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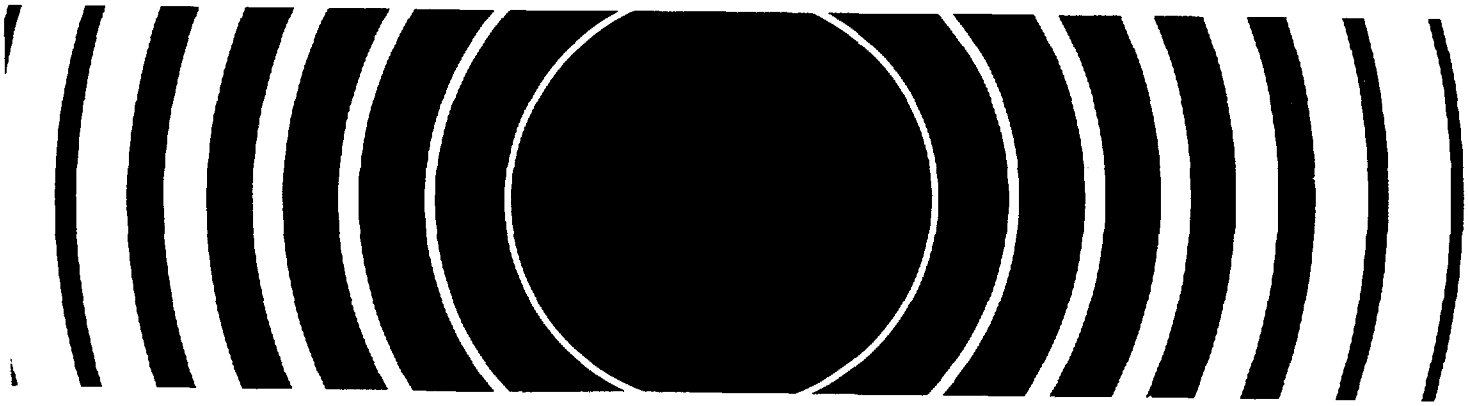

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40 CFR Part 197

DRAFT BACKGROUND INFORMATION DOCUMENT FOR 40 CFR PART 197:

ENVIRONMENTAL RADIATION PROTECTION STANDARDS FOR YUCCA MOUNTAIN, NEVADA

April 1996

U.S. Environmental Protection Agency
Office of Radiation and Indoor Air
Washington, D.C. 20460

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LIST OF ACRONYMS

ACOHP	Advisory Council On Historic Preservation
AEA	Atomic Energy Act
AEC	Atomic Energy Commission
AECB	Canadian Atomic Energy Control Board
AECL	Atomic Energy of Canada Limited
AFCN	Belgian Nuclear Inspection Agency
AGNEB	Swiss Interagency Working Group on Licensing of Nuclear Waste Facilities
AGRs	Advanced Gas-Cooled Reactors
AIRFA	American Indian Religious Freedom Act
ALARA	As Low As is Reasonably Achievable
ALI	Annual Limit on Intake
ANDRA	French Radioactive Waste Management Agency
ANL-W	Argonne National Laboratory - West
BEW	Swiss Energy Office
BfS	German Institute for Radiation Protection
BID	Background Information Document
BMFT	German Ministry for Research and Technology
BMU	German Ministry for Environment, Protection of Nature and Reactor Safety
BNFL	British Nuclear Fuels
BRGM	French Bureau of Geological and Mineral Research

BSS	Basic Safety Standards
BWRs	Boiling Water Reactors
CAA	Clean Air Act
CCDF	Complementary Cumulative Distribution Function
CEA	French Atomic Energy Commission
CEC	Council of the European Communities
CEN	Belgian Nuclear Research Center
CFR	Code of Federal Regulations
CLAB	Swedish Centralized Spent Fuel Storage Facility
CRPPH	Committee on Radiation Protection and Public Health
CRWM	Committee on Radioactive Waste Management
DACs	Derived Air Concentrations
DOD	U.S. Department of Defense
DOE	U.S. Department of Energy
DSIN	French Directorate for the Safety of Nuclear Installations
EDE	Effective Dose Equivalent
EDI	Swiss Department of Interior
EIA	Environmental Impact Assessment
EIR	Swiss Institute for Reactor Research
EIS	Environmental Impact Statement
EnPA	Energy Policy Act
EPA	U.S. Environmental Protection Agency

ERA	Energy Reorganization Act
EURATOM	European Atomic Energy Community
EVED	Swiss Department of Transport, Communications, and Energy
FEIS	Final Environmental Impact Statement
FERC	Federal Energy Regulatory Commission
FFTF	Fast Flux Test Facility
FRC	Federal Radiation Council
GTCC	Greater-Than-Class-C
GW (e)	Gigawatt - Electric
HEU	Highly Enriched Uranium
HI	Human Intrusion
HSK	Swiss Nuclear Safety Division
HTGR	High-Temperature Gas-Cooled Reactors
IAEA	International Atomic Energy Agency
ICPP	Idaho Chemical Processing Plant
ICRP	International Commission on Radiological Protection
INEL	Idaho National Engineering Laboratory
IPSN	French Institute for Nuclear Protection and Safety
IRG	Interagency Review Group
JAERI	Japan Atomic Energy Research Institute
JNFL	Japan Nuclear Fuel Services Limited
KASAM	Swedish Consultative Committee for Nuclear Waste Management

KSA	Swiss Commission for the Safety of Nuclear Installations
LMFBR	Liquid-Metal Fast-Breeder Reactor
LWRs	Light Water Reactors
MCLs	Maximum Contaminant Levels
MCLGs	Maximum Contaminant Level Goals
MFRP	Midwest Fuel Recovery Plant
MITI	Japanese Ministry of International Trade and Industry
MPC	Multi-Purpose Canister
mrem	Millirem
MRS	Monitored Retrievable Storage
mSv	Microsieverts
MTHM	Metric Tons of Heavy Metal
MTIHM	Metric Tons of Initial Heavy Metal
MWd	Megawatt Days
NAGRA	Swiss Cooperative for the Storage of Radioactive Waste
NAS	National Academy of Sciences
NCRP	National Council on Radiation Protection and Measurements
NEPA	National Environmental Policy Act
NHPA	National Historic Preservation Act
NIREX	British Nuclear Industry Radioactive Waste Executive
NPRM	Notice of Proposed Rulemaking
NRC	U.S. Nuclear Regulatory Commission

NSC	Japanese Nuclear Safety Commission
NUCEF	Japanese Nuclear Fuel Cycle Engineering Facility
NWPA	Nuclear Waste Policy Act
NWPAA	Nuclear Waste Policy Amendments Act
NWPO	Nuclear Waste Project Office
OECD/NEA	Organization for Economic Cooperation and Development/Nuclear Energy Agency
OCRWM	Office of Civilian Radioactive Waste Management
OMB	Office of Management and Budget
ONDRAF	Belgian Agency for Radioactive Waste and Fissle Materials
ORNL	Oak Ridge National Laboratory
PA	Programmatic Agreement
PARCLAY	Belgian Preliminary Demonstration Test for Clay Disposal
PBF	Power Burst Facility
PNC	Japanese Power Reactor and Nuclear Fuel Development Corporation
PUREX	Plutonium-Uranium Extraction
PWRs	Pressurized Water Reactors
QAP	Quality Assurance Plan
R&D	Research and Development
RADWASS	Radioactive Waste Safety Standards Radioactive Waste Management
RBOF	Receiving Basin for Off-site Fuels
RETF	Japanese Recycling Equipment Testing Facility

ROD	Record of Decision
RSK	German Reactor Safety Commission
SAB	Science Advisory Board
SAR	Safety Analysis Report
SCP	Site Characterization Plan
SDWA	Safe Drinking Water Act
SGN	French Agency Providing Architectural and Engineering Services
SHP	Japanese Steering Committee on High-Level Radioactive Waste
SKB	Swedish Nuclear Fuel and Waste Management Company
SKI	Swedish Nuclear Power Inspectorate
SKN	Swedish Board for Spent Nuclear Fuel
SRS	Savannah River Site
SSI	Swedish Institute for Radiation Protection
SSK	German Committee on Radiological Protection
STA	Japanese Science and Technology Agency
TBM	Tunnel Boring Machine
THORP	Thermal Oxide Reprocessing Plant
TRU	Transuranic
TSPA	Total System Performance Assessments
UK	United Kingdom
USDWs	Underground Sources Drinking Waters
WIPP LWA	Waste Isolation Pilot Plant Land Withdrawal Act

WVDP	West Valley Demonstration Project
ZWILAG	Swiss Cooperative of Nuclear Utility Operators

CHAPTER 1

INTRODUCTION

The U.S. Environmental Protection Agency (EPA) is responsible for developing and issuing environmental standards and criteria to ensure that public health and the environment are adequately protected from potential radiation impacts. The EPA is proposing in 40 CFR Part 197 site-specific environmental standards to protect public health from releases from radioactive materials disposed of or stored in the potential repository to be constructed at Yucca Mountain in Nevada.[1] These standards provide the basic framework to control the long-term storage and disposal of three types of radioactive waste:

1. Spent nuclear reactor fuel, if disposed without reprocessing;
2. High-level radioactive waste from the reprocessing of spent nuclear fuel; and
3. Other radioactive materials that may be placed into the potential repository.

The other radioactive materials that could be disposed of in the Yucca Mountain repository include highly radioactive low-level waste, known as greater-than-Class-C waste, and excess plutonium resulting from the dismantlement of nuclear weapons. However, the plans for placement of these materials are very uncertain and therefore, for the purpose of the present rulemaking, the information presented in this Background Information Document is limited to spent nuclear fuel and high-level radioactive waste.

This document presents background information and provides a discussion of the technical analyses conducted by EPA in developing 40 CFR Part 197.

1.1 EPA'S REGULATORY AUTHORITY FOR THE RULEMAKING

The proposed standards governing environmental releases from the Yucca Mountain repository have been developed pursuant to the Agency's authorities under the Energy Policy Act (EnPA)

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[1] It is important to note that no decision has been made regarding the acceptability of Yucca Mountain for storage or disposal. However, for the purposes of this document, the description of Yucca Mountain as "potential" will not be used but is intended.

of 1992 (Public Law 102-486). Section 801 of this Act (EnPA92) directed EPA to promulgate standards to ensure protection of public health from high-level radioactive waste in a deep geologic repository to be built at Yucca Mountain (EnPA92). EPA must set standards to ensure protection of the health of individual members of the public. The EnPA also required EPA to contract with the National Academy of Sciences (NAS) to advise the Agency on the technical bases for the Yucca Mountain standards. These standards will apply only to the Yucca Mountain site and are to be developed based upon and consistent with the findings and recommendations of the NAS:

> ...the Administrator shall, based upon and consistent with the findings and recommendations of the National Academy of Sciences, promulgate, by rule, public health and safety standards for protection of the public from releases from radioactive materials stored or disposed of in the repository at the Yucca Mountain site. Such standards shall prescribe the maximum annual effective dose equivalent to individual members of the public from releases to the accessible environment from radioactive materials stored or disposed of in the repository. (EnPA92)

1.2 THE NATIONAL ACADEMY OF SCIENCES RECOMMENDATIONS

In the EnPA, the Congress asked the Academy to address three issues in particular:

- whether a health-based standard based upon doses to individual members of the public from releases to the accessible environment will provide a reasonable standard for protection of the health and safety of the general public;
- whether it is reasonable to assume that a system for post-closure oversight of the repository can be developed, based upon active institutional controls, that will prevent an unreasonable risk of breaching the repository's engineered or geologic barriers or increasing exposure of individual members of the public to radiation beyond allowable limits; and
- whether it will be possible to make scientifically supportable predictions of the probability that the repository's engineered or geologic barriers will be breached as a result of human intrusion over a period of 10,000 years. (EnPA92)

To address these questions, the Academy assembled a committee of 15 members representing a range of scientific expertise and perspectives. The Committee conducted a series of five technical meetings; more than 50 nationally and internationally known scientists and engineers were invited to participate. In addition, the Committee received information from the Nuclear Regulatory Commission (NRC), the Department of Energy (DOE), and EPA, Nevada State and county agencies, and private organizations, such as the Electric Power Research Institute.

The Committee's conclusions and recommendations are contained in its final report entitled, *Technical Bases for Yucca Mountain Standards*, which was issued on August 1, 1995. In this report, the Committee offered the Agency several general recommendations as to the approach EPA should take in developing 40 CFR 197. Specifically, the NAS recommended:

- The use of a standard that sets a limit on the risk to individuals of adverse health effects from releases from the repository. 40 CFR 191[2] contains an individual-dose standard, and it continues to rely on a containment requirement that limits the releases of radionuclides to the accessible environment. The stated goal of the containment requirement was to limit the number of health effects to the global population to 1,000 incremental fatalities over 10,000 years. We do not recommend that a release limit be adopted.

- That compliance with the standard be measured at the time of peak risk, whenever it occurs. (Within the limits imposed by the long-term stability of the geologic environment, which is on the order of one million years.) The standard in 40 CFR 191 applies for a period of 10,000 years. Based on performance assessment calculations provided to us, it appears that peak risks might occur tens or hundreds of thousands of years or even farther into the future.

- Against a risk-based calculation of the adverse effect of human intrusion into the repository. Under 40 CFR 191, an assessment must be made of the frequency and consequences of human intrusion for purposes of demonstrating compliance with containment requirements. In contrast, we conclude that it is not possible to assess the frequency of intrusion far into the future. We do recommend that

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[2] In 1985, EPA promulgated 40 CFR Part 191, "Environmental Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes (EPA85a)." These are generally-applicable environmental standards promulgated under EPA's authority under the Atomic Energy Act of 1954. As a result of court action these standards remanded back to EPA and were subsequently repromulgated in 1993. (See Sections 1.3.1 and 1.3.4 for more detail).

the consequences of an intrusion be calculated to assess the resilience of the repository to intrusion.

The NAS Committee also recommended that the resolution of policy issues be done through a rulemaking process that allows opportunity for wide-ranging input from all interested parties (NAS95).

The Committee also addressed each of the specific questions posed to it by the Congress in the EnPA. With regard to the first issue, protecting human health, the NAS Committee recommended:

- The use of a standard that sets a limit on the risk to individuals of adverse health effects from releases from the repository.
- The critical-group approach be used in the Yucca Mountain standards.
- Compliance assessment be conducted for the time when the greatest risk occurs, within the limits imposed by long-term stability of the geologic environment.

The NAS also concluded that an individual-risk standard would protect public health, given the particular characteristics of the site, provided that policy makers and the public are prepared to accept that very low radiation doses pose a negligibly small risk. A necessarily important component to the development of a standard for Yucca Mountain is the means of assessing compliance. The NAS Committee concluded the following with respect to this consideration:

- Physical and geologic processes are sufficiently quantifiable and the related uncertainties sufficiently boundable that the performance can be assessed over time frames during which the geologic system is relatively stable or varies in a boundable manner. The geologic record suggests that this time frame is on the order of 10^6 years. The Committee further concluded that the probabilities and consequences of modifications by climate change, seismic activity, and volcanic eruptions at Yucca Mountain are sufficiently boundable that these factors can be included in performance assessments that extend over this time frame.
- However, it is not possible to predict on the basis of scientific analyses the societal factors required for an exposure scenario. Specifying exposure scenarios therefore requires a policy decision that is appropriately made in a rulemaking process conducted by EPA.

With respect to the second and third questions posed by the Congress in Section 801 of the EnPA, the NAS Committee concluded:

- It is not reasonable to assume that a system for post-closure oversight of the repository can be developed, based on active institutional controls, that will prevent an unreasonable risk of breaching the repository's engineered barriers or increasing the exposure to individual members of the public to radiation beyond allowable limits.
- It is not possible to make scientifically supportable predictions of the probability that a repository's engineered or geologic barriers will be breached as a result of human intrusion over a period of 10,000 years. (NAS95)

1.3 HISTORY OF EPA'S RULEMAKING

1.3.1 Legislative History

As well as its authority to set site-specific standards for Yucca Mountain, EPA also has the authority to set generally applicable environmental standards for radioactive releases under the Atomic Energy Act (AEA) of 1954, as amended, and the EPA Reorganization Plan No. 3 of 1970 (NIX70). The basic authority under the AEA, as transferred to the EPA by Reorganization Plan No 3, 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. (AEA54)

In 1982, the Nuclear Waste Policy Act (NWPA) (Public Law 97-425) established formal procedures regarding the evaluation and selection of sites for geologic repositories, including procedures for the interaction of State and Federal governments. The Act established provisions for the selection of at least two independent repository sites. Further, the NWPA limited the quantity of spent fuel to be disposed in the initial repository to 70,000 metric tons of heavy metal (MTHM)[3], or a quantity of solidified high-level radioactive waste resulting from the reprocessing of such a quantity of spent fuel, until a second repository is in operation (NWP83). The NWPA also 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. (NWP83)

In September 1985, EPA published 40 CFR Part 191, "Environmental Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes (EPA85a)." These standards were to apply to all sites for the deep geologic disposal of high-level radioactive waste. In 1987, the U.S. Court of Appeals for the First Circuit responded to a legal challenge by remanding Subpart B of the 1985 standards to the Agency for further consideration.

In December 1987, Congress enacted the Nuclear Waste Policy Amendments Act (NWPAA). The 1987 Amendments Act redirected the nation's nuclear waste program to consider Yucca Mountain as the prime site for the first high-level waste and spent nuclear fuel repository (NWP87). Activities at all other potential sites were to be phased out. If the Yucca Mountain site is found to be suitable, the President is required to submit a recommendation to Congress to develop a repository at this location. In the event that site characterization activities indicate that Yucca Mountain is an unsuitable site for the repository, the Secretary of Energy is required to inform Congress and the State of Nevada of its findings. The NWPAA prohibits DOE from conducting site-specific activities for a second repository unless authorized to do so by Congress. However, the NWPAA does require a report from the Secretary of Energy on the need for a second repository no later than January 1, 2010.

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 (required under the original Act, as amended by

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[3] This is a measure of the uranium content of the spent fuel to be emplaced in the repository.

Public Law 100-507) a report outlining their recommendations on November 1, 1989 (NWP88, RMR89).

In October 1992, the Waste Isolation Pilot Plant Land Withdrawal Act (WIPP LWA) was enacted. While reinstating certain sections of the Agency's 1985 disposal standards, the Act specifically exempted the Yucca Mountain site from these generic disposal standards (WIP92). However, the EnPA directed the EPA to set site-specific radiation protection standards for the Yucca Mountain disposal system (EnPA92).

1.3.2 The Development of EPA's Role in the Federal Program

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 waste, regardless of whether it is produced by commercial or national defense activities. In 1949, the Atomic Energy Commission (AEC) initiated work aimed at developing systems for converting high-level liquid waste into a stable form. Then, in 1955, at the request of the AEC, an NAS Advisory Committee was established to consider the disposal of high-level radioactive waste within the United States. Its report, issued in 1957, recommended that:

1. The AEC continue to develop processes for the solidification of high-level radioactive liquid waste; and
2. Naturally-occurring salt formations be used as the medium for the long-term isolation of the solidified waste. (NAS57)

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 waste in salt formations (McCL70).

In 1968, the AEC again requested the NAS to establish a Committee on Radioactive Waste Management (CRWM) to advise the AEC on its long-range radioactive waste management plans and to evaluate the feasibility of disposing of solidified radioactive waste in bedded salt. The CRWM convened a panel to discuss the disposal of radioactive waste in salt mines. Based on the recommendations of the panel, the CRWM concluded that the use of bedded salt was satisfactory for the disposal of radioactive waste (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 alternative sites (DOU72).

In 1976, the Federal government intensified its program to develop and demonstrate a permanent disposal method for high-level radioactive waste. The Office of Management and Budget (OMB) established an interagency task force on commercial wastes in March 1976. The task force defined the scope of the responsibility of each Federal agency's activities on high-level management, including the preparation of environmental standards for high-level waste by the EPA (LYN76, ENG77a, ENG77b).

Shortly after the interagency task force was formed, the Federal Energy Regulatory Commission (FERC) published a status report on the management of commercial radioactive waste. The report, issued in May of 1976, emphasized the need for coordination of administration policies and programs relating to energy and called for an accelerated comprehensive government radioactive waste program plan. The report also recommended that an interagency task force be formed to coordinate activities among the responsible Federal agencies.

Subsequent to its findings, FERC established a nuclear subcommittee to coordinate Federal nuclear policy and programs. The EPA was given the responsibility of establishing general environmental standards governing waste disposal activities, including standards for high-level radioactive waste to be delivered to Federal repositories for long-term management (FER76).

In October of 1976, after the OMB interagency task force proposed its plan for high-level waste management, 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 had the facilities for the long-term management of nuclear waste from commercial power plants. He also reported that experts had concluded that the most practical method for disposing of high-level radioactive waste would be in geologic repositories located in stable formations deep underground. The EPA was charged with the responsibility of issuing general environmental standards governing nuclear waste facility releases to the biosphere above natural background radiation levels (FOR76). These standards were to place a numerical limit on long-term radiation releases outside the boundary of the repository.

1.3.3 Early Federal Action

In December 1976, the EPA announced its intent to develop environmental radiation protection criteria for radioactive waste 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). Based on issues raised during workshop deliberations, EPA published a Federal Register Notice on November 15, 1978 (43 FR 53262) (EPA78c) to propose criteria for radioactive wastes and to solicit public comments on possible recommendations for Federal Radiation Guidance. In this Notice, EPA presented a set of criteria to address six key waste control decision issues: 1) the types of materials to be categorized as radioactive wastes and subject to control; 2) the efficacy of engineered controls and natural barriers to isolate wastes; 3) the usefulness of social institutions in providing control, especially their viability over time; 4) the potential health risks of wastes (over various time intervals and with differing levels of control); 5) the unacceptability of various levels of risk; and 6) other considerations such as retrievability and communication of waste disposal sites to succeeding generations to assure continued isolation. As proposed, EPA intended that the initial set of six criteria—each addressing one of the six key issues—would serve collectively as the basis for developing environmental standards for different radioactive waste sources.

During this time, President Carter established the Interagency Review Group (IRG) to develop recommendations for an administrative policy to address the long-term management of nuclear waste 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 waste, spent nuclear fuel, and transuranic waste (DOE79). In a message to Congress in February 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 reiterated that the EPA was responsible for creating general criteria and numerical standards applicable to nuclear waste management activities (CAR80).

In March 1981, the EPA withdrew the proposed "Criteria for Radioactive Wastes" because it considered the implementation of generic disposal guidance too complex given the many different types of radioactive waste (EPA81).

In 1982, Congress enacted the NWPA, which established the current national program for the disposal of spent nuclear fuel and high-level waste. The Act assigned DOE the responsibility of siting, building and operating an underground geologic repository for the disposal of these wastes and directed the EPA to "promulgate generally applicable standards for the protection of the general environment from off-site releases from radioactive material in repositories" (NWP83). In that same year, under the authority of the AEA, 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).

After the first comment period on the proposed rule ended in May 1983, the EPA held two public hearings on the proposed standards—one in Washington, DC, and one in Denver, CO. During a second public comment period, EPA 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 (EPA85b).

In parallel with its public review and comment effort, the EPA conducted an independent scientific review of the technical bases 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 to September 1983 and released a final report in February 1984 (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 in May of 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 (EPA85c).

In February 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 1984 deadline mandated by the NWPA for promulgation of final 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 August 15, 1985 (EPA85d, EPA85e).

1.3.4 40 CFR Part 191

The 1985 EPA standards for the management and disposal of spent nuclear fuel, high-level and transuranic waste were divided into two main sections, Subparts A and B (EPA85a). Subpart A, which addressed the management and storage of waste, limited radiation exposure to any member of the general public to 25 millirem (mrem) to the whole body and 75 mrem to any critical organ for disposal facilities operated by the Department of Energy, but not regulated by the NRC or an Agreement State. For facilities regulated by the NRC or an Agreement State, the Standards endorsed the annual dose limits given in the environmental standards for the uranium fuel cycle (40 CFR Part 190): 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any critical organ (EPA77c).

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:

- Limits on cumulative releases of radioactive materials into the environment over 10,000 years;
- Assurance requirements to compensate for uncertainties in achieving the desired level of protection;
- Individual exposure limits based on the consumption of ground water and any other potential exposure pathways for 1,000 years after disposal; and
- Ground water protection requirements in terms of allowable radionuclide concentrations and associated doses for 1,000 years after disposal. (EPA85a)

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 ground water concentration for beta or gamma emitters was 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 (pCi/L). For radium-226 and radium-228 alone, the concentration limit was 5 pCi/L. Appendix A of the standards provided acceptable radionuclide-specific 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 (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 ground water protection requirements; and
3. Arbitrary standards, 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 judgments. The Court's action resulted in the remand of Subpart B. The Court began by looking at the definition of "underground injection." In the view of the Court, the method envisioned by DOE for disposal of radioactive waste in underground repositories 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 ground water. 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" was considered when doses higher than that allowed by the Primary Drinking Water Regulations could occur. Section 191.15 permitted an annual dose of 25 mrem to the whole body and 75 mrem to any critical organ from all pathways. Existing EPA regulations promulgated under the SDWA allowed an annual dose of 4 mrem from drinking water. Although the Court recognized that an exposure level less than 4 mrem could result from the ground water pathway, it rejected this possibility because the Agency stated that radioactivity could eventually be released into the ground water system near the repository and that substantially higher doses could result. Therefore, the Court decided that a large fraction of the 25 mrem limit could be received through the ground water exposure pathway. Accordingly, the Court found that the high-level waste standards should either have been consistent with the SDWA or the Agency should have justified the adoption of a different standard.

The Court also noted that the Agency was not necessarily incorrect in promulgating the proposed standards. However, it noted that the Agency never acknowledged the interrelationship of the SDWA and high-level waste 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 was not inherently flawed, but rather that the administrative record and the Agency's explanations did not adequately support this choice. The criterion was remanded for reconsideration and the Agency was directed to provide a more thorough explanation for its basis.

Finally, the Court found that the Agency did not provide sufficient opportunity for notice and comment on Section 191.16 (Ground Water Protection Requirements), which was added to Subpart B after the standards were proposed. This section was remanded for a second round of notice and comment. There were, however, no rulings issued on technical grounds about Section 191.16.

In August 1987, the Justice Department petitioned 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.

In 1992, the WIPP LWA reinstated Subpart B of 40 CFR Part 191, except Sections 191.15 and 191.16, and required the Administrator to issue final disposal standards no later than six months after enactment. On December 20, 1993, EPA issued amendments to 40 CFR Part 191 which eliminated section 191.16 of the original rule; altered the individual protection requirements; and added Subpart C on ground water protection (EPA93). The amended standards represent the Agency's response to the above legislation and to the issues raised by the court pertaining to individual and ground water protection requirements. In so doing, EPA did not revisit any of the regulations reinstated by the WIPP LWA.

The WIPP LWA also exempted Yucca Mountain from the generic disposal standards set forth under 40 CFR Part 191, Subpart B. Pursuant to specific provisions in the EnPA, EPA was charged with setting site-specific environmental radiation standards for Yucca Mountain and has responded to this mandate in its proposed rule, 40 CFR Part 197.

1.4 PURPOSE AND SCOPE OF THE BACKGROUND INFORMATION DOCUMENT

This document provides the necessary background information and technical analyses in support of the proposed rule 40 CFR Part 197. 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 discusses the radioactive source term characterization, movement of radionuclides from the repository at Yucca Mountain through the appropriate environmental exposure pathways, and calculations of doses received by members of the general public performed to date.

Chapter 2 provides a brief history of the evolution of radiation protection activities in the United States as well as current U.S. regulatory programs and strategies. A summary of key international programs for high-level waste disposal is presented in Chapter 3. Chapter 4 describes U.S. programs for the management and disposal of high-level radioactive waste and spent nuclear fuel. Current and projected inventories of spent nuclear fuel and DOE defense high-level radioactive waste are presented in Chapter 5. Chapter 6 describes the methodology used by EPA for dose and risk estimation. Chapter 7 provides descriptions of the natural features of the Yucca Mountain site, the concepts under consideration for the engineered features of a potential repository at the site, and analyses to date concerning safety performance of a disposal system at the site. Chapter 8 describes the physical and human environment in the Yucca Mountain region, current conditions of human radiation exposure in the region, and concepts that could be used to evaluate the consequences of radioactivity release from a repository at Yucca Mountain.

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CHAPTER 2

HISTORY OF RADIATION PROTECTION IN THE UNITED STATES AND CURRENT REGULATIONS

2.1 INTRODUCTION

Radiation from cosmic rays and the naturally-occurring radioactivity contained in the earth make up the natural radiation background environment in which all life forms have evolved. Society's recognition of radiation began in 1895 with the discovery of X-rays; naturally-occurring radioactivity was observed in 1896. These discoveries marked the beginning of the study and use of radioactive substances in science, medicine, and industry.

The discovery of radioactivity led rapidly 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.

Today the use of radiation, be it naturally-occurring or man-made, is widespread and reaches every segment of our society. Common examples include:

- Nuclear reactors generate electricity, power ships and submarines, produce radioisotopes used for research, medical, industrial, space and national defense applications, and are used as research tools for nuclear engineering and physics.
- Particle accelerators produce radioisotopes and radiation and are used to study the structure of matter, atoms, and common materials.
- Radioisotopes are used in nuclear medicine, biomedical research, and medical treatment.

- X-rays and gamma rays are used as diagnostic tools in medicine, as well as in diverse industrial applications, such as industrial radiography, luggage x-ray inspections, and non-destructive materials testing.

- 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 contain radioactive materials.

It was soon recognized that the use of radioactive materials would have to be controlled to protect the public, workers, and the environment from radiation exposures. The following sections present a brief history of the evolution of radiation protection activities, and their principles and concepts. U.S. regulatory programs and strategies are also discussed. Chapter 3 presents a summary of key international programs.

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

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.

In 1928, at the Second International Congress of Radiology meeting in Stockholm, Sweden, 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 act in concert with the Commission in developing and disseminating recommendations on radiation protection. This suggestion led to the formation of the U.S. Advisory Committee on X-Ray and Radium Protection in 1929. In 1964, the Committee was Congressionally chartered as the National Council on Radiation Protection and Measurements (NCRP).

Throughout their existence, the ICRP and the NCRP have worked closely together to develop radiation protection recommendations that reflect the current understanding of the risks associated with exposure to ionizing radiation (ICR34, ICR38, ICR51, ICR59, ICR65). Neither organization has official status, in that they do not have authority to issue or enforce regulations. However, their recommendations serve as the basis for the radiation protection regulations adopted by the regulatory authorities in the United States and most other nations.

The International Atomic Energy Agency (IAEA) was chartered in July 1957 as an autonomous intergovernmental organization under the aegis of the United Nations. The IAEA gives advice and technical assistance to Member States on nuclear power development, health and safety issues, radioactive waste management, and on a broad range of other areas related to the use of radioactive material and atomic energy in industry and government. As is the case for ICRP and NCRP, Member States do not have to follow IAEA recommendations. However, funding for international programs dealing with the safe use of atomic energy and radioactive materials can be withheld if Member States do not comply with IAEA recommendations. In addition, in matters related to safeguarding special nuclear material, the full weight of the UN can be brought to bear to "enforce" UN resolutions pertaining to the use of nuclear materials for peaceful purposes. Many of the IAEA recommendations adopt ICRP recommendations with respect to the Commission's radiation protection philosophy and numerical criteria.

In 1977, the ICRP released recommendations that are in use today. ICRP Publication No. 26 (ICR77) adopted the weighted, whole-body dose equivalent (defined as the effective dose equivalent) concept for limiting occupational exposures. This approach reflected the increased understanding of the differing radiosensitivities 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 [ICR59].)

ICRP No. 26 defined the goal of radiation protection as the prevention or limitation of effects from radiation exposure and the assurance that practices involving radiation exposure are justified. The concept of collective dose equivalent for populations was also discussed. The ICRP No. 26 recommendations represented 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... ." 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 (ICR77). The ICRP recommended an annual individual dose limit of 100 mrem (1 mSv) from all radiation sources. However, the Commission also recognized that an annual individual dose limit of 500 mrem (5 mSv) may be permissible, provided that the average annual effective dose equivalent over a lifetime does not exceed the principal limit of 100 mrem (1 mSv) (ICR85a). No dose limits for populations were proposed; the Commission felt that the system of dose limitation specified in ICRP No. 26 would likely ensure that the average dose equivalent to the population would not exceed 50 mrem per year.

In 1979, the ICRP issued Publication No. 30 (ICR79) establishing the Annual Limit on Intake (ALI) system for limiting the intake of radionuclides by workers. The ALI is the activity of a given nuclide that 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 (ICR84) clarifying the use of ALIs and Derived Air Concentrations (DACs) for members of the public.

In 1985, the ICRP issued a statement (ICR85a) refining 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 its 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.

The Commission has also published guidance for waste disposal (ICR85b) and for general radiological protection (ICR91). The first of these, "Radiation Protection for the Disposal of Solid Radioactive Waste," emphasizes an individual risk approach that considers both the probability of a breach of a disposal site and its consequence to the critical group.

In 1987, the NCRP issued Report No. 91 (NCR87), which acknowledged the assumptions and the basic thrust of the recommendations in ICRP Reports 26 and 30. In discussing risk estimates, the NCRP noted in its report that new data were becoming available that might require changes in the current estimates. However, the value recommended in ICRP No. 26 of 10^{-4} per rem 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 (NCR87). Similar to the 1985 ICRP statement, 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 IAEA issued reports 96 and 99 in its Safety Series (IAE89a, IAE89b). These documents presented criteria and guidance for the underground disposal of nuclear waste. Safety Series No. 99, "Safety Principles and Technical Criteria for the Underground Disposal of High-Level Radioactive Wastes," set 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." The report went on to state 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 (i.e., 100 mrem/yr)[4] for prolonged exposures for individuals in the critical group (defined as the members of the public whose exposure is

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[4] The ICRP has adopted the International system of units (SI). Under this system, 1Sv equals 100 rem. As such 1 mSv equals 100 mrem.

relatively homogeneous and is typical of individuals receiving the highest effective dose equivalent or dose equivalent from a given radiation source). Finally, it suggested a risk upper bound of 10^{-5} per year for an individual for disruptive events.

In 1990, the ICRP issued Publication 60, which broadened its recommendations to include a wider range of exposure scenarios than had been previously addressed. Publication 60 also gave support to new concepts in the field of radiation exposure protection, most notably the ALARA (as low as reasonably achievable) concept of worker protection optimization. The ALARA principle suggests dose limits should be set at the lowest levels reasonably possible for a given scenario. In recent years, several international organizations, including the Council of the European Communities (CEC) and the Organization for Economic Cooperation and Development/Nuclear Energy Agency's (OECD/NEA's) Committee on Radiation Protection and Public Health (CRPPH), have worked to interpret this principle and develop guidelines for its practical use (NEA94). The formality with which the ALARA principle has been adopted varies widely internationally. In many cases, the ALARA principle is being applied only as part of a non-quantified conceptual framework within which protection measures are implemented; in other countries, the application of the ALARA approach to worker safety is becoming increasingly formalized (OEC95a).

In recent years, the IAEA has been developing a number of new international safety standards and guidance documents. Foremost among these is "International Basic Safety Standards for Protection Against Ionizing Radiation and for the Safety of Radiation Sources," known as BSS (Basic Safety Standards, Safety Series 115-I). The BSS was approved by the IAEA Board of Governors in 1994 and published as an interim document in December, 1995. A joint effort of the Food and Agricultural Organization of the United Nations, the International Labor Organization, the OECD/NEA, the Pan-American Health Organization, and the World Health Organization, the BSS is notable primarily for its movement toward an integrated approach to managing exposure risk in which potential but unlikely events (like accidents) are evaluated along with comparatively normal, likely scenarios for exposure. Previously, safety assessment had focused only on comparatively normal, likely scenarios (OEC95a). IAEA has also been developing a comprehensive set of safety standards regarding radioactive waste management called Radioactive Waste Safety Standards (RADWASS). RADWASS includes a safety fundamental document entitled "The Principles of Radioactive Waste Management," and a safety standard document entitled "Establishing a National Safety Standard for Radioactive

Waste Management," both of which were submitted to the IAEA Board of Governors for review in 1995. Three other safety standards (S-2, S-3 and S-6) addressing predisposal management of radioactive waste, near surface disposal of radioactive waste, and decommissioning are under review (OEC95b). The entire RADWASS series is currently under review to ensure harmonization with Safety Series Publications and BSS documents.

In recent years, the CEC has been developing directives on safety standards for radiological exposures established under European Atomic Energy Community (EURATOM) agreements. In accordance with ICRP recommendations, the CEC suggested in 1993 that doses to members of the public be limited to 100 mrem per year from all sources except medical and that occupational doses be limited to 2,000 mrem annually. The CEC is also expected to propose criteria for the shipment of radioactive waste among member countries and for the export of radioactive waste to non-member countries (OEC93).

2.3 FEDERAL RADIATION COUNCIL GUIDANCE

The Federal Radiation Council (FRC) was established in 1959 by Executive Order 10831. The Council arose as a direct result of new information that became available in the 1950s on the effects of radiation. Prior to that time, only non-governmental radiation advisory bodies (i.e., ICRP and NCRP), whose recommendations were not binding on users of radiation or radioactive materials, existed. The FRC was established as an official government entity and included representatives from all Federal agencies concerned with radiation protection. The Council served as the primary coordinating body for all radiation activities conducted by the Federal government (FRC60a) and was responsible for:

> ...advising the President with respect to radiation matters, directly or indirectly affecting health, including providing guidance to 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 President Eisenhower in 1960 (FRC60b). The guidance established exposure limits for members of the general public. These included the yearly radiation exposure of 0.5 rem per year for the whole body of individuals in the general population and an average gonadal dose of 5 rem in 30 years for the general population (exclusive of natural background and the purposeful medical exposure of patients).

The guidance also established occupational exposure limits, which differed only slightly from those recommended by the NCRP and ICRP at the time (NCR54, NCR59) and included:

- Whole body, head and trunk, active blood forming organs, gonads or lens of the eyes are not to exceed 3 rem in 13 consecutive weeks, and the total accumulated dose is limited to 5 rems multiplied by 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 consecutive weeks or 30 rem per year.
- Hands, forearms, feet, and ankles are not to exceed 25 rem in 13 consecutive 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 in 13 consecutive weeks or 15 rem per year.

In addition to the formal exposure limits, the guidance also established as Federal policy that any radiation exposure should be justified and that "...every effort should be made to encourage the maintenance of radiation doses as far below this guide as practicable...." Both of these concepts had previously been proposed by the ICRP. 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.

Following the issuance of this initial guidance, the FRC continued to provide guidance on a number of radiation protection matters. In 1970, the Council was dissolved, and its functions were transferred to the Environmental Protection Agency under authority of Reorganization Plan No. 3 (NIX70).

2.4 ENVIRONMENTAL PROTECTION AGENCY

Since its creation in 1970, the EPA has issued regulatory standards regarding radiation hazards from a number of different sources, including underground mining (EPA71), uranium fuel cycle operations (EPA77), uranium and thorium mill tailings (EPA83), radionuclide air emissions (EPA89b), and management and disposal of spent nuclear fuel and high-level and transuranic radioactive wastes (EPA93). Recently, EPA issued compliance criteria for the WIPP (EPA96). EPA is currently developing a standard for the disposal of contaminated soil at decommissioned sites, including Federal facilities.

The Agency has also exercised its authority to issue Federal guidance to limit radiation exposures to workers (EPA83, EPA87), as well as to the general public. In December 1994, EPA issued proposed Federal guidance to update the previous Federal Radiation Protection Guidance for Exposure to the General Public which was originally adopted in 1960 and 1961 (EPA94). The Agency is now finalizing these new recommendations.

EPA has also provided extensive technical information regarding the assessment of risk from radiation hazards. Specific examples of such information include radionuclide intake limits, occupational radiation doses, biological parameters, and dose conversion factors (EPA88). This information has been used extensively in the development of EPA standards and guidance, as well as specific site assessments.

In addition to its responsibility to provide Federal guidance on radiation protection, the EPA has various statutory responsibilities for regulating exposure to radiation. The standards and regulations that EPA has promulgated and proposed with respect to controlling radiation exposures are summarized in the following paragraphs. Their applicability to EPA's proposed standards under 40 CFR Part 197 is also discussed.

2.4.1 <u>Environmental Radiation Exposure</u>

The Atomic Energy Act (AEA) of 1954, as amended, and Reorganization Plan No. 3 granted the EPA the authority to establish generally applicable environmental standards for exposure to radiation (AEA54, NIX70). The AEA is the cornerstone of current radiation protection activities and regulations. In 1977, pursuant to this authority, 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. 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 standards also set 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 standards cover all exposures resulting from radiation and radionuclide releases to air and water from operations of fuel-cycle facilities. The development of these standards 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.

2.4.2 Environmental Impact Assessments

In 1969, Congress passed the National Environmental Policy Act (NEPA), which declared a national policy that encouraged a productive and enjoyable harmony between the public and the environment (NEP70). The Act recognized the profound impact of human activity on the interrelations of all components of the natural environment and sought to promote efforts to prevent or eliminate damage to the environment. To this end, the national policy is geared towards increasing the understanding of the ecological systems and natural resources important to the United States. In addition, the Act established a Council on Environmental Quality to assist the President in determining the state of the environment and developing environmental policy initiatives.

The Act also directed all Federal agencies to use a systematic, interdisciplinary approach to ensure the integrated use of natural, social, and environmental sciences in support of plans and decisions that have a potential impact on the environment. Specifically, it mandated that a detailed Environmental Impact Statement (EIS) be submitted for any major action proposed by a Federal agency or for legislation that would significantly affect the quality of the environment. The EIS must describe any adverse environmental effects that the proposal would cause, alternatives to the proposed action, effects of the project on the long-term productivity of the environment, and any irreversible and irretrievable commitment of resources involved in the proposed action. The EIS must also be prepared through consultation with any Federal agency having jurisdiction or special expertise regarding the project and its environmental impact.

