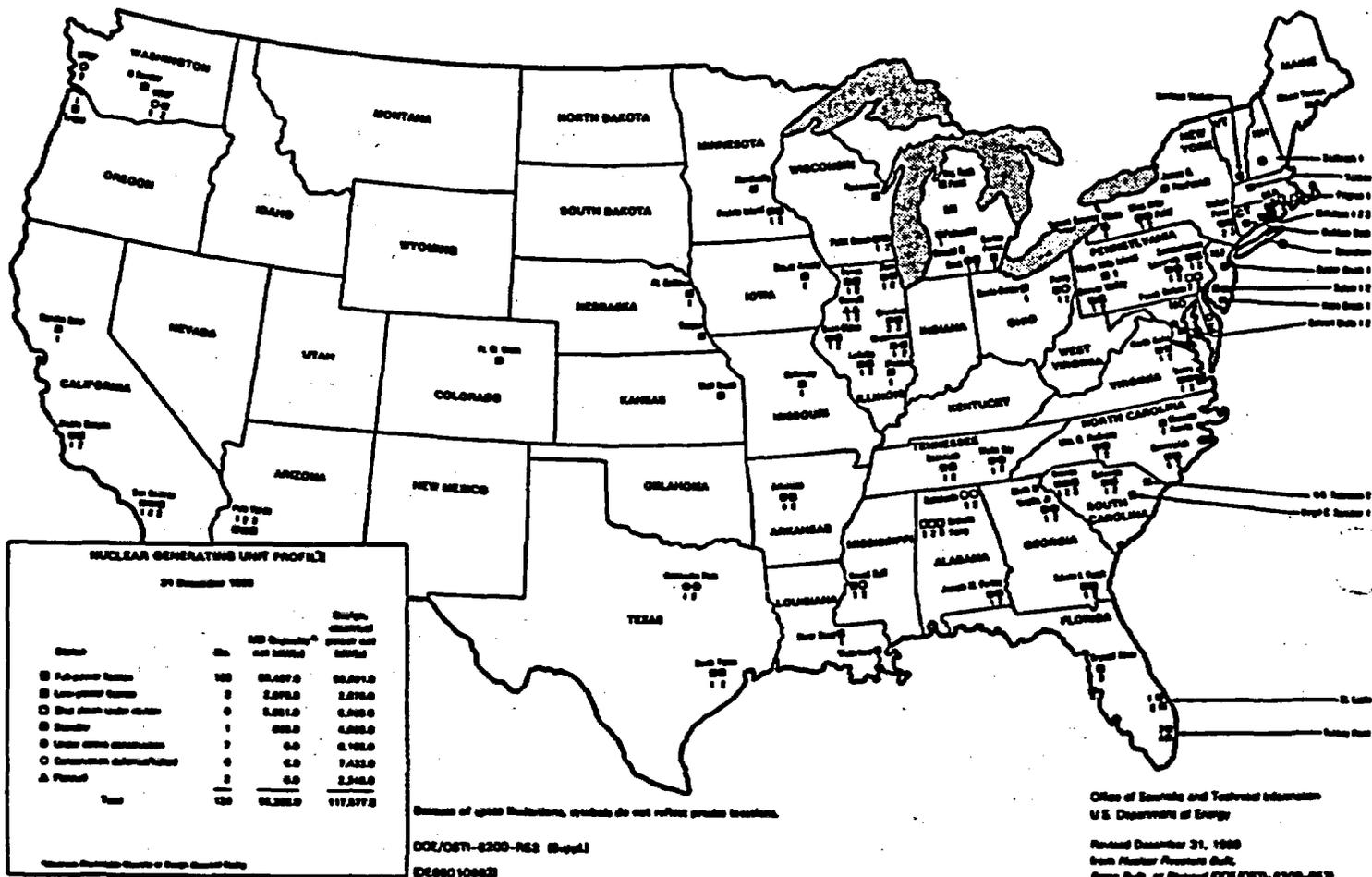
Another topic of financial interest is the relative cost of operating nuclear power plants in comparison with power plants fueled by other means. Since nuclear power plants are relied on for baseload capacity, as opposed to peaking capacity, such comparisons are generally established in comparison with large coal plants which are the main alternative for baseload power. One study of this issue by G.R. Corey (1984), based on comparisons of six large coal and six large nuclear baseload power plants in 1979,

FIGURE A-1:

Locations of Existing and Planned Commercial Reactors as of December 31, 1988

COMMERCIAL NUCLEAR POWER REACTORS IN THE UNITED STATES

31 DECEMBER 1988



Source: ORNL89

Table A-3. Nuclear Capacities and Market Shares for the Top Ten Nuclear Utilities

Utilities	Capacity (Net MWe)	Percent of Total Industry Capacity
Commonwealth Edison 1	11713	12.21
Public Service E& and Philadelphia Electric	6426	6.70
Tennessee Valley Authority	5497	5.73
Duke Power	4880	5.09
Arizona Public Service	3799	3.96
Virginia Electric and Power	3392	3.54
Connecticut L&P	3196	3.33
Caroline P&L	3105	3.24
Florida P&L	3010	3.14
Georgia Power	2603	2.71
Total for Top Ten Utilities	47621	49.66
Total for All Utilities	95901	100

1 These utilities have joint ownership of four generating units and, in addition, each has sole ownership of one unit.

Source: EIA89

found that the nuclear plants provided power for only 16.9 mils per kWh (bus-bar costs per books) compared to 30.2 mils per kWh for the coal plants. Even when the figures were adjusted to a 60 percent capacity factor, as the nuclear units were more heavily used due to their cost-efficiency, the nuclear plants still cost only 16.8 mils per kWh compared to 27.3 mils per kWh for the coal plants. The author concludes that the most likely conclusion is that nuclear power would enjoy a cost advantage in the future in the 15 to 20 percent range.

Moreover, sensitivity analysis on the cost of waste disposal found that doubling of these costs from the original assumption of 2 mils per kWh to 4 mils per kWh would only reduce the advantage by 2 percent. This is not surprising as one author has put the cost of waste fuel management at only four percent of total fuel cycle costs (TOM).

A.1.4 Short Term Industry Prospects

The commercial nuclear power industry is currently in a period of stagnant growth. While twenty-eight nuclear power plants were ordered in 1974, only 14 were ordered in the years 1975-1978, and none since 1978. Of the 14 plants ordered since 1974, 10 were canceled, two were rejected by New York State, and the remaining two have not yet entered the construction phase.

The absence of new orders for nuclear plants over the last decade has important implications for short-term growth in the commercial nuclear power industry. This is due mainly to the long lead times required to construct new nuclear power facilities. For example, for the 38 nuclear units in the construction pipeline in 1984 the total construction lead-time, from the application for a construction permit to commercial operation, is projected by the electric utilities to average 14 years (DOE88).

Given the relatively long period of time that is required between the order and operation of a nuclear power plant, any increases in nuclear power generation over the next decade must originate through operation of plants already in the construction pipeline or through increased capacity utilization at existing plants. Thus DOE forecasts predict nuclear capacity to increase from 93.7 gigawatts at the end of 1987 to between 102.9 gigawatts (lower reference case) and 106.5 gigawatts (upper reference case) (DOE88). Even the optimistic case results in only a 14 percent increase in nuclear capacity over the entire 13 year period.

A.1.5 Long Term Industry Prospects

While the short-term levels of nuclear generating capacities are quite predictable due to the fixed level of capital plant and the long lead-times associated with new construction, forecasts of

long-term capacity are much less predictable. According¹, DOE has constructed three quite different forecasts of future generating capacities through 2020. The pessimistic case, "no new orders," assumes that only plants currently under active construction are ever built, that all units close after their 40 year operating licenses expire, and that if any new orders are placed they will not result in operable capacity through 2020. This scenario results in close to a halving of nuclear capacity to only 51 gigawatts by 2020.

The other two nuclear power generating capacity forecasts modeled by DOE rest on several assumptions more favorable to the industry. These include a satisfactory solution of waste disposal issues, more predictability in licensing, reasonable construction lead-times of 7 years, economic competitiveness with alternate baseload capacity, financial protection through some form of liability coverage, and modest improvements in technology.