The Final EIS prepared by the Department of Energy for the Yucca Mountain site must comply with NEPA requirements.

2.4.3 Ground Water Protection

The Safe Drinking Water Act (SDWA) was enacted to assure safe drinking water supplies and to protect against endangerment of underground sources of drinking waters (USDWs). Under the authority of the SDWA, 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 limit chosen for man-made beta and photon emitters is 4 mrem/yr to the whole body or organ for the most exposed individual.

In 1991, the EPA issued a Notice of Proposed Rulemaking (NPRM) under 40 CFR Parts 141 and 142 to update the 1976 interim regulations for radionuclide water pollution control (EPA91). The NPRM, under the SDWA, 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, gross beta, and photon emitters. As proposed, MCLGs are not enforceable health goals. In contrast, 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, 20 pCi/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 allow doses of between 4 mrem/yr and 20 mrem/yr to individuals drinking the contaminated water.

2.4.4 Radionuclide Air Emissions

In December 1979, the EPA designated radionuclides as hazardous air pollutants under Section 112 of the Clean Air Act (CAA) Amendments of 1977 (Public Law 95-95) (EPA79). In April 1983, the EPA proposed standards regulating radionuclide emissions from four source categories, one of which included U.S. Department of Energy (DOE) facilities. The rule established annual airborne emission limits for radioactive materials and specified that annual doses resulting from such emissions should not exceed 25 mrem to the whole body and 75 mrem to any critical organ for members of the general public. The EPA also proposed not to regulate several other categories of facilities, including high-level radioactive waste disposal facilities. EPA based its decision with respect to high-level waste disposal facilities on estimated releases from conceptual repositories that indicated that the airborne exposure pathway would not cause doses high enough to warrant regulation.

In October 1984, following a court order, 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 waste.

In December 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 order in 1985 by issuing standards for selected sources (EPA85f, EPA85g). As a result of the decision in *National Resources Defense Council Inc. v. EPA,* November 1987, the Agency submitted a motion to the court requesting a voluntary remand of its national emission standards for the four original categories of emission sources proposed in April of 1983. 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) determine what is safe, based exclusively on health risk, and 2) adjust the level of safety downward to provide an ample margin of safety.

In March 1989, the EPA issued a proposed rule for regulating radionuclide emissions under the CAA following the re-examination of the regulatory issues associated with the use of Section 112 (EPA89a). The rule proposed four policy alternatives to control emissions and risks from 12 categories of sources. Each of the four approaches considered 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 maximum individual lifetime fatal cancer risk must not exceed 1×10^{-4}.
- Incidence-Based Approach — Based on the best estimate of the total incidence of fatal cancer. The proposed acceptable level of incidence must not exceed one fatal cancer per year per source category.
- Maximum Individual Risk Approach (10^{-4} or less) — Only risk indicator considered is the best estimate of the maximum individual lifetime risk of fatal cancer. The 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 maximum individual lifetime risk, however, must not exceed 1×10^{-6}.

Consistent with the two-step process established by the Court, the Agency determined an ample margin of safety after ascertaining a safe level based solely on health risks. In reaching its final decision, the EPA considered all health risk measures, as well as technological feasibility, costs, uncertainties, economic impacts of control technologies, and any other relevant information.

In its radionuclide emission standards, EPA considered a lifetime risk to an individual of approximately 1 in 10,000 as acceptable. 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.

In its final rule, EPA concluded that there was no need to establish air emission standards for high-level waste disposal repositories since anticipated operations at the site would be governed by 40 CFR Part 191. Radioactive materials received at such facilities are sealed in containers. Normal operations do not require additional processing or handling because spent fuel or high-level waste is 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, are covered by 40 CFR Part 191 and must comply with Subpart I of the National Emission Standards for radionuclides[5] (EPA85g). Consequently, the Agency believed there to be an ample margin of safety since the likelihood of releases, and attendent risks, are very low.

2.4.5 Disposal of High-Level Radioactive Waste and Spent Nuclear Fuel

Congress passed the Nuclear Waste Policy Act (NWPA) of 1982 to provide for the development of repositories for the disposal of high-level radioactive waste and spent nuclear fuel, and to establish a program of research, development and demonstration regarding this disposal (NWP83). The Act established a schedule for the siting, construction and operation of repositories that would provide a reasonable assurance that the public and environment would be adequately protected from the hazards posed by high-level radioactive waste. The Secretary of Energy was charged with nominating candidate sites for a repository and following a number of steps through a process of Presidential and Congressional approval, site characterizations, public participation, and hearings. The Act also required the Secretary to adhere to NEPA in considering alternatives and to prepare an EIS for each candidate site. Initially the Act called for the development of two mined geologic repositories. The first repository was to be selected after characterizing sites in volcanic rock at Yucca Mountain, Nevada and Hanford, Washington or in salt domes in Texas. The second repository was to be located in the eastern United States in crystalline rock. EPA was charged with the responsibility of promulgating generally applicable standards for the protection of the general environment from off-site releases from radioactive material in repositories. The NRC, in turn, was responsible for promulgating technical requirements and criteria consistent with EPA's standards for use in approving or disapproving applications for the construction, use, and closure of the repository. The Act also discussed interim waste storage requirements, as

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[5] Subpart I of the National Emission Standard can be found in 40 CFR 61.101 and is entitled "National Emission Standard for Radionuclide Emissions from Facilities Licensed by the Nuclear Regulatory Commission (NRC) and Federal Facilities Not Covered by Subpart H." Subpart H of the National Emission Standard addresses radionuclide standards for DOE facilities.

well as the payment of benefits to affected States and tribal groups to allow them sufficient resources to fully participate in the process.

In 1987, the NWPA was amended to reflect a redirection of the nuclear waste program. The generic nature of the original act was changed to reflect the selection of the Yucca Mountain site in Nevada as the only candidate site for the repository (NWP87). The State of Nevada was also identified as the affected community. All site-specific activities at other candidate sites were phased out, and the Final EIS, necessary for compliance with the NEPA, was to be prepared specifically for the Yucca Mountain site without further consideration of alternative sites. The redirection charged DOE with reporting to Congress on the potential social, economic, and environmental impacts of locating the repository at Yucca Mountain.

2.4.5.1 Generic Disposal Standards for High-Level and Transuranic Wastes

The Waste Isolation Pilot Plant Land Withdrawal Act (WIPP LWA) of 1992 reinstated all of the disposal standards remanded by the First Circuit Court of Appeals in 1987 except the three aspects of the individual and ground water protection requirements that were the subject of the court remand (WIP92). It then put the Agency on a schedule for issuing the final disposal standards for high-level and transuranic wastes, which were published in December of 1993. The law also provided an extensive role for EPA in reviewing and approving various phases of DOE activities at the WIPP and required EPA to certify whether the WIPP repository would meet the final 40 CFR Part 191 standards. Finally, and of greatest importance to the current rulemaking, the WIPP LWA exempted radioactive waste disposal activities at Yucca Mountain from compliance with the generic standards set forth under the 40 CFR Part 191 standards.

2.4.5.2 Site-Specific Disposal Standards for High-Level Radioactive Waste

The Energy Policy Act (EnPA) of 1992 addressed energy efficiency throughout the United States in different situations and for various types of fuel. Title VIII of the Act dealt specifically with high-level radioactive waste. Section 801 of the EnPA assigned EPA the responsibility of promulgating public health and safety standards for protection of the public from releases from radioactive materials stored or disposed of in the repository at the Yucca Mountain site. EPA is to prescribe a maximum annual effective dose equivalent to individual members of the public from releases to the accessible environment from radioactive materials stored or disposed of in the repository (EnPA92). The Act also requires that the standards developed be based upon and consistent with the findings and recommendations of the NAS. Specifically, the NAS was charged with considering: the use of a dose-based standard, the reasonableness of post-closure oversight in preventing breaches, and the predictability of human intrusion over a period of 10,000 years. NAS's findings and recommendations were published on August 1, 1995 in its report *Technical Bases for Yucca Mountain Standards*. These standards will apply only to Yucca Mountain.

2.5 NUCLEAR REGULATORY COMMISSION

The NRC was created as an independent agency by the Energy Reorganization Act (ERA) of 1974 (ERA74), which abolished the AEC and moved the AEC's regulatory function to NRC. This Act, coupled with the AEA, as amended, provided the foundation for regulation of the nation's commercial nuclear power industry. NRC regulations are issued under the U.S. Code of Federal Regulations Title 10 Chapter 1.

The mission of the NRC is to ensure adequate protection of public health and safety, the national defense and security, and the environment in the use of nuclear materials in the United States. The NRC's scope of responsibility includes regulation of commercial nuclear power reactors; nonpower research, test, and training reactors; fuel cycle facilities; medical, academic, and industrial uses of nuclear materials; and the transport, storage, and disposal of nuclear materials and waste. In addition to licensing and regulating the use of by-product, source, and special nuclear material, the NRC is also responsible for assuring that all licensed activities are conducted in a manner that protects public health and safety. The NRC assures that none of the operations of its licensees expose an individual of the public to more than 100 mrem/yr from all pathways (NRC91).

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. These facilities are prohibited from releasing radioactive effluents in amounts that would result in doses greater than the 25 mrem/yr limit imposed by that standard. Currently, NRC-licensed facilities are also required to operate in accordance with the requirements of the CAA (40 CFR Part 61), which limits radionuclide emissions into the air (EPA89b).[6]

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.

2.5.1 Fuel Cycle Licensees

The NRC licenses and inspects all commercial fuel cycle facilities involved in the processing and fabrication of uranium ore into reactor fuel. NRC 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 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.

2.5.2 Radioactive Waste Disposal Licenses

The NWPA, as amended, specifies a detailed approach for high-level radioactive waste disposal. DOE has operational responsibility and the NRC has licensing responsibility for the transportation, storage, and geologic disposal of the waste. The disposal of high-level radioactive waste requires a determination of acceptable health and environmental impacts that may occur over a period of thousands of years. Current plans call for the ultimate disposal of waste in solid form in a licensed, geologic disposal system. The NWPA, as amended, designates Yucca Mountain, Nevada as the candidate site for the high-level waste repository.

The EnPA provides additional direction to the NRC as to its role in the licensing of a specific disposal site located at Yucca Mountain. Section 801 of the EnPA requires the Commission to modify its technical requirements and criteria under section 121(b) of the NWPA of 1982, as necessary, to be consistent with EPA's standards for the Yucca Mountain site. The NRC's

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[6] Pursuant to Section 112(d)(9) of the CAA Amendments of 1990, EPA is proposing to rescind Subpart I as it applies to NRC-licensed facilities. The NRC is proposing to adopt a constraint level rule which would limit radionuclide airborne emissions to 10 mrem/yr.

requirements and criteria shall assume that engineered barriers and post closure oversight provided by the DOE will be sufficient to: 1) prevent any activity at the site that poses an unreasonable risk of breaching the repository's engineered or geological barriers and 2) prevent any increase in the exposure of individual members of the public to radiation beyond allowable limits (EnPA92). In order to meet the requirements of the EnPA, the NRC is prepared to revise, or if appropriate, draft new sections to its existing 10 CFR Part 60 regulations.

NRC regulations governing deep geologic disposal are contained in 10 CFR Part 60 entitled *Disposal of High-level Radioactive Wastes in Geologic Repositories*. These regulations are summarized below. In addition, the NRC promulgates (under 10 CFR Part 71) packaging criteria for the transportation of spent nuclear fuel and high-level and transuranic radioactive wastes. Under 10 CFR Part 72, the NRC licenses independent facilities for storage of spent nuclear fuel.

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 EIS in order to obtain a license to construct a repository (NRC81, NRC83). The EIS must meet the requirements of 10 CFR Part 51, *Environmental Protection Requirements for Domestic Licensing and Related Regulatory Functions*.

The SAR must 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, an assessment of the waste isolation properties of the proposed site, and a description of the engineered features of the repository. A program of site characterization field work is required to support the preparation of the SAR. DOE's general plan for the program of characterization is presented in its Site Characterization Plan (SCP) (DOE88a). This plan contains the description of the studies to be conducted, their sequencing and possible interference, and the impact of the studies on the ability of the site to isolate and contain the waste. Before beginning site characterization, the SCP received extensive review by the NRC, the State of Nevada, and other interested parties. Progress during site characterization and any changes to the plans for site characterization are reported in semiannual progress reports that are also reviewed by the NRC and other interested parties. The SAR will be prepared using the information developed during site characterization and using the results of engineering design studies for the repository.

Upon receipt of the SAR, the NRC will conduct a safety review. The planned repository will be evaluated against the technical criteria specified in 10 CFR Part 60. The NRC will determine whether: (1) there is reasonable assurance that performance of a repository at the site will be in compliance with applicable radiological protection standards; (2) there is reasonable assurance that the activities proposed in the application do not jeopardize national defense and security; and (3) after weighing the environmental, economic, and technical benefits against environmental costs and considering available alternatives, construction authorization is appropriate. Any authorization issued by the NRC may be contingent on DOE meeting a number of conditions deemed necessary to protect the nation's environmental goals.

After construction has been completed, DOE will update the SAR and EIS. This information will be reviewed by the NRC to determine if an operating license to receive, possess, and dispose of nuclear waste should be granted. At this stage, the NRC will confirm that construction has been completed in conformance with the construction authorization provisions and that the repository does not pose an unreasonable risk to public health and safety. At the end of the operating period, DOE will submit an application to amend its operating license. This application and the associated revised information will be reviewed by the NRC to determine if the repository may be permanently closed.

At each stage of the licensing process, the SAR is reviewed to determine if the technical criteria specified in Subpart E of 10 CFR Part 60 are satisfied. These technical criteria include performance objectives and other requirements, such as land ownership and control, siting, and design criteria, intended to ensure that 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, levels, and releases conform to 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 waste. Other requirements deal with land ownership and waste package design criteria.

In general, the performance objective for high-level waste repositories provides for protection against radiation exposures and releases by requiring that the repository be designed so that radiation exposures, levels, and releases to unrestricted areas meet the applicable environmental standards contained in Subpart A of 40 CFR Part 191. However, a high-level waste repository built at the Yucca Mountain site must meet site-specific environmental standards contained in 40 CFR Part 197, to be developed by EPA, pursuant to its authority under the EnPA of 1992.

2.5.3 Repository Licensing Support Activities

The current NRC repository licensing program consists of both proactive and reactive activities. Proactive activities include developing and reviewing regulatory requirements and guidance to identify and resolve regulatory and technical 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.

The NRC staff is currently developing and implementing performance assessment models using Yucca Mountain site data. The models will assist the NRC in performing a technical assessment of the site, as well as identifying areas of regulatory and technical uncertainty during the license application review process. The uncertainties identified must be addressed in a timely fashion so that the NRC can meet the three-year license review schedule mandated by Public Law 97-425 (NWP83).

These activities have produced licensing review plans in anticipation of the DOE submissions. They include review of the SCP, Study Plan, and Quality Assurance Plan (QAP).

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

Reactive activities of the repository licensing program consist of pre-licensing reviews that follow DOE's sequence and schedule of activities. To date, the NRC has reviewed a number of the QAPs proposed by DOE and its contractors for Yucca Mountain. Any quality assurance issues identified must be resolved before significant data collection activities are performed at the Yucca Mountain site.

The next major activity planned during the licensing process for Yucca Mountain will involve the NRC's review of DOE's strategies, assumptions, and programs. The NRC also plans to conduct a completeness review of the more detailed Study Plans. However, only a sample (about 20 percent) of the approximately 100 Study Plans will be reviewed in this manner. During the site characterization phase, 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. The review 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 results of work performed to date, and topical and issue resolution reports that summarize the site characterization work for specific licensing topics. These reviews will be used to evaluate compliance with NRC regulations.

All concerns identified by the NRC will be tracked by its staff. The tracking system now being implemented will focus not only on the issues identified, but also on DOE's progress towards their resolution. The system also provides a licensing record of all NRC and DOE actions related to resolving specific issues.

2.6 DEPARTMENT OF ENERGY

The DOE operates facilities for the production and testing of nuclear weapons; the management and disposal of radioactive waste generated in national defense activities; research and development; and for the storage of spent nuclear fuel. 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 from the generation of electricity by commercial nuclear reactors and high-level radioactive waste from defense activities. The facilities developed by the DOE for the management and disposal of these wastes must be licensed by the NRC. The Yucca Mountain site in Nevada is the candidate location for disposal of these wastes.

DOE is responsible for operating its facilities in a manner that is environmentally safe and sound, as stated in DOE Order 5400.1 (DOE88b). In meeting this mandate, DOE has issued a number of orders specifying environmental standards and procedures. Many of these orders are currently under review to determine their conformance with NRC and EPA regulations and standards and will be revised in accordance with the applicable NRC or EPA guidance. Key DOE orders pertaining to the management of radioactive and hazardous materials include:

- DOE Order 1540.2 (DOE86), which establishes administrative procedures for the certification and use of radioactive and other hazardous materials packaging by the DOE.
- 1540.3A (DOE92), which specifies DOE's policies and responsibilities for coordinating and planning base technology for radioactive material and transportation packaging systems.
- DOE Order 5480.3 (DOE85), which establishes requirements for the packaging and transportation of hazardous materials, hazardous substances, and hazardous wastes.
- DOE Order 5440.1E (DOE91), which establishes procedures for implementing the requirements of the NEPA (NEP70). The Order requires new facilities and existing facilities with proposed modifications to submit EISs with their proposed facility design or design modification. In addition, the facilities are subject to extensive design criteria reviews to determine compliance.

In addition to the above Orders, in March 1993, DOE published a Notice of Proposed Rulemaking for 10 CFR Part 834, entitled *"Radiation Protection of the Public and the Environment"* (58 FR 16268) (DOE93). The proposed rule contains DOE's internal primary standards for the protection of the public and environment against radiation. The requirements would be applicable to control of radiation exposures from normal operations under the authority of DOE and DOE contractor personnel.

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CHAPTER 3

HIGH-LEVEL WASTE DISPOSAL PROGRAMS IN OTHER COUNTRIES

As in the United States, countries that are committed to use nuclear power or in which nuclear power already makes up a significant fraction of their total electrical generating capacity are establishing long-term programs for the safe management and disposal of spent reactor fuel and high-level radioactive waste. Such programs include adopting a national strategy, assigning the technical responsibility for research and development activities to designated agencies, selecting disposal strategies and development activities, and setting the appropriate regulatory standards to protect public health and the environment. Management strategies may include spent fuel storage at and away from reactor sites, spent fuel reprocessing, high-level waste vitrification and storage, and ultimate high-level waste disposal in deep geologic media. Typically, the objective of such geologic disposal programs is to immobilize and isolate radioactive waste 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.

Some of the nations with major commitments to nuclear power have begun activities concerned with disposal of high-level waste by isolation in a deep geologic formation. No nation other than the United States plans to dispose of intact reactor spent fuel; all anticipate that at some time in the future, perhaps as much as 50 years, the spent fuel will be reprocessed to recover fissile materials and the high-level waste containing fission products will be solidified using a vitrification process similar to that planned for use with defense-related high-level waste in the United States (see Chapter 4 of this BID). Disposal of the solidified high-level waste would then be accomplished by emplacement in a deep geologic formation within the nation's borders.

Further, no other nation has at this time identified a specific candidate location for disposal of high-level waste, analogous to the U.S. candidate Yucca Mountain site in Nevada. Other countries are, to varying degrees, engaged in technical evaluations of the potential suitability of indigenous geologic formations for disposal. Some nations, such as France, have alternative geologic formations, such as clay and granite, that might be used for disposal, and those alternatives are being evaluated. Others, such as Canada, have only one type of geologic formation considered potentially suitable for disposal, and its evaluations are focused on that geology. In addition, several countries, such as Canada and Sweden, have established underground research facilities and extended their research programs to include participation by other nations with similar candidate geologies. For example, the Swedish research facility is in a crystalline rock formation and its research program includes participation by Japan.

The disposal strategies for all nations include the expectation that waste isolation will be maintained by reliance on a combination of engineered and natural barriers between the emplaced waste and the environment. Countries other than the United States have to date focused their efforts on characterizing the natural barriers provided by varying geologic formations and have not yet developed detailed engineered barrier design concepts such as the options under consideration in the U.S. (see Chapter 7). However, in response to a mandate from the national government in the 1970s, the Swedish commercial nuclear waste program developed an engineered barrier concept involving emplacement of spent fuel in a copper matrix contained within a highly-robust copper canister. The viability of this concept to maintain wastes in isolation for one million years in Sweden's geologic formations was then demonstrated, as required by the governmental directive. Sweden has not, however, committed to the use of this engineered barrier concept. The actual engineered barrier design selected in Sweden, as in other nations, will be based on findings concerning the performance of natural barriers provided by the geologic formations.

Various nations and international agencies, in addition to the United States, have begun to give consideration to regulations and regulatory standards for high-level waste disposal. Some nations have developed broad risk or dose criteria, and some have supplemented such criteria with additional qualitative technical criteria concerning features of the disposal system. International organizations such as the Nuclear Energy Agency in Paris, France, provide opportunities for discussion of regulatory criteria and also provide programs of common interest, such as comparison of performance assessment computer codes. There are, however, no international standards for high-level waste disposal accepted by all nations.

Although the performance standards and criteria for the various national regulations are similar, each nation has established specific requirements to meet its needs. Current information concerning the provisions of national and international criteria and objectives for the safety of long-lived radioactive waste disposal is presented in Table 3-1. Regulatory requirements are still evolving, and the information Table 3-1 is subject to change.

Table 3-1. National and International Criteria and Objectives for the Disposal of Long-Lived Radioactive Wastes (OEC95a)

Organization/ Country/ Reference	Main Objective/ Objective/Criteria	Other Main Features	Criteria for Judging Human Intrusion (HI) Scenarios	Comments
NEA (1984) [7]	For HLW: max. indiv. risk $< 10^{-5}$/y (all sources)		Indiv. risk/dose - best criterion to judge long-term acceptability	No consensus on ALARA/ optimization
ICRP (Pub. 46, 1985) [8]	For HLW, for individuals (all sources): 1 mSv/y (normal evolution scenarios); 10^{-5}/y (probabilistic scenarios)	Both probability and dose should be taken into account in ALARA	Future human activities should be treated probabilistically	ALARA useful, notably to compare alternatives, but may not be the most important siting factor
IAEA (Safety Series 99, 1989) [9]	Idem ICRP Publication 46		Future human activities are randomly disruptive events that usually are examined probabilistically	Also includes qualitative technical criteria on disposal system features and role of safety analysis and quality assurance
CANADA (Reg. Document R-104, 1987) [3]	For HLW: max. indiv. risk objective: $< 10^{-6}$/y	Period of time for demonstrating compliance: 10^4y No sudden and dramatic increase for times $> 10^4$y	Main criteria applicable to all exposure scenarios; no criteria specific to HI scenarios	Additional qualitative, non-prescriptive requirements and guidelines in regulatory documents
FINLAND (Decision of the Council of State, 1991) [10]	For LLW and ILW: max. indiv. dose < 0.1 mSv/y, with max. indiv. dose < 5 mSv/y from accident conditions caused by possible natural events or human actions		Max. indiv. dose < 5 mSv/y from possible human actions	For spent fuel or HLW, proposed criterion for max. indiv. dose < 0.1 mSv/y
FRANCE (Basic Safety Rule, RFS III.2.f, 1991) [11]	For ILW and HLW: max. indiv. dose < 0.25 mSv/y for normal evolution scenarios; for altered evolution scenarios, risk may be considered (probability of scenario times effect of exposure)	Beyond 10^4 y, dose limit is considered as a "reference" level	Assumptions (French Basic Safety Rule, Appendix 2): Date of HI occurrence > 500 y; Existence of repository and location forgotten; Level of technology same as present day	Technical criteria for siting established in 1987
GERMANY (Section 45, para. 1 of Radiation Protection Ordinance, 1989) [12]	For all waste types: max. indiv. dose < 0.3 mSv/y for all reasonable scenarios	Calculation of individual doses limited to 10^4y but isolation potential beyond 10^4y may be assessed		Additional qualitative technical criteria in guidelines and regulatory document

Table 3-1 (Continued)

Organization/ Country/ Reference	Main Objective/ Objective/Criteria	Other Main Features	Criteria for Judging Human Intrusion (HI) Scenarios	Comments
NORDIC COUNTRIES (Basic Criteria Document, 1993) [13]	For all waste types: max. indiv. dose < 0.1 mSv/y (normal scenarios): max. indiv. risk $< 10^{-6}$/y (disruptive events)	For HLW, additional criterion on "total activity inflow" limiting releases to biosphere, based on inflow of natural alpha radionuclides		Includes other qualitative criteria
SWITZERLAND (Reg. Document R-21, 1993) [14]	For all waste types: max. indiv. dose < 0.1 mSv/y at any time for reasonably probable scenarios; max. indiv. risk $< 10^{-6}$/y for unlikely scenarios	Repository must be designed in such a way that it can at any time be sealed within a few years without the need for institutional control	No criteria for HI scenarios except that for high consequences, probabilities can be taken into account	
UNITED KINGDOM [15]	For L/ILW: 10^{-6}/y target for indiv. risk from a single facility For HLW: no specific criteria but likely application of principles similar to existing objectives for L/ILW	No time frame for quantitative assessments specified	Main criterion for HI scenarios currently indiv. risk	ALARA to be used to the extent practical and reasonable

Characteristics of programs in eight nations with major commitments to nuclear power and existing activities concerning disposal of high-level wastes are summarized below. The descriptions address nuclear power utilization, waste disposal, and regulatory programs. Discussions are provided for programs in Belgium, Canada, France, Germany, Japan, Sweden, Switzerland, and the United Kingdom.

3.1 BELGIUM

3.1.1 Nuclear Power Utilization

In 1994, Belgium met about 56 percent of its electrical needs through nuclear power (EIA95). The Belgian nuclear power program relies on seven pressurized light water reactors, all of which are operated by Electrabel, a privately owned company.

From 1966 to 1974, Belgium reprocessed spent fuel at its Eurochemic facility. The company, Belgoprocess, was created in a consortium with foreign firms to reactivate the Eurochemic plant, but these efforts failed in the mid-1980s. Belgoprocess is now responsible for decommissioning the plant. Belgium currently ships spent fuel to France for reprocessing and stores it in reactor pools pending such shipments. By 1996, France will be returning to Belgium the high-level vitrified waste created from the reprocessing of Belgian fuel. It is estimated that by the year 2000, Belgium will have produced about 2,500 MTHM of spent fuel.

In 1985, a vitrification plant, PAMELA, began processing high-level waste from the Eurochemic plant. Vitrified high-level waste will be stored in an intermediate storage facility (recently constructed by the National Agency for Radioactive Waste and Fissile Materials, (ONDRAF)) for 50-70 years. Characterization of a potential site for a repository located in a deep clay formation at the Mol-Dessel site in the northeast corner of the country is progressing.

3.1.2 Disposal Programs and Management Organizations

The Belgian program to establish a radioactive waste repository was initiated in 1974 with the establishment of a government-sponsored research and development initiative. In 1982, the National Agency for Radioactive Waste and Enriched Fissile Material, ONDRAF, was established to implement and manage a multi-year national program addressing the long-term management and disposal of radioactive wastes, including spent fuel, high-level waste, and other reprocessing waste returned from French facilities. SYNATOM, an agency privatized in 1994, is responsible for uranium procurement, reprocessing spent fuels, off-site waste management, and disposal of packaged irradiated fuel assemblies. The Nuclear Research Center (CEN), under the Ministry of Economic Affairs, provides technical assistance in basic and applied R&D in nuclear energy and technology.

Belgium's waste disposal program takes a multi-barrier approach, relying significantly upon natural barriers. Engineered barriers are anticipated to last no more than a few thousand years, after which the surrounding barriers will provide primary containment. For this reason, engineered barriers have been designed to limit impact to surrounding geology, as much as to provide public health protection (NWT94).

ONDRAF intends to begin operation of a shallow land burial facility for low-level waste by the year 2000 and has established an underground laboratory in a deep clay formation at Mol-Dessel to evaluate the site's suitability as a high-level waste repository. Twenty years of research at Mol-Dessel has led to significant evolution of the design for the planned repository. The clay formation in which the site is situated is the only suitable geological medium that has been identified in Belgium. In 1980, a repository conceptual design was developed for a clay site, and, by 1985, an underground research laboratory at Mol-Dessel (Project HADES) began operation. In recent years, the repository's design has been altered. The original HADES design was conceived according to the perceived thermal tolerance of the host rock and was intended to allow for retrieval of vitrified waste containers for a long period of time. The new design does not permit easy retrieval and allows for more homogeneous dispersion of heat. The new design is also believed to be simpler to construct and less damaging to the surrounding clay layer (NWT95).

The underground research laboratory at Mol-Dessel extends to a depth of 224 meters. Research conducted there includes 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. Over the next few years a preliminary demonstration test for clay disposal (PARCLAY) will be launched at the Mol-Dessel site. PARCLAY will be used to investigate the thermal effects of final disposal on clay and will include the construction of a new 1:1 scale gallery. Based on the outcome of these studies, a larger underground facility might be constructed for a full-scale demonstration project. Assuming that the results of investigations at Mol-Dessel are favorable, repository construction could begin around 2025 and operation around 2030 (OEC95b).

3.1.3 Regulatory Organizations and their Regulations

Belgium does not currently have specific regulatory requirements or criteria governing the disposal of spent fuel or high-level waste. In 1994, the Federal Nuclear Inspection Agency (AFCN) was created to oversee inspection and surveillance of Belgium's nuclear facilities under guidance of the Ministry of Employment and Labor and the Ministry of Public Health and the Environment. The King of Belgium has the authority to grant, suspend, reject or withdraw authorization for the construction and operation of nuclear facilities (OEC95c).

3.2 CANADA

3.2.1 Nuclear Power Utilization

In 1994, Canada produced about 19 percent of its electrical needs through nuclear power (22 pressurized heavy-water cooled and moderated "CANDU" reactors (EIA95)). Canadian utilities currently produce a surplus of electric power, and only one new nuclear facility is planned before the year 2005 (OEC95a). Canada's nuclear power is produced by three provincial utilities, Ontario Hydro, Hydro Quebec, and New Brunswick Power.

Canada relies on the CANDU reactor design, which operates using natural uranium in a once-through fuel cycle. Currently, the program considers only direct disposal of spent fuel 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. Estimates indicate that Canada will have produced about 34,000 MTHM of spent fuel by the year 2000. The country's existing five sites (the Ontario Hydro utility has three sites) have adequate storage facilities for spent fuel, and there is little urgency to dispose of waste. Current time lines suggest that a disposal facility could be established by about 2025 (OEC95c).

3.2.2 Disposal Programs and Management Organizations

There is no national legislation specific to nuclear waste management in Canada. Under Canada's nuclear waste management policy, the producer/owner of waste bears responsibility and must ensure regulatory compliance. On behalf of the government of Ontario, Ontario Hydro (owner of 20 out of 22 total nuclear power units) is responsible for the development of technology for storage and transportation of spent fuel. Atomic Energy of Canada Limited (AECL), a Crown corporation reporting to Canada's Ministry of Natural Resources, is responsible for disposal of the spent fuel.

The country's nuclear waste management program, established in 1978, is currently under Federal review and assessment, and no site selection activities will occur until this review is complete. The review began in 1988. In 1994, an Environmental Impact Assessment (EIA) evaluating planned disposal programs was submitted to the Federal Environmental Assessment

Review Panel. A response to the EIA based upon the recommendations of the Panel is expected in 1996 (GAO94), and the Federal government will determine subsequent steps in the country's waste management program. AECL estimates that siting, licensing, and construction of a disposal facility will take 25 to 30 years and that the facility could, therefore, be in operation by 2025.

The Canadian disposal concept involves siting a repository in a granitic formation located in the Canadian Shield, a large region wrapped around the Hudson Bay stretching east to Labrador, south to Lake Ontario, and northwest to the Arctic Ocean. The repository would be located at depths of 500 to 1,000 meters. Spent fuel canisters would be inserted into floor cavities located in excavated disposal rooms and surrounded with a mix of bentonite and silica sand. A mix of glacial rock clay and crushed granite aggregate, along with engineered barriers, would be used to seal most of the remainder of the vault (OSC95).

In 1986, AECL established the Whiteshell underground research laboratory in undisturbed granitic rock at a depth of 240 meters at Lac du Bonnet in the Province of Manitoba. AECL has since deepened the facility to 440 meters. The purpose of the laboratory is to conduct large-scale, in-situ experiments in the type of rock envisioned under the Canadian disposal concept, demonstrating some of the components of the disposal concept (the facility is not a candidate repository site). The AECL is developing methodologies and analytical techniques to evaluate the geomechanical and geohydrological properties of granitic rock. The underground research laboratory was also recently used to study the possible effects of microbial activity in a disposal system. Other studies conducted at the laboratory include large block migration studies, container corrosion studies, and an alternate post-closure assessment case study.

3.2.3 Regulatory Organizations and their Regulations

The Atomic Energy Control Board (AECB) is the lead regulatory agency for assessing 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 high-level waste repository. Other provincial and Federal agencies operate under AECB in the regulation of some activities in the nuclear fuel cycle.

In 1987, the AECB issued a regulatory policy statement containing objectives, requirements, and guidelines on nuclear waste disposal and high-level waste repository siting (AECB87). The overall regulatory objective expressed in these documents is to ensure that there is only a small probability that radiation doses to the public associated with the repository will exceed a small fraction of natural background radiation doses. Under these regulations, predicted radiological risk to of death individuals from a waste repository must not exceed 1 x 10^{-6} annually. For the purpose of demonstrating compliance with the individual risk requirement, the time period need not exceed the first 10,000 years (OEC95a).

3.3 FRANCE

3.3.1 Nuclear Power Utilization

In 1994, France met approximately 75 percent of its electrical needs through nuclear power, having the highest per capita installed capacity in the world (OEC95c). The French nuclear power program relies on 56 units, the vast majority of which are light water reactors. 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 plans to continue building nuclear power plants, although some will act as replacements for old facilities. The overall contribution of nuclear power to the country's electricity production is not expected to exceed 80 percent (GAO94).

The French radioactive waste disposal program is based on a closed fuel cycle involving spent fuel reprocessing and recovery and re-use of plutonium in breeder and light water reactors. From 1976 through 1990, France reprocessed over 20,000 MTHM and oxide fuel. France has already begun to solidify high-level waste in glass and, ultimately, intends to dispose of it—as well as alpha-emitting transuranic waste (alpha waste)—in deep geological formations. Vitrification plants for France's two reprocessing plants, UP2 and UP3, entered service in 1990 and 1992. It is estimated that by the year 2000, France will accumulate about 3,000 cubic meters of high-level waste and 47,000 cubic meters of alpha waste. France also provides reprocessing services to foreign customers; in 1993, the international component comprised an estimated one-third of the country's reprocessing business.

3.3.2 Disposal Programs and Management Organizations

The French nuclear waste program has been entrusted to the National Radioactive Waste Management Agency (ANDRA). ANDRA was formed in 1979 as an arm of the French Atomic Energy Commission (CEA), but 1991 legislation made it an independent entity. ANDRA is responsible for all radioactive waste disposal activities and long-term waste management. Other organizations with key roles in the management of the country's high-level waste include Electricite de France (the national electric utility) and COGEMA (operator of spent fuel reprocessing and high-level waste immobilization and storage facilities).

In 1987, ANDRA identified four geological media for potential high-level waste disposal—clay, salt, granite, and schist—and, since then, has begun investigative work at a site in each medium. An underground research laboratory was to be established at one or more of the candidate sites; if found suitable, one of these was to have been converted to an operating repository to receive transuranic waste by 2000 and high-level waste by 2010. However, in light of the serious public protests at three of the sites under investigation, former Prime Minister Michel Rocard declared a one-year moratorium on siting activities to allow a reassessment of France's overall waste management strategy. The moratorium began in February of 1990, and, in January 1991, the Parliamentary Office for the Assessment of Technological Options published a report that recommended major changes to the program.

On December 30, 1991, the Parliament enacted a new Law on Radioactive Wastes. The 1991 law requires the government to submit a report to Parliament within 15 years that assesses the results of studies on partitioning and transmutation of actinides, the use of test facilities for retrievable and permanent storage of high-level waste, and the technologies for waste conditioning and surface storage. The report must also propose a bill authorizing an underground waste repository. At this time, no schedule has been set for developing such a repository. Instead, the Parliament will reassess the program based on the results of the 15-year research phase. The law states that, once the underground research laboratory is built, only research-level quantities of waste may be emplaced into it until the Parliament votes to convert the laboratory into a repository. While no direct disposal of spent fuel is envisioned, the law also requires that the government perform research on direct spent fuel disposal options.

The new bill allowed the government to resume site selection efforts for underground research laboratories. A waste "negotiator" was appointed to discuss proposed investigations with local and regional officials. About 30 localities subsequently expressed interest in hosting a laboratory. In 1994, the government's Bureau of Geological and Mineral Research (BRGM) investigated these regions and eliminated those with adverse geology. In early 1994, the Negotiator announced the selection of four new regions as candidates to host a repository: 1) the southern region of the Vienne "departement," in west-central France; 2) the area surrounding Marcoule in the Gard departement; 3) the Meuse departement, bordering south-eastern Belgium; and 4) the northern Haute Marne departement, north of Dijon. Two other localities were selected as secondary choices because their local governments had not voted on their candidacy; the primary four localities all voted in favor of their candidacy.

Whereas ANDRA had previously considered four types of potential host rock, the four selected regions represent only two: clay and granite. Additional research on suitability has been underway since their selection. ANDRA is currently working to identify suitable site locations within each region. It is expected that two sites will be selected by early 1996 and that a license application for underground research laboratories will follow shortly thereafter (OEC95b). Operation of a repository is not expected before 2020.

The 1991 law includes additional provisions designed to ease public concern about France's high-level waste management program, including the creation of a policy of openness concerning the country's high-level waste disposal program and a requirement that government grants and jobs for the host municipality accompany the underground research laboratories.

3.3.3 Regulatory Organizations and their Regulations

Agencies with regulatory responsibilities include the Directorate for the Safety of Nuclear Installations (DSIN) within the Ministry of Industry; the CEA and its subsidiary, the Institute for Nuclear Protection and Safety (IPSN); the BRGM; and SGN (architect and engineering services).

DSIN, France's principal nuclear regulatory authority, issued "Fundamental Safety Rule III.2.f." (DSI91) pertinent to high-level and alpha waste disposal, on June 10, 1991. The rule requires that:

- the impact of a deep geologic disposal facility be as low as reasonably achievable

- individual dose equivalent due to the facility be limited to 0.25 mSv (25 mrem) per year for likely events

- the stability of geologic barriers be demonstrated for at least 10,000 years; and

- high-level waste packages prevent the release of radioactive contents during the period when short- and medium-lived radionuclides dominate total radioactivity

In preparation for the underground laboratory phase, the IPSN, within CEA, is independently preparing facilities to evaluate the long-term safety requirements of a repository on behalf of the regulatory authority DSIN.

3.4 GERMANY

3.4.1 Nuclear Power Utilization

In 1994, Germany satisfied about 30 percent of its electrical needs through nuclear power (EIA95). The German nuclear power program relies primarily on pressurized light water reactors (14 units) and boiling water reactors (7 units), although research and development activities and demonstration projects are also evaluating alternate reactor designs (high temperature gas-cooled reactors and liquid metal fast breeder reactors) for power production. It is estimated that by the year 2000, Germany will have generated about 9,000 MTHM of spent fuel. Germany has historically planned to dispose of spent fuel in deep geological formations only after reprocessing, as stipulated in a 1976 amendment to Germany's Atomic Energy Law. Plans for a domestic reprocessing facility were abandoned in 1989 and German utilities chose instead to ship their spent fuel to France and Britain for reprocessing. Resulting vitrified waste is currently returned to Germany and stored in metal casks for planned subsequent disposal. A 1994 amendment to its Atomic Energy Law, however, legalized the *direct* disposal of spent fuel elements as well. German utilities are currently considering both management options (Atomic Energy Law, Article 4, amendment of section 9a(1)) (GER94).

3.4.2 Disposal Programs and Management Organizations

The German government's Institute for Radiation Protection (BfS) is responsible for the design, construction and operation of waste disposal facilities. Vitrified high-level waste returned from foreign reprocessors will eventually be disposed at the Gorleben facility, a salt dome located in Lower Saxony, if the site proves acceptable. Under current plans, spent fuel will also be directly disposed at Gorleben. The newly legalized option of direct disposal has required modification of plans at the facility. Until a repository is in operation, vitrified waste will be stored at Gorleben and the Ahaus facility in Northrhine-Westphalia. Expansions at both the Gorleben and Ahaus storage facilities are currently proposed. A former salt mine at Asse, which served until 1978 as a repository for low-level (125,000 containers) and intermediate-level (1,300 drums) radioactive wastes, now serves as an underground research laboratory for high-level waste disposal.

Federal agencies are generally positive about the suitability of Gorleben as a repository site. However, while research to date appears to generally support the suitability of the Gorleben site, the project has faced increasing opposition from the government of Lower Saxony. As of June 1995, the BfS was seeking 30 million Deutschmarks from the state mining authorities (who act on behalf of the state Ministry of the Environment) in compensation for delays allegedly caused by their opposition. In addition, conservation laws of Lower Saxony required the planting of thousands of trees and bushes to conceal the site of the headframe of two shafts (OEC95b).

The Gorleben facility will be situated at depths ranging from 250 to 3,000 meters. The geology of the site has been widely investigated by exploratory drilling and by geophysical measurements. Construction of an underground research laboratory was initiated in 1986, but all work was stopped for over a year in 1987 because of a construction fatality. As of 1995, two shafts had been sunk to depths of 600 and 620 meters (emplacement at approximately 870 meters is anticipated (LOM95)). Current areas of emphasis include hydrogeological investigation and seismic measurements (OEC95b). Construction of the repository could start at the turn of the century, and the facility is scheduled to remain operational for about 50 years. It is anticipated that the site will receive about 550 MTHM of vitrified waste, 200 metric tons of directly disposed spent fuel, and about 6,690 containers of low-level and intermediate-level waste per year (LOM95).

Repository design emphasizes the role of the surrounding geology as a barrier. It is anticipated that the salt dome's formations will move over time and encapsulate the waste. The use of steel and iron canisters is intended primarily to contain waste in the short-term. The possibility of direct spent fuel disposal has required additional research and design development. The German program is investigating drift emplacement of heavily shielded Pollusk casks for directly-disposed spent fuel (OEC93).

3.4.3 Regulatory Organizations and their Regulations

Key German agencies include the Federal Ministry for Environment, Protection of Nature and Reactor Safety (BMU), the Federal Ministry for Research and Technology (BMFT), the BfS, the Federal Institute for Geosciences and National Resources, and the host state's ministry for environmental protection. As the primary federal supervisory authority, BMU receives advice from two committees of independent experts, the Reactor Safety Commission (RSK) and the Committee on Radiological Protection (SSK).

In Germany, the institutional and legal framework for the regulation of nuclear facilities is based on the joint participation of Federal and state governments. State governments serve as licensing authorities for all nuclear waste facilities, although the Federal government has the authority to override these decisions. The Federal government retains primary responsibility for waste disposal; the Atomic Energy Law and the Radiation Protection Ordinance (GER94) establish the principles and requirements regarding the safe utilization and application of atomic energy and radioactive materials, including the disposal of radioactive waste. Under the Radiation Protection Ordinance, dosage limits are set at 0.3 mSv (30 mrem) per year for "all reasonable scenarios" (OEC95a).

German regulators are developing safety regulations that the Gorleben facility will be required to meet through a site-specific safety assessment. It is expected that this safety assessment will be required to demonstrate that potential exposure to radiation from disposed waste will be kept within the range of natural radiation for a period of about 10,000 years and that integrity of the repository system will be maintained over a longer period of time (GAO94).

3.5 JAPAN

3.5.1 Nuclear Power Utilization

In 1994, Japan produced about 31 percent of its electrical needs through nuclear power provided by 49 reactors (EIA95). It is anticipated that this figure will increase to approximately 33 percent by the year 2000 and to about 42 percent by the year 2010 (AEC94). The Japanese nuclear power program currently relies primarily upon light water reactors, although 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). By 1994, the country's first prototype fast breeder reactor, Monju, had reached criticality (OEC95b), but a December, 1995 coolant leak dealt a setback to the project.

Japan's spent fuel is currently reprocessed in France and England. However, both countries have exercised their option to return vitrified residue to Japan; the first return delivery from France took place in February, 1995. Domestically, the Power Reactor and Nuclear Fuel Development Corporation (PNC) has operated a small reprocessing plant since 1977, where roughly 720 tons of spent fuel had been reprocessed as of 1993. Furthermore, at the Rokkasho site in Aomori prefecture, a private utility consortium, Japan Nuclear Fuel Services Limited (JNFL), plans to begin operating a large commercial-scale plant shortly after the year 2000 (AEC94). It is estimated that by the year 2000, Japan will have discharged about 20,000 MTHM of spent fuel from its reactors. Vitrified high-level waste will be stored 30 to 50 years for cooling before disposal in a geologic repository.

3.5.2 Disposal Programs and Management Organizations

As noted above, Japan's 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, and eventual disposal in a suitable deep geological formation. Japanese nuclear utilities are responsible for storing high-level waste and funding its disposal; JNFL is responsible for low-level waste disposal activities at Rokkasho (OEC95c). Two government-sponsored organizations—PNC and the Japan Atomic Energy Research Institute (JAERI)—are responsible for research and development addressing the fuel cycle, waste management, and disposal. In 1993, the Steering Committee on High-Level Radioactive Waste (SHP) was created to spearhead planning for disposal of the country's high-level waste.

Radioactive waste is managed in accordance with Japan's Long Term Program for the Development and Utilization of Nuclear Energy (AEC94). In 1994, the Atomic Energy Commission (AEC) issued the most recent update to the long-term disposal plan, placing particular emphasis on the disposal of high-level waste and adding new details to the country's plans and timetables for this effort. The 1994 update establishes a procedure for implementation of a deep geologic repository and provides guidelines on storage, vitrification, and geologic disposal. The plan also adds new clarity to the roles of Japan's nuclear-related organizations, which can be summarized as follows:

- Government Research and Development Organizations (PNC and JAERI): PNC is the lead organization implementing the research and development program in various areas of the fuel cycle and geologic disposal, while JAERI performs research in support of the government's safety evaluation of geological disposal, as well as research on advanced waste management technologies.
- Utilities and their Consortia: Utilities are responsible for funding high-level waste disposal programs and for contributing to related research and development work.
- Government Agencies: Government agencies are responsible for oversight and overall coordination of disposal. While it has not yet been decided what entity will implement or license the disposal project, the AEC's 1994 update of the country's long-term disposal plan suggests that this duty will be delegated by the year 2000 and that SHP is responsible for studying the matter (SHP has not been designated the implementing entity).

The 1994 plan also lays out a five-step process to develop a high-level waste repository. The first phase, selection of effective formations, was completed in 1984. Subsequent steps as established by the AEC include: 1) establishment of an implementing organization by the year 2000; 2) selection of candidate disposal sites, subject to government cooperation and community acceptance; 3) demonstration of disposal technology at the candidate site, followed by license application; and 4) establishment of necessary laws and policies for the disposal implementation and safety. The plan calls for the repository to be operational by 2030 to 2045. Japanese authorities have determined that high-level waste disposal should be possible in any geologic formation excluding unconsolidated media (e.g., soil and sand). Because of geological heterogeneities in Japan, geological characterization is expected to be difficult, causing uncertainties in predicting the performance of natural barriers. Thus, Japan is assigning a major role to the engineered barrier system, while defining a small number of critical natural characteristics for the site which are expected to be achievable in various geological settings.

Considerable research and development has been underway in recent years. Most research is conducted by PNC and regular plans are submitted to AEC; the most recent progress report was submitted in 1992, and a subsequent report is expected before the year 2000 (OEC95b). PNC operates an underground test facility in both sedimentary and crystalline rock environments. The test facility is located in the Tono Uranium Mine in central Japan. Major experiments in the mine include a ground water flow investigation; studies on the effects of excavation on the mechanical and hydraulic behavior of the repository; natural analogue studies and evaluations of the chemical durability of simulated waste glasses; and the corrosion rates of candidate overpack materials. Since 1988, PNC has also conducted major tests in the Kamaishi iron ore mine in northern Honshu. Work at this mine is currently guided by a 5-year research plan, submitted by PNC in 1993 and characterized by work in a deeper gallery. Major investigations at Kamaishi have included detailed fracture mapping, cross-hole hydraulic and geophysical testing, drift excavation-effect studies, in-situ stress measurements, single-fracture flow tests, and observations of seismic activity.

Several new facilities were started in recent years, including: 1) the Tokai Vitrification Facility (the first vitrification facility in the country, where operation began in 1995); 2) the Nuclear Fuel Cycle Engineering Facility (NUCEF), where construction was completed in 1995; and 3) the Recycling Equipment Testing Facility (RETF), begun in 1995 to develop reprocessing techniques for spent fuel from fast breeder reactors (OEC95b).

3.5.3 Regulatory Organizations and their Regulations

The Atomic Energy Basic Law of 1955 established the AEC and the principles and requirements for the safe utilization and application of atomic energy and radioactive materials, including the disposal of radioactive waste. In addition to the AEC, other key agencies or organizations include the Nuclear Safety Commission (NSC), the Ministry of

International Trade and Industry (MITI), and the Science and Technology Agency (STA). Regulatory requirements for the high-level waste repository have not yet been established, nor have formal individual dose limits been issued.

3.6 SWEDEN

3.6.1 Nuclear Power Utilization

Following a 1980 national referendum, the Swedish Parliament decided to phase out nuclear power plants by the year 2010. Although the Swedish government maintains this commitment, as of 1994, the country remained dependent on nuclear fuel for approximately 51 percent of its electrical power needs. Sweden's nuclear power is produced with nine boiling water reactors and three pressurized water reactors (EIA95).

By 2010, Sweden will have produced nearly 8,000 MTHM of spent fuel. In the 1970s, Swedish utilities had entered into agreements with other countries to reprocess foreign sources of spent fuel; however, this approach was abandoned following the 1980 referendum and the utilities have since sold their reprocessing contracts or traded high-level waste from reprocessing for other spent fuel. A joint utility consortium, the Swedish Nuclear Fuel and Waste Management Company (SKB), manages the disposal of radioactive waste. In 1985, SKB began operating a centralized spent fuel storage facility (CLAB) that will eventually hold all of Sweden's spent fuel for about 40 years. As of 1994, this facility was filled to about 45 percent capacity (SKB94). The facility is situated in an underground granite cavern at a depth of 30 meters, near an existing nuclear power plant (Oskarshamn). The license application for a demonstration encapsulation facility at the site is anticipated by the end of 1996 (SAI95). As a result, all spent fuel will be directly disposed.

3.6.2 Disposal Programs and Management Organizations

Nuclear waste management activities in Sweden are guided by the Act Concerning Nuclear Activities and the Act Concerning the Management of Natural Resources. Every three years, SKB is required to provide Swedish regulators with a research and development plan for activities related to the management and disposal of the country's radioactive waste. The 1992 plan was approved contingent upon additional details regarding deposition and canister design (SKB94).

As outlined in the 1993 plan, Sweden's reference disposal concept for spent fuel is to encapsulate it in high-integrity copper canisters and emplace the canisters in a repository built in crystalline rock at a depth of about 500 meters, backfilling the deposition holes with highly-compacted bentonite and the tunnels and shafts with a mixture of sand and bentonite. SKB is evaluating alternative concepts such as deep boreholes and tunnel emplacement, as well as alternative canister designs. Canisters are expected to consist of a steel insert (for mechanical protection) inside of a copper sleeve (for corrosion protection).

SKB's 1995 R&D Programme (SKB95) sets forth the following schedule for establishing a deep geologic repository:

- SKB is currently conducting feasibility studies, planned at a total of five to ten municipalities.
- Feasibility studies are expected to be completed by 1997, after which time two municipalities, and locations within both, will be selected for site investigation.
- One site will be selected around 2001, and deposition of encapsulated fuel is planned for 2008, when a small portion (approximately 800 tons) of Sweden's nuclear fuel will be deposited.

Feasibility studies were completed in Storumann in May, but a local referendum rejected further research there. A feasibility study at a second municipality, Malaa, was scheduled for release in late 1995 or early 1996. Other sites where feasibility studies are underway or being considered include four of the five municipalities with existing nuclear facilities (Varberg, Oskarshamn, Nykoping, and Osthammer). The fifth site with an existing nuclear facility, Kavlinge, is currently not considered a candidate.

The OECD/NEA conducted an international research project in an underground research laboratory at Sweden's Stripa mine from 1980 to 1991 (OEC95b). SKB has recently completed construction of a second laboratory under the island of Äspö, 2 km north of Oskarshamn, at a depth of 450 meters. The Äspö site will be used to test methods of site selection and characterization, and to research disposal technologies for later use in Sweden's deep geologic repository.

3.6.3 Regulatory Organizations and their Regulations

Key government entities with direct responsibilities in waste management include the Swedish Nuclear Power Inspectorate (SKI), the National Institute for Radiation Protection (SSI), and the Swedish Consultative Committee for Nuclear Waste Management (KASAM). All operate under the supervision of the Ministry of the Environment and Natural Resources. The National Board for Spent Nuclear Fuel (SKN), a former public entity with regulatory responsibilities, was absorbed into SKI in the early 1990s.

As of 1995, safety requirements for management and disposal of high-level waste were in development. These requirements are the responsibility of SKI, in cooperation with SSI. Both SKI and SSI favor a total systems approach, without specifying detailed sub-system quantitative criteria in early phases of repository development. Proposed guidelines for the deep repository would require that:

- Radiation doses to individuals be limited to 1 mSv/yr for a reasonably predictable period of time, after which radionuclide flow is to be limited to a level corresponding to naturally occurring flows;
- A passive multi-barrier approach be used;
- Future safety of the facility requires no further controls after the facility is sealed; and
- The repository be designed to not restrict future attempts to change the repository or retrieve the waste. (SKB95)

Established general principles for the management of nuclear waste state that:

- Radiation protection take into consideration, in addition to human health, issues of biodiversity and natural resource use;
- Radiation protection be independent of whether doses arise today or in the future, or whether they originate within or outside the country;

- The disposal of nuclear waste pose a risk no greater risk than that of other portions of the nuclear fuel cycle; and

- All activities must be justified, protection must be optimized, and the individual must be protected by dose limits. (SKB95)

3.7 SWITZERLAND

3.7.1 Nuclear Power Utilization

In 1994, Switzerland's five nuclear power plants supplied about 37 percent of the country's electrical power needs (EIA95). The Swiss nuclear power program relies on a mix of pressurized and boiling light water reactors (three PWRs and two BWRs). Although there is currently a moratorium on construction of new nuclear plants, capacity increases at existing plants has kept supply high, and a 10 percent increase by the year 2000 is planned (OEC95c).

It is estimated that by the year 2000, the Swiss will have produced about 1,800 MTHM of spent fuel. Switzerland currently ships its spent fuel to France and Britain for reprocessing but maintains the options of spent fuel management both with and without reprocessing in the future.

3.7.2 Disposal Programs and Management Organizations

The responsibility for establishing radioactive waste disposal facilities in Switzerland lies with the National Cooperative for the Storage of Radioactive Waste (NAGRA), a joint government and utility cooperative agency. NAGRA was established in 1972 to manage the disposal of radioactive wastes, including spent fuel, high-level waste and other reprocessed waste returned from the French and British reprocessing facilities. Waste conditioning and interim storage of reprocessed waste, high-level waste and spent fuel is the responsibility of ZWILAG, a cooperative comprised of nuclear utility operators.

Overall nuclear policy is governed by the Swiss Atomic Law, to which two major changes were proposed in 1994. The first, a provision to simplify permitting for field work and repository projects, was rejected. The second, an initiative designed to reduce the possibility of nuclear proliferation, was approved. The overall goal of the Swiss high-level waste program is to establish the viability of a repository in Switzerland by the year 2000, although commissioning of a repository will not occur before 2020 to allow a 40-year spent fuel/high-level waste cooling period. Participation in any international repository projects that may develop is also under consideration.

The Swiss have historically considered two rock types, crystalline rock and sedimentary rock, as potential host media for a high-level waste repository. In 1984, NAGRA launched studies in crystalline rock by drilling seven deep boreholes into the crystalline basement of northern Switzerland and, subsequently, conducted geological and safety assessment studies. In 1994, NAGRA released a synthesis of this research (Kristallin I), expressing optimism about the use of crystalline rock as a host rock; specifically, NAGRA is considering crystalline rock formations in northern Switzerland as viable sites for a repository and is planning additional field work there (OEC95b). In support of crystalline rock studies, a new three-year Phase IV study was launched at the Grimsel Rock Laboratory in 1994. The Swiss have also considered two sedimentary rock types, opalinus clay and freshwater molasse, and have conducted field research on both formations. In January 1994, the Safety Inspectorate, NAGRA, and representatives of the relevant government agencies identified opalinus clay as the preferred sedimentary host rock option (OEC95b). A site in Benten, just north of Zurich, has been identified for seismic survey and the construction of an 800-meter borehole to further examine the feasibility of a repository in clay.

By the year 2000, NAGRA must submit a program—the Siting Feasibility Project—for government approval that demonstrates the feasibility of siting a repository in one or more of the crystalline or sedimentary media under consideration. NAGRA intends to select which medium or media by 1997.

3.7.3 Regulatory Organizations and their Regulations

Key organizations or agencies with direct responsibilities in waste management include: the Nuclear Safety Division (HSK) of the Federal Energy Office (BEW) within the Federal Department of Transport, Communications, and Energy (EVED); the Federal Commission for the Safety of Nuclear Installations (KSA); the Federal Department of Interior (EDI); and the Institute for Reactor Research (EIR). An interagency working group (AGNEB) was also established to coordinate activities in support of government decisions on the licensing of nuclear waste facilities.

In November 1993, HSK released the current guidelines for management of nuclear waste in the country, entitled Radiation Protection for the Disposal of Nuclear Waste (HSK93). Dosage is limited to 0.10 mSv (10 mrem) per year for reasonably probable scenarios, and annual risk is limited to 10^{-6} for unlikely scenarios. Candidate repositories must produce a system capable of meeting these requirements in order for their application to be considered (GAO94); furthermore, all repositories must be designed to be sealed at any time within a few years, after which it must be possible to dispense with institutional controls.

3.8 UNITED KINGDOM

3.8.1 Nuclear Power Utilization

In 1994, the United Kingdom (UK) met about 26 percent of its electrical needs through nuclear power (EIA95). During the 1960s and 1970s, the UK depended primarily on a series of Magnox (Magnesium-clad, Uranium metal-fueled) reactors (20 units in operation as of 1995), but began to use advanced gas-cooled reactors (AGRs) during the 1970s and 1980s (14 units in operation as of 1995). One pressurized water reactor (PWR) was commissioned in 1995. Use of a fast reactor was explored as well, but a prototype facility in Dounreay was closed in 1994.

British Nuclear Fuels (BNFL), a government-owned corporation, reprocesses spent fuel at its Sellafield facility on behalf of both domestic and foreign utilities. Spent metallic fuel from the country's Magnox reactors is reprocessed at the Sellafield facility at a rate of approximately 400 cubic meters annually (NIR94). In March 1994, BNFL began operating the Thermal Oxide Reprocessing Plant (THORP) at Sellafield to reprocess spent fuel produced by the country's AGR and PWR reactors and by international customers. THORP is the country's first commercial-scale reprocessing plant for oxide fuels.

In 1994, the Board of Trade and the Secretary of State for the Environment placed portions of the British nuclear program under review. In May 1995, at the conclusion of the review, it was announced that the country's comparatively modern facilities (7 AGR stations and the

Sizewell PWR) and all future facilities were to be privatized (nuclear power had been excluded from the 1990 privatization of the electric utility industry). Under plans for privatization, scheduled to begin in July 1996, all AGR and PWR stations will be grouped under a new company called British Energy plc. Current owners of these stations, Nuclear Energy and Scottish Nuclear, will become wholly-owned subsidiaries of British Energy. Magnox stations will be transferred to a new company called Magnox Electric plc (Magco), a government-owned company responsible for operation of and liabilities resulting from these stations. It is expected that Magco will ultimately become a subsidiary of BNFL, after privatization of Nuclear Energy and Scottish Nuclear.

Since 1952, over 30,000 MTHM of metal fuel from the Magnox reactors have been reprocessed in the UK. It is estimated that by the year 2000, Britain will have about 4,000 cubic meters of high-level waste destined for storage or disposal due to the reprocessing of some 60,000 metric tons of spent fuel. High-level waste is currently stored in an air-cooled facility at Sellafield.

3.8.2 Disposal Programs and Management Organizations

The responsibility for managing the storage and disposal of radioactive waste lies with its producers. BNFL has the lead responsibility for management of high-level waste from reprocessing. In 1990, BNFL began operating a vitrification plant at Sellafield. In 1982, the government established the Nuclear Industry Radioactive Waste Executive (NIREX) to develop and operate intermediate- and low-level radioactive waste 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, UK NIREX.

Historically, the UK's radioactive waste disposal strategy has postponed the development of a high-level waste disposal facility, considering deep disposal of low- and intermediate-level wastes a higher priority. NIREX is currently in its seventh year of research at a potential disposal site near Sellafield, and current plans suggest that a repository for low- and intermediate-level wastes could be operational there by the year 2010.

Eventually, the high-level waste is planned for deep disposal. Current plans call for continued reprocessing of spent fuel, solidification of high-level waste, and surface storage for about 50 years. Under this schedule, the need for a high-level waste repository is not expected before the year 2040. Vitrified high-level waste would then be disposed in deep geologic media.

The UK 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 develop a high-level waste disposal and repository strategy using concepts that best fit British needs.

3.8.3 Regulatory Organizations and their Regulations

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 waste.

The regulatory functions are performed by the Nuclear Installations Inspectorate, which is part of the Health and Safety Executive; the Radiochemical Inspectorate of the Department of the Environment; the Ministry of Agriculture, Fisheries, and Food; the UK Atomic Energy Authority; and the Secretaries of State of Scotland and Wales. The government also takes advice from several independent experts and advisory committees, including the Radioactive Waste Management Advisory Committee and the National Radiological Review Board. The current system of regulation is expected to remain unchanged following the industry's privatization mandated nuclear review announced in 1994 (OEC95c).

Exposure limits are based on recommendations of the National Radiological Protection Board. While no current limits pertaining to high-level waste exposure have been set, indications are that they will be similar to those set for low- and intermediate-level wastes (10^{-6} per year for individual risk from a single facility (OEC95a)). The British waste management philosophy favors the use of broad safety goals over prescriptive regulatory approaches, placing the burden of compliance upon the operator.

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CHAPTER 4

U.S. PROGRAMS FOR THE MANAGEMENT AND DISPOSAL OF SPENT NUCLEAR FUEL AND HIGH-LEVEL RADIOACTIVE WASTE AND THE EVALUATION OF YUCCA MOUNTAIN

4.1 INTRODUCTION

As discussed in Chapter 1, DOE is ultimately responsible for the management and disposal of spent nuclear fuel and high-level radioactive waste. Fulfillment of this responsibility involves at present four principal program activities in the Department: 1) receipt, transport, interim storage, and disposal of spent fuel from commercial nuclear power reactors, under the Office of Civilian Radioactive Waste Management (OCRWM); 2) management and disposal of DOE spent fuel, which originates from DOE production and research operations and from naval propulsion reactors; 3) solidification and disposal of high-level waste generated by reprocessing operations for spent fuel from DOE's production reactors at Hanford and Savannah River; 4) storage and disposition of fissile materials from dismantled nuclear weapons. Disposition of weapon from dismantled weapons materials is included here because, under alternatives currently being evaluated, these materials could be treated and disposed like high-level waste or they could be used as a fuel in reactors and eventually become part of the spent fuel inventory for disposal.

The United States, like other nations with major radioactive waste management and disposal programs, has selected isolation of highly-radioactive wastes by engineered emplacement in deep geologic formations as the disposal method. This method of disposal was chosen after comprehensive evaluation (DOE80) of alternatives including disposal by ejection into space; elimination by transmutation to other materials; disposal in ice sheets; and emplacement within ocean seabed sediments.

The DOE and its predecessor agencies, the AEC and the Energy Research and Development Administration, were for more than 15 years, beginning in the 1960s, engaged in a search for one or more disposal sites. A framework for this and other program efforts was provided by the NWPA, which was amended in 1987 to focus the DOE program on evaluation of the Yucca Mountain site in Nevada as the only candidate location for disposal. If the Yucca Mountain site is found suitable for disposal and if the U.S. Nuclear Regulatory Commission

(NRC) licenses an engineered repository at the site, this location would be used for disposal of commercial spent nuclear fuel and high-level waste from DOE's production operations.

Other radioactive wastes have been considered for disposal in a repository at Yucca Mountain. These candidate wastes include DOE spent fuel, fissile materials from dismantled nuclear weapons, and low-level radioactive wastes known as Greater-Than-Class-C (GTCC) wastes because their radioactivity levels exceed the NRC's limits for Class C wastes as established in the 10 CFR Part 61 regulations. Decisions concerning disposition of these radioactive materials have not been made, and the NWPA limits the contents of a Yucca Mountain repository to "...70,000 metric tons of heavy metal or a quantity of solidified high-level radioactive waste resulting from the reprocessing of such a quantity of spent fuel until such time as a second repository is in operation" (NWPA, Section 114(d)). As detailed in Chapter 7 of this BID, DOE currently assumes that a repository at Yucca Mountain would contain approximately 63,000 metric tons of spent commercial reactor fuel and defense high-level wastes that are the equivalent of 7,000 metric tons of heavy metal.

A waste disposal repository at the Yucca Mountain site would involve mine-like excavations, about 300 meters below the crest of the mountain. This depth would place the waste several hundred meters above the water table. The waste would be contained in canisters designed to augment the isolation from the human environment provided by the geologic formation. Under current DOE plans, if the site is licensed for disposal, emplacement would begin about 2020.

Disposal operations would continue for several decades, after which the repository would remain open for several more decades for monitoring of safety performance. If disposal safety is confirmed, the repository would be sealed approximately a century from now. After closure and sealing, a system of markers would be established. It is expected that active institutional control of the site would be maintained for as long as possible.

4.2 DOE OFFICE OF CIVILIAN RADIOACTIVE WASTE MANAGEMENT (OCRWM)

OCRWM in DOE was established by Congress specifically to provide management and disposal of spent fuel from commercial nuclear power reactors. Under a 1985 Presidential Executive Order, the repository established by OCRWM will also be used for disposal of high-level waste from DOE's production operations.

The OCRWM charter includes responsibility for receipt of spent fuel from reactors at the reactor sites, interim storage of spent fuel as necessary prior to disposal, transport of spent fuel to the site(s) for interim storage and disposal, and siting, design, licensing, and operation of a central interim storage facility. Alternative designs for a central interim storage facility (known historically as a Monitored Retrievable Storage (MRS) facility) have been developed by DOE but, as of early 1996, DOE has not established a site for a central storage facility.

DOE/OCRWM activities are currently focused on the evaluation of the Yucca Mountain site as a location for disposal of spent fuel and high-level waste. In accordance with the Site Characterization Plan (DOE88a), characterization of the Yucca Mountain site is proceeding with surface-based and sub-surface activities. DOE is currently using a Tunnel Boring Machine (TBM) to access the geologic formations at the proposed disposal horizon in order to obtain characterization information important to evaluating disposal safety performance.

The first phase of excavation using the TBM is expected to be finished in 1996. Detailed examination of the disposal-horizon rock characteristics and hydrologic conditions will then be done, and a suite of in-situ experiments on technical issues, such as the effect of thermal loading on geologic formations, will be performed.

In 1992, OCRWM began development of the Multi-Purpose Canister (MPC) System as the method for containing the commercial spent fuel throughout the various management and disposal operations (DOE94a). The basic concept is to place the spent fuel in a container which then would be placed in overpacks tailored for the storage, transport, and disposal operations. Objectives of the strategy include reduction of the number of handling operations by use of canisters with capacities much larger than past technology (e.g., by about a factor of five) and standardization of technology used for the many spent fuel configurations.

During recent years the MPC system has been the basis for design studies concerning engineered features of the repository. It has also been a reference concept for OCRWM's Total System Performance Assessments (TSPA; DOE94b, DOE95a). Specific design parameters for the canisters have not been selected, nor has the emplacement configuration (spacing and orientation) been selected. The TSPA evaluations have examined the effects of alternative engineered features of the repository on safety performance in terms of potential waste radioactivity release and transport to the environment.

As a result of Fiscal Year 1996 budget constraints, OCRWM has recently abandoned continued development of the MPC system. Design work under current contracts will be completed, but further development will be delayed. This situation may ultimately affect significantly the engineered features of the repository. A strategy for future engineered design work has not yet been selected.

Information concerning natural features of the Yucca Mountain site and region, and of the repository engineered concepts under consideration, is presented in Chapter 7 of this document. This information is abstracted from DOE reports that present and interpret data from site characterization and design development activities. The DOE reports are in turn compilations of information contained in detailed topical, technical reports such as the U.S. Geological Survey open-file reports on geologic features of the site.

The OCRWM program has to date produced thousands of documents concerning its mission and activities. Future documents are expected to emphasize information concerning natural features of the site and region. Past site characterization activities focused on the mountain and the proposed repository horizon because the previous EPA standard, 40 CFR Part 191, required evaluation of radionuclide releases over a period of 10,000 years across the accessible environment boundary 5 kilometers from the repository. Future work may include a greatly increased effort to characterize geologic and hydrologic features of the region beyond the five-kilometer accessible environment boundary.

4.3 DOE MANAGEMENT AND DISPOSAL OF DEFENSE WASTES

The DOE has produced significant amounts of radioactive material that may eventually be disposed in a repository at Yucca Mountain (see Chapter 5). The radioactive material is high-level waste, produced as part of the Department's defense programs. Other waste produced by these defense programs (e.g., transuranic (TRU) waste) will be managed and disposed of separately. Following enactment of the NWPA, the DOE studied various options for disposal of the defense program wastes and concluded that disposal in a geologic repository was feasible. The DOE further concluded that these materials could be sent to the same repository being considered for disposal of commercial spent fuel (DOE85).

During the last 40 years, the DOE and its predecessor agencies generated, transported, received, stored, and reprocessed spent fuel at facilities throughout its nationwide complex. The spent fuel came from: nuclear weapons production reactors; U.S. Navy nuclear propulsion program power reactors; government, university, and test reactors; special-case commercial reactors; and research reactors. The DOE operated production reactors at the Hanford and Savannah River Sites to provide special nuclear materials and other isotopes for defense programs. These production reactors are no longer operating. However, the Naval Nuclear Propulsion Program and some test and research reactors are still operating.

The DOE has reprocessed spent nuclear fuel—more than 100,000 metric tons of heavy metal (MTHM)—at the Idaho National Engineering Laboratory (INEL), Hanford Site, and Savannah River Site to recover fissile material (uranium-235 and plutonium-239) and other nuclides needed for national defense or research and development programs. These reprocessing operations generated large quantities of high-level radioactive waste. This waste exists as liquid, sludge, solids and calcine and is stored primarily at its reprocessing site.

In April 1992, the DOE began to phase out spent nuclear fuel reprocessing. As a result, approximately 2,700 metric tons of spent fuel exist today in the DOE inventory. This spent nuclear fuel is in a wide range of enrichments and physical conditions and is stored at several locations throughout the United States. In addition to this inventory, the DOE estimates that over the next 40 years it will generate another 100 metric tons of heavy metal from defueling DOE and naval reactors.

From 1990 to 1995, the DOE was embroiled in a series of controversies concerning the management of the spent fuel inventories.[7] In 1993, a Federal judge enjoined shipments of spent nuclear fuel to the INEL. In May 1995, the DOE issued a Record of Decision (ROD) concerning spent fuel management at the INEL (DOE95b). The ROD documents decisions made as a result of information and analyses contained in a Final Environmental Impact Statement (FEIS) for the management of spent fuel at the INEL (DOE95c).[8] The ROD, as well as the Spent Fuel FEIS, details the management of spent fuel and high-level waste at other DOE facilities.

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[7] *See, e.g.,* ID Dept. Health and Welfare v. U.S., F.2nd 149 (9th Cir. 1992).

[8] Hereinafter referred to as the Spent Fuel FEIS.

As documented in the FEIS published in April 1995, the DOE decided to regionalize spent fuel management by fuel type at the INEL, Hanford, and Savannah River reprocessing sites (DOE95c). This decision calls for the future fuel type distribution to be:

- The Hanford production reactor fuel will remain at the Hanford Site.
- Aluminum clad fuel will be sent to the Savannah River Site.
- Non-aluminum clad fuels, including spent fuel from the Fort St. Vrain reactor and naval spent fuel, will be sent to the INEL.

4.3.1 High-Level Waste

High-level waste, which is generated by reprocessing spent reactor fuel and irradiated targets, generally contains more than 99 percent of the nonvolatile fission products produced during reactor operation. This high-level waste is stored where it was generated, namely the INEL, the Hanford Site, and the Savannah River Site. A limited quantity of high-level waste is stored at the West Valley Demonstration Project, near West Valley, New York. This high-level waste is under the management control of the DOE Idaho Operations office.

The DOE high-level waste at the INEL, which is stored at the Idaho Chemical Processing Plant (ICPP), resulted from reprocessing nuclear fuels from Naval Nuclear Propulsion Program reactors and special research and test reactors. The acidic liquid portion of this waste is stored in tanks, although the bulk of the material has been converted to a granular solid (calcine).

At the Savannah River and Hanford Sites, the acidic liquid waste from reprocessing production reactor fuel has been made alkaline with caustic soda and is now stored in tanks. During storage, this alkaline waste separates into a liquid and a sludge. The liquid volume is sometimes reduced by evaporation, leaving a salt cake in the tanks that holds the evaporator concentrates.

These high-level wastes are planned to be treated and converted to solid form as a glass (i.e., vitrified). The vitrified wastes will be placed in steel containers and disposed of at Yucca Mountain if a repository is licensed at the site.

Hanford Site

The DOE placed approximately 350 million liters of liquid high-level waste in 149 single-shell tanks from 1944 to 1980. This liquid waste and the later waste that was added came from chemical reprocessing operations that separated plutonium from spent fuel in the Hanford 200 area. Other tank waste was generated in smaller volumes from research and development programs, laboratory processes, and plutonium finishing facility operations.

All of the fuel reprocessing methods generated acidic waste streams. Sodium hydroxide or calcium carbonate was added to the waste before it was transferred to the tanks to neutralize the acid and minimize tank corrosion. The tanks currently contain moderate to strong alkaline solutions. Additional post-processing of the waste to recover plutonium and uranium, or to reduce the volume of high-level waste, has resulted in the addition of ferrocyanide and some organic compounds listed as hazardous.

Double-shell tanks continue to receive waste generated by decommissioning and cleanup of Hanford Site facilities. This includes: effluents associated with the deactivation program for the PUREX Plant; waste from B-Plant maintenance activities; laboratory waste; and miscellaneous waste streams from ion-exchanger resin regeneration.

The DOE is considering a program to treat and remediate some of this tank waste. This program would include retrieval of all the waste from Hanford's 149 single-shell tanks. DOE plans to separate tank contents into high-level and low-level components, thereby reducing the amount of high-level radioactive waste. All remaining high-level liquid waste would then be vitrified and placed in stainless steel canisters for storage on site until a geologic repository is available for disposal. This tank remediation effort would be complete by 2035 (DOE95d). Disposal at the Yucca Mountain site is expected if the site is found suitable and a repository is licensed for disposal.

The tanks now contain a mixture of salt cake, liquid, and sludges with both radioactive and hazardous components. Sludge consists primarily of solids (hydrous metal oxides) precipitated from the neutralization of acid waste. Salt cake consists of the various salts formed from the evaporation of water from the waste. Liquids exist as supernatant (liquid above solids) and interstitial liquid (liquid filling the void between solids) in the tanks.

The tank waste is mostly inorganic, containing sodium hydroxide; salts of nitrate, nitrite, carbonate, aluminate, and phosphate; and hydrous oxides of aluminum, iron, and manganese. The radioactive components consist primarily of long-lived fission products and shorter-lived radionuclides, such as strontium-90 and cesium-137, and isotopes of uranium, plutonium, and americium. Some tanks contain the chelating agents EDTA and HEDTA; some tanks contain halogenated and nonhalogenated organic contamination; others contain mixed waste with detectable levels of lead, chromium, and cadmium.

Idaho National Engineering Laboratory

Most of the high-level waste generated at the INEL came from spent nuclear fuel reprocessing at the Idaho Chemical Processing Plant (ICPP). The ICPP stopped fuel reprocessing in April 1992 (DOE95d).

The DOE stores three types of high-level waste at the ICPP: high-level liquid waste, sodium-bearing liquid waste, and calcined solid waste. The liquid waste is stored in stainless-steel tanks until it can be processed through a calcining facility. Once the liquid has dried to form a calcine, it is stored in heavily shielded steel bins.

Savannah River Site

Most of the Savannah River Site high-level waste was generated as acidic liquid. Sludge formed during subsequent treatment with caustic soda (DOE95d). Salt cake results when the supernatant liquor is concentrated in evaporators.

Tank farms at the Savannah River Site contain 24 single-shell and 27 double-shell tanks for storing high-level waste. The DOE plans to remove the liquid waste from these tanks by 2035 (DOE95d). The removal process includes these process steps involved in vitrifying the waste:

- The salt solution is removed from the tanks and chemically treated in the In-Tank Precipitation Facility to precipitate radionuclides. The decontaminated filtrate is stripped of benzene and transferred to the Saltstone Facility for disposal. The concentrated precipitate is treated to remove nitrites and then stored.

- Sludge from the tanks is transferred to a tank at the Extended Sludge Processing Facility, where it is washed in water and sodium hydroxide to remove salts and aluminum. The washed sludge is stored in a tank until it is transferred to the Defense Waste Processing Facility.
- At the Defense Waste Processing Facility, which is not yet in full operation, the washed sludge and the precipitate from the In-Tank Precipitation Facility are combined with glass frit to form glass logs. The process removes organics from the precipitate and mercury from both precipitate and sludge. The DOE expects that the vitrification process will solidify the waste, which will then be placed in stainless steel canisters.

4.3.2 Spent Nuclear Fuel

The DOE reprocessed most of its spent nuclear fuel in the facilities at INEL, the Hanford Site, and the Savannah River Site. However, some spent nuclear fuel remains because of U.S. government decisions to stop reprocessing. Most of this fuel came from the Hanford Site N-Reactor, a dual-purpose reactor designed to produce plutonium for use in nuclear weapons and to generate electricity for commercial use. Smaller amounts of spent nuclear fuel associated with nuclear weapons production is stored at the Savannah River Site. Spent nuclear fuel from the Naval Nuclear Propulsion Program is stored at the INEL and, for short times, at some naval nuclear shipyards. Finally, the DOE will assume responsibility for fuel from some special-case commercial nuclear reactors, foreign research reactors, and certain domestic research and test reactors. The sections that follow discuss the nature and quantity of this spent nuclear fuel, as well as DOE's plans to manage it. Most of the discussion that follows is derived from the Spent Fuel FEIS (DOE95c).

Hanford Site

The Hanford Site produced plutonium for use in nuclear weapons from the start of the Manhattan Project until DOE halted production in 1989. Hanford's production reactors (including the N-Reactor and the Single Pass reactor) generated 2,100 MTHM of the existing DOE inventory of spent nuclear fuel. The Hanford Site also stores spent nuclear fuel from the Shippingport reactor, and miscellaneous special-case commercial and nuclear fuels.

N-Reactor Spent Nuclear Fuel

N-reactor spent nuclear fuel is stored in three facilities; DOE's interim plans for management of this fuel include possible relocation to a storage facility in the 200 Area. There is a total of about 2,096 metric tons of spent N-reactor fuel at Hanford, which comprises all but about one percent of the spent fuel inventory at the site. Sources of the other spent fuel at the site included single-pass Hanford production reactors, the Fast Flux Test Facility, Shippingport Core II, and miscellaneous test facilities.

Idaho National Engineering Laboratory

Six major facility areas at the INEL store spent fuel: Argonne National Laboratory-West; Idaho Chemical Processing Plant; Naval Reactors Facility; Power Burst Facility; Test Area North; and the Test Reactor Area. Spent nuclear fuel is kept in a variety of dry and wet configurations. The INEL stores about 10 percent of the DOE's current inventory of spent nuclear fuel, i.e., about 300 tonnes.

Argonne National Laboratory-West (ANL-W) generates spent nuclear fuel from research and development activities related to advanced reactor designs. The DOE has also used ANL-W to store small quantities of spent nuclear fuel from other facilities. Storage facilities include both wet (including molten sodium) and dry configurations.

The Idaho Chemical Processing Plant (ICPP) was used by DOE from 1956 until 1992 to reprocess spent nuclear fuel, including fuel from the Naval Nuclear Propulsion Program. The ICPP also stored spent fuel, including most fuel types except those used in nuclear weapons production. The ICPP maintains both wet and dry storage configurations.

The Naval Reactors Facility maintains small quantities of spent nuclear fuel from the Naval Nuclear Propulsion Program and from test reactors at the Expended Core Facility. All spent nuclear fuel is currently stored in water-filled pools. The Navy is planning to construct a dry storage facility.

The Power Burst Facility (PBF) is used to store a small amount of spent nuclear fuel in a wet configuration. The DOE plans to remove this fuel because the facility is uneconomical to operate.

Test Area North is used by DOE to store special case commercial fuel. This includes debris from the Three Mile Island Unit 2 core and DOE experimental fuel similar to commercial fuel. Test Reactor Area houses small amounts of special case commercial, foreign, and Power Burst Facility spent nuclear fuel. All spent nuclear fuel is stored in water-filled pools.

Savannah River Site

The DOE has about 200 MTHM, or about 8 percent of the total spent nuclear fuel inventory, in storage at the Savannah River Site. This fuel is stored in the Receiving Basin for Off-site Fuels (RBOF), in three reactor disassembly basins, and in basins in the F- and H-Canyons.

The F- and H-Area Canyons are among the only remaining operable chemical separation facilities of their kind in the DOE complex. Each canyon has an associated storage basin that serves as an interim staging area where spent nuclear fuel awaits chemical separation. The basins contain 13 reactor fuel assemblies (H-Area) and aluminum-clad targets (F-Area).

The DOE has stored most of the remaining aluminum-clad spent nuclear fuel from Savannah River Site reactors in water-filled concrete basins. Reactor disassembly basins for the K-, P-, and L-Reactors contain spent nuclear fuel and target material. These basin structures were built in the 1950s and were not intended for prolonged storage of radioactive materials.

The RBOF has been receiving fuels of United States origin since 1964, including fuel manufactured in the United States but irradiated in foreign reactors. About 30 percent of the fuels in the RBOF consist of uranium clad in stainless steel or Zircaloy, which Savannah River Site facilities cannot process without modifications.