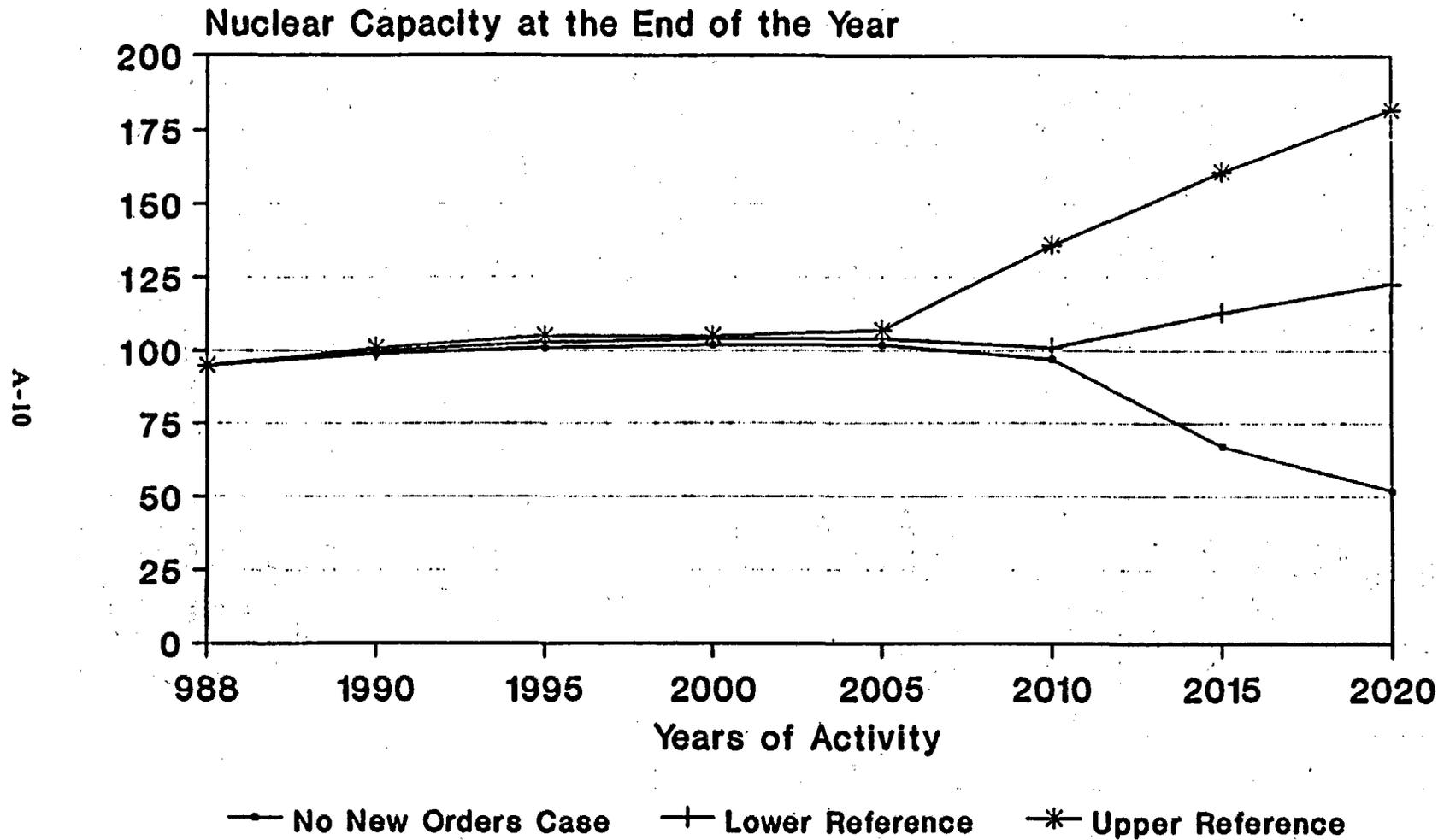
Within these assumptions the lower reference case assumes that only a limited amount of newly ordered nuclear capacity will become operable between 2005 and 2010. Assuming that these new orders are completed successfully, some other utilities will begin ordering new capacity. However, only those utilities with currently successful nuclear programs would replace retiring units with nuclear units.

The upper reference case is based on a number of additional assumptions that are very favorable to the growth of nuclear power. These include: strong growth in electric demand; movement away from coal based on environmental concerns such as acid rain and global warming; high economic growth and labor participation; and high capacity factors for nuclear plants.

The three DOE/EIA long term forecasts for nuclear power are illustrated in Figure A-2. Under the no new orders case capacity will decrease to 52 Net GWe in 2020. Capacity under the lower reference case will be 123 Net GWe, while under the upper reference case it will be 182 Net Gwe.

Comparisons between the DOE forecasts and those made by five other organizations are provided in Table A-4. Both short- and long-term forecasts are considered. Most of the alternative forecasts are only available through the year 2000. In general these estimates are similar too, although slightly higher, than the DOE forecasts. The one forecast available for 2010, developed by Data Resources, Inc. is at the midpoint between DOE's No New Orders and Lower Reference Cases. In the following section, which provides estimates of waste streams that would result from the projections of nuclear capacity and generation, the Upper Reference Case is not considered. This is because these forecasts have not been available in the DOE documents.

Figure A-2: Historical and Projected Domestic Nuclear Capacity, 1982-2020



Source: EIA89

Table A-4. Comparisons of Projections of U.S. Commercial Nuclear Capacity at the end of the year, 1990-2000 (Net Gigawatts - Electric)

Source	Projected Capacity				
	1990	1995	2000	2010	2020
Energy Information Association					
1989 Commercial Nuclear Power					
No New Orders Case	99	101	102	97	52
Lower Reference Case	100	103	104	101	123
Upper Reference Case	101	105	105	136	182
1988 Commercial Nuclear Power					
No New Orders Case	97	102	103	97	51
Lower Reference Case	97	102	103	103	116
Upper Reference Case	103	105	107	138	189
DRI	102	106	106	99	-
NERC	103	106	-	-	-
NAC	101	105	107	-	-
NEI	105	108	108	-	-
NUKEM	103	108	110	-	-

Source: EIA89, p. 16.

This reflects the low probabilities that are associated with this scenario.

A.1.6 Quantities of Wastes Generated at Commercial Nuclear Power Plants

For a full discussion of this matter, see Chapter 2, Section 2.1.1.

A.2 High Level Waste Producer Profiles

A.2.1 Savannah River Site

The Savannah River Site (SRS), located in Aiken, South Carolina, makes Plutonium 239 and Tritium for use in nuclear weapons, and Plutonium 238, which is used as a heat source in thermoelectric generators such as the one on the Voyager spacecraft. The SRS has been a major source of nuclear materials for defense programs and, incidentally, for space, medical, and energy applications. The SRS contains two nuclear fuel reprocessing facilities, three production reactors, and a heavy water rework plant. All three reactors were out of service because of safety concerns as of September 1989, and none is expected to restart before late 1990. The site covers about 300 square miles and its FY 1989 budget was \$1.7 billion, of which \$300 million is for capital projects.

The primary contractor who manages the operation for the Savannah River Site is the Westinghouse Savannah River Company. In all, the site employs about 16,000 people. The Westinghouse company employs 12,000; Bechtel (subcontractor for construction) employs 2,000, the DOE employs 400, and the Wackenhut Corporation (subcontractor for security) employs 1,000. The U.S. Forest Service (which manages the extensive woodlands) and The Savannah River Ecology Lab, run by the University of Georgia, which coincidentally turns out radio-pharmaceutical products, together employ 200 people. The DOE, the Forest Service, and the subcontractors all manage themselves.

In December 1988, the DOE estimated that the Savannah River site had 128,300 cubic meters of high-level nuclear waste (HLW) on-site. The site is projected to have only 49,300 cubic meters of HLW by 2020.

In July 1989, DOE estimated that it would cost \$5.3 billion FY 1988 dollars to retrieve, process, immobilize, and store the HLW until it can be moved to a permanent disposal site: \$3.6 billion of this would be spent processing accumulated waste and \$1.6 billion on wastes generated after immobilization begins (from 1992 until 2020). Construction for reprocessing facilities at Savannah started in 1983, and immobilization is planned to start

in 1992, with all wastes that have been accumulated by the start of immobilization being immobilized by 2008.