Other Generator/Storage Locations

The DOE has in its possession, or has title to, a small amount of spent nuclear fuel in many other locations throughout the United States. These locations include both DOE and non-DOE facilities. The Oak Ridge National Laboratory (ORNL) stores less than 1 MTHM of spent nuclear fuel. This fuel is left over from research on fuel elements removed from commercial or demonstration reactors, as well as fuel removed from reactors that operated at the ORNL.

Under the Spent Fuel FEIS, this fuel will be transferred to either the INEL, Hanford Site or Savannah River Site.

Besides the ORNL, the DOE is responsible for spent nuclear fuel from research and test reactors at the Brookhaven, Los Alamos, Sandia, and Argonne-East Laboratories. These facilities have a total of about 2 MTHM in storage.

- Non-DOE Research Reactors — the DOE has title to the spent nuclear fuel that is stored at or is generated by 57 small research reactors. These reactors operate at universities, commercial establishments and other government agencies such as the Department of Defense. These reactors have a current inventory of less than 5 MTHM, and will generate very little additional spent nuclear fuel by 2035.

- Commercial Power Reactors — The DOE has possession of 125 spent nuclear fuel assemblies and 20 complete or sectioned spent nuclear fuel rods from several commercial power reactors that supported DOE-sponsored research and development programs. This fuel is stored at the West Valley Demonstration Project in West Valley, New York and at the Babcock and Wilcox Lynchburg Technology Center in Campbell County, Virginia. Other commercial spent nuclear fuel is already stored at the INEL, Hanford Site, and Savannah River Site.

- Foreign Research Reactors — the DOE has accepted limited amounts of spent nuclear fuel from foreign reactors (WCM95a). In some cases, this fuel was manufactured by the DOE. The DOE will, under the Spent Fuel FEIS, continue to receive and store spent fuel from foreign sources (DOE95c).

All the spent nuclear fuel listed above, if not already transferred, will be shipped to the INEL, the Hanford Site, or the Savannah River Site, according to the Spent Fuel FEIS.

Spent Nuclear Fuel Management Options

The Spent Fuel FEIS says that most spent nuclear fuel in the possession of the DOE will be stored until a geologic repository is available. Some of this storage will be wet, but some fuels, such as those with aluminum cladding, may be placed in dry storage. The DOE is also considering options to stabilize some of its corroding spent nuclear fuel (WCM95b). One of these options is chemical separation at the Savannah River Site.

4.4 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.

4.4.1 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 waste. The Congress has provided for public participation in the NWPA and in the NWPAA (NWP87). Specific provisions of the NWPA, as amended, govern the notification of potentially affected States and Indian tribes (Section 116(a)).

4.4.2 Programs in the State of Nevada

In 1985, the State of Nevada Agency for Nuclear Projects/Nuclear Waste Project Office (NWPO) was created by the Nevada Legislature to oversee Federal high-level nuclear waste activities in the State. Since then, it has dealt primarily with the technical and political issues associated with DOE characterization of the Yucca Mountain site.

4.5 NATIVE AMERICAN TRIBES

Native American Tribes have a unique sovereign status in U.S. law; this status was recognized by the NWPA and the NWPAA. This government-to-government relationship between the Federal government and the tribes obligates the Federal government to interact directly and specifically with tribes in areas where repository or MRS siting activities will occur. The NWPA, as amended, under Section 2(2), defines an affected tribe as any tribe:

> (A) within whose reservation boundaries monitored retrievable storage facility, test and evaluation facility, or a repository for high-level waste or spent fuel is proposed to be located; or (B) 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... (NWP83)

As noted above, specific provisions 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 tribes. The means for an affected tribe to disapprove of the site selection and designation process is given in Section 118(a). An affected 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). Since the passage of the NWPAA, no tribes have been designated as affected tribes. However, to ensure compliance with the American Indian Religious Freedom Act (AIRFA), the National Historic Preservation Act (NHPA) and related statutes, and the National Environmental Policy Act (NEPA), the DOE is cooperating with Indian tribes that have current or traditional religious or cultural ties to the Yucca Mountain site or that may be located near the transportation routes to or around the site (DOE88b).

In 1985 and in keeping with the NHPA, the Advisory Council On Historic Preservation (ACOHP) issued guidelines to provide a basis for discussing which tribes should be involved in the Yucca Mountain cultural resource study (STO90). The guidelines contributed to the Yucca Mountain Project's Programmatic Agreement (PA), which was jointly produced by DOE and the ACOHP. The PA requires that DOE consult with tribal groups having traditional cultural ties to the Yucca Mountain area prior to land-disturbing activities to assure that cultural or religious values are preserved to the extent practical. The PA further stipulates that when such activities are thought to have a negative effect that cannot be avoided, the DOE will consult further with the tribal groups and others to identify ways to mitigate those effects.

DOE has established the Yucca Mountain Site Characterization Project, which led to the Cultural Resources Program to meet resource preservation requirements set forth in the PA. The preliminary site characterization (DOE87) identified the ethnic and tribal affiliations of the tribal groups most likely to have traditional ties to cultural resources located in the Yucca Mountain region. These groups consist of Southern Paiute, Western Shoshone, and Owens Valley Paiute/Shoshone people from Nevada, Utah, Arizona and California. Extensive ethnographic research led to the identification of sixteen tribes, containing members of the three involved ethnic groups, that potentially could be involved in the Yucca Mountain Cultural Resource Project:

1. Benton Paiute Indian Tribe, California
2. Timbisha Shoshone Tribe, California
3. Bishop Paiute Indian Tribe, California
4. Big Pine Indian Tribe, California
5. Fort Independence Indian Tribe, California
6. Lone Pine Indian Tribe, California
7. Yomba Shoshone Tribe, Nevada
8. Duckwater Shoshone Tribe, Nevada
9. Pahrump Paiute Indian Tribe, Nevada
10. Las Vegas Paiute Indian Colony, Nevada
11. Las Vegas Indian Center, Nevada
12. Chemehuevi Tribe, California
13. Colorado River Indian Tribes, Arizona
14. Moapa Paiute Tribe, Nevada
15. Paiute Indian Tribe of Utah
16. Kaibab Paiute Tribe, Arizona

All sixteen tribes requested that they be included in the project. The DOE informs tribes of the status of the Program through a cooperative agreement with the National Congress of American Indians. Through this group, the tribal governments have established a consulting relationship with the DOE in which they can express the concerns of the tribal peoples.

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CHAPTER 5

QUANTITIES, SOURCES, AND CHARACTERISTICS OF SPENT NUCLEAR FUEL AND HIGH-LEVEL WASTE IN THE UNITED STATES

5.1 INTRODUCTION

This chapter presents current and projected inventories of spent nuclear fuel and DOE defense high-level radioactive waste. Current planning calls for both of these waste forms to be disposed in the Yucca Mountain repository. The waste inventories cited are from the most recent and reliable Federal government information sources publicly available (DOE94a, DOE95a, DOE95b, DOE95c, DOE95d, DOE95e, DOE95f). The waste forms are inventoried by mass or volume and radioactivity content.

5.2 COMMERCIAL 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, e.g., 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 98 percent of the spent fuel from commercial power reactors is stored at reactor sites. Spent fuels from one-of-a-kind reactors are currently stored at the Hanford Site and INEL. Spent fuels from the Fort St. Vrain HTGR, the N reactor, the Fast Flux Test Facility (FFTF), the Shippingport reactor, and the damaged Three Mile Island Unit 2 reactor are stored at INEL. Other types of special spent fuel are stored at the Savannah River Site (SRS) (DOE95a). The fuels at these DOE facilities are government-owned and are not scheduled for reprocessing in support of DOE/defense activities.

The fuel for commercial light-water reactors consists of uranium dioxide encased in zirconium alloy (zircaloy) or stainless steel tubes. During reactor operation, fission of the uranium-235 produces energy, neutrons, and radioactive isotopes known as fission products. The neutrons produce further fission reactions and thus sustain the chain reaction. The neutrons also convert a portion of the uranium-238 into plutonium-239, which can also undergo fission. In time, the fissile uranium-235, which originally constituted some 3 to 4 percent of the enriched fuel, is depleted to such a level that power production becomes inefficient. Once this occurs, the fuel bundles are deemed "spent" and are removed from the reactor. In the United States, reprocessing of commercial spent fuel to recover the unfissioned uranium-235 and the plutonium for reuse as a fuel resource is not currently taking place, nor is it expected to occur.

The radioactive materials associated with spent fuel fall into three categories: a) fission products, b) actinide elements (atomic numbers of 89 and greater), 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 high radiological decay rates, their decay-heat production, and their potential biological hazard. Such fission products include: strontium-90; technetium-99; iodine-129 and -131; cesium isotopes, such as cesium-134, -135, and -137; tin-126; and krypton-85 and other noble gases.

Activation products include tritium (hydrogen-3), carbon-14, cobalt-60, and other radioactive isotopes created by neutron activation of fuel assembly materials and impurities in cooling water or in the spent fuel. The actinides include uranium isotopes and transuranic elements (i.e., isotopes with an atomic number greater than 92, including plutonium, curium, americium, and neptunium). The exact radionuclides composition of 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.

5.2.1 Commercial Spent Fuel Inventory and Projection

By the end of 1994, there were 29,811 MTIHM[9] of spent fuel in inventory from commercial reactor operation. Approximately 29,000 MTIHM is stored at reactor sites. The remainder is stored at the West Valley Demonstration Project (WVDP) (27 MTIHM) at West Valley, New York; the Idaho National Engineering Laboratory (INEL) (43 MTIHM) at Idaho Falls, Idaho;

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[9] Commercial spent nuclear fuel reported in DOE95e is in units of metric tons of *initial* heavy metal (MTIHM) to avoid difficulties arising from the need to estimate ranges of varied heavy-metal content that result from different levels of enrichment and reactor fuel burnup. A metric ton (tonne) is 1,000 kilograms, corresponding to about 2,200 pounds.

and at the Midwest Fuel Recovery Plant (MFRP) (674 MTIHM) at Morris, Illinois (DOE95e). The historical (1970-1994) and projected (1995-2030) spent fuel inventories and accumulated radioactivities are given in Table 5-1. Projections of nuclear capacity are based on the DOE/EIA Low Case assumptions, which forecast an increase in the installed nuclear capacity from 99.1 GW(e) in 1994 to 100.3 GW(e) in 2000 and a decrease to 2.3 GW(e) by 2030, as shown in Table 5-2 (DOE95f). The Low Case scenario also assumes no reprocessing of spent fuel, that currently-licensed reactors will be retired when their initial license-terms expire, and that no new advanced LWRs will be available before 2015. Further, DOE/EIA projections assume that burnup levels of spent fuel will increase from their current average of 33,065 and 39,989 MWd/MTIHM to 42,000 and 54,000 MWd/MTIHM for BWRs and PWRs, respectively. This increase is predicted over the time period from 1994 to 2020.

Table 5-1. Historical and Projected Mass and Radioactivity of Commercial Spent Fuel (DOE94a, DOE95e, DOE95f)

End of Calendar Year	Mass Accumulated (MTIHM)[a]	Radioactivity Accumulated (10^6 Ci)[b]
1970	55	215
1975	1,567	3,315
1980	6,558	10,137
1985	12,684	14,228
1990	21,547	22,910
1994	29,811	26,661
1995[c]	30,800	25,600
2000[c]	42,300	32,600
2005[c]	52,000	36,900
2010[c]	61,800	39,800
2015[c]	71,300	36,700
2020[c]	77,100	34,700
2025[c]	81,900	32,100
2030[c]	84,100	24,700

[a] Metric tons initial heavy metal refers to the original mass of the actinide elements of the fuel.

[b] A Curie of radioactivity corresponds to 3.7 x 10^{10} disintegrations per second.

[c] Projections beyond 1994 are based on the DOE/EIA Low Case.

A summary inventory of DOE spent nuclear fuel is given in Table 5-3. Spent nuclear fuel in this listing includes material from fuels other than those discharged from production reactors.

Table 5-2. Historical and Projected* Installed Nuclear Electric Power Capacity (DOE94a)

End of Calendar Year	Total GW(e)	End of Calendar Year	Total GW(e)
1960	0.3	1995*	99.1
1965	0.4	2000*	100.3
1970	5.8	2005*	100.3
1975	38.3	2010*	91.1
1980	51.9	2015*	61.4
1985	78.5	2020*	46.7
1990	99.6	2025*	22.0
1994	99.1	2030*	2.3

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

It includes nuclear fuel that has been withdrawn from or resides in storage at a reactor following irradiation and that has not been reprocessed. Also included are some defective fuel elements and special nuclear forms. This table also includes some commercially generated nuclear fuels and fuels from foreign reactors and university research reactors, all of which are stored at DOE facilities.

5.3 DEFENSE HIGH-LEVEL RADIOACTIVE WASTE

High-level radioactive wastes are the highly radioactive materials resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing, and any solid material derived from such liquid waste (EPA85, NWP83). Commercial high-level radioactive waste currently stored at the West Valley Demonstration Project will be converted to a solid form (glass) prior to disposal (NRC88).

High-level waste is generated by the chemical reprocessing of spent research and production reactor fuel, irradiated targets, and naval propulsion fuel. The fission products, actinides, and neutron-activated products of particular importance are the same for high-level waste as those listed for the spent fuel assemblies (DOE88, DOE95e).

Weapons program reactors were operated mainly to produce plutonium. Reprocessing to recover the plutonium was an integral part of the weapons program. Naval propulsion reactor fuel elements were also reprocessed to recover the highly enriched uranium that remained after

Table 5-3. DOE Spent Nuclear Fuel Inventory (DOE95a)

Generator or Storage Site	Existing (1995)		Future Increases (through 2035)		Total (2035)	
	MTHM[a]	Percent	MTHM	Percent	MTHM	Percent
DOE Sites						
Hanford Site	2132.44	80.6	0.00	0.0	2132.44	77.8
Idaho National Engineering Laboratory	261.23	9.9	12.92	13.5	274.14	10.0
Savannah River Site	206.27	7.8	0.00	0.0	206.27	7.5
Oak Ridge Reservation	0.65	<0.1	1.13	1.2	1.78	<0.1
Other DOE Sites	0.78	<0.1	1.50	1.6	2.28	<0.1
Naval Nuclear Propulsion Reactors	0.00[b]	0.0	55.00	57.6	55.00	2.0
Foreign Research Reactors	0.00	0.0	21.70	22.7	21.70	0.8
Non-DOE Domestic						
Domestic Research and Test Reactors[c]	2.22	<0.1	3.28	3.4	5.50	0.2
Special-Case Commercial SNF at non-DOE locations[d]	42.69	1.6	0	0	42.69	1.6
Total[e]	2646.27		95.53		2741.80	
Percent of 2035 total	96.5		3.5		100.0	

[a] MTHM = metric tons of heavy metal.

[b] The existing inventory of Naval Nuclear Propulsion Program spent nuclear fuel (10.23 MTHM) stored at the INEL is included in the INEL total.

[c] Includes research reactors at commercial, university, and government facilities.

[d] The total inventory of spent nuclear fuel from special case commercial reactors is 186.41 MTHM. The 42.69 MTHM listed here is that stored at the Babcock & Wilcox Research Center, Fort St. Vrain Reactor, and West Valley Demonstration Project. The remaining special-case commercial spent nuclear fuel is stored at the INEL, the Oak Ridge Reservation, and the Savannah River Site, and is included in the totals for those locations.

[e] Numbers may not sum due to rounding.

use. DOE decided in 1992 to phase out the domestic reprocessing of irradiated nuclear fuel, so minimal amounts of high-level waste will be added to the current inventory.

High-level radioactive waste that is generated by the reprocessing of spent reactor fuel and targets contains more than 99 percent of the non-volatile fission products produced in the fuel or targets during reactor operation. It generally contains about 0.5 percent of the uranium and plutonium originally present in the fuel. Most of the current high-level waste inventory, which is the result of DOE national defense activities, is stored at the SRS, INEL, and the Hanford Site. These high-level wastes have to date been through one or more treatment steps (e.g., neutralization, precipitation, decantation, evaporation). They will be solidified, using a vitrification process, for disposal.

The DOE defense high-level waste 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 SRS and the Hanford Site, the acidic liquid waste from reprocessing defense reactor fuel is or has been made alkaline by the addition of caustic soda and stored in tanks. During storage, this alkaline waste separates 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.

Both alkaline and acidic high-level wastes were generated at West Valley. The alkaline waste was generated by reprocessing commercial power reactor fuels and some Hanford N-Reactor fuels, whereas acidic waste was generated by reprocessing a small amount of commercial fuel containing thorium.

Projected volumes and total radioactivity for high-level waste stored at the Hanford Site, INEL, SRS and WVDP are given in Table 5-4. Projected inventories for each site are based on specific assumptions and are subject to change. New treatment methods and waste forms are possible and may affect the future projections. Since all sites are progressing toward closure, there should be minimal amounts of waste added to the current inventory. Interim storage of DOE high-level waste will be required and will most likely be at the site where the waste is produced. Current DOE policy states that DOE high-level waste will not be accepted at the geologic repository until six years after initial receipt of commercial spent nuclear fuel (DOE94b).

5.3.1 High-Level Waste Inventories at the Hanford Site

The alkaline high-level waste (239,000 m^3) located at Hanford is stored in underground carbon-steel tanks. Currently 155,800 m^3 is solid (salt cake and sludge) and 83,200 m^3 is

Table 5-4. Historical and Projected Cumulative Volume and Radioactivity of High-Level Waste Stored in Tanks, Bins, and Capsules By Site (DOE95d, DOE95e)

End of Calendar Year	Volume, 10^3 m^3					Radioactivity, 10^6 Ci				
	Hanford	INEL	SRS	WVDP	Total	Hanford	INEL	SRS	WVDP	Total
1980	219.4	11.4	96.7	2.2	329.7	576.7	53.4	699.0	33.5	1,362.6
1981	219.4	12.0	105.7	2.2	339.3	550.2	63.6	982.0	32.7	1,628.5
1982	213.3	11.5	115.0	2.2	342.0	437.1	71.6	828.8	31.9	1,369.4
1983	229.4	9.7	111.4	2.2	352.7	427.5	64.8	776.2	31.2	1,299.7
1984	225.6	10.1	125.6	2.2	363.5	470.2	58.6	795.9	30.5	1,355.2
1985	222.1	10.1	122.7	2.2	357.1	519.0	69.4	841.4	29.7	1,459.5
1986	226.4	9.5	127.8	2.2	365.9	534.6	60.6	794.7	29.1	1,419.0
1987	239.7	11.9	127.6	2.2	381.4	478.2	62.5	734.0	28.4	1,303.1
1988	243.4	11.0	128.4	2.1	384.9	447.4	67.0	664.4	27.9	1,206.7
1989	244.8	12.0	122.0	2.4	381.1	419.3	68.4	598.9	27.3	1,113.9
1990	253.6	12.0	131.7	1.2	398.5	399.3	63.2	561.6	26.7	1,050.8
1991	256.4	10.4	127.9	1.7	396.5	384.2	59.4	537.6	26.2	1,007.4
1992	258.7	11.2	126.9	1.6	398.3	372.1	50.8	632.4	25.9	1,081.2
1993	261.7	10.5	129.3	2.0	403.5	361.4	52.5	606.0	25.3	1,045.3
1994	238.9	11.0	126.3	2.2	378.4	348.0	51.6	534.5	24.7	958.8
1995	237.3	11.4	122.1	0.9	371.7	339.9	50.5	502.2	24.1	916.7
2000	232.1	9.8	98.7		340.6	302.4	44.1	352.7		699.2
2005	229.6	9.5	75.2		314.2	269.1	38.9	239.5		547.5
2010	197.9	9.8	51.7		259.4	232.2	34.6	147.0		413.8
2015	134.4	10.4	28.2		173.0	128.9	30.8	71.7		231.3
2020	70.8	7.2	4.7		82.8	45.5	26.9	10.7		83.1
2025	26.3	5.8			32.0	15.2	13.8			29.0
2030	2.4	4.1			6.5		3.9			3.9

liquid; waste volumes change with time because of on-going waste management activities. There are approximately 350 million curies of total radioactivity contained in the waste, which has been accumulating since 1944. The high-level waste 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 waste, converted to solids as strontium fluoride and cesium chloride, placed in double-walled capsules, and stored in water basins. A total of 2,217 capsules were manufactured and 1,933 remain. (A portion of these capsules have been used outside the facility or have been dismantled.)

5.3.2 High-Level Waste Inventories at INEL

About 11,000 m^3 of high-level waste, containing approximately 50 million curies of total radioactivity, is currently stored at INEL; this volume consists of 7,200 m^3 of acidic liquid waste (1,306 m^3 is high-level waste; the remainder is sodium waste that was treated as high-level waste) and 3,800 m^3 of solid materials. Liquid high-level waste was generated at INEL primarily by the reprocessing of spent fuel from the national defense (naval propulsion nuclear reactors) and reactor testing programs; a small amount was also generated by reprocessing fuel from non-defense research reactors. This acidic waste is stored underground in large, high-integrity, stainless steel tanks. Waste that has been converted to a calcine is stored in retrievable stainless steel bins housed in reinforced concrete vaults. Greater than 90 percent of the total radioactivity is contained in the calcine.

5.3.3 High-Level Waste Inventories at SRS

Approximately 126,300 m^3 of alkaline high-level waste that has accumulated at the SRS over the past three decades is currently stored underground in carbon-steel tanks. The current inventories consist of alkaline liquid, sludge, and salt cake that were generated primarily by the reprocessing of nuclear fuels and targets from plutonium production reactors. The sludge is formed after treatment with caustic agents. Salt cake results when the supernatant liquor is concentrated in waste treatment evaporators. The high-level waste consists of 58,100 m^3 of liquid and 68,200 m^3 of solid material having a total radioactivity of approximately 500 million curies.

5.3.4 High-Level Waste Inventories at WVDP

About 2,180 m^3 of high-level waste is stored at the WVDP Facility and consists of 2,040 m^3 of liquid alkaline waste and 140 m^3 of solid waste (consisting of alkaline sludge and inorganic zeolite ion-exchange medium). The alkaline waste is stored in an underground carbon-steel tank and the zeolite waste is stored in an underground carbon-steel tank covered by an aqueous alkaline solution. Reprocessing at the WVDP plant was discontinued in 1972, and no additional high-level waste has been generated since then.

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NRC88	U.S. Nuclear Regulatory Commission, Code of Federal Regulations, Title 10, Part 60, *Disposal of High-Level Radioactive Wastes in Geologic Repositories*, as amended, October 1988.
NWP83	*Nuclear Waste Policy Act of 1982*, Public Law 97-425, January 7, 1983.

CHAPTER 6

DOSE AND RISK ESTIMATION

6.1 INTRODUCTION

Ionizing radiations emitted by the radioactive decay of nuclides released into the environment pose a risk of inducing excess cancers or heritable genetic effects in exposed human populations. Exposure can occur through several "pathways," including inhalation, ingestion, or external irradiation by radionuclides in the air or deposited on the ground (see Chapter 7). The risk of a health effect being induced in an exposed individual by a given environmental exposure is calculated by first estimating the radiation dose to sensitive tissues in the individual, as a function of age. Depending on the radionuclide in question, its chemical form, and the exposure pathway, its distribution will vary within the body and with time, leading to a variation in radiation dose with organ and across time. The dose per unit exposure is referred to as a "dose conversion factor" (DCF). From the tissue specific doses, the risks of a radiation induced cancer, cancer death, or genetic effect are calculated using age- and organ-specific "risk factors." The dose conversion and risk factors are generally calculated from models, as outlined below. The number of excess cancers in a population are projected using a life-table calculation (BUN81, EPA94), which corrects for competing causes of death.

6.2 DOSE ESTIMATION

The risk of inducing a cancer in a specific tissue or organ increases with the absorbed dose, i.e., the amount of ionization and excitation energy per unit mass, deposited in that tissue or organ. The risk of inducing a genetic effect, likewise, increases with dose to the testes or ovaries. The absorbed dose, D, is expressed in gray (Gy) or rad, where 1 Gy = 100 rad. The risk also depends on the the density of ionizations (the number of ionizations per unit pathlength) produced by the radiation. Accordingly, a derived quantity called the effective dose is introduced, which is expressed in units of sieverts (Sv) or rem. The effective dose in a tissue is given by $Q \times D$, where Q is a quality factor (unitless) defined for a specific type of radiation. We are here concerned only with: (1) "low-LET" radiation from beta particles, gamma rays, or energetic X-rays, for which Q is taken to be unity and (2) "high-LET" alpha particles for which Q is taken to be 20 (ICR91, EPA94). In the case of low-LET radiation, 1 Sv = 1 Gy, and 1 rem = 1 rad. Since Q depends only on the type of radiation and not the dose, it follows that 1 Sv = 100 rem.

When the exposure is external, the dose calculation is a straightforward application of radiation physics. The radiation doses to target organs in an idealized "reference man" are calculated from the decay properties of the radionuclides and the well understood interactions of radiation with matter (ICR79, EPA89).

For ingested or inhaled radionuclides, the dosimetry modeling is more complex. It is necessary to incorporate biokinetic information to describe the distribution and retention of the radionuclide (and any radioactive decay products) in the body as a function of time after intake. The irradiation of target tissues by internally deposited radionuclides is further complicated by the need to consider the cross irradiation of one tissue by radionuclides deposited in another tissue. Dosimetry models for internally deposited radionuclides have been developed by the International Commission on Radiological Protection (ICR79, ICR80, ICR81, ICR88). The set of dosimetry models employed by EPA were developed at the Oak Ridge National Laboratory. For the most part, these models are very similar to those recommended by the ICRP. The main difference is that while the ICRP models assume that any decay product has the same biokinetic behavior as its parent, the EPA models assume that a decay product of an inhaled or ingested radionuclide behaves biokinetically as if the decay product itself had been inhaled or ingested.

Detailed descriptions of the EPA dosimetry models for individual radionuclides have been published (SUL81). A discussion of the uncertainties in the estimates of dose from internally deposited radionuclides is included in the Background Information Document for the Radionuclide NESHAPs rulemaking (EPA89).

6.3 CANCER RISK ESTIMATION

From 1984 until 1994, EPA's estimates of risk from low-LET radiation were based on the 1980 National Academy of Sciences' BEIR III Report (NAS80, EPA84, EPA89). Subsequent to the publication of BEIR III, important new data have become available, especially revised dosimetry and further epidemiological follow-up on the Japanese atomic bomb survivors. Following an extensive review of the available information, carried out in consultation with its Science Advisory Board, the Agency has published a revised methodology for estimating radiogenic cancer risks (EPA94).

The revised risk model incorporates age- and organ-specific risk coefficients for low-LET radiation based on data obtained from the Japanese atomic bomb survivors up through 1985, supplemented by organ-specific data from other sources (e.g., breast cancer induction in fluoroscopy patients). For most cancer sites, EPA's methodology involves an averaging of two sets of coefficients, reflecting two different ways of projecting risk from the atomic bomb survivors to the U.S. population, which have significantly different baseline rates of specific cancers (LAN91, EPA94).

Aside from breast cancer, for which there is good epidemiological evidence that the dose response is approximately linear and independent of fractionation (NAS90), it was assumed that the risks at low doses and dose rates are reduced by a "dose, dose rate effectiveness factor" (DDREF) of 2 compared to the acute high dose exposures experienced by the bomb survivors. The value of 2 for the DDREF is consistent with ICRP recommendations (ICR91). For low dose (or dose rate) conditions, the calculated risk of a premature cancer death attributable to uniform, whole-body, low-LET irradiation is about 5.1×10^{-2}/Gy. Neglecting nonfatal skin cancers, which are usually not serious, the corresponding incidence risk estimate is 7.6×10^{-2}/Gy. A preliminary analysis of the uncertainties indicated that the whole-body mortality risk estimate could be about a factor of 2 times higher or 2.5 times lower (EPA95).

High-LET (alpha particle) risks are presumed to increase linearly with dose and to be independent of dose rate. Except for leukemia and breast cancer, a relative biological effectiveness (RBE) factor of 20 is adopted for estimating the risk of high-LET radiation relative to that for low-LET radiation at low dose or dose rate conditions. Again the RBE value of 20 is consistent with the recommendations of the ICRP (ICR91). In view of epidemiological data on people ingesting or being injected with alpha-emitting radionuclides that deposit in bone, an effective RBE of 1 was adopted for leukemia; for breast cancer, the high-LET RBE of 10 is used to be consistent with the DDREF of 1 adopted for this site.

The lifetime excess risks of cancer incidence and mortality, for constant dose rates, and for constant exposure rates of 300 different radionuclides, were projected from a life-table calculation using 1980 U.S. vital statistics. The organ-specific risk estimates for constant, low dose rate conditions are shown in Table 6-1; for comparison, risk estimates from the 1989 NESHAPs rulemaking are also shown (EPA89). The radionuclide-specific risk estimates for mortality and incidence, respectively, are exhibited in Tables 6-2a and 6-2b. The values in these tables are applicable to the estimation of: (1) individual risks under constant lifetime dose-rate conditions or (2) excess cancer incidence or mortality in a population due to low individual exposures (EPA89, EPA94). Default assumptions for inhalation class and ingestion f_1 are listed in Table 6-3 (EPA94).

Table 6-1. EPA Low Dose, Low Dose Rate Risks (10^{-4} per Gy)

Cancer Site	Mortality		Morbidity	
	NESHAPs	Revised	NESHAPs	Revised
Esophagus	9.1	9.0	9.1	9.5
Stomach	46.0	44.4	60.1	49.3
Colon	22.9	98.2	42.9	178.5
Liver	49.6	15.0	49.6	15.8
Lung	70.1	71.6	74.5	75.4
Bone	2.5	0.9	2.5	1.3
Skin	--	1.0	--	1.0
Breast	55.4	23.1	142.0	46.2
Ovary	--	16.6	--	23.8
Bladder	11.8	24.9	21.4	49.7
Kidney	5.9	5.5	21.4	8.4
Thyroid	6.4	3.2	64.3	32.1
Leukemia	44.8	49.6	44.8	50.1
Remainder	67.8	123.1	90.5	173.4
Total	392.1	486.0	623.0	714.4

Notes for Table 6-1.

The Dose and Dose Rate Effectiveness Factor (DDREF) is 1 for breast and 2 for all other sites. These risk coefficients are applicable to all doses less than 200 mGy and for total doses greater than 200 mGy from dose rates less than 0.1 mGy/min. The revised model morbidity estimate shown for skin is for fatalities only. The entire morbidity risk for skin would be 500 times greater. The thyroid morbidity risk includes only malignant neoplasms and does not include benign tumors or nodules.

Table 6-2a. Radionuclide Mortality Risk Coefficients

	Internal		External	
Nuclide	Ingestion (Bq^{-1})	Inhalation (Bq^{-1})	Submersion (per Bq y/m^3)	Gnd Surface (per Bq y/m^2)
H-3	1.28E-12	1.73E-12		
Be-7	1.38E-12	3.56E-12	2.83E-09	5.88E-11
C-11	1.01E-12	8.36E-13	5.83E-08	1.20E-09
C-14	1.93E-11	1.31E-13		
C-15	1.58E-14	2.06E-14	2.68E-07	3.10E-09
N-13	7.01E-13	6.36E-13	5.83E-08	1.20E-09
O-15	2.23E-13	2.45E-13	5.83E-08	1.20E-09
F-18	2.34E-12	1.56E-12	5.65E-08	1.17E-09
Na-22	1.50E-10	8.95E-11	1.29E-07	2.42E-09
Na-24	2.57E-11	1.52E-11	2.82E-07	4.20E-09
Si-31	8.42E-12	8.09E-12	5.35E-11	9.35E-13
P-32	1.14E-10	6.21E-11		
P-33	1.42E-11	7.91E-12		
S-35	6.92E-12	3.33E-12		
Cl-36	4.02E-11	2.36E-11	3.27E-16	3.60E-17
Cl-38	4.72E-12	4.06E-12	1.01E-07	1.57E-09
Ar-41		1.05E-14	7.74E-08	1.35E-09
K-40	2.28E-10	1.33E-10	9.49E-09	1.60E-10
K-42	2.36E-11	1.65E-11	1.70E-08	2.83E-10
Ca-45	3.84E-11	5.77E-11	5.92E-19	1.56E-19
Ca-47	1.05E-10	1.04E-10	6.41E-08	1.13E-09
Sc-46	8.87E-11	2.73E-10	1.22E-07	2.26E-09
Sc-47	4.44E-11	3.83E-11	6.10E-09	1.35E-10
Sc-48	1.03E-10	7.99E-11	2.04E-07	3.67E-09
V-48	1.17E-10	1.32E-10	1.76E-07	3.21E-09
Cr-51	2.16E-12	3.59E-12	1.77E-09	3.82E-11
Mn-52	9.59E-11	8.34E-11	2.08E-07	3.81E-09
Mn-54	3.32E-11	7.31E-11	5.05E-08	9.73E-10
Mn-56	1.46E-11	1.26E-11	1.09E-07	1.88E-09
Fe-55	6.24E-12	1.08E-11	8.87E-13	1.71E-13
Fe-59	9.56E-11	1.37E-10	7.20E-08	1.28E-09
Co-57	1.65E-11	6.47E-11	6.65E-09	1.50E-10
Co-58	4.69E-11	1.07E-10	5.83E-08	1.14E-09
Co-58m	1.49E-12	1.93E-12	1.33E-12	2.75E-13
Co-60	3.36E-10	1.46E-09	1.51E-07	2.65E-09
Ni-59	3.04E-12	8.00E-12	1.48E-12	3.22E-13
Ni-63	8.85E-12	2.01E-11		
Ni-65	9.51E-12	8.80E-12	3.34E-08	5.78E-10
Cu-64	8.29E-12	9.33E-12	1.09E-08	2.22E-10
Zn-65	1.78E-10	1.94E-10	3.52E-08	6.37E-10
Zn-69	1.29E-12	2.67E-12	3.40E-13	7.16E-15
Zn-69m	2.36E-11	2.57E-11	2.36E-08	4.95E-10
Ga-67	1.28E-11	9.81E-12	7.97E-09	1.77E-10
Ga-72	7.38E-11	4.39E-11	1.77E-07	2.97E-09
Ge-71	2.03E-13	1.45E-12	3.52E-12	9.03E-13
As-73	1.28E-11	3.09E-11	1.97E-10	8.14E-12
As-74	6.62E-11	9.71E-11	4.43E-08	9.00E-10
As-76	1.01E-10	9.16E-11	2.55E-08	4.91E-10
As-77	2.52E-11	2.40E-11	4.86E-10	1.04E-11
Se-75	1.21E-10	9.46E-11	2.17E-08	4.73E-10
Br-82	2.54E-11	1.45E-11	1.58E-07	3.00E-09
Kr-83m		7.81E-16	3.41E-12	8.26E-13
Kr-85		6.08E-15	1.28E-10	2.63E-12
Kr-85m		5.93E-15	8.83E-09	1.95E-10
Kr-87		2.68E-14	5.23E-08	8.65E-10
Kr-88		4.60E-14	1.34E-07	2.08E-09
Kr-89		3.68E-14	1.19E-07	1.95E-09
Kr-90		3.86E-14	7.82E-08	1.39E-09
Rb-82	2.51E-13	3.00E-13	6.29E-08	1.29E-09
Rb-86	1.38E-10	8.28E-11	5.75E-09	1.05E-10
Rb-87	7.27E-11	4.43E-11		
Rb-88	3.47E-12	3.46E-12	4.22E-08	6.78E-10
Rb-89	1.97E-12	1.73E-12	1.33E-07	2.23E-09
Sr-82	4.06E-10	1.68E-10	1.20E-11	2.89E-12
Sr-85	2.39E-11	2.07E-11	2.92E-08	6.04E-10
Sr-85m	3.38E-13	1.51E-13	1.21E-08	2.64E-10
Sr-89	1.63E-10	7.18E-11	8.32E-12	1.57E-13
Sr-90	9.64E-10	1.51E-09		
Sr-91	4.42E-11	1.46E-11	4.16E-08	7.85E-10
Sr-92	3.14E-11	8.53E-12	8.09E-08	1.39E-09
Y-90	2.25E-10	1.89E-10		
Y-91	2.02E-10	4.05E-10	2.18E-10	3.86E-12
Y-91m	6.76E-13	7.02E-13	3.06E-08	6.24E-10
Y-92	3.15E-11	3.92E-11	1.53E-08	2.81E-10
Y-93	8.76E-11	7.55E-11	5.60E-09	9.90E-11
Zr-93	8.55E-12	1.29E-10		
Zr-95	5.99E-11	1.33E-10	4.39E-08	8.62E-10
Zr-97	1.57E-10	9.39E-11	1.09E-08	2.01E-10
Nb-93m	1.00E-11	1.07E-10	4.42E-12	6.78E-13
Nb-94	1.08E-10	1.79E-09	9.47E-08	1.84E-09
Nb-95	3.50E-11	6.27E-11	4.59E-08	8.96E-10
Nb-95m	4.60E-11	4.32E-11	3.34E-09	7.48E-11

Table 6-2a. (Continued)

	Internal		External	
Nuclide	Ingestion (Bq^{-1})	Inhalation (Bq^{-1})	Submersion (per Bq y/m^3)	Gnd Surface (per Bq y/m^2)
Nb-97	3.34E-12	5.38E-12	3.92E-08	7.82E-10
Nb-97m	6.41E-14	8.41E-14	4.36E-08	8.54E-10
Mo-99	3.88E-11	8.60E-11	9.15E-09	1.82E-10
Tc-95	1.22E-12	7.59E-13	4.72E-08	9.21E-10
Tc-95m	2.22E-11	4.38E-11	3.91E-08	7.85E-10
Tc-96	3.95E-11	3.86E-11	1.51E-07	2.90E-09
Tc-96m	4.66E-13	4.83E-13	2.50E-09	4.90E-11
Tc-97	2.83E-12	8.02E-12	3.05E-11	4.13E-12
Tc-97m	2.10E-11	4.49E-11	4.33E-11	3.71E-12
Tc-99	2.49E-11	6.75E-11	2.63E-14	6.22E-16
Tc-99m	8.87E-13	7.66E-13	7.02E-09	1.56E-10
Ru-97	9.18E-12	7.83E-12	1.28E-08	2.82E-10
Ru-103	5.09E-11	9.66E-11	2.76E-08	5.70E-10
Ru-105	1.83E-11	1.82E-11	4.62E-08	9.24E-10
Ru-106	5.28E-10	2.75E-09		
Rh-103m	1.66E-13	3.26E-13	6.59E-12	7.46E-13
Rh-105	2.91E-11	2.38E-11	4.37E-09	9.40E-11
Rh-105m	1.98E-14	2.11E-14	1.51E-09	3.66E-11
Rh-106	8.78E-14	1.18E-13	1.21E-08	2.41E-10
Pd-100	5.76E-11	6.71E-11		
Pd-101	5.89E-12	4.82E-12		
Pd-103	1.58E-11	2.23E-11	6.03E-11	6.62E-12
Pd-107	3.14E-12	3.61E-11		
Pd-109	5.05E-11	4.17E-11	3.94E-11	8.11E-13
Ag-105	2.67E-11	4.75E-11		
Ag-108	1.69E-13	2.42E-13	1.02E-09	2.07E-11
Ag-108m	1.03E-10	1.49E-09	9.41E-08	1.90E-09
Ag-109m	6.55E-15	8.87E-15	2.04E-10	7.24E-12
Ag-110	5.94E-14	8.10E-14	1.80E-09	3.59E-11
Ag-110m	1.43E-10	6.78E-10	1.65E-07	3.11E-09
Ag-111	1.03E-10	1.01E-10	1.48E-09	3.19E-11
Cd-109	1.38E-10	4.29E-10	7.25E-11	6.22E-12
Cd-115	1.10E-10	9.45E-11	1.17E-08	2.40E-10
Cd-115m	2.19E-10	3.64E-10	1.33E-09	2.44E-11
In-113m	1.50E-12	1.43E-12	1.44E-08	3.07E-10
In-114	1.10E-13	1.49E-13	1.85E-09	3.71E-11
In-114m	3.18E-10	5.39E-10	5.15E-09	1.08E-10
In-115	8.14E-10	5.00E-09		
In-115m	5.46E-12	4.09E-12	8.92E-09	1.93E-10
Sn-113	5.66E-11	1.46E-10	3.92E-10	1.26E-11
Sn-121	1.84E-11	1.20E-11		
Sn-121m	3.08E-11	1.63E-10	1.23E-11	4.38E-13
Sn-125	2.51E-10	2.32E-10	1.88E-08	3.39E-10
Sn-126	3.26E-10	9.07E-10	2.36E-09	5.89E-11
Sb-122	1.33E-10	1.04E-10	2.57E-08	5.21E-10
Sb-124	1.65E-10	2.70E-10	1.16E-07	2.05E-09
Sb-125	4.59E-11	1.10E-10	2.43E-08	5.01E-10
Sb-126	1.51E-10	1.62E-10	1.63E-07	3.24E-09
Sb-126m	1.60E-12	1.62E-12	9.22E-08	1.85E-09
Sb-127	1.28E-10	1.16E-10	3.88E-08	7.78E-10
Sb-129	2.92E-11	1.95E-11	8.69E-08	1.63E-09
Te-125m	4.08E-11	6.14E-11	3.42E-10	1.48E-11
Te-127	1.32E-11	9.46E-12	2.74E-10	5.79E-12
Te-127m	1.02E-10	2.99E-10	1.12E-10	4.74E-12
Te-129	2.90E-12	3.71E-12	3.08E-09	6.38E-11
Te-129m	1.84E-10	2.87E-10	1.94E-09	4.02E-11
Te-131	2.59E-12	2.67E-12	2.44E-08	4.91E-10
Te-131m	8.77E-11	7.06E-11	8.59E-08	1.63E-09
Te-132	1.66E-10	1.45E-10	1.19E-08	2.66E-10
I-122	5.12E-13	5.69E-13	5.59E-08	1.14E-09
I-123	2.24E-12	1.31E-12	8.58E-09	1.94E-10
I-125	7.24E-11	4.78E-11	3.86E-10	1.71E-11
I-126	1.38E-10	9.01E-11	2.65E-08	5.38E-10
I-129	5.06E-10	3.35E-10	2.98E-10	1.56E-11
I-130	2.04E-11	1.19E-11	1.26E-07	2.50E-09
I-131	1.04E-10	6.72E-11	2.16E-08	4.55E-10
I-132	6.30E-12	4.00E-12	1.37E-07	2.64E-09
I-133	3.53E-11	2.11E-11	3.52E-08	7.06E-10
I-134	3.87E-12	2.62E-12	1.60E-07	2.99E-09
I-135	1.12E-11	6.80E-12	9.74E-08	1.69E-09
Xe-122		5.60E-14	3.23E-09	7.54E-11
Xe-123		1.39E-14	3.63E-08	7.08E-10
Xe-125		1.93E-14	1.38E-08	3.01E-10
Xe-127		7.37E-15	1.44E-08	3.21E-10
Xe-129m		1.09E-14	1.01E-09	3.78E-11
Xe-131m		7.86E-15	3.65E-10	1.42E-11
Xe-133		7.87E-15	1.64E-09	4.58E-11
Xe-133m		9.81E-15	1.54E-09	3.99E-11
Xe-135		1.44E-14	1.39E-08	3.00E-10
Xe-135m		3.81E-15	2.45E-08	5.05E-10
Xe-137		3.17E-14	1.10E-08	2.16E-10
Xe-138		4.21E-14	7.33E-08	1.23E-09

Table 6-2a. (Continued)

Nuclide	Internal		External	
	Ingestion (Bq^{-1})	Inhalation (Bq^{-1})	Submersion (per Bq y/m^3)	Gnd Surface (per Bq y/m^2)
Cs-131	3.28E-12	1.91E-12	2.51E-10	1.39E-11
Cs-134	8.49E-10	5.01E-10	9.22E-08	1.82E-09
Cs-134m	9.17E-13	7.09E-13	1.04E-09	2.71E-11
Cs-135	8.27E-11	4.78E-11		
Cs-136	1.38E-10	8.10E-11	1.30E-07	2.47E-09
Cs-137	5.71E-10	3.34E-10		
Cs-138	3.96E-12	3.21E-12	1.49E-07	2.52E-09
Ba-131	2.64E-11	8.35E-12	2.56E-08	5.46E-10
Ba-133	4.53E-11	7.90E-11	2.02E-08	4.52E-10
Ba-133m	4.17E-11	9.49E-12	3.03E-09	7.19E-11
Ba-137m	5.35E-14	3.90E-14	3.52E-08	7.01E-10
Ba-139	5.72E-12	3.59E-12	1.96E-09	4.19E-11
Ba-140	1.78E-10	5.49E-11	1.07E-08	2.23E-10
La-140	1.44E-10	9.75E-11	1.44E-07	2.49E-09
Ce-141	5.86E-11	8.99E-11	4.07E-09	9.21E-11
Ce-143	8.91E-11	7.48E-11	1.46E-08	3.14E-10
Ce-144	4.42E-10	2.60E-09	9.32E-10	2.18E-11
Pr-142	1.05E-10	8.44E-11	3.64E-09	6.02E-11
Pr-143	9.88E-11	1.10E-10	5.33E-16	1.04E-17
Pr-144	1.91E-12	3.37E-12	2.10E-09	3.49E-11
Pr-144m	7.46E-13	1.44E-12	2.01E-10	7.63E-12
Nd-147	8.82E-11	9.47E-11	7.23E-09	1.59E-10
Nd-149	7.60E-12	9.89E-12	2.12E-08	4.53E-10
Pm-147	2.12E-11	1.84E-10	1.91E-13	4.27E-15
Pm-148	2.17E-10	2.00E-10	3.46E-08	6.24E-10
Pm-148m	1.52E-10	6.26E-10	1.17E-07	2.33E-09
Pm-149	8.27E-11	6.83E-11	6.59E-10	1.40E-11
Sm-147	5.12E-10	1.69E-07		
Sm-151	6.99E-12	1.15E-10	3.02E-14	3.48E-15
Sm-153	6.04E-11	4.17E-11	2.45E-09	6.57E-11
Eu-152	9.17E-11	1.69E-09	6.79E-08	1.28E-09
Eu-154	1.47E-10	2.01E-09	7.51E-08	1.39E-09
Eu-155	2.54E-11	2.23E-10	2.81E-09	6.94E-11
Eu-156	1.65E-10	1.83E-10	8.59E-08	1.46E-09
Gd-153	2.01E-11	7.03E-11	4.18E-09	1.15E-10
Gd-159	3.93E-11	2.48E-11	2.04E-09	4.62E-11
Tb-158	6.54E-11	1.53E-09	4.55E-08	8.62E-10
Tb-160	1.16E-10	2.37E-10	6.48E-08	1.22E-09
Dy-165	5.54E-12	5.54E-12	1.34E-09	2.93E-11
Dy-166	1.40E-10	1.49E-10	1.48E-09	4.04E-11
Ho-166	1.14E-10	7.92E-11	1.54E-09	3.04E-11
Er-169	3.17E-11	2.93E-11	9.54E-14	4.33E-15
Er-171	2.53E-11	1.65E-11	2.03E-08	4.46E-10
Tm-170	1.12E-10	2.41E-10	2.18E-10	5.82E-12
Tm-171	8.79E-12	4.23E-11	2.33E-11	6.94E-13
Yb-169	5.21E-11	8.31E-11	1.45E-08	3.55E-10
Yb-175	3.60E-11	3.19E-11	2.18E-09	4.71E-11
Lu-177	4.42E-11	4.22E-11	1.87E-09	4.21E-11
Hf-181	8.30E-11	1.26E-10	3.05E-08	6.45E-10
Ta-182	1.07E-10	3.61E-10	7.70E-08	1.41E-09
W-181	4.29E-12	1.47E-12	1.47E-09	4.17E-11
W-185	3.07E-11	7.29E-12	1.46E-12	3.26E-14
W-187	3.77E-11	9.25E-12	2.74E-08	5.59E-10
Re-183	2.82E-11	5.77E-11	7.12E-09	1.76E-10
Re-186	4.80E-11	5.47E-11	1.04E-09	2.44E-11
Re-187	1.63E-13	4.41E-13		
Re-188	4.69E-11	4.89E-11	3.31E-09	6.85E-11
Os-185	2.86E-11	9.57E-11	4.08E-08	8.25E-10
Os-191	4.58E-11	5.38E-11	3.44E-09	8.36E-11
Os-191m	7.50E-12	6.88E-12	1.89E-10	5.45E-12
Os-193	6.56E-11	5.26E-11	3.65E-09	7.95E-11
Ir-190	7.68E-11	8.70E-11	7.98E-08	1.65E-09
Ir-192	9.84E-11	2.42E-10	4.64E-08	9.80E-10
Ir-194	1.06E-10	8.49E-11	5.28E-09	1.07E-10
Pt-191	2.30E-11	7.09E-12	1.51E-08	3.37E-10
Pt-193	2.43E-12	1.38E-12	1.70E-12	4.31E-13
Pt-193m	3.76E-11	9.62E-12	4.56E-10	1.22E-11
Pt-197	3.21E-11	7.82E-12	1.14E-09	2.74E-11
Pt-197m	5.32E-12	2.07E-12	4.06E-09	9.42E-11
Au-196	2.04E-11	2.00E-11	2.60E-08	5.65E-10
Au-198	7.99E-11	7.02E-11	2.29E-08	4.83E-10
Hg-197	1.79E-11	1.33E-11	2.99E-09	7.60E-11
Hg-203	4.02E-11	6.10E-11	1.27E-08	2.77E-10
Tl-202	1.83E-11	1.08E-11	2.58E-08	5.51E-10
Tl-204	3.54E-11	2.10E-11	4.84E-11	1.23E-12
Tl-207	2.61E-13	3.16E-13	1.32E-10	2.51E-12
Tl-208	3.95E-13	3.42E-13	2.33E-07	3.56E-09
Tl-209	3.08E-13	2.82E-13	1.28E-07	2.27E-09
Pb-203	1.62E-11	5.66E-12	1.62E-08	3.61E-10
Pb-209	3.45E-12	1.45E-12		
Pb-210	1.46E-08	3.59E-08	5.49E-11	2.41E-12
Pb-211	7.15E-12	2.61E-10	3.00E-09	5.99E-11

Table 6-2a. (Continued)

	Internal		External	
Nuclide	Ingestion (Bq^{-1})	Inhalation (Bq^{-1})	Submersion (per Bq y/m^3)	Gnd Surface (per Bq y/m^2)
Pb-212	2.99E-10	8.68E-10	7.98E-09	1.77E-10
Pb-214	6.02E-12	1.56E-10	1.39E-08	2.98E-10
Bi-206	1.11E-10	9.61E-11	1.98E-07	3.72E-09
Bi-207	7.82E-11	1.95E-10	9.16E-08	1.74E-09
Bi-210	1.10E-10	1.28E-09		
Bi-211	4.42E-13	4.48E-11	2.66E-09	5.69E-11
Bi-212	1.22E-11	9.35E-10	1.11E-08	2.06E-10
Bi-213	8.90E-12	7.92E-10	7.90E-09	1.65E-10
Bi-214	4.45E-12	3.74E-10	9.48E-08	1.64E-09
Po-210	5.88E-09	5.14E-08	5.13E-13	9.95E-15
Po-212	1.10E-21	1.52E-19		
Po-213	1.57E-20	2.00E-18	1.83E-12	3.57E-14
Po-214	5.16E-19	7.12E-17	5.02E-12	9.74E-14
Po-215	1.13E-17	1.15E-15	8.46E-12	1.77E-13
Po-216	1.59E-15	7.56E-14	8.75E-13	1.69E-14
Po-218	1.10E-12	9.44E-11		
At-217	1.98E-16	1.32E-14	1.39E-11	2.80E-13
Rn-219		1.74E-12	3.22E-09	6.91E-11
Rn-220		4.42E-12	3.01E-11	6.16E-13
Rn-222		3.21E-11	2.22E-11	4.59E-13
Fr-221	3.04E-12	2.05E-10	1.70E-09	3.75E-11
Fr-223	7.91E-12	1.17E-11	2.34E-09	5.85E-11
Ra-223	3.91E-09	9.12E-08	7.18E-09	1.61E-10
Ra-224	2.46E-09	5.70E-08	5.57E-10	1.21E-11
Ra-225	2.88E-09	6.01E-08	2.63E-10	1.16E-11
Ra-226	5.26E-09	6.62E-08	3.71E-10	8.21E-12
Ra-228	4.48E-09	2.22E-08	2.48E-18	5.19E-19
Ac-225	2.19E-09	1.06E-07	7.03E-10	1.68E-11
Ac-227	8.17E-09	1.78E-06	6.41E-12	1.95E-13
Ac-228	2.55E-11	8.33E-10	5.58E-08	1.05E-09
Th-227	6.11E-10	1.10E-07	5.72E-09	1.28E-10
Th-228	9.84E-10	2.41E-06	1.01E-10	2.69E-12
Th-229	9.94E-10	1.91E-06	4.42E-09	1.05E-10
Th-230	6.39E-10	4.22E-07	1.88E-11	7.72E-13
Th-231	2.70E-11	2.18E-11	5.60E-10	1.71E-11
Th-232	5.61E-10	4.73E-07	8.51E-12	5.31E-13
Th-234	2.88E-10	3.89E-10	3.78E-10	9.65E-12
Pa-231	2.86E-09	5.85E-07	1.65E-09	3.81E-11
Pa-233	7.05E-11	1.01E-10	1.17E-08	2.58E-10
Pa-234	3.37E-11	2.91E-11	1.17E-07	2.25E-09
Pa-234m	1.16E-13	1.61E-13	6.90E-10	1.31E-11
U-232	1.40E-09	1.35E-06	1.23E-11	8.11E-13
U-233	7.44E-10	3.63E-07	1.21E-11	4.38E-13
U-234	7.38E-10	3.58E-07	6.72E-12	6.12E-13
U-235	7.53E-10	3.33E-07	8.29E-09	1.84E-10
U-236	6.99E-10	3.39E-07	5.12E-12	5.47E-13
U-237	6.01E-11	5.99E-11	6.96E-09	1.65E-10
U-238	7.31E-10	3.19E-07	4.36E-12	4.81E-13
U-240	8.29E-11	6.98E-11	2.79E-11	2.90E-12
Np-236	1.43E-11	8.96E-11	6.80E-09	1.61E-10
Np-237	6.65E-09	8.04E-07	1.13E-09	3.05E-11
Np-238	6.92E-11	9.80E-11	3.36E-08	6.24E-10
Np-239	6.42E-11	4.61E-11	8.96E-09	2.03E-10
Np-240	3.42E-12	3.26E-12	6.81E-08	1.34E-09
Np-240m	5.76E-13	7.24E-13	1.94E-08	3.82E-10
Pu-236	1.56E-09	3.39E-07	4.66E-12	7.14E-13
Pu-238	6.64E-09	6.76E-07	3.32E-12	6.11E-13
Pu-239	7.12E-09	6.82E-07	3.83E-12	2.89E-13
Pu-240	7.11E-09	6.81E-07	3.26E-12	5.86E-13
Pu-241	1.19E-10	6.55E-09		
Pu-242	6.76E-09	6.47E-07	2.78E-12	4.86E-13
Pu-243	5.80E-12	6.29E-12	1.16E-09	2.83E-11
Pu-244	6.95E-09	6.53E-07	1.94E-12	4.11E-13
Am-241	7.39E-09	8.96E-07	8.46E-10	2.61E-11
Am-242	2.25E-11	2.52E-10	7.03E-10	1.78E-11
Am-242m	6.74E-09	8.08E-07	2.01E-11	2.00E-12
Am-243	7.36E-09	8.89E-07	2.43E-09	6.21E-11
Cm-242	6.60E-10	7.81E-08	3.62E-12	6.51E-13
Cm-243	5.58E-09	6.75E-07	6.82E-09	1.55E-10
Cm-244	4.66E-09	5.70E-07	3.04E-12	5.80E-13
Cm-245	7.54E-09	9.12E-07	3.69E-09	8.68E-11
Cm-246	7.48E-09	9.08E-07	2.49E-12	5.10E-13
Cm-247	6.93E-09	8.34E-07	1.79E-08	3.80E-10
Cm-248	2.90E-08	3.39E-06	2.26E-12	4.11E-13
Cf-252	3.50E-09	6.59E-07	2.79E-12	4.37E-13

Table 6-2b. Radionuclide Incidence Risk Coefficients

	Internal		External	
Nuclide	Ingestion (Bq^{-1})	Inhalation (Bq^{-1})	Submersion (per Bq y/m^3)	Gnd Surface (per Bq y/m^2)
H-3	1.93E-12	2.59E-12		
Be-7	2.33E-12	4.81E-12	4.31E-09	8.94E-11
C-11	1.21E-12	9.14E-13	8.87E-08	1.83E-09
C-14	2.79E-11	1.89E-13		
C-15	1.79E-14	2.18E-14	4.05E-07	4.69E-09
N-13	8.25E-13	6.84E-13	8.87E-08	1.83E-09
O-15	2.57E-13	2.60E-13	8.87E-08	1.83E-09
F-18	2.94E-12	1.77E-12	8.59E-08	1.77E-09
Na-22	2.17E-10	1.32E-10	1.96E-07	3.68E-09
Na-24	3.74E-11	2.03E-11	4.28E-07	6.36E-09
Si-31	1.36E-11	8.90E-12	8.13E-11	1.42E-12
P-32	1.65E-10	7.92E-11		
P-33	2.11E-11	1.07E-11		
S-35	1.12E-11	5.01E-12		
Cl-36	6.04E-11	3.51E-11	6.32E-16	6.95E-17
Cl-38	5.58E-12	4.41E-12	1.54E-07	2.37E-09
Ar-41		1.27E-14	1.17E-07	2.05E-09
K-40	3.39E-10	2.02E-10	1.44E-08	2.43E-10
K-42	3.48E-11	2.04E-11	2.57E-08	4.30E-10
Ca-45	5.46E-11	6.79E-11	1.13E-18	2.98E-19
Ca-47	1.80E-10	1.41E-10	9.74E-08	1.71E-09
Sc-46	1.55E-10	3.53E-10	1.85E-07	3.43E-09
Sc-47	7.97E-11	5.44E-11	9.42E-09	2.08E-10
Sc-48	1.80E-10	1.13E-10	3.10E-07	5.58E-09
V-48	2.04E-10	1.85E-10	2.67E-07	4.88E-09
Cr-51	3.74E-12	4.70E-12	2.71E-09	5.83E-11
Mn-52	1.62E-10	1.19E-10	3.16E-07	5.79E-09
Mn-54	5.30E-11	9.98E-11	7.68E-08	1.48E-09
Mn-56	2.32E-11	1.41E-11	1.65E-07	2.85E-09
Fe-55	9.50E-12	1.51E-11	1.71E-12	3.30E-13
Fe-59	1.59E-10	1.91E-10	1.09E-07	1.94E-09
Co-57	2.62E-11	7.78E-11	1.03E-08	2.33E-10
Co-58	7.62E-11	1.40E-10	8.85E-08	1.73E-09
Co-58m	2.56E-12	2.41E-12	2.55E-12	5.31E-13
Co-60	5.11E-10	1.86E-09	2.30E-07	4.03E-09
Ni-59	5.00E-12	1.08E-11	2.85E-12	6.21E-13
Ni-63	1.49E-11	2.74E-11		
Ni-65	1.52E-11	9.71E-12	5.07E-08	8.78E-10
Cu-64	1.42E-11	1.13E-11	1.65E-08	3.38E-10
Zn-65	2.68E-10	2.70E-10	5.34E-08	9.68E-10
Zn-69	1.67E-12	2.82E-12	5.18E-13	1.09E-14
Zn-69m	4.11E-11	3.16E-11	3.59E-08	7.54E-10
Ga-67	2.26E-11	1.39E-11	1.23E-08	2.72E-10
Ga-72	1.29E-10	5.87E-11	2.69E-07	4.51E-09
Ge-71	3.19E-13	1.58E-12	6.79E-12	1.74E-12
As-73	2.22E-11	3.69E-11	3.21E-10	1.39E-11
As-74	1.14E-10	1.25E-10	6.74E-08	1.37E-09
As-76	1.77E-10	1.17E-10	3.87E-08	7.46E-10
As-77	4.46E-11	3.10E-11	7.43E-10	1.58E-11
Se-75	1.77E-10	1.33E-10	3.33E-08	7.26E-10
Br-82	3.83E-11	2.12E-11	2.41E-07	4.56E-09
Kr-83m		9.40E-16	6.48E-12	1.58E-12
Kr-85		7.75E-15	1.94E-10	3.99E-12
Kr-85m		7.42E-15	1.36E-08	3.00E-10
Kr-87		3.26E-14	7.93E-08	1.31E-09
Kr-88		5.96E-14	2.03E-07	3.15E-09
Kr-89		4.34E-14	1.81E-07	2.96E-09
Kr-90		4.33E-14	1.19E-07	2.12E-09
Rb-82	2.84E-13	3.17E-13	9.56E-08	1.95E-09
Rb-86	1.92E-10	1.14E-10	8.74E-09	1.59E-10
Rb-87	9.96E-11	6.12E-11		
Rb-88	3.96E-12	3.69E-12	6.39E-08	1.03E-09
Rb-89	2.34E-12	1.87E-12	2.02E-07	3.38E-09
Sr-82	6.97E-10	2.40E-10	2.29E-11	5.51E-12
Sr-85	3.78E-11	3.08E-11	4.44E-08	9.19E-10
Sr-85m	4.87E-13	1.93E-13	1.85E-08	4.04E-10
Sr-89	2.79E-10	9.94E-11	1.26E-11	2.39E-13
Sr-90	1.10E-09	1.61E-09		
Sr-91	7.62E-11	2.11E-11	6.32E-08	1.19E-09
Sr-92	5.48E-11	1.27E-11	1.23E-07	2.12E-09
Y-90	4.05E-10	2.68E-10		
Y-91	3.64E-10	5.01E-10	3.32E-10	5.86E-12
Y-91m	9.98E-13	8.09E-13	4.65E-08	9.49E-10
Y-92	5.26E-11	4.36E-11	2.32E-08	4.27E-10
Y-93	1.55E-10	9.40E-11	8.51E-09	1.50E-10
Zr-93	1.41E-11	1.42E-10		
Zr-95	1.06E-10	1.75E-10	6.68E-08	1.31E-09
Zr-97	2.80E-10	1.28E-10	1.65E-08	3.05E-10
Nb-93m	1.79E-11	1.17E-10	8.30E-12	1.27E-12
Nb-94	1.87E-10	2.22E-09	1.44E-07	2.79E-09
Nb-95	6.09E-11	8.40E-11	6.97E-08	1.36E-09
Nb-95m	8.27E-11	6.08E-11	5.12E-09	1.15E-10

Table 6-2b. (Continued)

	Internal		External	
Nuclide	Ingestion (Bq^{-1})	Inhalation (Bq^{-1})	Submersion (per Bq y/m^3)	Gnd Surface (per Bq y/m^2)
Nb-97	4.74E-12	5.75E-12	5.96E-08	1.19E-09
Nb-97m	8.85E-14	9.03E-14	6.62E-08	1.30E-09
Mo-99	6.13E-11	1.21E-10	1.39E-08	2.78E-10
Tc-95	1.84E-12	9.13E-13	7.17E-08	1.40E-09
Tc-95m	3.36E-11	5.67E-11	5.95E-08	1.20E-09
Tc-96	6.17E-11	5.25E-11	2.29E-07	4.41E-09
Tc-96m	7.05E-13	6.11E-13	3.81E-09	7.53E-11
Tc-97	4.27E-12	9.31E-12	5.69E-11	7.72E-12
Tc-97m	3.24E-11	5.30E-11	7.56E-11	6.78E-12
Tc-99	3.79E-11	7.81E-11	4.14E-14	9.78E-16
Tc-99m	1.51E-12	9.43E-13	1.09E-08	2.42E-10
Ru-97	1.59E-11	1.10E-11	1.96E-08	4.34E-10
Ru-103	8.98E-11	1.24E-10	4.20E-08	8.67E-10
Ru-105	3.12E-11	2.17E-11	7.02E-08	1.40E-09
Ru-106	9.32E-10	3.11E-09		
Rh-103m	2.21E-13	3.45E-13	1.20E-11	1.37E-12
Rh-105	5.22E-11	3.30E-11	6.66E-09	1.43E-10
Rh-105m	2.92E-14	2.50E-14	2.34E-09	5.79E-11
Rh-106	9.80E-14	1.25E-13	1.84E-08	3.66E-10
Pd-100	1.01E-10	9.60E-11		
Pd-101	1.01E-11	6.18E-12		
Pd-103	2.85E-11	2.92E-11	1.09E-10	1.21E-11
Pd-107	5.66E-12	3.94E-11		
Pd-109	9.00E-11	5.37E-11	5.98E-11	1.23E-12
Ag-105	4.41E-11	6.28E-11		
Ag-108	1.88E-13	2.55E-13	1.55E-09	3.14E-11
Ag-108m	1.64E-10	1.90E-09	1.43E-07	2.89E-09
Ag-109m	7.32E-15	9.34E-15	3.31E-10	1.22E-11
Ag-110	6.60E-14	8.53E-14	2.74E-09	5.45E-11
Ag-110m	2.28E-10	8.69E-10	2.50E-07	4.72E-09
Ag-111	1.85E-10	1.42E-10	2.26E-09	4.86E-11
Cd-109	2.16E-10	5.00E-10	1.31E-10	1.13E-11
Cd-115	1.97E-10	1.33E-10	1.77E-08	3.65E-10
Cd-115m	3.84E-10	4.60E-10	2.02E-09	3.71E-11
In-113m	2.24E-12	1.56E-12	2.20E-08	4.68E-10
In-114	1.23E-13	1.57E-13	2.81E-09	5.64E-11
In-114m	5.56E-10	6.83E-10	7.87E-09	1.66E-10
In-115	9.43E-10	5.60E-09		
In-115m	9.24E-12	4.74E-12	1.36E-08	2.96E-10
Sn-113	1.01E-10	1.79E-10	6.29E-10	2.10E-11
Sn-121	3.31E-11	1.66E-11		

	Internal		External	
Nuclide	Ingestion (Bq^{-1})	Inhalation (Bq^{-1})	Submersion (per Bq y/m^3)	Gnd Surface (per Bq y/m^2)
Sn-121m	5.40E-11	2.02E-10	2.08E-11	7.39E-13
Sn-125	4.53E-10	3.23E-10	2.85E-08	5.14E-10
Sn-126	5.73E-10	1.15E-09	3.74E-09	9.36E-11
Sb-122	2.38E-10	1.47E-10	3.91E-08	7.91E-10
Sb-124	2.91E-10	3.57E-10	1.76E-07	3.12E-09
Sb-125	8.02E-11	1.41E-10	3.69E-08	7.64E-10
Sb-126	2.63E-10	2.27E-10	2.47E-07	4.92E-09
Sb-126m	1.97E-12	1.74E-12	1.40E-07	2.82E-09
Sb-127	2.29E-10	1.63E-10	5.89E-08	1.18E-09
Sb-129	5.02E-11	2.32E-11	1.32E-07	2.47E-09
Te-125m	6.78E-11	7.70E-11	5.94E-10	2.57E-11
Te-127	2.31E-11	1.17E-11	4.17E-10	8.81E-12
Te-127m	1.62E-10	3.53E-10	1.94E-10	8.24E-12
Te-129	4.00E-12	3.96E-12	4.70E-09	9.74E-11
Te-129m	3.18E-10	3.60E-10	2.97E-09	6.18E-11
Te-131	1.05E-11	6.71E-12	3.72E-08	7.49E-10
Te-131m	2.38E-10	2.27E-10	1.31E-07	2.48E-09
Te-132	3.29E-10	2.26E-10	1.83E-08	4.10E-10
I-122	5.84E-13	6.05E-13	8.49E-08	1.73E-09
I-123	1.47E-11	7.94E-12	1.33E-08	3.00E-10
I-125	6.98E-10	4.62E-10	6.63E-10	2.94E-11
I-126	1.30E-09	8.52E-10	3.97E-08	8.06E-10
I-129	4.98E-09	3.30E-09	5.03E-10	2.64E-11
I-130	1.31E-10	7.06E-11	1.91E-07	3.80E-09
I-131	9.77E-10	6.31E-10	3.24E-08	6.82E-10
I-132	1.79E-11	9.51E-12	2.09E-07	4.02E-09
I-133	2.86E-10	1.63E-10	5.35E-08	1.07E-09
I-134	6.25E-12	3.74E-12	2.43E-07	4.54E-09
I-135	6.14E-11	3.19E-11	1.48E-07	2.56E-09
Xe-122		8.34E-14	4.99E-09	1.18E-10
Xe-123		2.41E-14	5.53E-08	1.08E-09
Xe-125		3.25E-14	2.12E-08	4.65E-10
Xe-127		1.10E-14	2.22E-08	4.94E-10
Xe-129m		1.55E-14	1.65E-09	6.31E-11
Xe-131m		1.12E-14	5.99E-10	2.39E-11
Xe-133		1.12E-14	2.62E-09	7.38E-11
Xe-133m		1.38E-14	2.39E-09	6.32E-11
Xe-135		2.01E-14	2.12E-08	4.59E-10
Xe-135m		5.09E-15	3.72E-08	7.68E-10
Xe-137		3.76E-14	1.67E-08	3.29E-10
Xe-138		5.58E-14	1.11E-07	1.87E-09

Table 6-2b. (Continued)

Nuclide	Internal		External	
	Ingestion (Bq^{-1})	Inhalation (Bq^{-1})	Submersion (per Bq y/m^3)	Gnd Surface (per Bq y/m^2)
Cs-131	4.87E-12	2.87E-12	4.35E-10	2.42E-11
Cs-134	1.28E-09	7.80E-10	1.40E-07	2.76E-09
Cs-134m	1.23E-12	8.39E-13	1.63E-09	4.32E-11
Cs-135	1.22E-10	7.33E-11		
Cs-136	2.09E-10	1.26E-10	1.98E-07	3.76E-09
Cs-137	8.54E-10	5.18E-10		
Cs-138	4.75E-12	3.51E-12	2.26E-07	3.82E-09
Ba-131	4.59E-11	1.30E-11	3.91E-08	8.38E-10
Ba-133	7.29E-11	1.09E-10	3.09E-08	6.96E-10
Ba-133m	7.47E-11	1.51E-11	4.68E-09	1.12E-10
Ba-137m	6.58E-14	4.24E-14	5.34E-08	1.07E-09
Ba-139	8.23E-12	4.13E-12	3.02E-09	6.46E-11
Ba-140	3.18E-10	8.57E-11	1.62E-08	3.41E-10
La-140	2.56E-10	1.38E-10	2.18E-07	3.78E-09
Ce-141	1.06E-10	1.17E-10	6.31E-09	1.43E-10
Ce-143	1.60E-10	1.04E-10	2.24E-08	4.81E-10
Ce-144	7.99E-10	2.91E-09	1.46E-09	3.42E-11
Pr-142	1.89E-10	1.12E-10	5.52E-09	9.14E-11
Pr-143	1.78E-10	1.51E-10	8.09E-16	1.59E-17
Pr-144	2.18E-12	3.55E-12	3.18E-09	5.29E-11
Pr-144m	8.72E-13	1.52E-12	3.39E-10	1.29E-11
Nd-147	1.59E-10	1.31E-10	1.11E-08	2.45E-10
Nd-149	1.23E-11	1.14E-11	3.24E-08	6.94E-10
Pm-147	3.82E-11	2.02E-10	2.96E-13	6.65E-15
Pm-148	3.90E-10	2.83E-10	5.25E-08	9.47E-10
Pm-148m	2.68E-10	7.96E-10	1.78E-07	3.54E-09
Pm-149	1.49E-10	9.64E-11	1.01E-09	2.13E-11
Sm-147	6.78E-10	1.87E-07		
Sm-151	1.24E-11	1.25E-10	5.54E-14	6.42E-15
Sm-153	1.09E-10	5.88E-11	3.88E-09	1.05E-10
Eu-152	1.55E-10	2.14E-09	1.03E-07	1.94E-09
Eu-154	2.53E-10	2.47E-09	1.14E-07	2.12E-09
Eu-155	4.46E-11	2.59E-10	4.43E-09	1.10E-10
Eu-156	2.95E-10	2.50E-10	1.30E-07	2.22E-09
Gd-153	3.56E-11	8.66E-11	6.67E-09	1.85E-10
Gd-159	7.03E-11	3.34E-11	3.14E-09	7.13E-11
Tb-158	1.14E-10	1.90E-09	6.92E-08	1.31E-09
Tb-160	2.06E-10	3.08E-10	9.85E-08	1.86E-09
Dy-165	8.80E-12	6.06E-12	2.06E-09	4.53E-11
Dy-166	2.55E-10	2.11E-10	2.36E-09	6.48E-11
Ho-166	2.05E-10	1.10E-10	2.38E-09	4.70E-11
Er-169	5.73E-11	4.08E-11	1.52E-13	7.63E-15
Er-171	4.41E-11	2.03E-11	3.12E-08	6.85E-10
Tm-170	2.03E-10	2.96E-10	3.47E-10	9.29E-12
Tm-171	1.58E-11	4.97E-11	3.75E-11	1.12E-12
Yb-169	9.27E-11	1.08E-10	2.27E-08	5.57E-10
Yb-175	6.49E-11	4.54E-11	3.33E-09	7.21E-11
Lu-177	7.97E-11	5.94E-11	2.89E-09	6.51E-11
Hf-181	1.48E-10	1.67E-10	4.65E-08	9.85E-10
Ta-182	1.90E-10	4.47E-10	1.17E-07	2.14E-09
W-181	7.36E-12	2.17E-12	2.37E-09	6.71E-11
W-185	5.50E-11	1.15E-11	2.27E-12	5.07E-14
W-187	6.65E-11	1.43E-11	4.17E-08	8.52E-10
Re-183	4.26E-11	6.94E-11	1.12E-08	2.76E-10
Re-186	8.12E-11	7.06E-11	1.63E-09	3.82E-11
Re-187	2.47E-13	5.10E-13		
Re-188	8.63E-11	6.25E-11	5.07E-09	1.05E-10
Os-185	4.87E-11	1.25E-10	6.21E-08	1.26E-09
Os-191	8.22E-11	7.30E-11	5.41E-09	1.32E-10
Os-191m	1.34E-11	8.97E-12	3.03E-10	8.84E-12
Os-193	1.18E-10	7.24E-11	5.59E-09	1.22E-10
Ir-190	1.34E-10	1.21E-10	1.22E-07	2.52E-09
Ir-192	1.74E-10	3.04E-10	7.06E-08	1.49E-09
Ir-194	1.89E-10	1.13E-10	8.04E-09	1.62E-10
Pt-191	4.05E-11	1.12E-11	2.32E-08	5.21E-10
Pt-193	4.38E-12	2.13E-12	3.28E-12	8.32E-13
Pt-193m	6.77E-11	1.56E-11	7.26E-10	1.96E-11
Pt-197	5.74E-11	1.23E-11	1.79E-09	4.32E-11
Pt-197m	8.79E-12	2.72E-12	6.28E-09	1.47E-10
Au-196	3.52E-11	2.81E-11	3.97E-08	8.65E-10
Au-198	1.43E-10	9.83E-11	3.49E-08	7.36E-10
Hg-197	3.20E-11	1.88E-11	4.75E-09	1.21E-10
Hg-203	7.15E-11	8.18E-11	1.94E-08	4.24E-10
Tl-202	2.74E-11	1.64E-11	3.94E-08	8.43E-10
Tl-204	5.33E-11	3.12E-11	7.68E-11	1.95E-12
Tl-207	2.90E-13	3.34E-13	2.01E-10	3.81E-12
Tl-208	4.73E-13	3.68E-13	3.54E-07	5.41E-09
Tl-209	3.80E-13	3.04E-13	1.95E-07	3.45E-09
Pb-203	2.79E-11	8.38E-12	2.50E-08	5.56E-10
Pb-209	5.66E-12	1.85E-12		
Pb-210	1.82E-08	4.51E-08	9.09E-11	4.16E-12
Pb-211	9.13E-12	2.79E-10	4.56E-09	9.11E-11

Table 6-2b. (Continued)

	Internal		External	
Nuclide	Ingestion (Bq^{-1})	Inhalation (Bq^{-1})	Submersion (per Bq y/m^3)	Gnd Surface (per Bq y/m^2)
Pb-212	4.87E-10	1.04E-09	1.23E-08	2.74E-10
Pb-214	7.94E-12	1.68E-10	2.12E-08	4.56E-10
Bi-206	1.92E-10	1.37E-10	3.01E-07	5.66E-09
Bi-207	1.37E-10	2.54E-10	1.39E-07	2.65E-09
Bi-210	1.97E-10	1.38E-09		
Bi-211	4.93E-13	4.71E-11	4.05E-09	8.69E-11
Bi-212	1.68E-11	9.88E-10	1.69E-08	3.13E-10
Bi-213	1.19E-11	8.36E-10	1.20E-08	2.51E-10
Bi-214	5.27E-12	3.94E-10	1.44E-07	2.49E-09
Po-210	8.80E-09	5.79E-08	7.79E-13	1.51E-14
Po-212	1.22E-21	1.60E-19		
Po-213	1.81E-20	2.11E-18	2.79E-12	5.42E-14
Po-214	5.74E-19	7.50E-17	7.63E-12	1.48E-13
Po-215	1.35E-17	1.21E-15	1.29E-11	2.70E-13
Po-216	2.38E-15	7.98E-14	1.33E-12	2.57E-14
Po-218	1.37E-12	9.96E-11		
At-217	2.43E-16	1.39E-14	2.11E-11	4.26E-13
Rn-219		1.87E-12	4.91E-09	1.05E-10
Rn-220		5.18E-12	4.58E-11	9.35E-13
Rn-222		4.96E-11	3.38E-11	6.97E-13
Fr-221	3.91E-12	2.17E-10	2.61E-09	5.76E-11
Fr-223	1.21E-11	1.60E-11	3.66E-09	9.23E-11
Ra-223	6.33E-09	9.74E-08	1.11E-08	2.50E-10
Ra-224	4.04E-09	6.08E-08	8.54E-10	1.86E-11
Ra-225	4.25E-09	6.44E-08	4.40E-10	1.96E-11
Ra-226	7.98E-09	7.35E-08	5.72E-10	1.27E-11
Ra-228	6.64E-09	2.60E-08	4.77E-18	9.99E-19
Ac-225	3.85E-09	1.12E-07	1.10E-09	2.65E-11
Ac-227	9.52E-09	1.91E-06	1.01E-11	3.24E-13
Ac-228	4.38E-11	8.84E-10	8.49E-08	1.60E-09
Th-227	1.09E-09	1.16E-07	8.78E-09	1.97E-10
Th-228	1.70E-09	2.55E-06	1.58E-10	4.34E-12
Th-229	1.53E-09	2.05E-06	6.90E-09	1.66E-10
Th-230	1.01E-09	4.66E-07	3.00E-11	1.33E-12
Th-231	4.85E-11	2.97E-11	8.91E-10	2.83E-11
Th-232	8.85E-10	5.21E-07	1.39E-11	9.57E-13
Th-234	5.20E-10	5.15E-10	5.98E-10	1.54E-11
Pa-231	4.02E-09	6.54E-07	2.54E-09	5.92E-11
Pa-233	1.27E-10	1.33E-10	1.80E-08	3.97E-10
Pa-234	5.76E-11	3.51E-11	1.78E-07	3.42E-09
Pa-234m	1.29E-13	1.70E-13	1.05E-09	1.98E-11
U-232	2.19E-09	1.43E-06	2.01E-11	1.47E-12
U-233	1.21E-09	3.82E-07	1.91E-11	7.46E-13
U-234	1.20E-09	3.77E-07	1.13E-11	1.13E-12
U-235	1.22E-09	3.51E-07	1.28E-08	2.84E-10
U-236	1.14E-09	3.57E-07	8.72E-12	1.02E-12
U-237	1.08E-10	8.43E-11	1.08E-08	2.58E-10
U-238	1.15E-09	3.36E-07	7.46E-12	8.97E-13
U-240	1.48E-10	9.06E-11	4.84E-11	5.36E-12
Np-236	2.52E-11	1.05E-10	1.06E-08	2.53E-10
Np-237	7.98E-09	9.33E-07	1.79E-09	4.92E-11
Np-238	1.23E-10	1.26E-10	5.11E-08	9.48E-10
Np-239	1.15E-10	6.52E-11	1.38E-08	3.15E-10
Np-240	4.79E-12	3.55E-12	1.04E-07	2.04E-09
Np-240m	6.54E-13	7.64E-13	2.95E-08	5.80E-10
Pu-236	2.08E-09	3.61E-07	8.30E-12	1.34E-12
Pu-238	7.98E-09	7.42E-07	6.05E-12	1.16E-12
Pu-239	8.53E-09	7.51E-07	6.29E-12	5.28E-13
Pu-240	8.52E-09	7.51E-07	5.93E-12	1.11E-12
Pu-241	1.40E-10	7.59E-09		
Pu-242	8.11E-09	7.13E-07	5.04E-12	9.20E-13
Pu-243	9.97E-12	7.22E-12	1.82E-09	4.46E-11
Pu-244	8.46E-09	7.20E-07	3.63E-12	7.81E-13
Am-241	8.87E-09	1.04E-06	1.36E-09	4.26E-11
Am-242	3.96E-11	2.81E-10	1.10E-09	2.84E-11
Am-242m	7.90E-09	9.43E-07	3.42E-11	3.69E-12
Am-243	8.84E-09	1.03E-06	3.85E-09	9.92E-11
Cm-242	1.03E-09	8.54E-08	6.63E-12	1.23E-12
Cm-243	6.79E-09	7.81E-07	1.05E-08	2.40E-10
Cm-244	5.69E-09	6.58E-07	5.60E-12	1.10E-12
Cm-245	9.05E-09	1.06E-06	5.75E-09	1.36E-10
Cm-246	8.97E-09	1.05E-06	4.65E-12	9.67E-13
Cm-247	8.35E-09	9.68E-07	2.72E-08	5.78E-10
Cm-248	3.54E-08	3.95E-06	4.13E-12	7.76E-13
Cf-252	4.86E-09	7.01E-07	5.04E-12	8.22E-13

Table 6-3. Default Inhalation Clearance Class and Ingestion f_1 Values by Element

Element	Inhalation clearance class	Ingestion f_1
H	V	1.0E+00
Be	Y	5.0E-03
N	D	9.5E-01
C*	D	9.5E-01
O	D	9.5E-01
F	D	9.5E-01
Na	D	9.5E-01
Si	W	1.0E-02
P	D	8.0E-01
S	D	8.0E-01
Cl	D	9.5E-01
Ar	*	
K	D	9.5E-01
Ca	W	3.0E-01
Sc	Y	1.0E-04
V	W	1.0E-02
Cr	Y	1.0E-01
Mn	W	1.0E-01
Fe	W	1.0E-01
Co	Y	3.0E-01
Ni	W	5.0E-02
Cu	Y	5.0E-01
Zn	Y	5.0E-01
Ga	W	1.0E-03
Ge	W	9.5E-01
As	W	5.0E-01
Se	W	8.0E-01
Br	D	9.5E-01
Kr	*	
Rb	D	9.5E-01
Sr	D	3.0E-01
Y	Y	1.0E-04
Zr	W	2.0E-03
Nb	Y	1.0E-02
Mo	Y	8.0E-01
Tc	W	8.0E-01
Ru	Y	5.0E-02
Rh	Y	5.0E-02
Pd	Y	5.0E-03
Ag	Y	5.0E-02
Cd	Y	5.0E-02
In	W	2.0E-02
Sn	W	2.0E-02
Sb	W	1.0E-01
Te	W	2.0E-01
I	D	9.5E-01
Xe	*	
Cs	D	9.5E-01
Ba	D	1.0E-01
La	W	1.0E-03
Ce	Y	3.0E-04
Pr	Y	3.0E-04
Nd	Y	3.0E-04
Pm	Y	3.0E-04
Sm	W	3.0E-04
Eu	W	1.0E-03
Gd	W	3.0E-04
Tb	W	3.0E-04
Dy	W	3.0E-04
Ho	W	3.0E-04
Er	W	3.0E-04
Tm	W	3.0E-04
Yb	Y	3.0E-04
Lu	Y	3.0E-04
Hf	W	2.0E-03
Ta	Y	1.0E-03
W	D	3.0E-01
Re	W	8.0E-01
Os	Y	1.0E-02
Ir	Y	1.0E-02
Pt	D	1.0E-02
Au	Y	1.0E-01
Hg	W	2.0E-02
Tl	D	9.5E-01
Pb	D	2.0E-01
Bi	W	5.0E-02
Po	W	1.0E-01
At	D	9.5E-01
Rn	*	
Fr	D	9.5E-01
Ra	W	2.0E-01
Ac	Y	1.0E-03
Th	Y	2.0E-04
Pa	Y	1.0E-03
U	Y	5.0E-02
Np	W	1.0E-03
Pu	Y	1.0E-03
Am	W	1.0E-03
Cm	W	1.0E-03
Cf	Y	1.0E-03

*For ^{14}C, clearance class is *, f_1 is 1.0.