As of September 1989, design of the Defense Waste Processing Facility (DWPF), the vitrification facility, was about 99 percent complete, and construction, which began in 1983, was about 96 percent complete. The DWPF is designed to produce a maximum of 410 glass-filled canisters annually, each containing about 165 gallons of vitrified waste. There is also a canister storage building designed to hold 5 years of glass waste production, and another storage building is planned for construction in fiscal year 1993. This will be necessary as the federal waste repository is not scheduled to receive defense waste until 2008 (GAO89).

In September 1989, the total cost of the DWPF was estimated at \$1.25 billion. Annual operating costs for the DWPF are currently estimated to be \$115 million to retrieve, pretreat, and vitrify the HLW.

Savannah River currently keeps its nuclear waste in 45 underground carbon steel waste storage tanks: 22 double-shell and 23 single-shell. Four different waste types have been produced by nuclear fuel reprocessing operations, due to either different nuclear fuels or different reprocessing techniques. In the waste treatment process, all four waste types will be blended and treated to separate the waste into low-level waste, which will be stored on-site, and high-level waste, which will be vitrified into a borosilicate glass waste form and stored on-site to await ultimate disposal in a geologic repository.

A.2.2 Hanford Reservation

The Hanford Site, established in 1943 to produce plutonium for nuclear weapons, occupies 560-square miles in southeastern Washington State. In February 1988, the Department of Energy announced that because of reductions in the estimated need for plutonium, production of plutonium at Hanford's N Reactor would cease. However, several other Hanford facilities will operate until the mid-1990s to process defense materials. Hanford also continues to conduct various other activities, such as research on advanced reactors, the environment and energy, and the management of radioactive waste.

Hanford is a lightwater uranium commercial fuel fabrication facility; its final product is complete fuel assemblies. They do fabrication and scrap recovery, produce commercial light water reactor fuels, and convert UF_6 to UO_2 . Eight of the nine plutonium production reactors are on standby; they have basically closed operations, although they may continue to process left-over spent fuel, and glassify cesium and strontium.

Hanford's current annual budget is \$850 million, but they are projecting this will go up to \$1 billion to cover the cost of cleaning up. The total reservation has between 12,000 and 13,000 employees. Four contractors, the DOE, and Pacific Northwest Laboratory work at the site. The Pacific Northwest Laboratory conducts research on the environment, defense, and developing new forms of technology for energy. Westinghouse Hanford is the primary contractor, which manages the site. Westinghouse Hanford had 8,500 employees in 1989. The Pacific Northwest Laboratory has approximately 2,600 employees on the site, Kaiser Engineers-Hanford has 900 employees, the DOE has 325, and the Hanford Environmental Health Foundation has 110 employees.

In July 1989, DOE estimated it would cost \$2.8 billion (in FY 1988 dollars) to retrieve, process, immobilize, and store the high-level waste until it can be moved to a permanent disposal site. In December 1988, the DOE estimated Hanford had a total of 243,500 cubic meters of high-level waste on-site, which was stored in 149 single-shell and 28 double-shell carbon steel tanks. The single-shell tanks contain about 165,800 cubic meters; the double-shell tanks contain about 77,700 cubic meters. DOE's current plans to immobilize high-level and other tank waste include only the waste in double-shell tanks, which is in four forms: sludge, saltcake, slurry (concentrated liquid waste with suspended solids) and liquid.

Hanford's high-level waste will be vitrified at a facility called the Hanford Waste Vitrification Plant (HWVP), which is not yet built. The HWVP plans to produce about 320 glass-filled canisters annually: at this rate, about six to nine years will be required to vitrify the waste backlog. About thirty five percent of HWVP's second of three design stages was completed in March 1989. Construction is scheduled to begin in July 1991 and radioactive waste is scheduled to be processed beginning in December 1999. The Department of Energy estimates that all accumulated wastes will be immobilized by 2008, and is undecided whether future wastes will be immobilized.

According to the Government Accounting Office, "As of March 1989, DOE's estimate for the construction of HWVP is \$965 million (in year-of-expenditure dollars)". (GAO89) The total cost, including related expenditures (i.e., research and development, environmental and safety design analysis, technical support, etc.), is about \$1.435 billion. The cost could go up, according to DOE, with funding delays, and due to uncertainty about the total volume of waste to be treated and the successful development of a process for treating three of the four waste types. This is a new pretreatment process that will reduce the amount of vitrified waste.

It is not known what types of waste, or how much of each type, the single-shell tanks contain. It is due to the single-shell

tanks that there is a large uncertainty associated with the volume of waste requiring vitrification at Hanford. The decisions on how much of the waste needs to be vitrified is supposed to be made in 2002 when a Record of Decision "will be issued based on a supplemental Environmental Impact Statement on the single-shell tank waste". (GAO89)

In addition to the tank waste, Hanford plans to dispose of 597 strontium and 1,341 cesium high-level waste capsules produced before 1985. Hanford's high-level waste was processed to remove most of the strontium and cesium in order to save tank space. These capsules will be packaged and shipped to the geologic repository without being vitrified (GAO89)

The DOE projects that HLW inventories at Hanford will be 243,300 cubic meters in 2020.

A.2.3 Idaho National Engineering Laboratory

Idaho National Engineering Laboratory (INEL), located near Idaho Falls, Idaho, was established in 1949 as a nuclear reactor testing site. INEL's activities now include nuclear fuels reprocessing, nuclear safety research and waste management, and development of advanced energy concepts. Argonne National Laboratory maintains the country's only breeder reactor at INEL. INEL processes primarily government fuel, but also processes commercial fuel. Their main source of high-level nuclear waste is the Idaho Chemical Processing Plant (ICPP), which reprocesses spent nuclear fuel, primarily from naval nuclear propulsion reactors and reactor-testing programs. A small amount of high-level waste is produced from reprocessing fuel from non-defense research reactors. INEL's annual budget, in 1989 dollars, is \$913 million.

The site employs a total of 10,700 people. Ten percent are employed by the Westinghouse Electric Corporation, thirteen percent by the Westinghouse Idaho Nuclear Company, eight percent by the construction subcontractor Morrison-Knudsen Co., Inc., seventeen percent by the U.S. Navy who maintains a reactor facility, three percent by the DOE, four percent by Rockwell International, six percent by Argonne National Laboratories, three percent by Protection Technologies Incorporation (the guard service), and thirty three percent by EG&G Idaho, Inc., the prime contractor who manages the site.

INEL's construction of high-level waste immobilization facilities will begin in 2002, and immobilization is scheduled to start in 2011. All wastes accumulated by 2002 should be immobilized by 2028, with future wastes processed after that. In July 1989, DOE estimated it would cost \$4.0 billion (in FY 1988 dollars) to retrieve, process, immobilize, and store the high-level waste until it can be moved to a permanent disposal site: \$2.70

billion for accumulated wastes and \$1.30 billion for wastes generated during the period after immobilization begins (from 2011-2037).

As of December 1988, the DOE estimated that INEL had 11,000 cubic meters of HLW. This volume includes about 3,400 cubic meters of calcinated waste and about 7,600 cubic meters of liquid waste awaiting calcination. The DOE projects that INEL will have 17,200 cubic meters of HLW by 2020.

Idaho Chemical Processing Plant waste is converted to calcine form prior to being immobilized, rather than to an alkaline form (as it is at Savannah River, Hanford, and West Valley). Because of this, it does not require the pretreatment steps of separating the waste into high- and low-level components. The calcined waste, however, is not in a form that is suitable for permanent disposal in the geologic repository because it could possibly be readily dispersed or dissolved if it were to come into contact with water.

DOE is currently looking at processes besides vitrification to immobilize the calcined waste for ultimate disposal. According to a DOE document, vitrifying INEL's current and future calcined wastes would produce a volume of immobilized waste greater than that at Savannah River and Hanford combined. Therefore, a major focus for waste form and technology selection at INEL is to minimize the volume of the final product. The preferred waste form for this calcined waste is a glass-ceramic product: a glass-ceramic-filled canister is estimated to accommodate about 2.5 times the volume of waste as a similar-sized glass-filled canister.