EPA's methodology for estimating risks from airborne radon decay products is outlined in the Technical Support Document for the 1992 Citizen's Guide to Radon (EPA92). The risk model employed is based on an extrapolation of results from epidemiological studies of radon-exposed underground miners, as described in two National Academy of Sciences reports (NAS88, NAS91; EPA92). The estimated risk for exposures to the U.S. population is 2.2×10^4 fatal lung cancers per working level month (WLM), where the WLM is the common unit of radon decay-product exposure. Under typical residential exposure conditions, it is assumed that 1 WLM corresponds to 170 hours exposure at 200 picocuries per liter (pCi/L) of radon gas (EPA92).

6.4 GENETIC EFFECTS

Genetic effects of radiation exposure are defined as stable, heritable changes induced in the germ cells (eggs or sperm) of exposed individuals, which are transmitted to and expressed only in their progeny across future generations.

The genetic risk of radiation exposure is more subtle than the somatic risk since it affects not the persons exposed, but only their progeny. Somatic effects are expressed in the exposed individual over the person's remaining lifetime, while about 30 subsequent generations (nearly 1,000 years) are needed for near complete expression of genetic effects. Genetic risk is incurred by fertile people when radiation damages the DNA of the germ cells. The damage, in the form of a mutation or a chromosomal change, is transmitted to, and may be expressed in, a child conceived after the radiation exposure. However, the damage may also be expressed in some subsequent generation(s) or never.

EPA treats genetic risk independently of somatic risk. Historically, research on genetic effects and the development of genetic risk estimates have proceeded independently of the research on somatic cell mutations and carcinogenesis (MUL50). Neither the dose-response models nor the risk estimates of genetic harm are derived from data on studies of carcinogenesis.

Although genetic effects may vary greatly in severity, the genetic risks considered by the Agency in evaluating the hazard of radiation exposure include only those "disorders and traits that cause a serious handicap at some time during lifetime" (NAS80). Genetic risk may result from one of several types of damage caused by ionizing radiation in the DNA within eggs and sperm. The types of damage usually considered are: dominant and recessive mutations in autosomal chromosomes, mutations in sex-linked (X-linked) chromosomes, chromosome aberrations (physical rearrangement or removal of part of the genetic message on the chromosome or abnormal numbers of chromosomes), and irregularly inherited disorders (genetic conditions with complex causes, constitutional and degenerative diseases, etc.).

Estimates of the genetic risk per generation are convention-ally based on a 30-yr reproductive generation. That is, the median parental age for conception of children is defined as age 30 (approximately one-half the children are produced by persons less than age 30, the other half by persons over age 30). Thus, the radiation dose accumulated from birth to age 30 is used to estimate the genetic risks. A basic assumption in assessing radiation genetic risk is that, at low doses and low dose-rates of low-LET radiation, there is a linear relationship between dose and the probability of occurrence of the genetic effect.

In the EPA Background Information Document for Radionuclides (EPA84), direct and indirect methods for obtaining genetic risk coefficients are described, and some recent estimates based on these methods are tabulated. Briefly, the direct method takes the frequency of mutation or occurrence of a heritable defect per unit dose observed in animal studies and extrapolates to what is expected for humans. These direct estimates are usually used for first generation effects estimates.

The indirect method, on the other hand, uses animal data in a different way. The estimated human spontaneous mutation rate per gene site is divided by the average radiation-induced mutation rate per gene observed in mouse studies to obtain the relative radiation mutation risk in humans. The inverse of this relative radiation mutation risk is the expected "doubling dose" for radiation-induced mutations in man. The doubling dose then, is the dose in rads which will double the current genetic malformation level in man. The doubling dose is usually the basis for estimating the number of both equilibrium effects and all future generation effects.

A doubling dose estimate assumes that the total population, both sexes, is equally irradiated, as occurs from background radiation, and that the population exposed is large enough that all genetic damage can be expressed in future offspring. Although it is basically an estimate of the total genetic burden across all future generations, it can also provide an estimate of effects occuring in the first generation. To obtain such an estimate, a fraction of the total genetic burden for each type of damage is assigned to the first generation based on population genetics data. For example, the BEIR III Committee geneticists estimated that one-sixth of the total genetic burden of x-linked mutations would be expressed in the first generation and five-sixths across all subsequent generations. EPA assessment of risks of genetic effects includes both first generation estimates and total genetic burden estimates.

Sankaranarayanan has published an extensive review of genetics and radiation genetics (SAN91). He concluded that the classical view of genetic expression, with dominant genes manifest in the heterozygous state and recessive genes expressed only when homozygous, is not true. Phenotype does not accurately reflect genotype for many reasons, including: (1) reduced penetrance and/or variable expressivity, (2) co-dominance, (3) "genetic compounds," and (4) somatic cell or germ-line cell mosaicism. Recently many new mechanisms of gene regulation and induction of genetic disease have been recognized. UNSCEAR might include the newly recognized mechanisms: genomic imprinting, uniparental disomy, cytoplasmic inheritance, anticipation, allelic expansion, gene amplification and transposable elements (UNS93).

The mechanisms listed above all interfere with our ability to discriminate genotype from phenotype, or even phenotype from phenocopy. If the genotype can not be adequately determined, then there is no assurance that phenotypic changes reflect gene changes, and the classical dose-response relationships developed may be illusionary.

In developing risk coefficients for genetic effects, EPA has employed traditional definitions of genetic effects and dose-response relationships. Although the newly recognized mechanisms of genetic change listed above have future implications for genetic risk assessment, there are no data upon which to base radiation risk coefficients for these kinds of damage at this time.

In the NESHAPs Environmental Impact Statement (EPA89) the EPA estimated the low dose-rate, low-LET doubling dose for genetic effects to be 1.0 Gy (100 rad). That is, 1.0 Gy per reproductive generation (considered to be 30 years) would double the rate of occurrence of congenital defects. However, at that time, the Agency indicated, based on limited human data, that the true doubling dose might be about three times greater. There is still no consensus on this point.

The BEIR V Committee extensively reviewed the data on genetic effects in both mice and humans (NAS90) and concluded that a best estimate of the low-LET, low dose-rate doubling dose in mice is 1.0 Gy. The Committee further estimated the lower one-sided 95 percent confidence limit on the doubling dose in humans to be 1.0 Gy (NAS90).

The ICRP, referencing the 1988 UNSCEAR Report as its basic source, estimated the doubling dose in man to be 1.0 Gy (ICR91).

The NRPB, like the ICRP, estimates the doubling dose in man to be 1.0 Gy (NRP93). In addition, the NRPB assumed the genetic risk in utero is the same as that after birth (NRP93).

Neel and Lewis reviewed untoward pregnancy outcomes (UPOs) in the Japanese A-bomb survivors and compared them to mouse genetic effects data (NEE90a). For UPOs with genetic basis, they estimated the doubling dose for high dose-rate in humans as 200 (169-223) rad and recommended a dose-rate factor of 2. The dose estimate was based primarily on DS86 dosimetry and "conjoint" parental dose [i.e. the sum of parental gonadal doses](NEE90b). The doubling dose for low dose-rate, low-LET radiation in man, in this case, would be 400 rad (NEE90a). In a companion analysis of mouse genetic data, they estimated a gametic doubling dose in mice of 135 (16-400) rad. The gametic doubling dose for a study where only one sex was irradiated provides an analog of the "conjoint" parental gonadal dose for comparison purposes. However, for mice, they recommended a dose-rate factor of 3 for low dose-rate, low-LET radiation, so the doubling dose would also be 400 rad in mice (NEE90a).

The United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) reviewed the recomendations of all the groups listed above and concluded that the doubling dose in humans is most likely between 1.7 and 2.2 Sv (170 and 220 rad) for acute exposure to low-LET radiation, but 4.0 Sv (400 rad) for chronic exposure (UNS93). However, the UNSCEAR report also continued to estimate the hereditary effects of exposure to ionizing radiation using a doubling dose of 1.0 Sv (100 rad), just as in earlier UNSCEAR reports (UNS86, UNS88).

The EPA will use a doubling dose of 100 rad (1.00 Sv) in this document, but again notes that the true doubling dose may be about 4 times greater.

Table 6-4 lists estimates from BEIR III (used by EPA), UNS88 (used by UNS93) and BEIR V.

Table 6-4. Incidence of Genetic Disease and Risk Estimates in Humans from 0.01 Gy of Low-LET Radiation from Application of the Indirect Method (UNS 93)

Genetic Disease	Incidence (per million live births)			Effect of 0.01 Gy per generation (effects per million live births)					
	BEIR III (1980)	UNSCEAR (1988)	BEIR V (1990)	UNSCEAR (1988)		BEIR V (1990)		NESHAPs (1989)	
				First Generation	Equilibrium	First Generation	Equilibrium	First Generation	Equilibrium
Autosomal dominant	10,000	10,000	--	15	100	--	--	5-65	40-200
Clinically severe	--	--	2,500	--	--	5-20	25	--	--
Clinically mild	9,600	--	7,500	--	--	1-15	75	--	--
X-linked	400	--	400	--	--	<1	<5	--	--
Autosomal recessive	1,100	2,500	2,500	0.05	15	<1	Very slow increase	Very few	Very slow increase
Chromosomal	6,000	--	--	--	--	--	--	<10	Increases only slightly
Structural anomalies	--	400	600	2.4	4	<5	Very little increase	--	--
Numerical anomalies	--	3,400	3,800	Probably very small	Probably very small	<1	<1	--	--
Congenital anomalies	90,000	60,000	20,000-30,000	Not estimated		10	10-100	Not estimated	20-900
Multifactorial diseases	Included in congenital anomalies	600,000	--	Not estimated		Not estimated		Included in congenital anomalies	
Heart disease		--	600,000						
Cancer		--	300,000						
Selected other		--	300,000						
Total	107,100	676,300	1,240,000	17	120	30	133	20	260

Estimates made by various expert panels of first generation genetic effects and equilibrium (or all generations) genetic effects are listed in Table 6-5. Equilibrium genetic effects are the expected rate of occurrence in a population continuously exposed to radiation for approximately 30 generations so that there is an "equilibrium" between mutations arising spontaneously, radiation-induced mutations and mutations eliminated each generation by selection pressures. Equilibrium effects are numerically equal to the expected number of genetic effects in all succeeding generations from acute radiation exposure of an individual or a small group to the same dose.

Table 6-5. Summary of Genetic Risk Estimates per 10^4 Live-born of 1.0 Gy (per 10^6 Live-born of 1 Rad) of Low-dose Rate, Low-LET Radiation in a 30-year Generation[a]

Source	Serious Hereditary Effects	
	First Generation	Equilibrium (all generations)
BEAR, 1956 (cited in NAS72)	-	500
BEIR I, 1972 (NAS72)	49[b] (12-200)[c]	300[b] (60-1500)
UNSCEAR, 1972 (UNS72)	9[b] (6-15)	300
UNSCEAR, 1977 (UNS77)	63	185
ICRP, 1980 (OFT80)	89	320
BEIR III, 1980 (NAS80)	19[b] (5-75)	260[b] (60-1100)
UNSCEAR, 1982 (UNS82)	22	149
UNSCEAR, 1986 (UNS86)[d]	17	104
UNSCEAR, 1988 (UNS88)[d]	18	115
NESHAPs, 1989 (EPA89)	20	260
ICRP, 1990 (ICR91)	23	260
BEIR V, 1990 (NAS90)[d]	29	153
UNSCEAR, 1993 (UNS93)[d]	18	120
NRPB, 1993 (NRP93)	30	240
NRC, 1993 (NRC93)	62	219

[a] Each of the 10^6 sperm and each of the 10^6 ova received a dose of 0.01 Gy during the 30-year reproductive generation leading to the 10^6 liveborn. That is, there were 2×10^6 gametes each receiving 0.01 Gy.

In view of the uncertainties in the new estimates of genetic effects, particularly in multifactorial effects (NEE90b, UNS93), the EPA genetic effects estimates will not be changed. The EPA estimate for equilibrium effects is about twice that of recent estimates by BEIR V and UNSCEAR because EPA included a value for equilibrium multifactorial effects where these others did not. The EPA estimates listed in Table 6-6 incorporate a dose-rate factor of 3 for low-LET radiation as reported in the 1977 UNSCEAR Report (UNS77).

The projected genetic effects attributable to a given population exposure depend on the population dynamics of future generations. However if a stationary population is assumed, the number of effects can be derived from Table 6-6. The dose in the table is that received by parents in the first 30 years of life, the assumed generation period. Since the average lifetime of a person in the 1980 stationary population is about 75 years, 40 percent (30/75) of the population dose is considered to be genetically significant. Thus, to calculate genetic risk coefficients comparable to the cancer risk coefficients in Table 6-1, the values in Table 6-6 should be multiplied by 0.4. On this basis, eight serious heritable disorders are expected in the first generation following a 10^4 person-Gy population exposure of low dose (or dose rate), low-LET radiation, and 104 such effects would be expected over all generations. From a comparison with Table 6-1, the number of serious genetic effects projected over all generations is then about 20 percent of the excess fatal cancers projected in the exposed population.

Table 6-6. Estimated Frequency of Genetic Disorders in a Birth Cohort Due to Exposure of Each of the Parents to 0.01 Gy (1 rad) per Reproductive Generation (30 yr).

Radiation	Serious Heritable Disorders (Cases per 10^6 Liveborn)	
	First Generation	All Generations
Low Dose Rate, Low-LET	20	260
High Dose Rate, Low-LET	60	780
High-LET	90	690

6.5 DEVELOPMENTAL EFFECTS

6.5.1 In-Utero Carcinogenesis

Studies of the effects of in utero X-ray exposures in the U.K. in the 1960s had shown increased childhhood cancer as a sequella. The BEIR III Committee reviewed the data and estimated that there was a risk of 25×10^{-4} excess fatal leukemias per year per Gy exposure (25×10^{-6} per rad) and 28×10^{-4} excess fatal cancers of other types (28×10^{-6} per rad) (NAS80). The risk starts at birth and continues for 12 years for leukemias and 10 years for solid tumors (NAS80).

Having reviewed additional data, the BEIR V Committee estimated that the risk was "... about 200 to 250 excess fatal cancer deaths/ $\times 10^{-4}$ per Gy [200 to 250 $\times$ 10^{-6} per rad] in the first 10 years of life,..." They also estimated one-half would be leukemias and one-quarter tumors of the nervous system (NAS90).

UNSCEAR estimated a risk of leukemia and solid tumors expressed during the first 10 years of life of 2×10^{-4} per rad (UNS86).

The NRPB estimated a cancer risk of 2.5×10^{-4} cases of leukemia and 3.5×10^{-4} cases of solid tumors per rad of in utero exposure (STA88). The NRPB in 1993 retained the same cancer risk estimates, but concluded about one-half the cases would be fatal and they would be expressed in the first 15 years of life (NRP93). However they also estimated the lifetime risk would be four times greater than that of the first 15 years (NRP93).

6.5.2 Brain Teratology

The ICRP published an excellent review of the biology and the possible mechanisms of occurrence of radiation-induced brain damage, in utero (ICR86). ICRP estimates: (1) for exposures from the 8th through the 15th week after conception, the risk of severe mental retardation is 4×10^{-1} per Gy (4×10^{-3} per rad), with a confidence interval of 2.5×10^{-1} to 5.5×10^{-1} (2.5×10^{-3} to 5.5×10^{-3}) and (2) for exposures from the 16th through the 25th week after conception, the risk of severe mental retardation is 1×10^{-1} per Gy (1×10^{-3} per rad). However, a threshold below 50 rad could not be excluded (ICR86). In a 1991 update, the ICRP concluded that there is a threshold of 0.6 to 0.7 Gy in the 16 to 25 week dose response [lower 95 percent bound, 0.21 Gy] (SCH91). At the same time, they estimated the lower 95 percent bound of a threshold [if one exists] for the 8 to 15 week dose response would be about 0.12 to 0.23 Gy (SCH91).

The BEIR V Committee estimated, from the same epidemiological data, that the frequency of induction of severe mental retardation would be 4.3×10^{-1} per Gy (4.3×10^{-3} per rad) exposure during weeks 8 to 15 of conception (NAS90).

Effects other than mental retardation and microcephaly have been noted in the Japanese A-bomb survivors. Schull et al (SCH88) reported that in individuals exposed prenatally between weeks 8 and 25 of gestation there is a progressive shift downward in IQ score with increasing exposure and that the most sensitive group is between 8 and 15 weeks gestational age at time of exposure. The BEIR V Committee estimated a 30 point loss in IQ per Gy exposure (0.3 points per rad) consistent with a linear nonthreshold relationship (NAS90). However, even if the effect is linear-non-threshold the response would be to small to be detectable at environmental exposure levels.

Much the same pattern was reported for average school performance, expecially in the earliest years of schooling (OTA88). Finally, a linear-nonthreshold relationship between exposure and incidence of unprovoked seizures in later life has been found to be consistent with the data for individuals exposed between 8 and 15 weeks gestational age (DUN88).

In 1986, the United Nations Scientific Committee on the Effects of Atomic Radiation also reviewed the question of mental retardation as a part of the overall review of the biological effects of prenatal radiation exposure (UNS86). UNSCEAR, like the ICRP, concluded there was a risk of severe mental retardation of 4×10^{-3} per rad over the period of 8 to 15 weeks after conception and of 1×10^{-3} per rad over the period 16-25 weeks after conception (UNS86).

NRPB reviewed available information including the 1988 UNSCEAR report to develop new health effects models (STA88). The NRPB estimated a mental retardation risk of 4.5×10^{-3} cases per rad of exposure during weeks 8 to 15 of gestation. In their 1993 review, the NRPB did not give a numerical estimate of the risk of mental retardation, but did estimate an IQ loss of 30 points per Gy (0.3 points per rad) (NRP93).

The question of a threshold for central nervous system effects, paricularly for the 8 to 15 week period of gestation, is unresolved. Apparent thresholds in the human data may merely be a reflection of the statistical uncertainty due to the small number of cases. If, as has been suggested, the effects are due to improper synaptognesis in the brain [temporal or spatial] (ICR86, OTA87), it should be noted that significant prolongation of cell cycle in matrix cells of the developing telencephalon in mice (exposed on day 13 of gestation) has been reported following exposures as low as 10 R (KAM78). Exposure of mice to 1 R on day 13 of gestation resulted in an increase in eye and brain abnormalities, but not a statistically significant increase (MIC78).

Among the Japanese A-bomb survivors receiving 0.10 Gy or more during weeks 8-15 of gestation, some suffered severe mental retardation but others did not. The probability increased with increasing dose. This led to the hypothesis that radiation-induced mental retardation is a "stochastic" (all-or-nothing) effect, analogous to the induction of radiogenic cancer or genetic effects (ICR86, UNS86). An alternative explanation, also consistant with the data, is that radiation exposure causes a reduction in intelligence, as measured by IQ, which is proportional to dose (UNS86, NRP93). Such a reduction in an individual with a low, but normal IQ, might cause the individual to be catagorized as "mentally retarded". A similar reduction in IQ in an individual with a high normal IQ might leave the individual with a reduced, but "normal" IQ. If the stochastic view is correct, there may or may not be a threshold, depending on the details of the mechanism of action. If the alternative view is correct, the projected IQ loss at environmental exposure levels is negligible.

6.5.3. Other Effects of Prenatal Irradiation

UNSCEAR estimated (1) a pre-implantation loss of 1×10^{-2} per rad during the first two weeks after conception and (2) a malformation risk of 5×10^{-3} per rad during weeks 2 to 8 after conception (UNS86).

For many of the teratologic effects observed, no threshold has been demonstrated. If a teratogenic effect of radiation is due to cell killing effects, then a threshold for that effect is probable. While early studies of radiation as a teratogen used high exposures and probably induced effects through cell killing, cell killing may not be required. Patrick cites Zwilling as follows: '... developmental anomalies appear to be caused by "failure of proper tissue interaction to occur."'(PAT78, ZWI63). For example, a somatic mutation in a single cell, perhaps through clonal expansion, could cause improper tissue interaction with no loss of cells; or, killing a single cell could cause release of a toxicant that causes an improper local interaction (RUS54, WEI54).

Jacobsen exposed pregnant mice to 0, 5R, 20R, or 100R on day 8 of gestation and scored skeletal abnormalities on day 19. His interpretation of the dose-effect curve was that it was linear or nearly so, and there was no evidence of a threshold for the types of damage studied (JAC70). He stated: "The observations made, and in particular that concerning the apparent absence of a threshold dose, indicate that it is not justified to assume that irradiation with doses of 5r and less is entirely without effect on the human embryo in early developmental stages." (JAC70). In another study, exposure of mice to 1 R on day 8 of gestation resulted in a significantly higher incidence of malformed and retarded fetuses compared to controls (MIC78). A 1981 review of data on the effects of ionizing radiation on the developing embryo/fetus reached essentially the same conclusions as Jacobsen (HHS81). Given the large number of experimental animals that would be required, direct evidence for a threshold below 5 rad will be difficult to provide.

6.5.4 Summary of Developmental Effects

EPA risk coefficients for estimating prenatal carcinogenic, teratologic, and nonstochastic effects in man (see Table 6-7 below) are, with one exception, the same as those published in the 1989 NESHAPs BID (EPA89). The first entry in the corresponding table in the NESHAPs BID lists "Fatal Cancer" as 6.0×10^{-4}. The entry should be for "Cancer Incidence." The fatal cancer risk is about half as great, 3×10^{-4}.

Table 6-7. Possible Effects of In utero Radiation Exposure

Type of Risk to Conceptus	Risk per Rad
Cancer Incidence	$6x10^{-4}$
Mental Retardation[a] (exposure at 8-15 weeks)	$4x10^{-3}$
Mental Retardation[b] (exposure at 16-25 weeks)	$1x10^{-3}$
Malformation[b] (exposure at 2-8 weeks)	$5x10^{-3}$
Pre-implantation Loss (exposure at 0-2 weeks)	$1x10^{-2}$

[a] A threshold for mental retardation following exposure at 8-15 weeks of gestational age may depend on the mechanism of action.

[b] A threshold is expected for mental retardation following exposure during the 16-25 week period of gestation and for many types of malformations following exposures at early gestational age.

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CHAPTER 7

CURRENT KNOWLEDGE AND DESCRIPTION OF YUCCA MOUNTAIN

7.1 SITE DESCRIPTION

7.1.1 Principal Physical Features of Natural Setting (Adapted from DOE95a)

7.1.1.1 Location

The Yucca Mountain site is located in Nye County, approximately 150 km northwest of Las Vegas, Nevada (Figure 7-1). The site is at the southwestern boundaries of the Nevada Test Site and the adjoining Nellis Air Force Base and about 50 km east of Death Valley National Monument. In geologic terms, the region is part of the southwestern Nevada volcanic field in the southern part of the Great Basin. The area is mapped on the following U.S. Geological Survey 7.5-minute topographic quadrangles: Amargosa Valley, Big Dune, Busted Butte, Crater Flat, East of Brady Mountain, and Pinnacles Ridge (ex-Tonopah Spring NW).

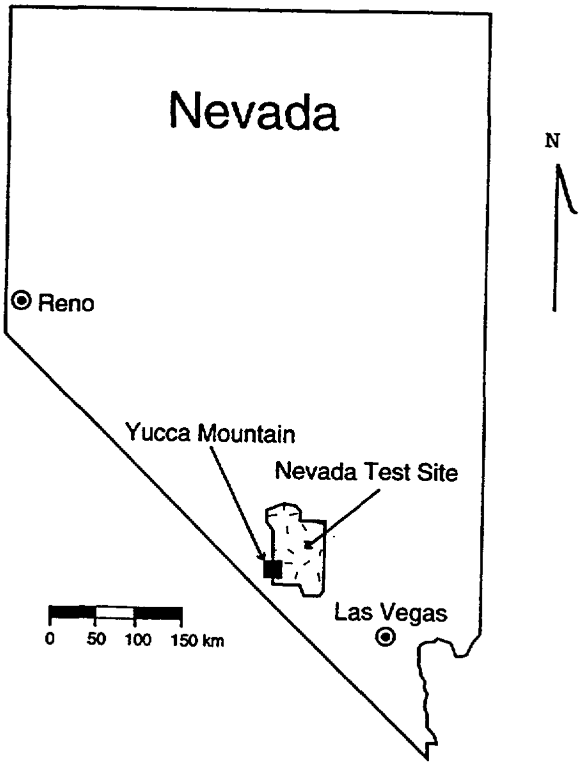


Figure 7-1. Location of Yucca Mountain (Source: DOE94a)

Yucca Mountain is an irregularly shaped upland, 6 to 10 km wide and about 40 km long. Its crest elevation ranges between 1,500 and 1,930 m above sea level, about 650 m above the floors of adjacent washes in Crater and Jackass Flats.

7.1.1.2 Climate

The Yucca Mountain area is situated in the south central and extreme southern National Weather Service climatological zones of Nevada. These two zones are differentiated mostly by elevation. Conditions at the lower elevations, typical of southwestern desert zones, are characterized by hot summers, mild winters, and limited precipitation. The higher elevations have less severe summer temperatures and greater but still limited amounts of precipitation. The climate in the Yucca Mountain area can be characterized as that of a mid-latitude desert, where summers are dominated by continental tropical air masses and winters by continental polar air masses. Mid-latitude deserts are also characterized by large annual and diurnal temperature fluctuations and significant variations in precipitation from year to year.

Precipitation in the Yucca Mountain area is associated with two distinct atmospheric circulation patterns. The first occurs in winter when fronts associated with Pacific air masses move toward the area from the west. The accompanying precipitation tends to be of low intensity and long duration (measured in days) and cover a large area. The second circulation pattern results in a late summer (July and August) peak with convective thunderstorms of high intensity, short duration (hours) and limited size. Precipitation in the vicinity of Yucca Mountain is approximately 170 mm/yr, with potential evapotranspiration estimated to be about 1,000 mm/yr (DOE94a).

Temperatures at the site vary widely on both a daily and annual basis. Winter months (November through March) are mild, with the daily high averaging at least 50° F. Freezing temperatures are frequent, but temperatures below 0° F are rare. Summer temperatures often exceed 100° F. Substantial differences between the average daily high and low temperatures are characteristic of the area due to high insolation rates and generally low relative humidities.

Meteorological data from the NTS and Las Vegas airport indicate prevailing winds from the north to northwest during the winter and from the southwest during the summer. On an annual basis, wind speed measured at the airport ranges between 4.6 and 6.6 mph 34 percent of the time, and 8.1 to 11.5 mph 29 percent of the time (DOE95b).

7.1.1.3 Geomorphology and Physiography

The Yucca Mountain Region includes the southern Great Basin in southern Nevada and an adjacent area in California (Figure 7-2). The Great Basin, the northern portion of the Basin and Range physiographic province, is bounded geologically by the margins of the Colorado Plateau to the east and southeast, by the Sierra Nevada and Transverse Ranges to the west and south, and the Snake River Plain and flood basalts of the Columbia Plateau to the north. Typical Great Basin topography consists of north-south mountain ranges separating narrow structural valleys with internal drainages. The Colorado River, flowing along the margin of the Colorado Plateau and topographically isolated from Yucca Mountain, provides the only external drainage.

Yucca Mountain is situated in the southern section of the Great Basin, in the Southwest Nevada Volcanic Field (SNVF). This area is bounded on the south by the Death Valley region and the Mojave Desert of California. Yucca Mountain is a narrow ridge which trends north-south and extends approximately 20 km from the southern margin of the Timber Mountain caldera complex. The mountain is composed of Miocene volcanic rocks erupted from the overlapping Silent Canyon, Claim Canyon, and Timber Mountain calderas between 11.5 and 14 mega-annum (Ma). The silicic volcanic tuffs that comprise Yucca Mountain are typical of mid-Tertiary basin and range extensional tectonics in southern Nevada.

The main drainage system for the Yucca Mountain area, including the Timber Mountain area, the Calico Hills, and the mesas lying to the south of Timber Mountain, is the Amargosa Valley. This drainage, east of Beatty, Nevada, carries runoff from the region south through the Tecopa basin and into the southern part of Death Valley. The Amargosa carries significant runoff only after extraordinarily heavy precipitation. There are no perennial streams or natural bodies of surface water on or adjacent to Yucca Mountain. The major drainages, Solitario Canyon on the west, Fortymile Wash on the east, and tributary drainages, are primarily along the east flank of the mountain and flow only briefly immediately after rainstorms.

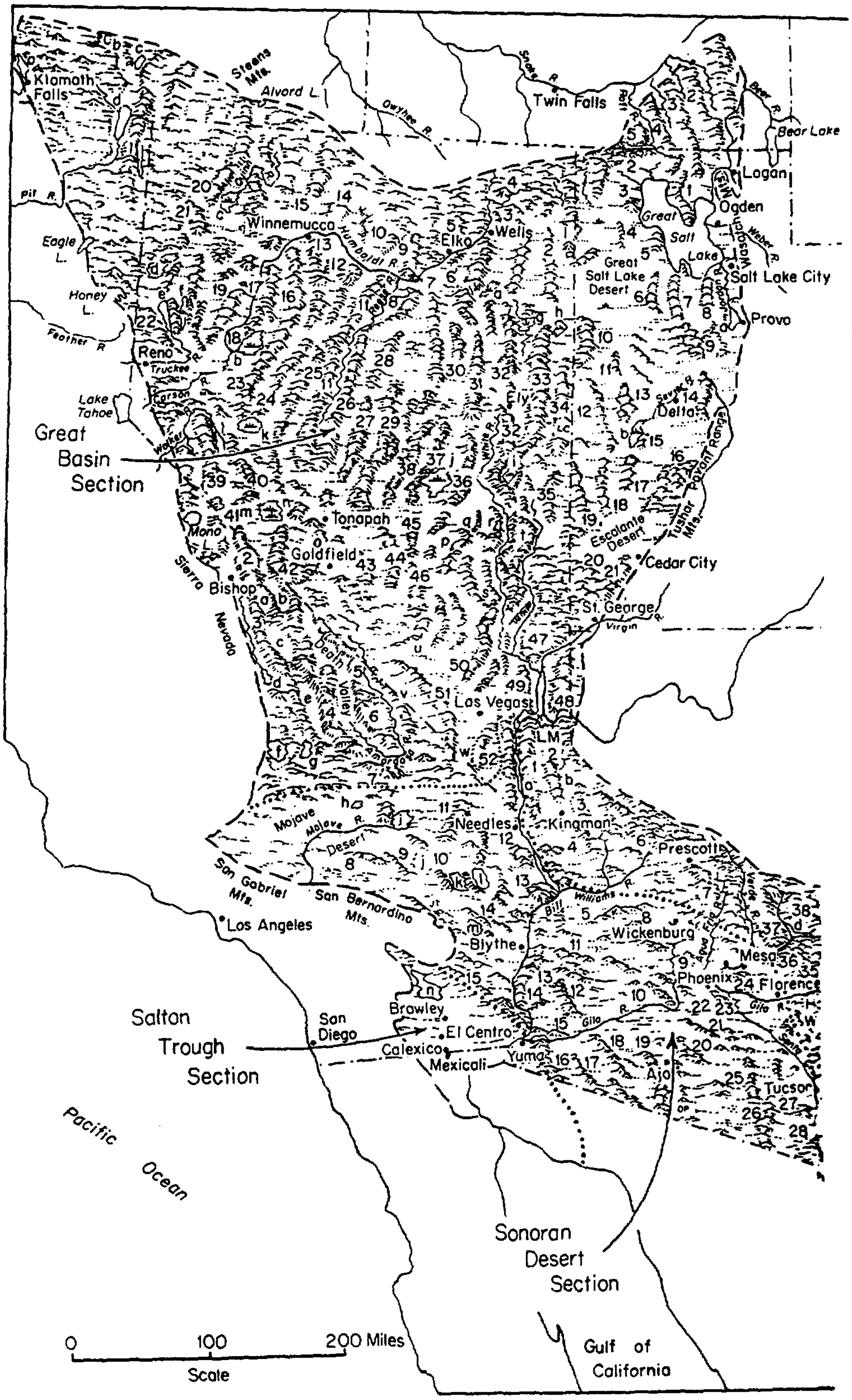


Figure 7-2. Physiographic Map of the Basin and Range Province (Source: HUN74)

Uplands in the Yucca Mountain area are composed of ridge crests, valley bottoms, and intervening hill slopes (DOE88) with dominantly north-trending enechelon ridges and valleys controlled by high-angled faults. The fault blocks, composed mostly of welded fine-grained volcanic rocks, are tilted eastward. As a result, the fault-bounded west-facing slopes are generally high, steep, and straight, whereas the east-facing slopes are more gentle and usually deeply dissected. Except where protected by a resistant rock layer capping the lip slopes, the ridge crests are mostly angular and eroded. Valleys range from shallow, straight, steeply sloping gullies and ravines to relatively steep, bifurcating, gently sloping valleys and canyons. Hill slopes are typically narrow and moderately steep near the crest, with progressively gentler slopes toward the valley floor.

Bedrock exposures are common at higher elevations. Many of the hill slopes have a discontinuous veneer of blocky talus, and wedges of colluvium cover the lower hill slopes.

7.1.2 Radioactive Waste Contents of Repository

7.1.2.1 Overview

The NWPA, in Section 114(d), limits the contents of a potential repository at the Yucca Mountain site to a total of 70,000 metric tonnes of heavy metal (MTHM) or equivalent. "Heavy metal" is nuclear fuel material such as uranium; a metric tonne is 1,000 kilograms, or about 2,200 pounds. Since the United States does not reprocess spent nuclear fuel from commercial power reactors, the principal waste form in the repository would be intact spent fuel subassemblies removed from the reactors and determined to be a waste for disposal.

The NWPA also required DOE to evaluate whether the repository could also contain high-level waste (HLW) from defense production operations. The HLW would be fission product waste made into glass logs and cast into containers approximately ten feet long and 1.5 feet in diameter.

The glass logs would be made by mixing approximately 30 weight percent of solidified HLW with 70 weight percent borosilicate glass frit in a melter that converts the mixture into waste form glass, which is then poured into the containers. A flow sheet for the HLW processing operations at the Savannah River Defense Waste Processing Facility is given in DOE94b.

DOE determined that it would be advantageous to commingle the commercial spent nuclear fuel (SNF) waste and the defense HLW in the repository. DOE also decided that the repository should contain about 63,000 MTHM of commercial SNF and 7,000 MTHM-equivalent of defense HLW.

The radioactivity content of the SNF wastes emplaced in the repository will depend on how the fuel was used in the reactor and for how long. Because of radioactive decay, it will also depend on how long ago the SNF was removed from the reactor. If emplacement begins in about 2010, some of the SNF will have been removed from reactor service nearly 50 years ago. DOE currently assumes the average age of the SNF at the time of emplacement will be on the order of 30 years.

The fission product content of the defense HLW has in some cases decayed since production operations during World War II. In general, the radioactivity content of the defense HLW canisters will depend on how the waste in its present form is treated and diluted by vitrification. The radioactivity content of a canister of defense HLW could be as much as a million times less than the radioactivity content of an SNF package. Depending on the size of the SNF waste package used, the repository could contain between 7,600 and 32,000 SNF packages. It would contain about 5,000 defense HLW packages.

Design details of the engineered portions of the repository, such as the numbers of packages and their emplacement configuration, have not been selected by DOE. The engineered features of the repository will contribute to waste isolation in combination with the waste isolation characteristics of the natural features of the repository, and the characteristics of the natural features, such as water inflow and geochemical conditions, are not yet well enough known to enable final selection of the preferred design for the engineered barrier system (EBS) of the repository. The DOE has identified and is characterizing the performance of several alternative engineered designs as described below and in Section 7.6 of this BID.

Details of the repository engineered designs under consideration by DOE, and of the analyses of waste isolation performance of the alternative repository systems, are contained in documents prepared by DOE as part of their iterative total-system performance assessment (TSPA) effort. A principal objective of the TSPA studies is to guide characterization of the natural features of the proposed repository site and selection of design features for a potential repository at the site. DOE completed its second TSPA iteration in 1993 (DOE94a) and its third iteration in 1995 (DOE95c). These documents, DOE94a and DOE95c, are principal sources of information in this section of the BID. Consistent with DOE practice, they are referred to herein as TSPA-93 and TSPA-95, respectively.

7.1.2.2 Spent Nuclear Fuel

Both pressurized water reactor (PWR) and boiling water reactor (BWR) spent fuel assemblies are to be emplaced in the repository. The spent fuel assemblies include not only the uranium oxide fuel but also the fuel cladding and support hardware, all of which are radioactive. An option to receiving intact fuel assemblies is receiving canisters of chopped fuel rods and support hardware.

To model the waste isolation performance of the repository, the radioactivity and heat-generating characteristics of the spent fuel must be specified. Assumptions used by DOE in its evaluations of alternative engineered features of the repository are described in TSPA-93 and TSPA-95.

DOE's studies have included consideration of technology and designs needed for transfer and handling operations, starting with SNF receipt at the reactors and ending with emplacement in the repository. Technologies considered include: 1) the transportable storage cask (TSC), 2) the multiple purpose canister (MPC), and 3) the Multi-Purpose Unit System (MPU). TSPA-95 used the large (21 PWR subassemblies) MPC as its reference waste package. Subsequently, further development of the MPC concept was terminated. The range of operations to be considered in selecting engineered technologies for management and disposal of SNF is shown in Figure 7-3. In this figure, the repository is designated as the Mined Geologic Disposal System (MGDS).

TSPA-93 assumed that the repository would contain 63,000 tonnes of SNF, composed of mixed BWR and PWR fuels, with burnups (in-reactor use of the fuel) of 30,000 Megawatt-days per tonne of uranium (MWd/MTU) and 40,000 MWd/MTU, respectively. For practical purposes, the tonnes of uranium (MTU) is equivalent to the tonnes of heavy metal (MTHM) and calculated from the number of PWR spent fuel assemblies per container and the mass of a PWR assembly. Both the BWR and PWR fuels are considered to have an age of 25 years before emplacement.