The glass-ceramic process is seen as being much less expensive than vitrifying the waste, since fewer canisters will need to be placed in the geologic repository for the former. DOE estimates that INEL could produce about 8,800 glass-ceramic canisters in the first 10 years of its solidification project. Vitrifying the same volume of waste could require as many as 23,000 canisters. According to the Government Accounting Office, "At DOE's 1987 estimated cost of about \$350,000 to transport and place each canister in the repository, vitrifying the waste could cost about \$5 billion more than the glass-ceramic product for just the first 10 years of production." (GAO89)

DOE estimates that if the glass-ceramic waste form and technology are selected, immobilizing the backlog of calcined HLW will be completed in about 2028 (if processing begins, as expected, in 2011). Estimated construction costs of the facility are from \$600 million to \$800 million (GAO89).

A.2.4

The West Valley Site

The West Valley Demonstration Project is located at the Western New York Nuclear Service Center in West Valley, NY. The Center is owned by the state of New York and is the only commercial nuclear fuel reprocessing facility to have operated in the U.S. Between 1966 and 1972 it was run by Nuclear Fuel Services, Incorporated. In 1976, Nuclear Fuel Services activities ended, and in 1980, the DOE was directed by the West Valley Demonstration Project Act (PL 96-368) to solidify the high-level waste at the site and move it to a geologic repository, as no other entity would accept responsibility for the nuclear wastes that remained at the site. The purpose of the project was to demonstrate techniques in solidifying HLW for final disposal, including developing containers for HLW permanent disposal, disposing of the low-level and TRU waste generated during the course of HLW solidification, and decontamination and decommissioning of the tanks, facilities, and hardware used in connection with the project according to NRC regulations. NY State was directed by this act to pay ten percent of the Project's cost and make available 200 acres of the 3,345 acre Center for the demonstration.

DOE assumed operational control of the Project in February 1982. The site includes a chemical reprocessing facility, a spent-fuel receiving and storage area, high-level waste storage tanks, a low-level waste treatment facility, an NRC-licensed land disposal area, and some support facilities.

A total of 806 people are employed at the West Valley Site. 583 are employed by Westinghouse, which runs management and operations, 11 are DOE representatives, 30 are Burke Security guards, 36 are temporary employees, 31 are Dames and Moore (the engineering contractor) employees, 55 are contract workers employed in a variety of tasks, 3 are employed by EG&G, and 2 are employed by the State of New York.

Construction for a West Valley HLW Immobilization Facility began in 1983, and immobilization is scheduled to start in 1992. As of December 1988, 2,130 cubic meters of HLW were stored on site. All accumulated wastes are expected to be immobilized by 2008, when the project is to be finished: the West Valley site is now simply a clean-up operation. It is estimated it will cost \$0.92 billion FY 1988 dollars to retrieve, process, immobilize, and store the high-level waste until it can be moved to a permanent disposal site. The DOE has projected that West Valley will have 210 cubic meters of HLW by 2020.

In 1983, when DOE began constructing the waste immobilization project, it estimated that all the waste would be solidified by 1990 for about \$473 million. These estimates were extended in the DOE January 1989 estimates, but the agency's recently

released 5-year plan for environmental restoration and waste management could reduce the time and money required to complete solidification.

During the facility's 6 years of operations, about 625 metric tons of spent nuclear fuel were reprocessed, generating 0.6 million gallons of HLW. This waste is stored in two single-shell tanks located in underground concrete vaults. One tank is made of carbon steel (which contains about ninety eight percent of the site's total waste) and the other is stainless steel (which contains the other two percent of waste which is liquid and still acidic). The Project's HLW is processed through vitrification. Low-level waste at the Site is solidified into a cement-like product and temporarily stored in drums, pending completion of an environmental impact statement.

DOE estimates that in vitrified form, the Project's high-level waste will fill about 300 stainless-steel canisters. The canisters will be stored on-site, pending shipment to the federal repository: DOE is studying the possibility of shipping these canisters to another federally owned facility for temporary storage as a means of accelerating closure of the Project.

DOE project activities at the West Valley Site were divided into two phases. Phase I consists of decontamination of existing facilities necessary to support the solidification effort, development of canisters to contain the waste for storage in the geologic HLW repository, and solidification of the HLW. Phase II consists of transporting the glass canisters to the repository, decontamination and decommissioning of the facilities used in the Project, and determining the proper disposal methods for low-level and other wastes generated by the Project.

In January 1989, DOE estimated that Phase I could cost approximately \$1.106 billion (year-of-expenditure dollars) and that its cost share will be \$995 million, with New York State providing the remaining \$111 million in funds, services, and credits for the value of the facilities provided by the project. Phase II would cost \$372 million to \$514 million (1988 dollars), and funding delays would probably be the only major factor affecting costs or scheduling. Table A-5. delineates the reasons for the growth of the DOE's cost estimate to clean up the West Valley Project.

A.3 Funding Mechanisms for High-Level Waste Management System

A.3.1 Commercial High-Level Waste

Section 302 of the Nuclear Waste Policy Act of 1982 (Public Law 97-425) establishes a nuclear waste fund for the payment of transportation and disposal costs of high-level waste or spent nuclear fuel. Section 302a authorizes the Secretary of Energy to

enter into contracts with anyone who holds title to or generates such waste and to accept ownership of the waste for a fee.

The fee is currently set at 1.0 mil (one-tenth of a cent) per kilowatt-hour for electricity generated by civilian nuclear power reactors and sold 90 days or more after the 1982 act was enacted (on January 7, 1983). The act also directs the Secretary of Energy to establish a one time fee per kilogram of heavy metal in spent nuclear fuel or in solidified high-level waste that was used to generate electricity in a civilian nuclear power reactor prior the date when the new 1.0 mil per kilowatt-hour fee of the Nuclear Waste Policy Act of 1982 goes into effect. This one-time fee is also equivalent to 1.0 mil per kilowatt-hour. After paying that fee, the previous owner of the spent fuel or solidified HLW will have no further financial obligation to the Federal Government for its long-term storage or permanent disposal. The contract for this one-time fee is described in the Federal Register for April 18, 1983, Article 8a, p.16-602.

The Secretary of Energy is directed to annually review the amount of fees established to evaluate whether they are adequate. This is done by the Office of Civilian Radioactive Waste Management.

Table A-5. West Valley Phase I Cost Growth Between February 1987 and January 1989

<u>Element of Cost Growth Amount</u>	<u>Dollars (in Millions)</u>
Project operating costs incurred during the 7.5-year schedule extension	259
Escalation (price increase) incurred during the 7.5-year schedule extension	159
Underestimate of vitrification system costs (\$60 million) and increased costs due to funding limitations (\$23 million)	83
Contingency fund increase	54
New work (e.g., environmental compliance, physical security upgrades)	51
Underestimate of low-level waste-processing system construction	12
<u>Total</u>	<u>618</u>

(OCRWM), also established in the 1982 act. If these fees are either insufficient or in excess, the Secretary shall propose an adjustment to the fee and submit it to Congress, where either

house will have 90 days to reject it before it automatically goes into effect.

The on-going fee was originally established as 1.0 mil per kilowatt hour based on gross amounts of waste generated by the facility. The commercial utilities, however, sued the DOE, and the DOE has published a Notice of Proposed Rulemaking that changes the basis computation from a fee based on the "net generation" of waste at a facility to a fee based on the "electricity generated and sold" by a facility. In order to determine the total electricity generated and sold by each facility, the DOE has proposed that each facility be required to:

- o Sum each owner's adjustment for "sales to ultimate consumer" and "sales for resale";
- o Multiply this summation by each owner's respective share of the plant, to calculate each owner's weighted energy adjustment factor;
- o Sum the owner's weighted energy adjustment factors to obtain the nuclear power plant's total energy adjustment factor; and
- o Multiply the total energy adjustment factor by the "net electricity generated" as currently reported on the Annex A form submitted to the Department (EXC90).