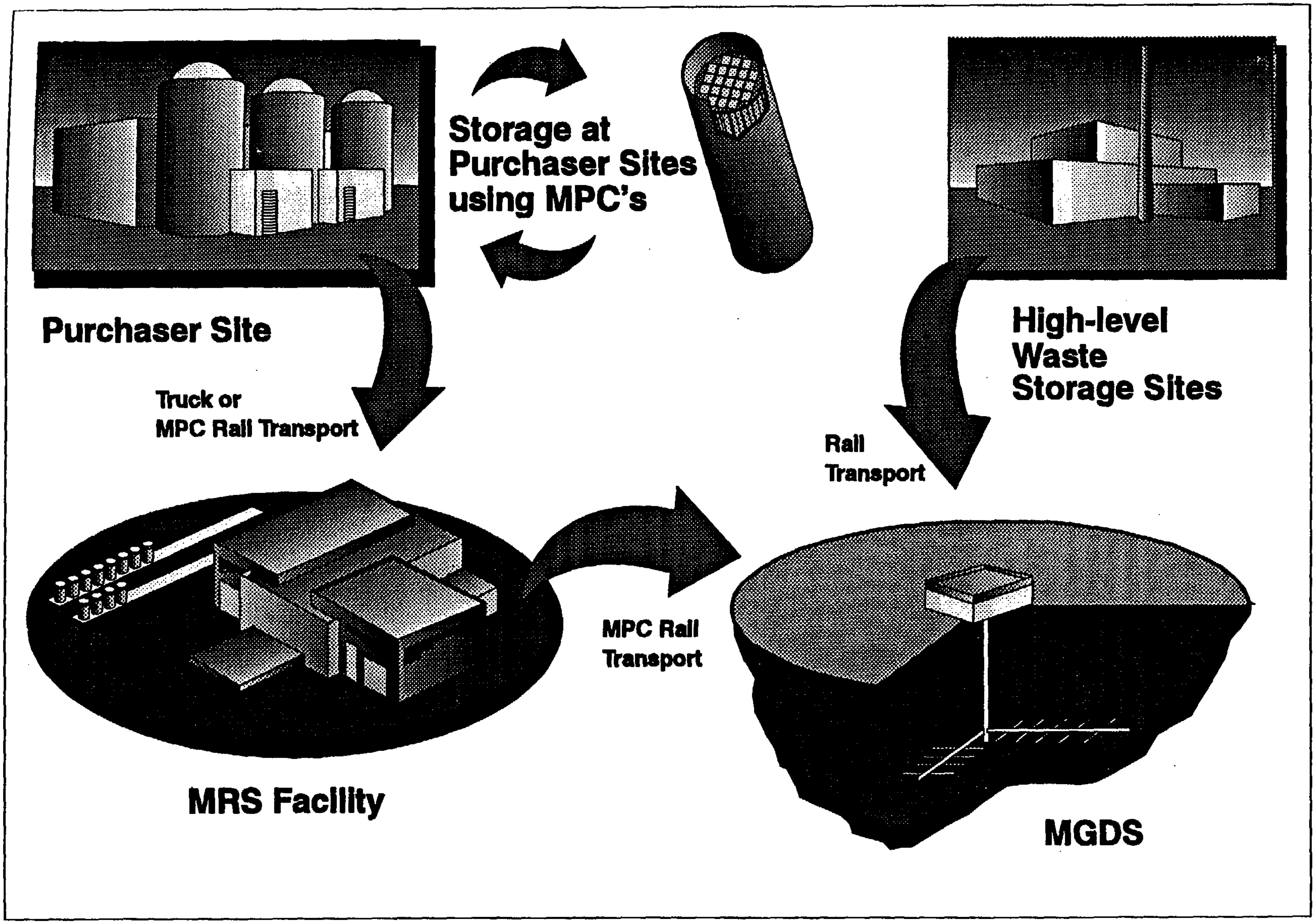


Figure 7-3. The Civilian Radioactive Waste Management System (Source: DOE94c)

Depending on the package used, waste could be emplaced (see Section 7.2.2) in the repository via boreholes or in-drift laydowns. Table 7-1 shows the relevant statistics for these package and fuel conditions.

Table 7-1. Spent Nuclear Fuel Emplacement Options
(Source: TSPA-93, Table 5-1 (DOE94a))

Reactor	SNF (MTU)	%	Average Age	Average Burnup (MWd/MTU)	No. Pkgs	
					Hybrid	Single
Borehole Emplacement						
BWR	22248	35.3	26.3	31550	28057	1215
PWR	40749	64.7	25.5	40461	-	2750
Totals	62996	100	-	-	32022	
In-Drift Emplacement						
BWR	22183	35.3	26.3	31533	-	3109
PWR	40646	64.7	25.5	40433	-	4531
Totals	62829	100	-	-	7640	

TSPA-95 also assumed that the repository would contain 63,000 tonnes of SNF, composed of 40,785 MTHM of spent PWR fuel and 22,210 MTHM of spent PWR fuel. The radioactivity content of the SNF was estimated by assuming that all SNF had been removed from the reactors thirty years before emplacement, and that the burnup was 39,651 MWd/MTHM for PWR fuel and 31,186 MWd/MTHM for BWR fuel.

7.1.2.3 Defense High Level Waste (DHLW)

High-level waste from spent fuel reprocessing operations has been generated and is now stored at four DOE sites: the Savannah River Site (SRS); the Hanford Site (HANF); the West Valley Demonstration Project (WVDP); and the Idaho Chemical Processing Plant (ICPP) at the Idaho National Engineering Laboratory. Quantities and characteristics of vitrified HLW from these sources are shown in Table 7-2. Assumptions used in the TSPA-93 analyses are shown in Table 7-3.

Table 7-2. Sources of High-Level Waste
(Source: TSPA-93, Table 5-6 (DOE94a))

Sources	Number of Canisters	Volume of HLW in Canister (m^3)	Density of HLW (Mg/m^3)	Mass of HLW (kg/ canister)	Mass of Radio-nuclides (kg/ canister)	Total Mass of Radio-nuclides (MT)[a]	Max. Radio-activity (Ci/can)[b]	Max. Total Radio-activity (MCi)
SRS	5,282	0.616	2.73	1,682	34	179	234,400	1,238
WVDP	275	0.7	2.71	1,900	70	19	114,700	31.5
ICPP	7,800	0.57	3.20	1,825	0.83	6.5	108,900	856
HANF	1,960	0.626	2.64	1,650	14	27	298,000	584
Totals	15,317	--	--	--	--	231.5	--	2,709.5

[a] Maximum values are approximately twice as great as average values.
[b] Mass in tonnes.

Table 7-3. HLW Waste Stream Used for TSPA Analyses
(Source: TSPA-93, Table 5-7 (DOE94a))

Source	Number of Canisters	Total Inventory (MTHM)	Maximum Total Radioactivity (MCi)
SRS	5,282	2,641	1,238
WVDP	275	577	31.5
ICPP	5,603	3,241	615
HANF	1,960	980	584
Totals	13,957	7,000	2,468.5

7.1.2.4 Key Parameter Values for the Repository Waste Contents

The radioactivity source term for repository performance assessments depends on the quantity of emplaced HLW, the geometry of placement, and the radioactivity/heat-generating characteristics of the HLW. The radioactivity and heat-generating characteristics are, in turn, a function of radionuclide composition, which is determined by fuel burnup, time out of reactor, and the radionuclide inventory of the emplaced defense high-level waste (DHLW). The configuration of SNF and DHLW in the repository emplacement panels determines the heat generation rate.

The radionuclide inventory of the repository will change with time after repository closure as a result of decay and ingrowth of daughters of radioisotopes originally placed into the repository. Inventories of some fission products, such as Cs-137 and Sr-90, which have approximately 30-year half-lives, will decay relatively rapidly. Some daughter products, such as Pb-210 and Ra-226, will not achieve peak values until after about 100,000 years have elapsed, and long-lived nuclides such as I-129 and Tc-99 will maintain virtually constant inventories for one million or more years. DOE developed the diagram shown in Figure 7-4 (taken from DOE95d) to illustrate how the inventory varies and depends on some of the key nuclides. The magnitude of the inventory was expressed as a calculated dose to illustrate the need for waste containment and isolation. A similar diagram which shows the time-variation of more of the nuclides was developed by NRC (NRC95a).

The physical characteristics of the waste packages are important in modeling the release of radionuclides. Key characteristics include waste disposal container design, fuel element cladding material, nominal SNF surface area (a function of number of rods, their size, and the fuel pellets' surface area), and the DHLW glass characteristic and surface area of the vitrified defense waste (see Section 6 of TSPA-95 (DOE95c)).

In modeling the release of radionuclides, only those with calculated release rates greater than 0.1 percent of the total NRC release limit established in the 10 CFR Part 60 regulations are considered (TSPA-95, Section 8.3.2 (DOE95c)). A consideration in release rate is the "retardation factor"; radionuclides with retardation coefficient (Kd) values greater than 10 or 20 ml/g contribute little to releases (TSPA-93, Section 5.4 (DOE94a)). For TSPA-95, 39 radionuclides were considered; for TSPA-93, 43 radionuclides were considered. A comparison of principal radionuclide inventories for TSPA-95 and TSPA-93 analyses is given in Table 7-4.

In recent years, commercial power plants have attempted to extend the refueling intervals. This results in higher burnup in the nuclear fuels and subsequently changes the potential repository waste contents. At higher fuel burnup, the activity of the radionuclide inventory and decay heat generated in the spent fuel increase (AIF85).

Historically, safety analyses of criticality control systems for transportation packages include an assumption that SNF loaded into the package is "fresh" or unirradiated. In other words,

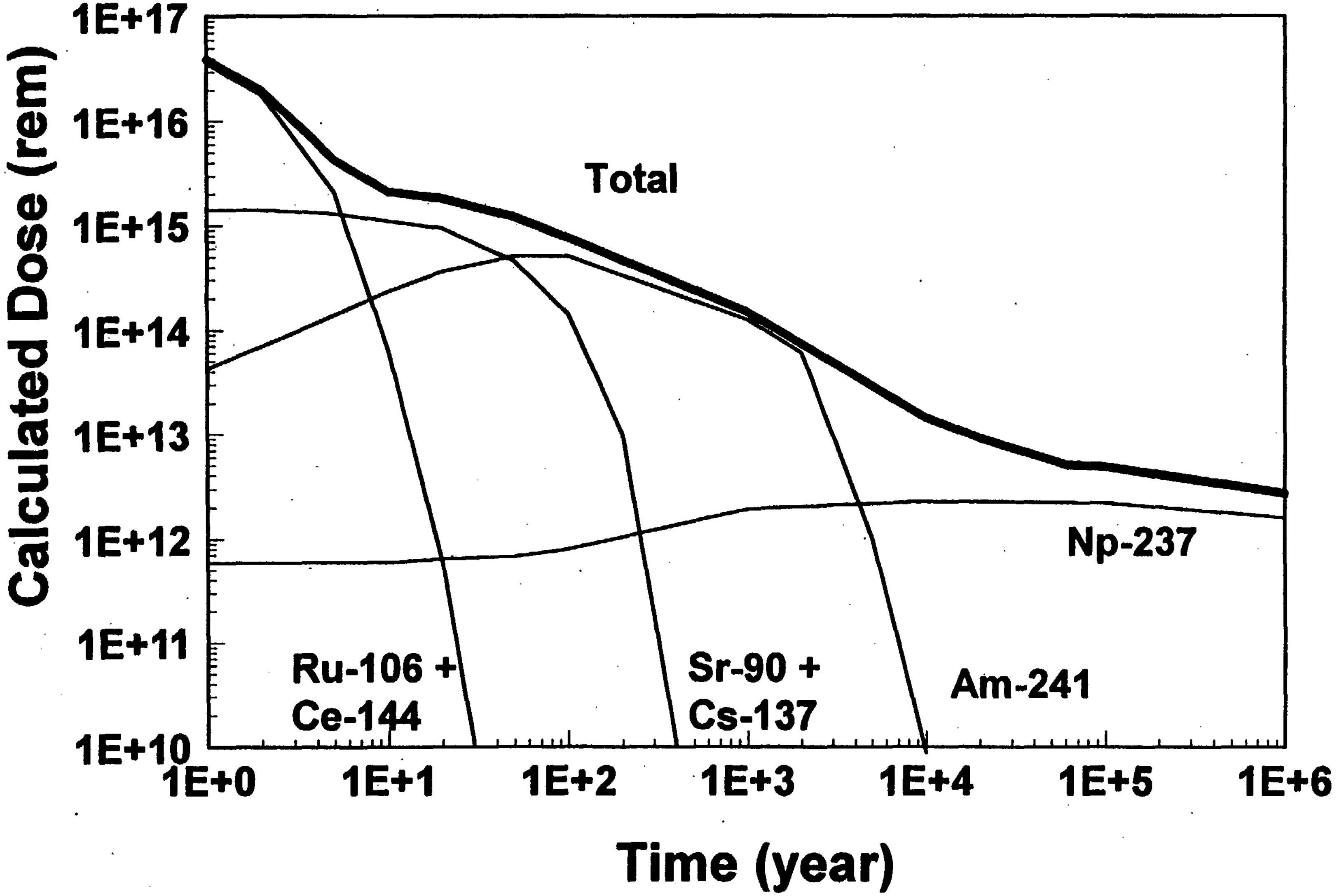


Figure 7-4. Radionuclide Burden for the Repository System (Source: DOE95d)

Table 7-4. Comparison of Selected Radionuclide Inventory in TSPA-95 with TSPA-93 Inventory (DOE95c, DOE94a)

Inventory	TSPA-1993[a,b] (Ci/container)	TSPA-1995[c] (Ci/container)
^{243}Am	2.74E+02	2.48E+02
^{14}C	1.44E+01	1.38E+01
^{135}Cs	5.52E+00	5.13E+00
^{129}I	3.62E+00	3.43E-01
^{237}Np	4.74E+00	4.35E+00
^{239}Pu	3.65E+00	3.56E+03
^{79}Se	4.67E+00	4.41E+00
^{126}Sn	9.01E+00	8.50E+00
^{99}Tc	1.47E+00	1.40E+02
^{234}U	1.39E+00	1.34E+01

a) (DOE94d)
b) TSPA-1993 inventory is for 30-year old fuel. Ci/container = Ci/MTHM x 9.2 MTHM/container.
c) TSPA-1995 inventory is for 30-year old fuel. Ci/container = Ci/MTHM x 7.94 MTHM/container.

the spent fuel is assumed to have its original, as-manufactured U-235 isotopic content. The "fresh" assumption is very conservative since the potential reactivity of the nuclear fuel is substantially reduced after being irradiated in the reactor core. Recently, DOE proposed to NRC the concept of "Burnup Credit," which takes credit for this reduction in nuclear fuel reactivity due to burnup of the fuel, instead of using the fresh fuel assumption in the criticality safety analysis (DOE95e). Burnup credit uses the actual physical composition of the fuel and accounts for the net reduction of fissile material and the buildup of neutron absorbers in the fuel as it is irradiated. Neutron absorbers include actinides and other isotopes generated as a result of fission process. Using only the change in actinide isotopes in the burnup credit criticality analysis is referred to as "Actinide-Only Burnup Credit."

DOE performed analyses and concluded that using burnup credit to maximize SNF transportation cask capacities is a justifiable concept that would result in public risk benefits and cost savings. Similar benefits would be realized for SNF storage and disposal packages. The use of burnup credit in the design of criticality control systems enables more spent fuel to be placed in a package. Increased package capacity in turn results in a reduced number of storage, shipping, and disposal containers for a given number of SNF assemblies. Several public and rate payer benefits result from an overall reduction in the number of packages because the total number of packages drives both cost and risk. Fewer shipments result in a lower risk of accidents associated with the handling and transportation of spent fuel, thus reducing both radiological and non-radiological risk to the public. The economic benefits of burnup credit result from lower storage, shipping, and disposal costs and reduced package handling operations at storage, shipping, and receiving facilities.

However, increased package capacity will change the radioactivity content of the emplaced HLW as well as the geometry of the placement. The design of criticality control systems would be a priority issue and require critical evaluations.

7.1.3 Proposed Conceptual Repository

7.1.3.1 General Layout

Although the layout for the Yucca Mountain repository has not been finalized, there is agreement on many baseline parameters needed to model the facilities. System facilities at a repository site will include rail and road access, waste receiving, safety and security control facilities, ventilation equipment, shops, warehousing, and facilities for interim storage of waste containers.

Access to the underground workings is by gentle-grade ramp, with access at the north and south ends of the repository (DOE88). The repository would be bisected by three adjacent unheated tunnels (drifts) about 46 m wide that provide ventilation and access for waste emplacement operations with a perimeter drift around 17 waste emplacement areas or panels. The emplacement panels are typically 426 m wide and 457-1200 m long. Associated with the main drift corridor is a thermal buffer zone 61 m wide (see Figure 7-5).

The three main center drifts and the thermal buffer zones break the thermal continuity of waste package heat generation. A similar break in continuity occurs between adjacent panels. These

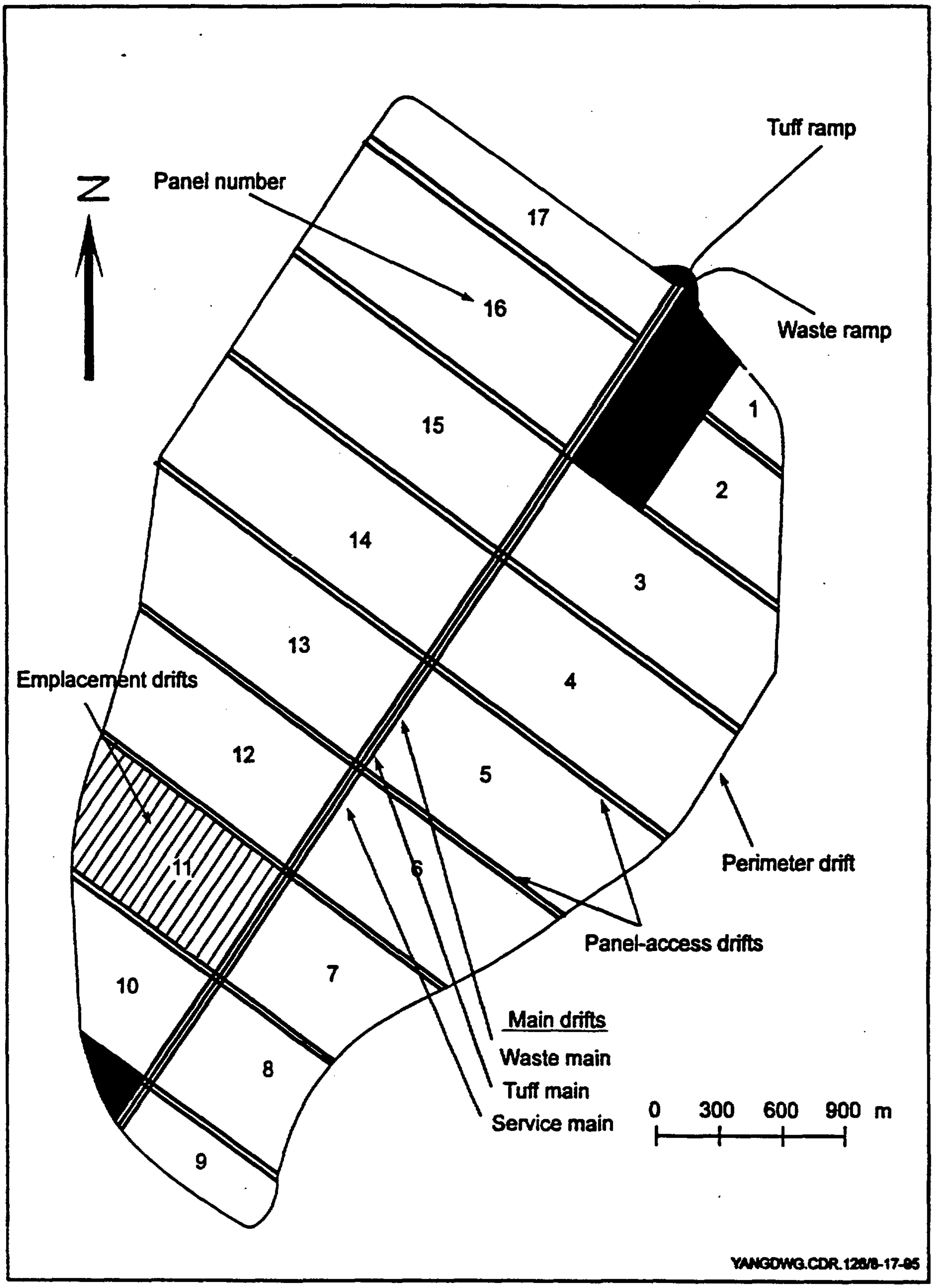


Figure 7-5. SCP-CDR Repository Layout (after DOE88)
(Source: TSPA-95, Figure 3.2-1 (DOE95c))

breaks in thermal continuity have a strong impact on coalescence and duration of far-field isothermal surfaces, making repository thermal analysis difficult.

As noted in the preceding section, the number of SNF waste packages could vary from 7,600 to 32,000, depending on the design selected. Additionally, the contents of a package can vary widely in thermal output. For example, assuming complete freedom to select BWR/PWR assemblies of various ages, the initial power output per package would be about 10 megawatts (MW), versus a youngest-fuel-first case in which the output would be about 14 MW.

The waste emplacement density, expressed as areal power density (APD), creates thermal effects that decrease with time because of radioactive decay. The repository is expected to remain open throughout the emplacement period, during which time drifts would continue to be ventilated. Each would eventually be sealed with backfill and engineered sealing structures.

7.1.3.2 Repository Size

The waste packages (about 25 to 40 m^2 each) will require an area of about 10 million m^2 in the repository. Six potential emplacement panels (see Figure 7-6) have been identified in the Tonopah Spring welded (TSw) geologic unit. They are well above the saturated zone and are not intersected by major known faults and fractures.

Performance modeling to date has shown that the repository performance will depend not only on the natural features of Yucca Mountain but also on the thermal loads imposed by waste package emplacement configurations. Two general thermal load cases have been postulated: a 20 to 40 tonnes of uranium per acre (MTU/Ac) low-thermal-load case and a high-thermal-load case of 80 to 100 MTU/Ac.

7.1.3.3 Waste Isolation Strategy

Waste isolation can be affected and achieved by seven basic factors:

1. Site location
2. Repository geology and stratigraphy

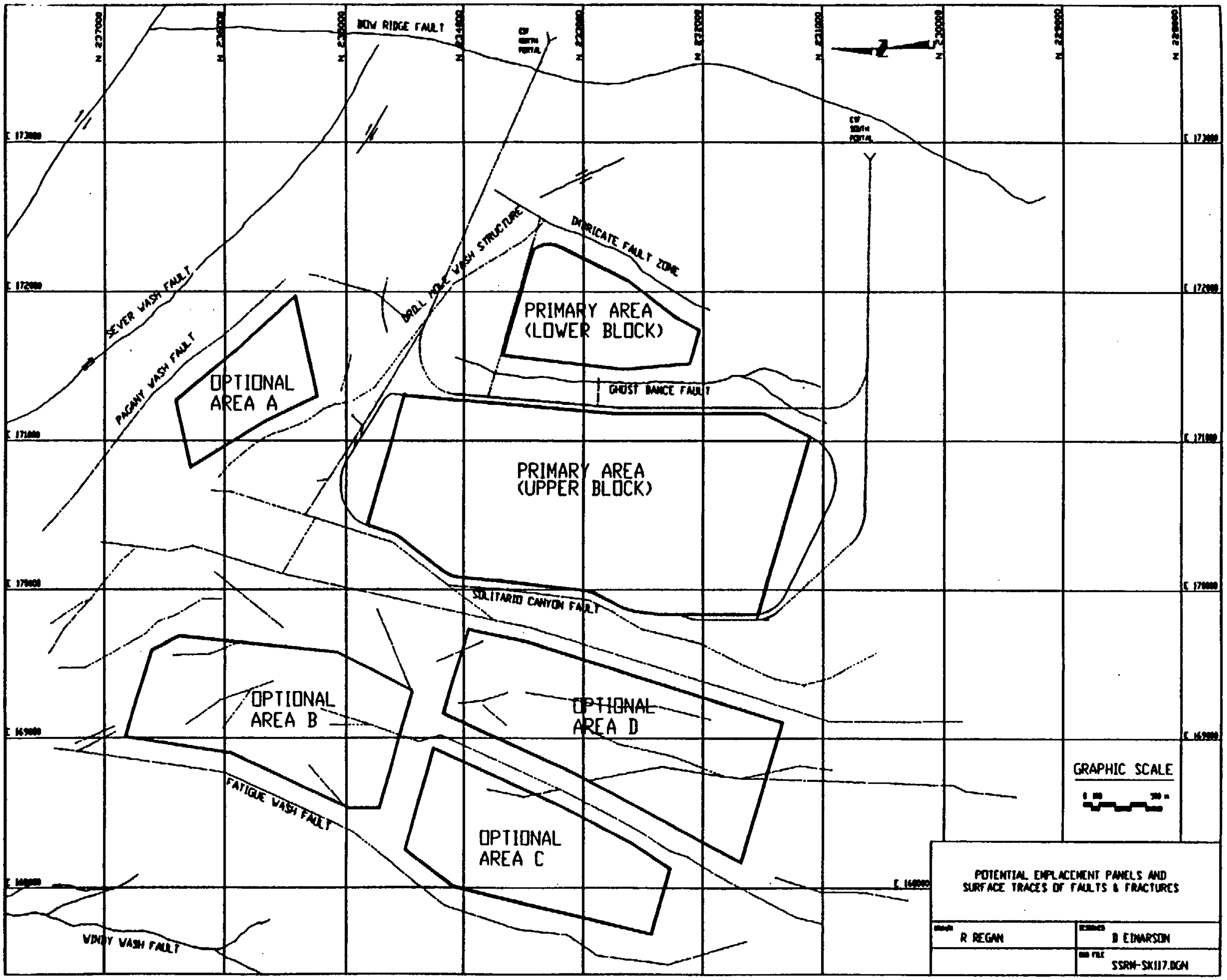


Figure 7-6. Potential Emplacement Panels (Source: DOE95c)

3. Host rock geochemistry and separation from the saturated zone
4. Hydrologic regime
5. Disposal panel location, size, and distance to the accessible environment
6. Repository thermal character
7. Engineered barrier system

A schematic of the natural barriers, which are the first four isolation factors listed above, is shown in Figure 7-7. The seven components of the waste isolation strategy are briefly described in the following sections, and a discussion of DOE's strategy for using the natural and engineered isolation barriers is given in Section 7.6.5.

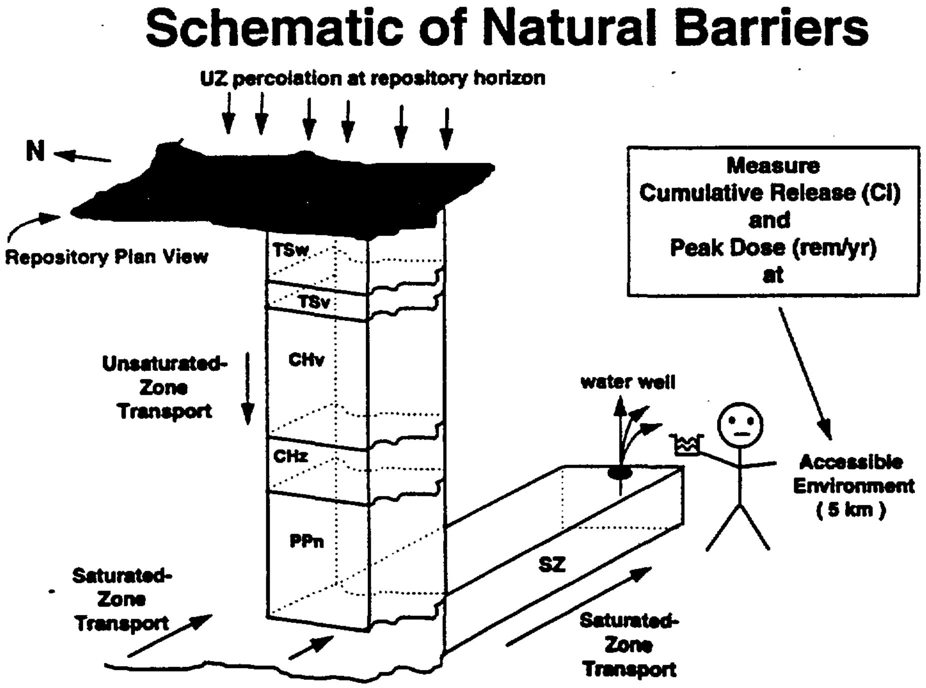


Figure 7-7. Schematic of Natural Barriers (Source: DOE95c, ES 8-1)

Site Location

The Yucca Mountain site is located in a desert region (precipitation about 170 mm/yr with estimated evapotranspiration of 1,000 mm/yr) with sparse population and low land use. Tectonic activity is low, and arid climatic conditions are expected to continue in the future (see Section 8.2).

Geology and Stratigraphy

The waste emplacement horizon for the Yucca Mountain site was chosen because of the sequence of unsaturated geologic units that make exposure of the waste to pervasive ground water unlikely. The unsaturated zone, above the water table, is approximately 600 m thick and provides ample room for waste emplacement.

Host Rock Geochemistry and Saturated Zone Chemistry

The repository would be located in the Yucca Crest, a relatively undisturbed structural block that consists of layered silicious rock (tuff), which is compacted, indurated volcanic ash. The saturated zone fluids beneath the proposed repository location have pHs in the range of 7 - 8. They are dilute sodium bicarbonate fluids with high concentrations of aqueous silica. Analyses of water compositions from the unsaturated zone indicate more acidic pHs of 6.4 - 7.5. The ground waters are relatively oxidizing.

Hydrologic Regime

The hydrologic regime of Yucca Mountain and its environs defines the relationship of the waste package emplacements within the unsaturated zone to the saturated zone and the impact on hydrology of the faults and fractures that have been found. DOE currently expects the water travel time from the repository to the saturated zone to be quite long, thus contributing to isolation. Conversely, the hydrologic regime can contribute to loss of isolation by acting to initiate aqueous corrosion of waste packages and their contents and then transporting released radionuclides to the accessible environment.

Disposal Panels: Size and Location

The disposal panels will be located to avoid intersection with major faults and fractures. Their size can be seen in Figure 7-6. These panels provide, along with the main access corridors and peripheral corridors, a waste isolation panorama conducive to monitoring to confirm analytical models and to detect repository abnormalities.

Repository Thermal Character

As noted in Section 7.1.3.2, various areal mass loadings for the emplaced waste produce different thermal characteristics. Modeling has demonstrated that release of radionuclides depends on the thermal environment of the repository. Thermally dependent factors important to waste isolation and repository performance include:

- corrosion initiation
- corrosion rates
- fuel matrix alteration and dissolution
- gas release and flow
- diffusion and advection of radionuclides or sorption of radionuclides

Repository temperatures, in the near term, are a function of waste package types and emplacement. Various configurations have been analyzed by DOE as follows:

TSPA-93
57 KW/Ac with vertically emplaced containers
114 KW/Ac with vertically emplaced containers
57 KW/Ac with horizontal containers
114 KW/Ac with horizontal containers

TSPA-95
25 and 83 MTU/Ac with the MPC package placed horizontally in drifts

Each configuration produces conditions that affect the timing and rate of release of radionuclides.

Repository Operations

Operational activities at the repository site will include construction of surface and subsurface facilities; excavation of tunnels and drifts for waste emplacement; waste receipt and storage prior to emplacement; waste handling for emplacement; decommissioning and sealing of the repository; and post-closure monitoring. Waste retrieval may also be required as a result of monitoring during operations or after closure.

DOE has developed detailed initial estimates of timelines for operational activities (DOE95f). Receipt of spent fuel waste for emplacement is expected to span a period of 25 years (DOE94a). The duration of emplacement may last well beyond the end of spent fuel receipt, depending on the thermal loading strategy chosen for the repository and the schedule for receipt and emplacement of defense HLW canisters. Overall, the operational period at the repository, including installation of seals, decommissioning of facilities, and initiation of post-closure monitoring, could span a century (DOE95f).

Engineered Barrier System (EBS)

By definition, the EBS is the waste package and the repository setting in which it is placed. Waste is contained until such time as the fuel cladding is breached after fuel canisters have failed, and the overpack metals have failed through corrosion. Radionuclides may still be retained for a time in the EBS because the backfill composition may retard their release into the unsaturated zone and from there to the accessible environment.

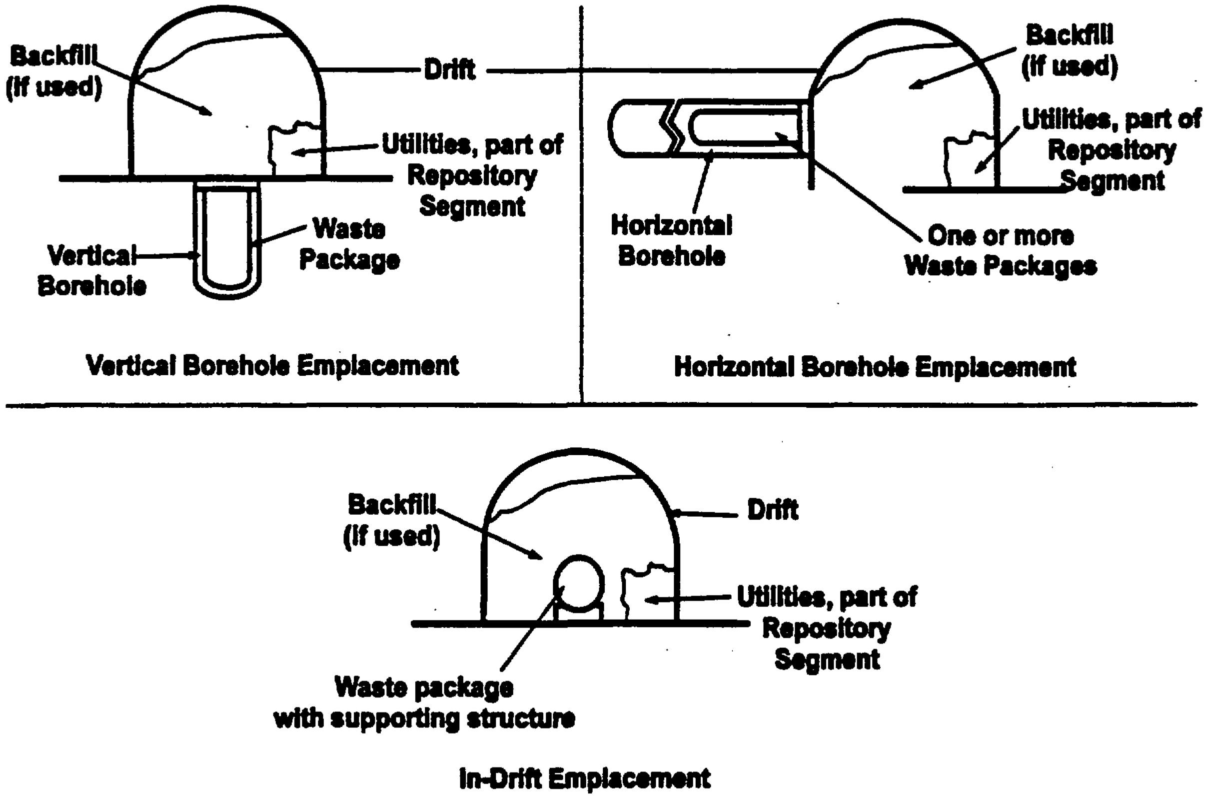
7.2 ENGINEERED SYSTEMS

7.2.1 Waste Package Designs

7.2.1.1 General Description

Waste package degradation depends on 1) waste package design (materials selected); 2) repository design (thermal load, backfill, size of emplacement drifts); 3) the near-field hydrologic regime; and 4) the corrosion characteristics of the waste package materials in the repository environment.

The waste package, as defined in existing NRC regulations 10 CFR 60.2, includes the waste form and any canisters, shielding, packing, and other absorbent materials immediately surrounding the individual waste container. It may be emplaced in horizontal or vertical boreholes in a drift or be placed on the floor of a drift on a support structure with the drift later back-filled (see Figures 7-8 and 7-9).



Vertical Borehole Emplacement

Horizontal Borehole Emplacement

In-Drift Emplacement

Figure 7-8. Borehole and In-Drift Emplacement Concepts from μ SCP-CDR (DOE88) (Source: DOE95c)

7.2.1.2 Waste Packages for SNF and DHLW

A Controlled Design Assumption Document (CDA) (DOE 95x) describes the current SNF and DHLW waste packages. CDA values, as follows, have been used by TSPA-95; other materials and thicknesses have been modeled in TSPA-93 and -91. The outer containment barrier, of corrosion allowance material (CAM), is mild steel, with an inner corrosion-resistant material (CRM) of Inconel 825 (Alloy 825). For the low thermal load case, a third barrier of moderately corrosion-resistant material (MCRM), such as Monel 400, is added over the CAM and CRM containers. Corrosion rates are generally low; thus, the wall thicknesses of the three containers directly affect the time of radionuclide release. The current assumed thickness of the inner CRM barrier (Alloy 825) for the large multipurpose canister (MRC) is 20mm (DOE95c) with an outer CAM barrier (mild steel) 100 mm thick for SNF and 50 mm thick for DHLW. The CDA did not specify the thickness of the third barrier (Monel 400); therefore, analyses in TSPA-95 deal only with two barriers for both SNF and DHLW of 20 mm and 100 mm thickness.

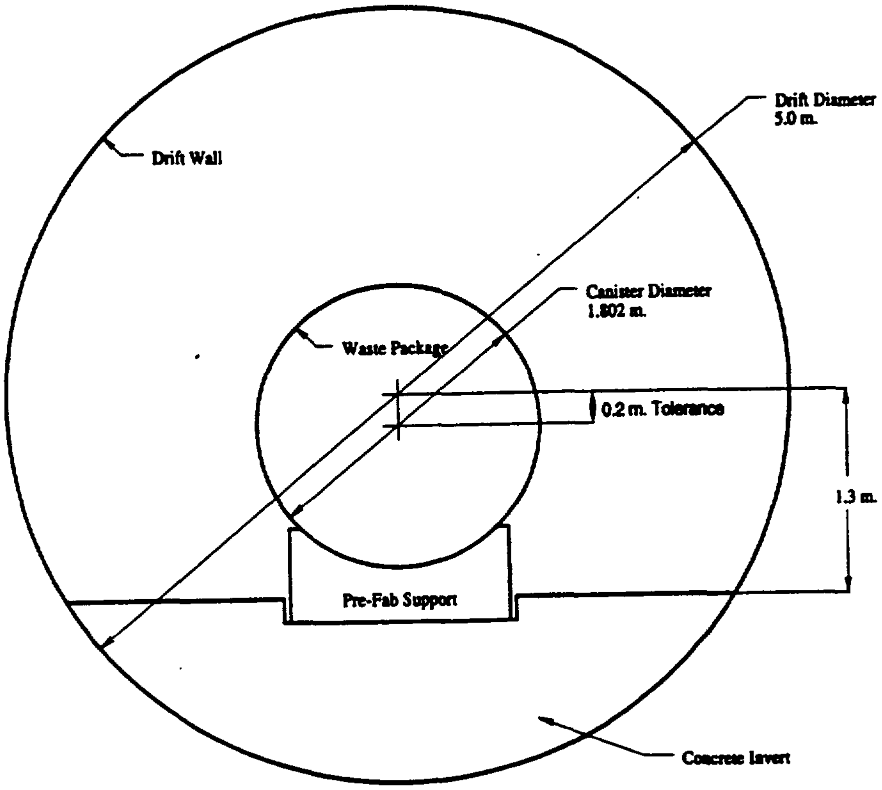


Figure 7-9. Emplacement Drift Design for the In-Drift Placement Option
(Source: DOE95c)

A September 1994 report, "Multi-Purpose Canister System Evaluation," (DOE94c) details three storage, transport, and disposed-waste-container systems: the MPC (multi-purpose canister), TSC (transportable storage cask) and the MPU (multi-purpose unit system). Design drawings for the 100-ton TSC metal cask are shown in Figures 7-10 and 7-11. Similar drawings for other design concepts are given in TSPA-95. These systems utilize stainless steel (SS) for parts of the canisters. TSPA-95 does not consider the SS MPC shell or the SS DHLW pour canister in its corrosion models.