The Nuclear Waste Fund, administered by the Treasury, will receive all deposits of such fees collected by the Secretary and other appropriations made by Congress or other agencies to take care of nuclear waste. The Secretary, according to Section 302.4(d), may only make expenditures from this fund for radioactive waste disposal activities including:

- "1) the identification, development, licensing, construction, operation, decommissioning, and post-decommissioning maintenance and monitoring of any repository, monitored retrievable storage facility or test and evaluation facility constructed under this Act;
- 2) the conducting of nongeneric research, development, and demonstration activities under this Act;
- 3) the administrative cost of the radioactive waste disposal program;
- 4) any costs that may be incurred by the Secretary in connection with the transportation, treating, or packaging of spent nuclear fuel or high-level radioactive waste to be disposed of in a repository, to

be stored in a monitored, retrievable storage site or to be used in a test and evaluation facility;

5) the costs associated with acquisition, design, modification, replacement, operation, and construction of facilities at a repository site, a monitored retrievable storage site or a test and evaluation facility site and necessary or incident to such repository, monitored retrievable storage facility or test and evaluation facility; and

6) the provision of assistance to States, units of general local government, and Indian tribes under sections 116, 118, and 219." (PL 97-425, Stat. 2259-2260)

A.3.2 Defense High-Level Waste

Defense high-level waste (DHLW) will be accepted for disposal in the geologic repository being developed by the OCRWM. "The defense high-level waste will be accepted by the OCRWM at its storage locations and transported directly to the repository in transportation casks that will be developed by the Civilian Radioactive Waste Management program and whose design will be certified by the Nuclear Regulatory Commission (NRC). The full cost for the transportation and the disposal of this waste, including interest, will be paid by the DOE's Office of Defense Programs". (DOE89a)

According to current plans, none of the DHLW will be shipped to the MRS facility: it will be processed and solidified into a borosilicate glass or ceramic waste-form and go directly to the repository for disposal. According to the DOE's Integrated Data Base Report for 1988, about 17,750 canisters (approximately 8,875 metric tons of heavy metal) of DHLW will be available for geologic disposal by the year 2030. The defense-waste fee will be based on cost estimates that will be annually refined as the program evolves and as, perhaps, quantities of defense HLW increase in the future. If the quantities of defense HLW increase and civilian HLW stays the same, greater portions of fixed program costs will be assigned to the defense waste.

According to the Department of Energy, "The Nuclear Waste Policy Act specifies that no waste, including defense waste, is to be accepted for disposal unless disposal fees have been paid and the DOE deposits, in the Nuclear Waste Fund, an amount of money equivalent to the fees paid by the civilian generators of the spent fuel." (DOE89a) The method used in estimating the fee to be paid for DHLW disposal was published in the August 1987 Federal Register. Each cost account in the analysis is grouped into one of three cost categories:

- 1) Direct costs. "According to this method, the costs of activities carried out solely for the disposal of a specific type of waste--civilian or defense--are directly assigned to the generators of the waste". (DOE89a) A large portion of the program cost is shared by the utilities and the Federal Government.
- 2) Common variable costs. Common costs for facilities and activities used for both defense and civilian waste are apportioned between the two types of waste. These cost-sharing factors are based on parameters that vary with the quantity of waste there is to be disposed of. The common variable costs are allocated to both defense and civilian wastes by applying one of two types of cost-sharing factors a: sharing based on the number of disposal containers (defense waste or spent fuel) accepted (i.e., the "piece count") or b: sharing based on the mined volume (i.e., the "areal dispersion") for the disposal of defense waste or spent fuel.

The "piece-count" factor is defined as the ratio of defense-waste containers to the total number of disposal containers. Case-specific waste logistics and information on disposal-container design are used to calculate this for each repository. The "areal-dispersion" factor is the ratio of the volume of rock excavated in the underground repository for the disposal of defense waste to the total volume excavated for all wastes. Data on the design of the repository, such as the distance between waste-emplacement boreholes and excavation requirements, combined with the waste logistics, were used to compute the areal-dispersion factor for each repository.

- 3) Common unassigned costs. These costs cannot be directly assigned to either type of waste based on the above cost-sharing measures, so they are assigned indirectly, based on the proportion of assigned waste which either the defense or the civilian sector produces.

The allocation of development-and-evaluation (D&E) costs are based on the specific D&E cost accounts. For example, waste-package D&E costs and the MRS D&E costs are assigned directly as civilian-waste costs. Allocating D&E costs in the transportation category is based on the proportion of defense-waste transportation costs to the total transportation costs. Allocation of other repository D&E costs are based on the proportion of the assignable defense-waste repository costs to the total assignable repository costs. Costs for administration by the Federal Government is also based on the proportion of the

total defense-waste assignable costs to the total-system assignable costs.

The May 1989 defense-waste cost estimates increased by \$2.9 billion from the June 1987 estimate, but this is offset by cost decreases of nearly \$0.8 billion that can be attributed to the passage of the 1987 Amendments Act, which lowered D&E costs by limiting the HLW repository to one site to be tested, Yucca Mountain. Thus, there was a net cost increase of nearly \$2.1 billion. In 1988 dollars, DHLW cost for a single repository waste disposal system, with intact fuel at a basic MRS is \$3.6 billion out of a total cost of \$23.8 billion. For a single repository system with the waste consolidated into canisters at an MRS, defense cost is \$3.6 billion out of a total cost of \$24.8 billion.

In a system which has two repositories, with intact fuel at a basic MRS, the defense cost is \$5.4 billion 1988 dollars out of a total cost of \$31.2 billion dollars. For a two repository system where the waste is consolidated into canisters at an MRS, the DHLW cost is \$6.4 billion out of a total \$32.0 billion cost. The upper reference for a two repository system where the waste is consolidated into canisters at an MRS has a DHLW cost of \$6,3 billion out of a total cost of \$32.7 billion.

Table A-6 shows the defense waste share of the HLW management system cost; each cost is grouped into one of the three cost categories.

Table A-6. Cost-Allocation Basis for Defense HLW¹

Development and Evaluation

MRS Facility.	Direct Cost (civilian).
Waste Package.	Direct Cost (civilian).
Transportation and Systems Integration.	Common unassigned cost (based on assignable transportation cost).
Other Repository D&E costs.	Common unassigned cost (based on assignable repository cost).
Government Administration.	Common unassigned cost (based on total assignable cost).

Repository

Management and Integration.	Common unassigned cost (based on assignable repository cost).
Site Preparation.	Common unassigned cost (based on assignable repository cost).
Waste-Handling Building.	Common variable cost (piece count).
Other Waste Handling. Common variable cost (piece count).	Common variable cost (piece count).
Balance of Plant: Facilities for change house, explosives, compressed air and steam, cooling, excavated material, and backfill.	Common variable cost (areal dispersion).
All Other.	Common unassigned cost (based on assignable repository cost).
Surface Shaft Facilities.	Common variable cost.

¹ DOE89a, Table 8-1.

Waste facility.+w	(piece count)
All other.	(areal dispersion)..
Shafts and Ramps	
Underground waste handling.	Common variable cost (piece count).
All other.	Common variable cost (areal dispersion).
Underground excavation.	
Development and Emplacement. Retrieval Operations. Spent-Fuel Facility	Direct cost (civilian waste)
Transport and Emplacement, Defense-Waste Facility.	Direct cost (defense waste)
Transport and Emplacement, Other Waste Facility.	Direct cost (civilian waste)
Transport and Emplacement Boreholes.	Common variable cost (piece count).
Waste Removal.	Common variable cost (piece count).
Backfill Waste-Handling Areas.	Common variable cost (piece count).
All Other Areas.	Common variable cost (areal dispersion).
Underground Service Systems.	Common variable cost (areal dispersion).
Waste-Package Fabrication Spent Fuel, Hardware, Civilian High-Level Waste.	Direct cost (civilian waste)
Defense Waste	Direct cost (defense waste)
All Other	

Common variable cost
(piece count)

Transportation

Civilian Waste	Direct Cost (civilian waste)
Defense Waste	Direct cost (defense waste)

MRS Facility

MRS Facility	Direct cost (civilian waste)
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Benefits

Benefits	Common unassigned cost
Repository	(based on assignable repository cost)
MRS Facility	Direct cost (civilian waste)

A.4. Transuranic Wastes

A.4.1. Transuranic Waste Producer Profiles

A.4.1.1 Hanford Reservation

The Hanford Reservation, which generates high-level nuclear wastes as is described in Section A.2.2, also generates TRU wastes. The waste is of three types: retrievable stored, contact-handled transuranic waste; retrievable stored, remote-handled TRU waste; and buried TRU waste.

Hanford annually produces 96 cubic meters of retrievable stored, contact-handled transuranic waste (TRU) as of December 31, 1988, and has accumulated a total of 9,876 cubic meters. In mass, the annual quantity is 9 kg and the total accumulated amount is 436 kg. In terms of radioactivity, the annual waste stream contains 100 Ci of radioactivity and the total stored amount contains 1,000 Ci.

Contact-handled TRU waste is stored in the TRU Waste Storage and Assay Facility, which has a high limit on the type of waste that it can store, and the Central Waste Complex, which has much

stricter DOE plutonium-content storage limits, since the Complex design consists only of several light-weight aluminum buildings. The Central Waste Complex is limited to 520 plutonium-equivalent curies of waste.

As of April 11, 1990 the TRU Waste Storage and Assay Facility contained approximately 252 cubic meters of TRU waste with a remaining storage capacity of approximately 83 cubic meters. The Central Waste Complex had an inventory of approximately 127 plutonium-equivalent curies of waste with a remaining storage capacity of approximately 393 plutonium-equivalent curies of waste.

With respect to retrievable stored, remote-handled TRU waste, Hanford annually generates 0.2 cubic meters and their current total is 137 cubic meters. This translates into 0.1 kg, annually, and 11 kg, total, with no radioactivity detected. Hanford has 109,000 cubic meters of buried TRU waste. This equals 346 kgs, and 600 Ci of radioactivity (ORNL89).

About 235 cubic meters of this waste has been stored in a third facility, the Alpha Caisson, which is now full. The remaining 113 cubic meters of waste is retrievably stored in drums and boxes in shallow trenches covered with dirt. All newly generated remote-handled TRU waste will be placed in highly shielded drums so that it can be treated as contact-handled waste and stored in the Central Waste Complex (GA090).

According to the GAO, "A DOE Richland official estimates that, if the WIPP does not open, Hanford's existing TRU waste storage capacity will be reached during the first half of calendar year 1991. According to Hanford officials, when it is apparent that storage space will be exceeded, Hanford could construct additional storage facilities as needed". (GA090)

A Waste Receiving and Processing facility is scheduled to be constructed during the 1990s. Processing and packaging capabilities for TRU waste in retrievable storage would also be provided in the Waste Receiving and Processing facility. Retrievably stored and newly generated waste would be inspected and certified to meet the WIPP's Waste Acceptance Criteria. (FEIS80) As of December 31, 1987, there was an estimated 31,960 cubic meters of TRU waste-contaminated soil at the Hanford site.

A.4.1.2 Idaho National Engineering Laboratory

The Idaho National Engineering Laboratory (INEL) was established in 1949 as a nuclear reactor testing site. INEL's activities now include nuclear fuels reprocessing, nuclear safety research and waste management, and development of advanced energy concepts. Argonne National Laboratory maintains the country's only breeder reactor at INEL. INEL mostly works with government fuel, but

also with some commercial fuel. They have an annual budget, in 1989 dollars, of \$913 million.

The site employs a total of 10,700 people. Ten percent are employed by the Westinghouse Electric Corporation, thirteen percent by the Westinghouse Idaho Nuclear Company, eight percent by the construction subcontractor Morrison-Knudsen Co., Inc., seventeen percent by the U.S. Navy who maintains a reactor facility, three percent by the DOE, four percent by Rockwell International, six percent by Argonne National Laboratories, three percent by Protection Technologies Incorporation (the guard service), and thirty three percent by EG&G Idaho, Inc., the prime contractor who manages the site.

Up until 1989, when the State of Idaho rescinded the order, the INEL was receiving TRU wastes from the Mound Facility, Bettis Laboratory, Argonne Laboratory, and the Rocky Flats Plant for interim storage. As of December 31, 1988, there was 1,065 cubic meters, annually, and 36,640 cubic meters, total, of retrievably stored, contact-handled TRU waste at INEL. In mass, this equalled 50.7 kg, annually, and 736 kg, total, and in radioactivity, this was 125,600 Ci, annually, and 877,000 Ci, total.

Contact-handled TRU waste at INEL constitutes 99 percent of its current inventory of TRU waste. As of March 28, 1990 INEL had 56 cubic meters of retrievably stored, remote-handled TRU waste, and an annual average remote-handled TRU generation rate of 1.133 cubic meters. At this rate, INEL could have adequate remote-handled storage space until the year 2060. Remote-handled waste is stored underground in steel pipe vaults at the Intermediate Level TRU Waste Storage Facility, which has a total storage capacity of 136 cubic meters.

As of December 31, 1989, buried TRU waste at INEL consisted of 57,100 cubic meters, with a mass of 357 kg, and no radioactivity measurable. (ORN89) As of December 31, 1987, there was an estimated 56,000-156,000 cubic meters of TRU waste-contaminated soil at INEL.

Since 1970, contact-handled TRU waste received at the Radioactive Waste Management Complex (RWMC) has been stored at the 56-acre Transuranic Storage Area (TSA), a controlled area surrounded by a security fence. Approximately 2.3 million cubic feet of TRU waste is currently stored at the TSA. However, about half of it is being reclassified as low-level waste.

This TRU waste is in interim 20-year retrievable storage. The RWMC includes (1) 2 fabric-covered buildings, (2) the Intermediate Level TRU Storage Facility for remote-handled TRU waste, and (3) 2 earth-covered storage area pads. As of February 1990, the contact-handled waste inventory was approximately

64,755 cubic meters: 13,035 in the two fabric covered buildings and 51,720 in the earth-covered storage pads.

The remaining storage capacity cannot be determined until EPA and INEL reach an agreement on a stacking configuration for the drums on the storage pads. EPA has contended, since January 1990, that INEL's current placement of these drums violates RCRA requirements since the stacking does not provide adequate aisle space to allow proper movement of inspecting personnel and emergency equipment between the drums. INEL is currently putting forward another stacking configuration plan, after the plan they proposed in May 1990 was rejected by the EPA.

If INEL is allowed to adopt a modified dense-pack configuration such as the one it proposed to the EPA in May 1990 (stacking 20 drums wide, 20-25 drums deep, and 5 drums high), "INEL would have an estimated remaining capacity of about 1,900 to 2,000 cubic meters. At the current on-site generation rate of about 5 cubic meters a year for contact-handled TRU waste..., INEL officials said that storage capacity should be adequate for hundreds of years," according to the GAO. (GAO90)

INEL says that if they are required to store TRU waste "under the worst-case RCRA stacking configuration (stacking 2 drums wide, 2 drums deep, and 3 drums high), there would not be enough physical storage space in existing facilities to accommodate the contact-handled TRU waste already in storage. According to INEL estimates, if all retrievably stored waste must be restacked using the modified dense-pack configuration, 20 new storage modules would be needed. However, if the worst case RCRA spacing configuration must be used, INEL estimates that 31 modules will be required." Each module is expected to cost between \$3.5 million and \$6.5 million. (GAO90)

A retrieval building is currently under conceptual design. Facilities are also being conceptually designed to provide for the storage, treatment, and repackaging of the retrieved waste to meet the WIPP Waste Acceptance Criteria (WAC). Treatment processes under construction include size reduction using mechanical plasma arc cutting to size-reduce metallic components, immobilization to stabilize free liquids or respirable/dispersible particulates, and shredding/compaction to shred and repackage boxed waste.

These facilities would be designed to ensure two levels of containment (in addition to the waste container) for all waste processing and repackaging areas. All air removed by the ventilation systems would pass through appropriate HEPA filtration systems for an estimated overall decontamination factor of 1,000.

In addition, a potential treatment facility, known as the Process Experimental Pilot Plant (PREPP), is being developed at the INEL. The PREPP is a research and development treatment facility designed to demonstrate the efficacy of a process to certify certain TRU waste. Eventually the PREPP treatment technology may be used in a production facility to certify (to the WIPP Waste Acceptance Criteria) a limited volume of TRU waste in retrievable storage." (FEIS80)

A.4.1.3 Los Alamos National Laboratory

The Los Alamos National Laboratory (LANL), located near Santa Fe, New Mexico, is operated by the University of California. Nuclear weapon technology is the laboratory's primary focus, and accounts for about half its effort, but their work has expanded from designing nuclear weapons to include non-nuclear defense programs, such as developing a neutral particle beam for the Strategic Defense Initiative, and non-defense programs, including contributions in energy technologies, computational science, public health and environmental protection. Sixty percent of weapons in the national nuclear stockpile were designed at Los Alamos, and they have responsibilities for nuclear warheads from the research phase through retirement. In this vein, they conduct nuclear tests at the Nevada Test Site.

The Laboratory also provides research and technical support for other nuclear weapons production sites. They have worked on technology to be transferred to the Rocky Flats Plant to improve ion exchange processing for recovery and purification of plutonium and other scrap materials, and they have developed in situ pyrochemical processing of salts used in the direct oxide reduction of plutonium oxide to metal, reducing waste and improving efficiency. There is one small research reactor at LANL. The Laboratory also works to increase the safety of nuclear waste storage and of civilian nuclear power. They are helping to characterize a possible location for a high-level nuclear waste repository in Nevada, and have created computer programs that predict and analyze reactor behavior during simulated accidents.

Los Alamos National Laboratory employs approximately 7,600 people: combined with its two major contractors, the complex employs approximately a total of 10,000. The FY 1990 operating budget was \$917 million (including capital).

According to the *Final Environmental Impact Statement* for WIPP, "Since 1971, Transuranic (TRU) waste has been packaged and stored at Los Alamos in either subsurface trenches or aboveground earth berms at the waste burial site. Two types of packaging have generally been used, 55-gal steel drums (sealed and coated with bituminous corrosion protection material) and plywood crates (sealed and coated with fiberglass-reinforced polyester).

The TRU waste facilities at Los Alamos include:

Existing Storage Facilities

TRU Waste Size Reduction Facility (expected to result in four-to-one reduction in the waste volume)

TRU Contaminated Solid Waste Treatment and Development Facility

TRU Waste Preparation Facility

TRU Waste Nondestructive Analysis and Examination Facility

TRU Waste Transportation Facility

TRU Waste Corrugated Metal Pipe Saw-Processing Facilities

Other Related Facilities: Liquid Waste Treatment Plant." (FEIS80)

The GAO states, "As of January 31, 1990, LANL had an inventory of 7,365.5 cubic meters of uncertified contact-handled TRU waste. With a current generation rate of approximately 8.8 cubic meters of waste annually and 580.5 cubic meters of remaining capacity, LANL should have storage space for its uncertified TRU waste until 2056." (GAO90) These estimates are conservative, as they do not take into account the storage savings to be achieved through use of the Size Reduction Facility, which is now in use. On the other hand they note, "However, if LANL was required to restack the waste containers for RCRA spacing requirements, rather than the current dense-pack configuration, LANL could reach storage capacity by the year 2008." (GAO90)

There is less unused capacity for certified contact-handled TRU waste. GAO states, "The total inventory of certified contact-handled TRU waste at LANL, as of January 13, 1990, was 304.2 cubic meters. With a current generation rate of about 191.2 cubic meters of waste per year, according to a LANL estimate, and about 543.9 cubic meters of remaining capacity, LANL should have space for its certified TRU waste until about November 1992." (GAO90)

The amount of retrievably stored, remote-handled TRU waste at LANL by January 1, 1989 was 11 cubic meters total, with none being generated annually as of that date, with no radioactivity and only 1 kg of mass. Buried TRU at LANL, as of the same date, consisted of 53 kgs, or 14,000 square meters, with 9,000 Ci radioactivity. (ORNLS9)

Approximately 5.4 cubic meters of remote-handled TRU waste will be generated through 1991, and no additional capacity is needed

for remote-handled TRU waste after this date, since LANL does not anticipate generating any additional remote-handled TRU waste that would require on-site storage. (GAO90)

It is estimated that approximately 40% of LANL's TRU waste is also mixed with hazardous waste. As of December 31, 1987, there was an estimated 1,140 cubic meters of TRU waste-contaminated soil at LANL.

A.4.1.4 Oak Ridge National Laboratory

Oak Ridge National Laboratory (ORNL) in Tennessee researches and develops energy conservation, fusion, fission, fossil and energy technologies. ORNL also conducts basic physical and life science research. Their primary source of nuclear waste comes from the research reactors division. There are four reactors, two of which are currently in operation and one of which is being worked on to bring it back into operation. One of the operating reactors is used in shielding studies, and the other is the highest neutron flux reactor in the world: it produces TRU elements and isotopes for medical use. Nuclear materials are also used throughout the laboratory in different types of scientific research performed there.

As of December 1989, there were 5,450 employees at ORNL; 3,005 employees of the Oak Ridge Gaseous Diffusion Plant; 6,924 at the Oak Ridge Y12 Plant; 1,403 employees at the Paducah Gaseous Diffusion Plant in Kentucky; and 2,356 employees at the Portsmouth Gaseous Diffusion Plant in Ohio.

As of October 3, 1989 ORNL generated 138 cubic meters annually of retrievably stored, contact-handled TRU waste at Oak Ridge, and had accumulated 510 cubic meters total. In terms of mass, this was 1.8 kg, annually, and 27 kg, total, while radioactivity was 9,700 Ci, annually, and 668,000 Ci, total. Contact-handled waste constitutes 70 percent of the solid TRU waste on this site.

ORNL currently stores contact-handled TRU waste in two buildings which are not in compliance with RCRA or DOE storage facility requirements and must be vacated by November 1992. Construction on a new storage facility, which will provide storage space for approximately 594 cubic meters of contact-handled TRU waste, is scheduled to begin in October 1991. ORNL projects that the new facility will handle already stored and newly generate contact-handled TRU waste through December 1995.

In terms of retrievably stored, remote-handled waste, there was no annual figure, but there was 1,304 cubic meters total, which equaled 106 kgs in mass, or 1,537,000 Ci in radioactivity. There was also a total of 6,200 cubic meters of buried TRU waste, which was 6 kgs in mass and 14,000 Ci in radioactivity. (ORNL89) This

waste is stored in concrete casks in a bunker-type facility or in shallow trenches.

However, the shallow trenches did not meet state or federal requirements and must be vacated by November 1992. ORNL plans to construct two new remote-handled TRU waste storage facilities, one starting in August 1990 and the other in July 1992, to replace the shallow trenches and provide additional storage capacity for newly generated waste. These facilities will add a remaining remote-handled TRU waste storage capacity of approximately 88 casks. At a generation rate of approximately 4.59 cubic meters a year, ORNL will not reach its remote-handled TRU waste storage capacity until June 2004.

As of December 31, 1987, there was an estimated 12,000 to 60,000 cubic meters of TRU waste-contaminated soil at Oak Ridge. Almost all of the ORNL TRU waste is considered mixed waste under current EPA guidelines.

A.4.1.5 Savannah River Site

The Savannah River Site, which generates high-level nuclear wastes as is described in Section A.2.1, also generates transuranic wastes.

As of December 31, 1988, 275 cubic meters of retrievably stored, contact-handled TRU were generated at the Savannah River Site annually and a total of 3,297 cubic meters had been accumulated. In terms of mass, the annual amount was 11.3 kg and the total was 192 kg, with radioactive levels of 147,000 Ci (annual) and 1,027,000 Ci (total). No retrievably stored, remote-handled TRU waste was generated or stored at the site, but there were 4534 cubic meters of buried TRU waste, which is a mass of 9 kg, and contains 38,000 Ci. As of December 31, 1987, there was an estimated 38,000 cubic meters of soil contaminated with TRU wastes at the Savannah River Plant. (ORNL89)

Stored TRU waste at the Savannah River Site is in retrievable storage on concrete pads or buried in shallow trenches. It is contained in concrete and steel boxes, concrete culverts, and galvanized steel drums covered with 4 ft. of earth or tornado netting, which has been in use since 1985.

As of May 1990, TRU waste stored at the SRS was equivalent to approximately 12.5 of the 13 existing RCRA-permitted storage pads for solid waste, and TRU waste temporarily stored at the generators awaiting completion of the pads was equivalent to .5 of the storage pads. Four new RCRA-permitted storage pads are to be completed in June, 1990. Therefore, as of June 1990, SRS is to have four pads for storing newly-generated TRU waste, excluding waste generated since May 8, 1990. Using SRS's projected annual waste generation rate of approximately 992 cubic

meters of mixed and TRU waste and approximately 857 cubic meters of large, bulky TRU waste, the remaining storage capacity at SRS is approximately 3,900 cubic meters.

The GAO estimates that capacity could be reached at the SRS's existing and soon-to-be completed TRU storage pads by June 1992. The storage capacity could significantly deteriorate if the South Carolina Department of Health and Environmental Control (SCDHEC) rescinds its waiver on requiring SRS to maintain aisles between the storage containers that allow for unobstructed movement of equipment to any area of the facility. The waiver was granted on condition that the stored drums contained no free liquids, but rainwater has seeped into the drums. Despite very slow movement on the part of the SRS to remove this water from the drums, the SCDHEC has not taken any action against the site, and will continue its waiver on the condition that the SRS is making a "good faith" effort to removed the rainwater. (GAO90)

A.4.1.6 Nevada Test Site

The Nevada Test Site (NTS) tests atomic weapons. NTS also has a gaseous spill facility. Commercial petroleum and gas companies use the gaseous spill facility 6 months out of the year to test the solvent efficiency of gaseous spills. They pay the government for this use.

NTS stores some very low-level nuclear waste, most of which comes from the Rocky Flats Plant and also from the Mound Plant. This waste is primarily composed of contaminated instruments, etc, and will eventually be placed in WIPP, according to current plans. NTS also stores low-level TRU waste from Lawrence Livermore National Laboratory (LLNL). LLNL is the only facility currently approved to ship TRU waste to NTS for storage.

The total site employment level at the end of January 1990 was 4,823. This includes 378 DOE Nevada personnel, 34 on-site DOE personnel, 3,434 RICO (Management and Operations Subcontractor) employees, 116 people who subcontract to RICO, 341 Holmes and Narver employees (who also do Management and Operations), 332 Wackenhut Security guards, and 98 Fenix and Scisson, Inc., employees (Management and Operations), as well as 57 Fenix and Scisson subcontractors. The Nevada Test Site area in the past included Yucca Mountain. Their current operating budget is \$922 million for the site and mountain together; this figure includes capital.

As of December 31, 1988, the Nevada test site had 10 annual cubic meters and 596 total cubic meters of retrievably stored, contact-handled TRU waste. In mass, this was 0.2 kg, annually, and 4 kg, total, with 400 Ci of radioactivity, annually, and 2,800 Ci, total. There was no retrievably stored, remote-handled or buried TRU waste at the site. (ORNL89)

NTS currently stores non-mixed TRU waste in 55-gallon drums and boxes in metal sea-land cargo containers on a pad built to RCRA specifications. NTS facilities have a TRU waste storage capacity of between 1,890 and 3,150 cubic meters, depending on the type of packaging used. As of October 31, 1989, the site had room for an additional 1,440 to 2,400 cubic meters of TRU waste. Again, this figures depends on the type of waste packaging used.

LLNL estimates it will ship approximately 21.2 cubic meters of TRU waste to the NTS annually. At this rate, if the WIPP does not open, the NTS will not exceed its current TRU waste storage capacity for approximately 100 years.

A.4.1.7 Rocky Flats Plant

The Rocky Flats Plant, located near Denver, Colorado, produces nuclear weapons. This process produces low-level transuranic waste, and the very low-level contaminated instruments to be disposed of that are sent to be stored at the Nevada Test Site. EG&G is the management and operations subcontractor at Rocky Flats: there are between 5,400 and 5,500 EG&G employees as of Spring 1990, and this number is expected to grow to 6,000 by the end of 1990. As well, there are 150 DOE employees, and 300-350 employees of J.A. Jones Construction Services.

The *Final Environmental Impact Statement* for WIPP states, "The Rocky Flats Plant Supercompaction and Repackaging Facility and TRU Waste Shredder are being proposed to process solid waste which is newly generated during routine production operations, maintenance activities, and laboratory support operations at the Rocky Flats Plant. Waste may be processed in permitted storage if appropriate. The Colorado Department of Health currently recognizes eight permitted storage areas at the Rocky Flats Plant for TRU mixed waste. The areas differ in size for a total permitted storage capacity of 1,601 cubic yards." (FEIS80)

Because the Rocky Flats Plant is expected to generate 1,188 cubic meters of TRU waste per year, it could reach its permitted storage capacity in FY 1992 -- even with the installation and operation of a waste compactor -- unless steps are taken to find alternative storage. Up until 1989, the Rocky Flats Plant was sending its TRU waste to the Idaho National Engineering Laboratory for interim storage. This practice stopped when the Idaho governor declared that only TRU waste produced in Idaho could be stored in Idaho.

A DOE task force is currently exploring options for temporarily storing mixed TRU waste from the Rocky Flats Plant if the plant's on-site capacity is reached. Options include sending the waste to WIPP during the test program, storing the waste at DOE interim storage sites, DOD sites, or newly created commercial TRU waste storage sites. These options will be pursued even if the WIPP

goes into operation, because the WIPP will only take a small volume of Rocky Flats waste during WIPP's early years of operation.

A.4.1.8 Argonne National Laboratory - East

Argonne National Laboratory, located in Argonne, Illinois, is operated by the University of Chicago; they have about 3900 employees, and a fiscal year 1989 budget of 325 million dollars. There are no reactors on site, but Argonne National Lab West, located at the Idaho National Engineering Lab, has the country's only breeder reactor.

Argonne's main work with radioactive materials is test tube work on transuranic elements; they also use accelerators. Argonne is a multidisciplinary laboratory which has programs in all fields, and almost all of the programs in the Department of Energy have some funding there. Their transuranic waste, comprising 7.4 cubic meters of contact-handled waste and 3.6 cubic meters of remote-handled waste, will be sent to WIPP once it becomes operational.

A.4.1.9 Lawrence Livermore National Laboratory

Lawrence Livermore National Laboratory (LLNL), located in Livermore, CA, is a research and development facility operated by the University of California which focuses on energy, biomedical, and environmental projects. LLNL has 8,300 career employees and about 2,000 current contract employees: however, it was announced in December 1989 that 500 of these contract employees will be laid off. Their fiscal year 1989 budget was \$951 million.

LLNL does not produce nuclear materials on site: there are no reactors. There is also no reprocessing done on-site: most of their 59 cubic meters of transuranic waste is sent to the Nevada Test Site to be stored temporarily until a permanent location such as WIPP can be found. LLNL estimates that they ship 100 55-gallon drums, or 21.2 cubic meters, of TRU waste a year from their site to the Nevada Test Site. A small amount of transuranic waste is also stored on site at LLNL: this is not reprocessed, just recycled.

A.4.1.10 Mound Plant

The Mound Plant, located in Miamisburg, Ohio, produces nuclear weapons components, and also assembles and tests radioisotopic thermoelectric generators. The plant receives tritium from off-site, but does not produce nuclear materials on-site. Most of their nuclear waste comes from the decommissioning and decontamination of their own plutonium 238 processing plants, which stopped operating in 1979.

The Mound Plant generally sends their extremely low-level waste to the Nevada Test Site (NTS), but stores most of their TRU waste on-site. The plant was sending its TRU waste to the Idaho National Engineering Laboratory for interim storage, until 1989, when the Idaho governor declared that only TRU waste produced in Idaho could be stored in Idaho. Recently, the Mound Plant has been unable to send its low-level waste to the NTS since they have not developed a new plan to comply with DOE transportation rules. However, they anticipate this as being only a temporary problem.

The plant's operating budget (including capital) is approximately \$210 million. The main contractor is EG&G Mound Applied Technologies, who employs 2,170 people full time, and subcontracts currently to 800; however, this number fluctuates widely according to their needs. The local DOE permitted employment level is 30 people, and 22 are current DOE employees.

A.4.2 Funding Mechanism for the Transuranic Waste Management System

The Department of Energy (DOE) does not attempt to do a Total Estimated Cost (TEC) for WIPP at all, as the institutional arrangements and appropriations from Congress for WIPP change drastically every year. Instead, the DOE is required to make out an annual budget. With WIPP, this budget projects three years into the future each time, although the projections often become outdated. Many times, appropriations do not even meet the estimated costs for annual budgets, and thus it has proved difficult for WIPP to project any type of long-term budget or cost projections, even for a three-year period. The DOE is currently working on the fiscal year (FY) 1990 budget. In FY 1989, WIPP received a \$145 million operating budget from Congress. This dropped to \$94 million for FY 1990 because WIPP was expected to be in the testing phase at this point. Since it was not, their current funding is lower than required, and WIPP has put in a FY 1991 request for \$120 million.

WIPP funds are granted by the Congress each year to the Department of Energy, and the DOE sends them directly to WIPP. The DOE is not required to explain their WIPP budget request at a more detailed level than that of the WIPP project as a whole; WIPP is treated like all other DOE programs. The amount of waste a DOE operation will give to WIPP has no effect on the allocations for DOE site budgets.

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