7.2.1.3 Corrosion of the Materials of Construction

Corrosion experts have developed various container degradation models for dry oxidation, general aqueous corrosion, and pitting (DOE95g). The models address the metals of the primary container and the overpack. The fuel cladding failure mode has been subjected to various assumptions ranging from instantaneous failure on breach of the inner container, to

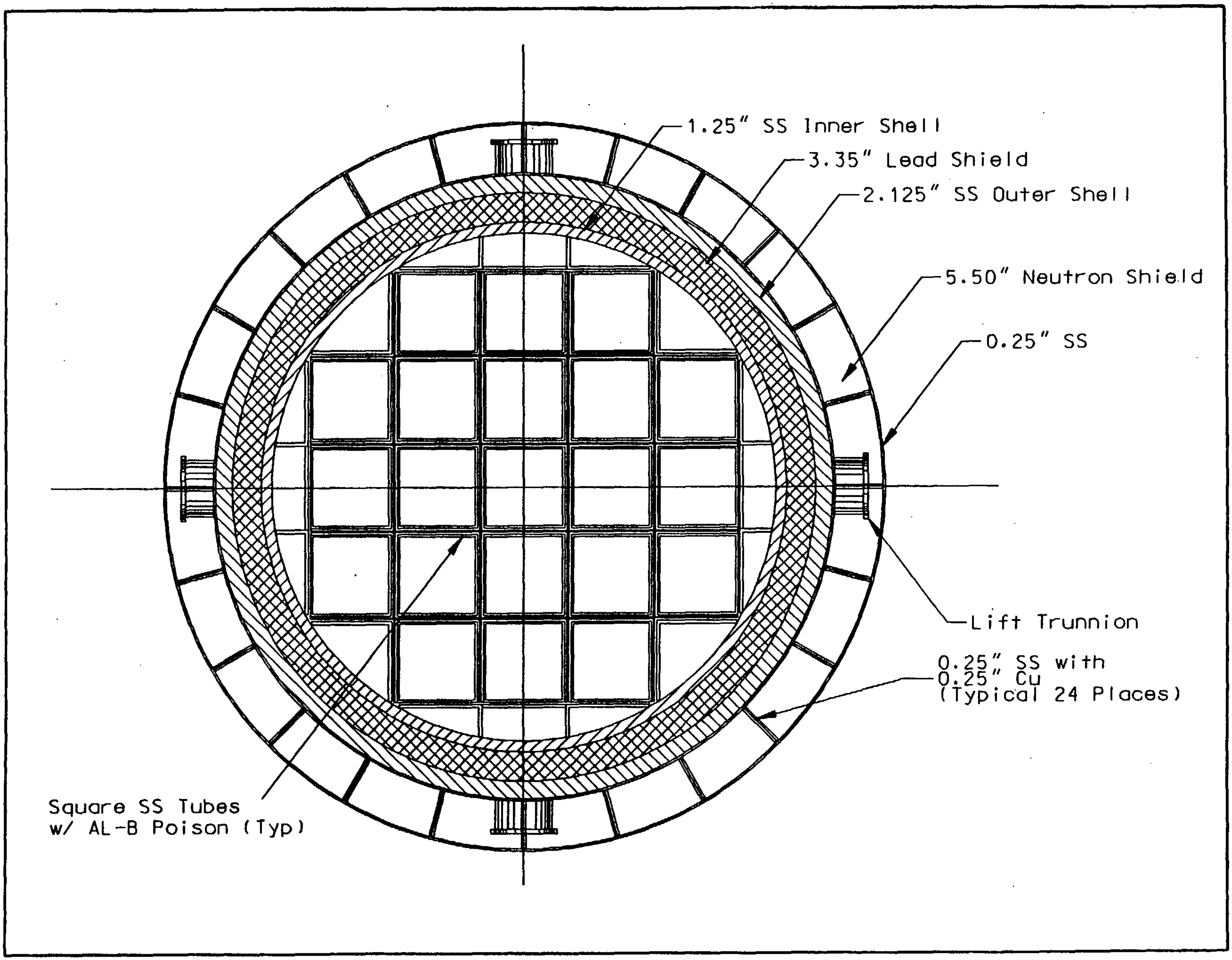


Figure 7-10. 100-Ton TSC Metal Cask for 21 PWR Fuel Assemblies (End Section View) (Source: DOE95c)

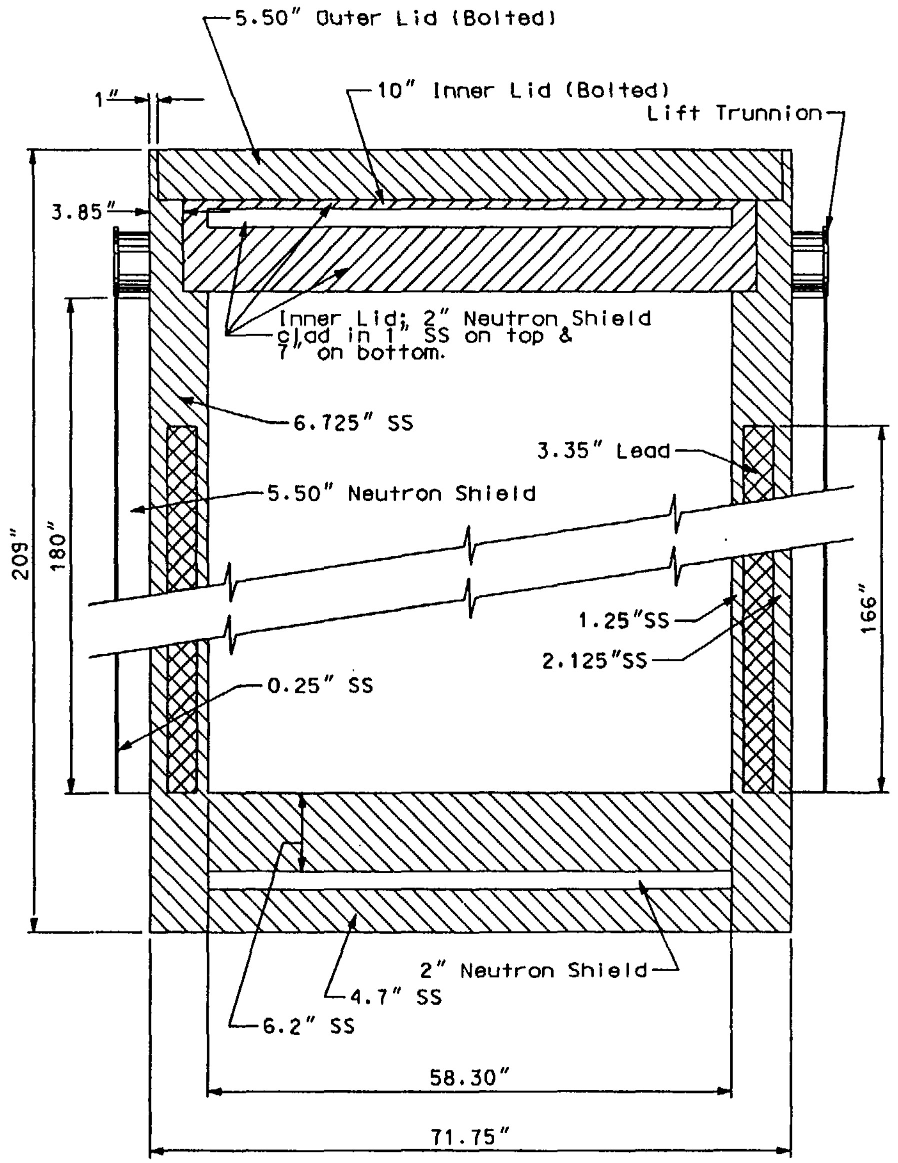


Figure 7-11. 100-Ton TSC Metal Cask (Side Section View) (Source: DOE95c)

some credit for cladding resistance to corrosion. Dissolution of spent fuel and vitrified high-level waste has also been modeled.

In general, although carbon steel has relatively low corrosion resistance, its corrosion rate is predictable and its cost is low, so it is used for the thick outer CAM barrier. Once penetrated, carbon steel serves as the sacrificial anode to cathodically protect the Alloy 825 inner barrier. Alloy 825 was chosen because it is highly resistant to uniform corrosion and, in many environments is resistant to localized corrosion such as pitting, crevice, and stress corrosion. Adequate long-term corrosion models for Monel 400 and 70/30 copper nickel alloy are not now available, but their potential is recognized. Stainless steel, used for the MPC shell and DHLW pour canister, was not considered in the TSPA-95 models.

7.2.2 Waste Emplacement Configurations

Three major waste emplacement concepts have been considered in repository models: vertical borehole emplacement, horizontal borehole emplacement and in-drift emplacement (see Figure 7-8). Factors considered in choosing an emplacement mode include heat generation from the SNF, maximum permissible canister centerline temperatures (premature failure of fuel rod cladding may result if very high centerline temperatures occur), and the overall repository thermal load.

DOE has evaluated (DOE95h) six emplacement modes for: 1) operational complexity during emplacement, 2) ease of retrieval, 3) safety, 4) thermal load flexibility and adjustment, 5) potential thermal management by ventilation, 6) constructability, 7) backfill emplacement, 8) stability of the emplaced waste and backfill excavation, 9) cost, and 10) long-term performance. Based upon the recent Controlled Design Assumption (CDA) document, the MPC waste package posited must be emplaced on the drift floor, as shown in Figure 7-9.

The large MPC package, 5.6 m long by 1.8 m in diameter, would weigh 66 tons and would be emplaced by a remotely controlled, rail-mounted gantry crane on permanent pedestals, prepositioned in the drift. Ventilation will be used during construction and package emplacement and can be directed to minimize localized heat spikes. Ventilation will also be needed for any required retrieval operations.

The waste package content and the density of waste package emplacements determine the heat generation characteristics for the repository panel. The near-field thermohydrology results in various processes that affect waste package performance, such as conductive and convective heat transfer, boiling and condensation, capillary adsorption, vapor pressure lowering, and thermal buoyancy-driven vapor flow.

TSPA-95, Section 4.2.6, notes that assuming an emplacement protocol of "Oldest Fuel First" (OFF), the 21 PWR MPC, containing fuel with an average age of 26 years and burnup of 39 GWd/MTU (heat output of 0.98KW/MTU), requires a distance between waste packages of 19 m when emplaced for an areal mass loading of 83 MTU/Ac in a nominal drift spacing of 22.5 m. For the 25 MTU/Ac case and a nominal drift spacing of 45 m, the waste package spacing becomes 19 m.

The location, number, and size of the postulated emplacement panels may change in response to more information being obtained from the Exploratory Studies Facility at Yucca Mountain (DOE88).

7.2.3 Engineered Barriers

7.2.3.1 General

The Engineered Barrier System (EBS) consists of the waste package and the engineered setting into which it has been placed. Thus, the EBS includes the drifts, vertical boreholes, and any airgap between the waste package and the host rock and backfill. It is designed for explicit hydrologic and/or geochemical properties. Figure 7-12 schematically depicts the Waste Package/EBS. Similar figures from TSPA-95 appear as Figures 7-8, 7-9, and 7-13.

In its strategy for waste containment and isolation (DOE95d), DOE identifies potential additional barriers to aqueous transport of nuclides. These barriers include the drift invert on which horizontally-emplaced waste packages would rest or special backfill materials placed under or around the waste package. DOE anticipates that these materials could be selected to provide transport-inhibiting properties at least as good as those of the host rock, but the effectiveness of the barriers wold have to be verified by measurements.

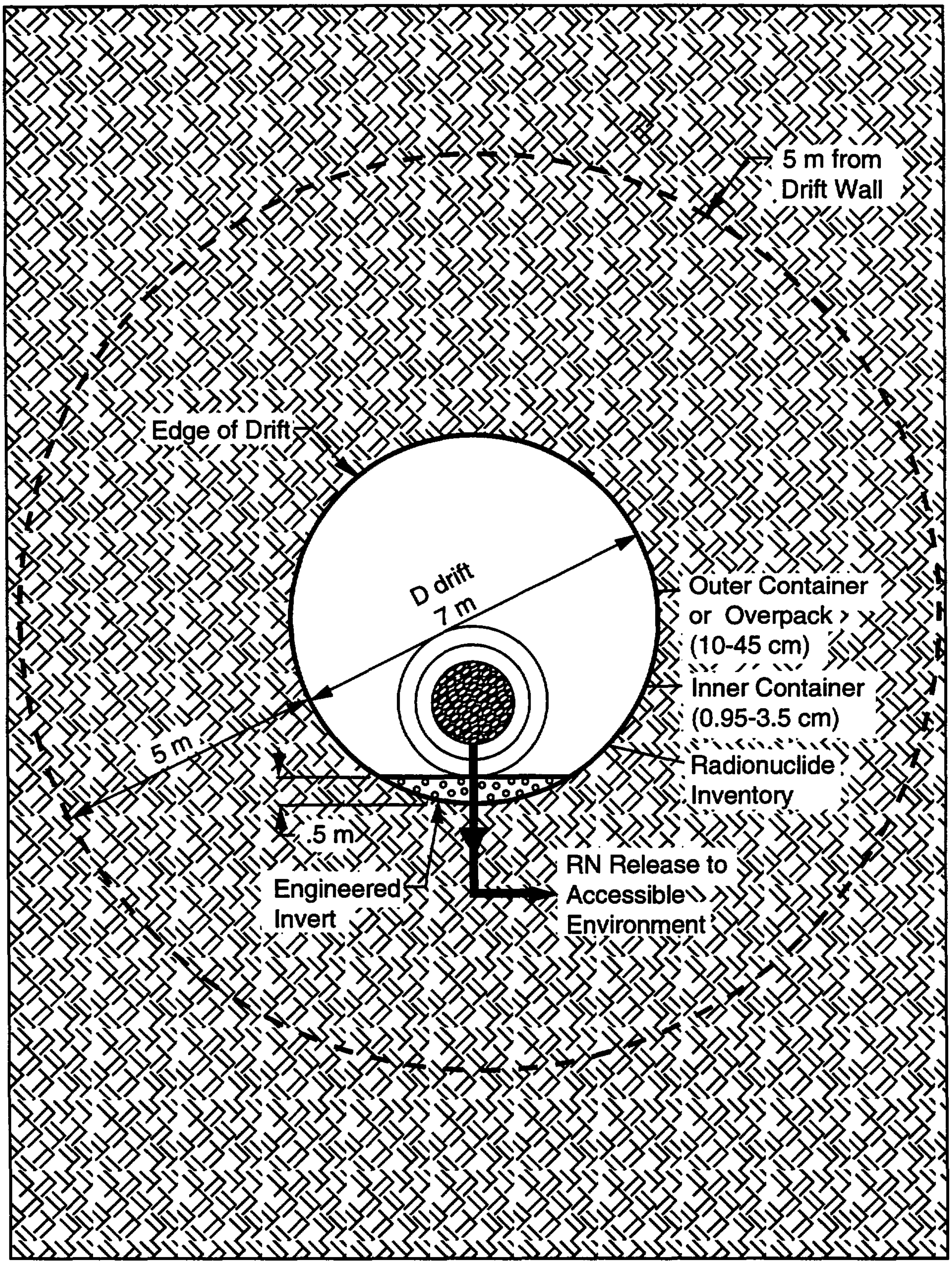


Figure 7-12. Waste Package/Engineered Barrier System Schematic (Source: DOE94d)

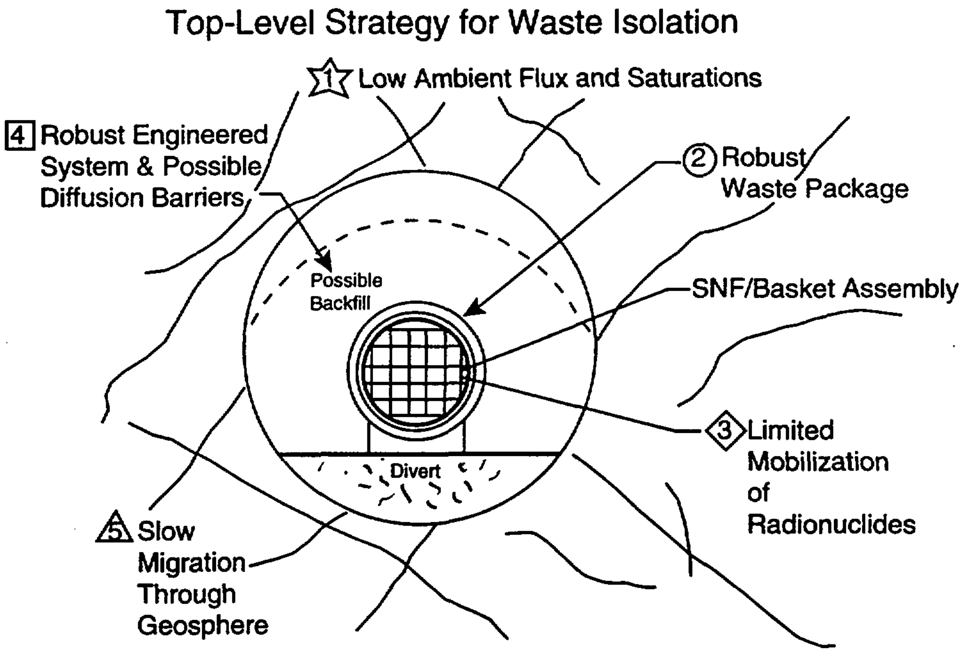


Figure 7-13. Engineered Barrier System Components (Source: DOE95c)

Release of radionuclides from spent nuclear fuel and defense high-level waste involves corrosion of containment metals, waste form alteration/dissolution, solubility constraints on the concentration of dissolved radionuclide species, transport in a mobile phase, diffusion through the degraded waste package, and the potential for advective transport in the localized flow of the near-field environment. Characteristics of these release-related factors are discussed below.

7.2.3.2 The Waste Package

The metals of the waste package are the first line of defense against the release of radionuclides from the EBS. They include the metallic containment barriers of fuel pellet cladding and waste disposal containers that have two or three layers of different metals: corrosion allowance material (CAM), corrosion-resistant material (CRM), and moderately corrosion-resistant material (MCRM).

7.2.3.3 Backfill

DOE is investigating backfill compositions around the emplaced waste containers in order to help mitigate radionuclide release. The backfill's possible chemical interaction and adsorption characteristics may retard the release of radionuclides in both aqueous and gaseous forms.

7.2.3.4 Corrosion of the Waste Package Materials

Temperature influences all the mechanisms for corrosion, such as dry oxidation, general aqueous attack, localized pitting, and fuel material alteration. The released radionuclides can be adsorbed by the minerals of the tuff and such adsorption is enhanced by temperature. The thermal character of the repository is a significant factor in corrosion and release and will depend on the waste package design and contents, spacing of drifts and waste packages, and the characteristics of the backfill.

7.2.4 Thermal Loading

7.2.4.1 General

Thermal loading refers to the temperatures and temperature gradients that develop as a result of the waste package emplacement configuration. The thermal loading is characterized by the heat emission and radioactivity content per unit area of the repository. Spacing of packages based on their MTU content provides the most uniform thermal conditions over the long-term performance period. Heat generated by a representative waste package as defined by DOE is shown in Figure 7-14. This heat generation rate then determines the waste package spacing and other waste package emplacement parameters.

7.2.4.2 Thermal Loads

DOE has investigated a wide range of repository thermal loadings and thermal designs. Most recently, TSPA-95 has used the assumption stated in the Controlled Design Assumption Document (DOE95f) Key Assumption 019: "Surface, subsurface and waste package/EBS designs will be robust and flexible and will accommodate a range of thermal loads from about 20 to about 100 MTU/Ac."

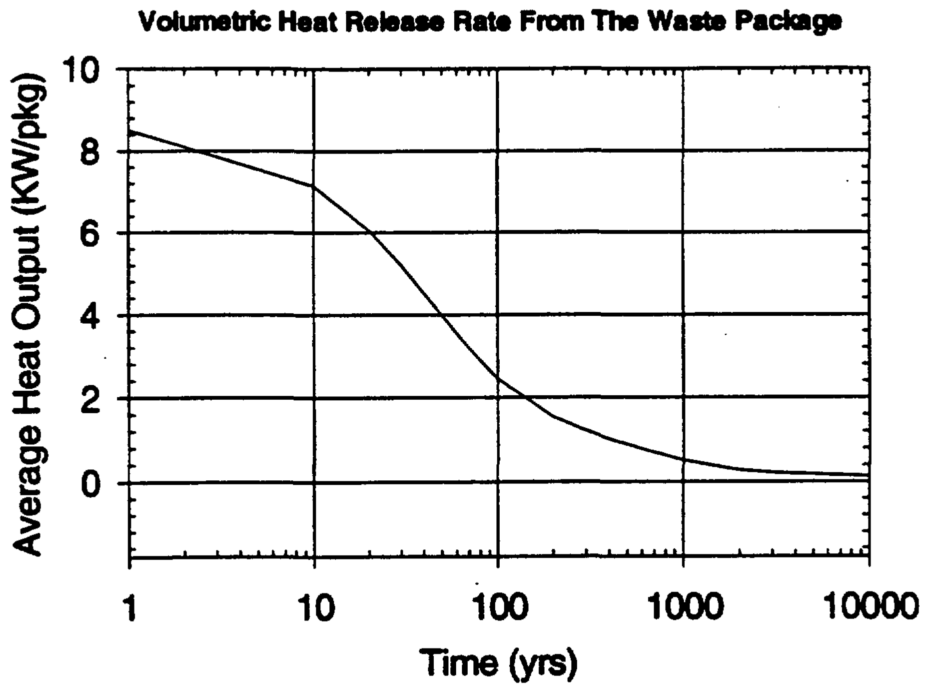


Figure 7-14. Heat Generation Rates of an Average Waste Package (Source: DOE95c)

Two cases were analyzed in TSPA-95 that reflect the design choices necessary for a "low" thermal load and a "high" thermal load:

1. Low thermal load (20-40 MTU/Ac). Either wide spacing of waste packages with moderately spaced drifts or waste packages spaced close together in widely spaced drifts will produce the low thermal loading (see Figure 3.2-2, TSPA-95). For 25 MTU/Ac an area of about 2,520 acres is required.

2. High thermal load (80-100 MTU/Ac). By reducing waste package spacing and with a proper choice of drift spacing, the high thermal load repository will result. For 83 MTU/Ac, an area of about 760 acres is required.

7.3 GEOLOGIC AND HYDROLOGIC CHARACTERISTICS OF THE SITE

7.3.1 Geologic History of the Region (Adapted from DOE95a)

In general terms, the region is characterized by a thick section of Precambrian and Paleozoic sedimentary rocks overlain by a sequence of Tertiary silicic volcanic rocks (see Figure 7-15).

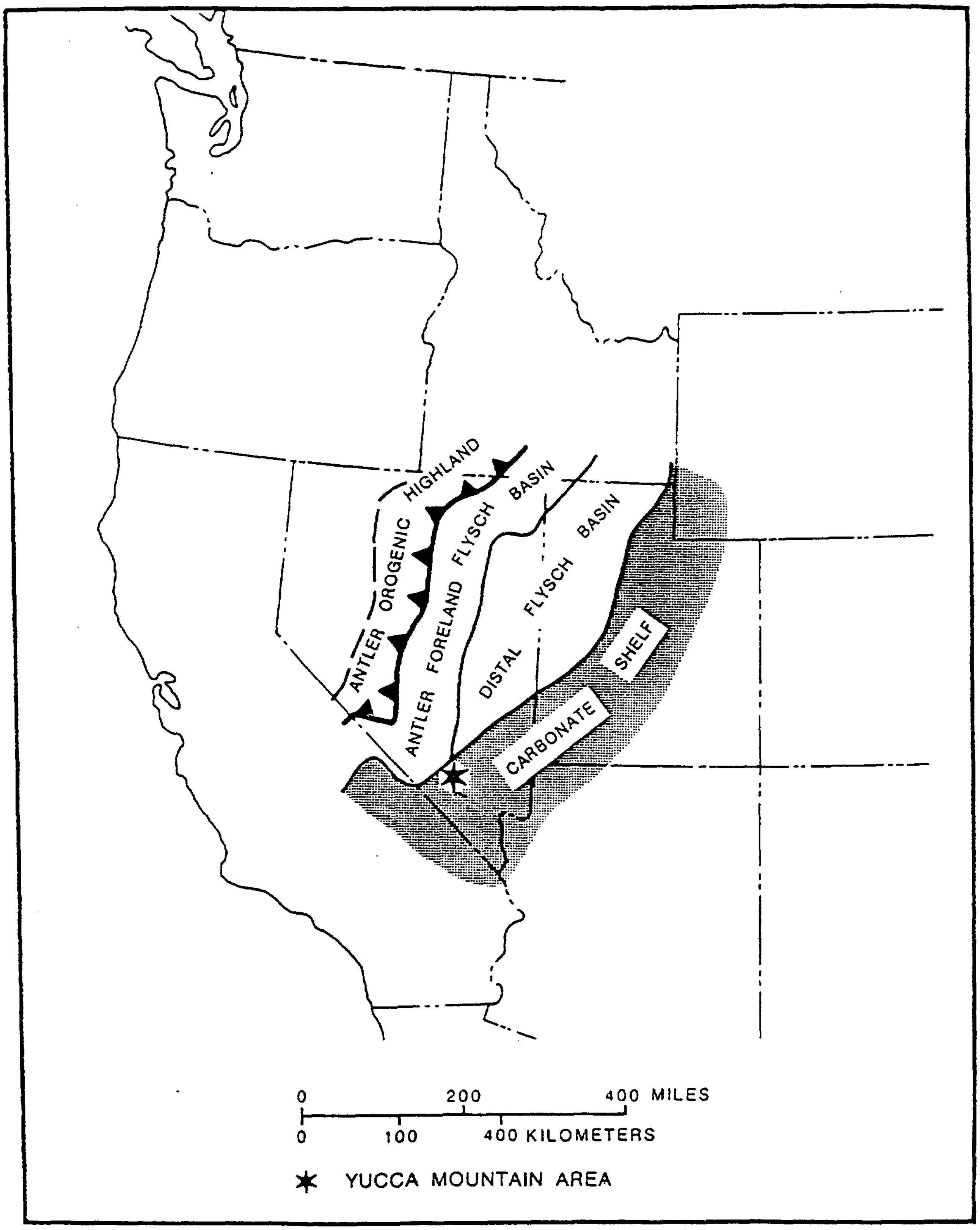


Figure 7-15. Late Devonian and Mississippian Paleogeography of the Great Basin (Source: Modified from DOE95a)

The older rocks have been folded and faulted in response to compressional tectonic events and the entire stratigraphic section subsequently deformed by extensional basin-and-range tectonics. Uplifted ranges, such as Yucca Mountain, are separated by basins partially filled with alluvial deposits.

A basement complex of older Precambrian metamorphic and younger Precambrian igneous rocks is presumed to underlie the area. These rocks are overlain by a westward-thickening prism of shallow marine late Precambrian and early Cambrian marine sediments, quartzite, siltstone, shale, and carbonate, interpreted as a rifted continental margin miogeosyncline, shown in Figure 7-16. These rocks are locally fossiliferous. Deposition continued through the Devonian, represented by carbonate and shale with interbedded quartzite and sandstone, thickening from up to 500 m in western Utah to at least 6,100 m in central Nevada.

In late Devonian and early Mississippian time the Antler Orogeny, a mountain-building event, abruptly changed the depositional environment of southern Nevada, forming a north-northeast trending highland adjacent to the Roberts Mountains Thrust, a pushing together of crustal segments or plates of the Earth surface. Large volumes of sediments eroded into a foreland basin in the eastern half of the Great Basin, forming thick flysch[10] deposits adjacent to the highlands and shallow-water shelf carbonates to the east, shown in Figure 7-17. Erosion and deposition continued through the Permian, decreasing as the mountain-building waned. In Mesozoic and early Tertiary time, these rocks were folded and displaced along thrust faults with extensive fracturing of the brittle rocks in the upper thrust plates. This faulting was accompanied by intrusion of granitic stocks, uplift, and erosion of the land surface (DUD 90).

Middle and late Cenozoic crustal uplifting and stretching occurred over an area 1,500 km long and 500 to 1,000 km wide. The stretching, estimated at 10 to 50 percent of the original width and locally as great as 100 percent, resulted in large north-northeast fractures with sliding and tilting of large crustal blocks, forming the characteristic structure and topography of the Great Basin.

Accompanying these crustal adjustments, volcanic eruptions in the vicinity of Yucca Mountain from a series of calderas deposited numerous thick beds of pyroclastics, tuff, and lava,

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[10] Flysch deposits are typified by the widespread sandstones, marls, shales, and clays occurring at the northern and southern borders of the Alps.

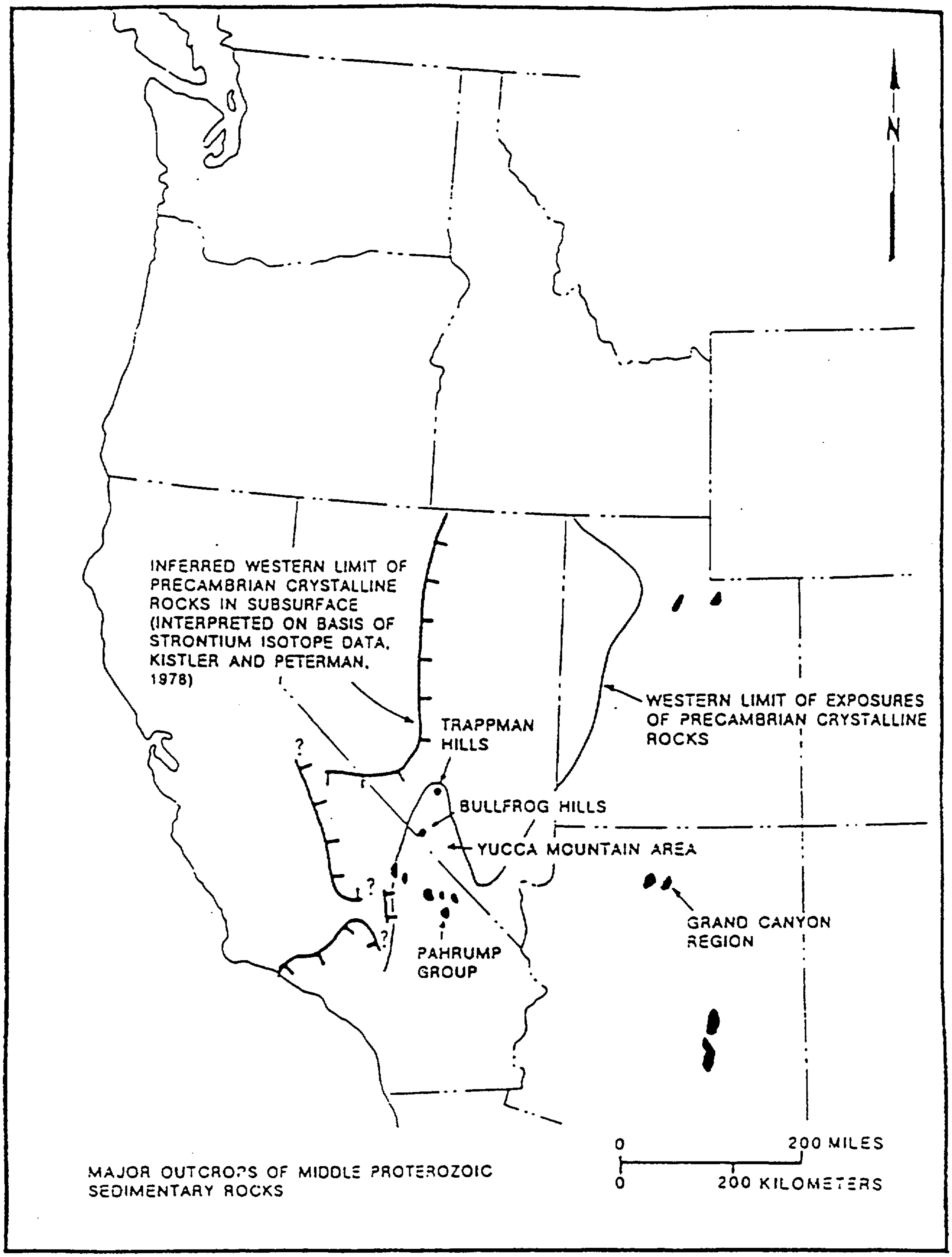

Figure 7-16. Distribution of Lower and Middle Proterozoic Crystalline Rocks and Middle Upper Proterozoic Restricted Deposits in the Great Basin (Source: Modified from DOE95a)

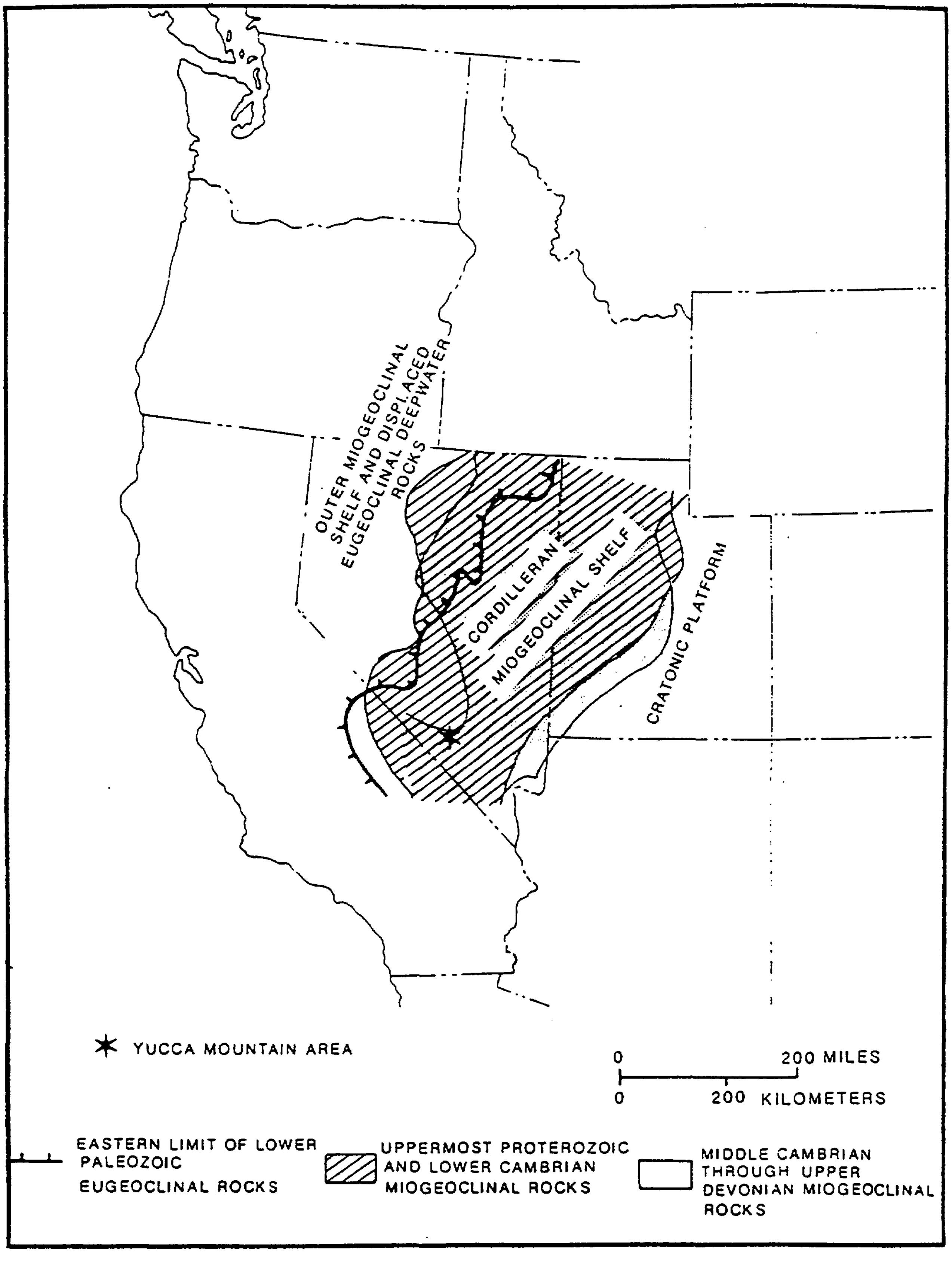

Figure 7-17. Latest Precambrian through Mid-Paleozoic Paleogeography of the Great Basin (Source: Modified from DOE95a)

aggregating up to 3 km in thickness near Yucca Mountain. The major episodes of silicic volcanism ceased about 7.5 million years ago; however, smaller basaltic eruptive centers have formed in the basins adjacent to Yucca Mountain as recently as 20,000 years ago.

7.3.2 Stratigraphy of Yucca Mountain Area (Adapted from DOE95a)

The stratigraphy of the southern Great Basin is highly varied, with formations ranging in age from Precambrian to Holocene (Figure 7-18). These rocks have been divided into eight general groups based on age, lithology, and history and are briefly described in Table 7-5.

At Yucca Mountain, the stratigraphy is dominated by mid-Tertiary rocks of volcanic origin that erupted from the southwestern Nevada volcanic field. The stratigraphic sequence can be divided into four general categories based on similarities in lithology, age, and history of deposition or emplacement: 1) Pre-Cenozoic rocks, 2) mid-Tertiary pyroclastic rocks, 3) younger basalt, and (4) late Tertiary to late Quaternary surficial deposits. These are discussed in the following sections.

7.3.2.1 Pre-Cenozoic Rocks

Pre-Cenozoic rocks, believed to consist primarily of Paleozoic sedimentary strata, underlie the pyroclastic rocks at Yucca Mountain, but little detailed information is available as to their thickness, lithology, and contact with overlying stratigraphic units. Exposures of complexly deformed Paleozoic rocks occur at scattered localities in the vicinity of Yucca Mountain, including the Calico Hills to the east, Bare Mountain to the west, and Striped Hill to the south. Similar strata may underlie the proposed repository site.

In the Calico Hills, exposures of carbonate rocks occur in the upper plate of a gently dipping thrust fault over a black shale sequence containing minor amounts of siltstone, sandstone, conglomerate, and limestone. These strata occur as structural slices and are locally highly folded, making correlation with stratigraphic units elsewhere in the region uncertain.

At Bare Mountain there is a varied sequence of pre-Cenozoic sedimentary and meta-sedimentary rocks, totaling about 6,650 m (21,800 ft) in thickness and ranging from Precambrian to Mississippian in age. Fourteen Paleozoic and two Proterozoic formations are

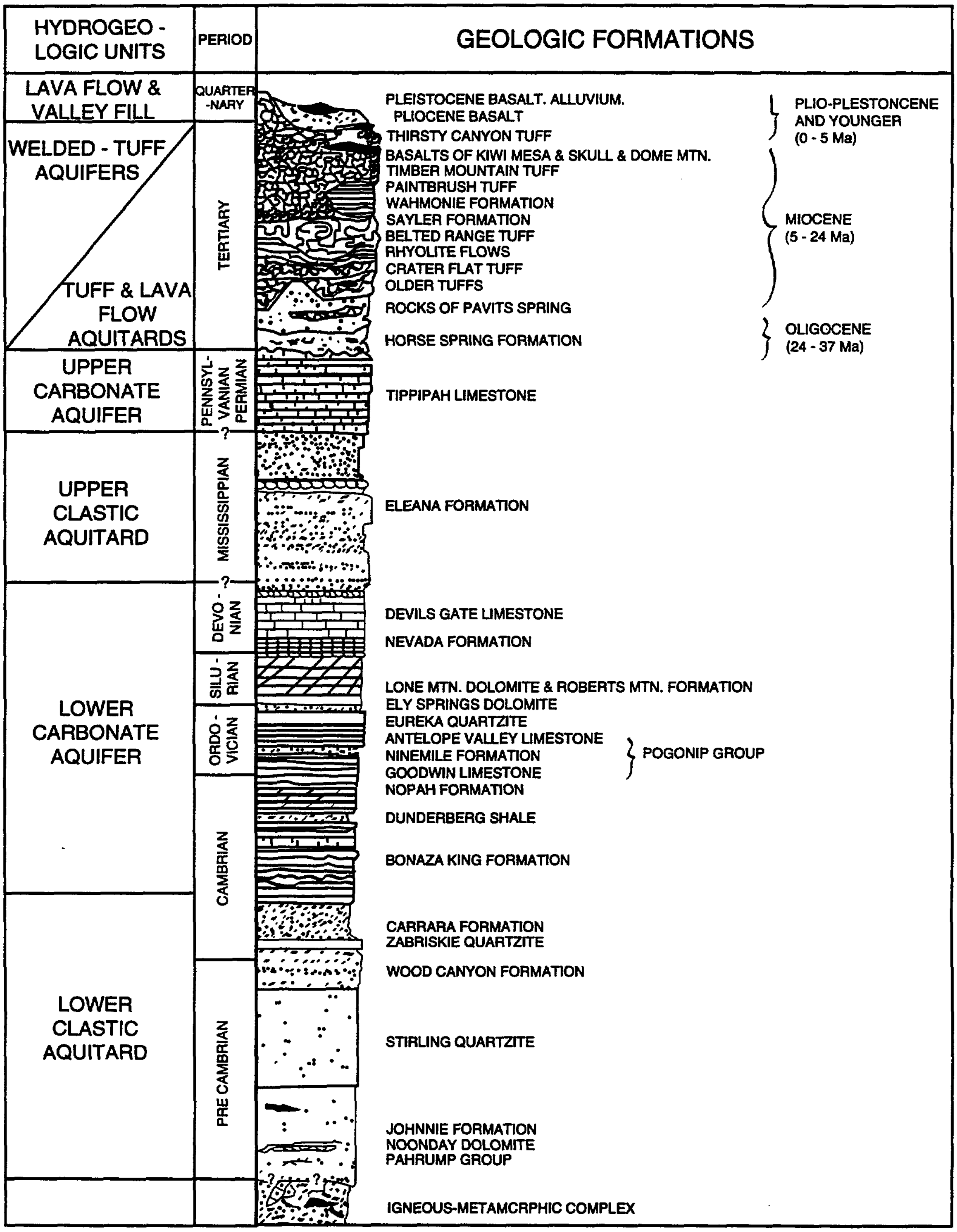

Figure 7-18. Generalized Regional Stratigraphic Column Showing Geologic Formations and Hydrogeological Units in the Nevada Test Site Area. (Modified from DOE95a)

Table 7-5. Stratigraphy of the Southern Great Basin

Older Precambrian Crystalline Rocks	These include extensive exposures of older Precambrian schist and gneiss and younger Precambrian igneous rocks in eastern Clark and southeastern Lincoln Counties, and outcrops of Precambrian granite, pegmatite, amphibolite, and gneiss in southern Lincoln County. Schist, gneiss, and gneissic quartz monzonite, possibly as young as late Proterozoic, are exposed in the Bullfrog Hills and Trapman Hills of southern Nye County.
Precambrian and Lower Cambrian Rocks	Late Precambrian and early Cambrian strata include a westward-thickening prism of quartzite, siltstone, shale, and carbonate interpreted as a rifted continental margin miogeosyncline. This prism has been divided into two depositional systems in Nevada: an eastern quartzite and siltstone system and a western siltstone, carbonate, and quartzite province.
Middle Cambrian through Devonian	Middle Cambrian through Devonian rocks exposed in the southern Great Basin consist of carbonates and shales, with interbedded quartzite and sandstone with thicknesses from up to 500 m in western Utah to at least 6100 m in central Nevada. Strata of middle Cambrian through Devonian age comprise the Lower Carbonate Aquifer.
Mississippian through Permian Sedimentary Rocks	Thick flysch* deposits result from erosion of the north-northeast trending highland formed during the Antler Orogeny in late Devonian and early Mississippian. This sedimentation continued through Permian time, declining as the orogeny waned.
Mesozoic Rocks	Mesozoic sedimentary rocks, locally present only in Clark County, consist of continental and marine sandstone, siltstone, and limestone of the Triassic and Jurassic Aztec Sandstone, Chinle Formation, and Moenkopi Formation. Approximately 30 separate Mesozoic to Tertiary granitic plutons are exposed in Esmeralda County, west of Yucca Mountain. These range in size from less than 1 km^2 to the 1000 km^2 Inyo Batholith.
Tertiary Sedimentary Rocks	Tertiary sedimentary rocks, such as the Esmeralda and Horse Spring Formations, crop out throughout the southern Great Basin. These consist of poorly to moderately consolidated alluvial deposits and fresh water limestones in variable thicknesses of up to 1000 m. They are commonly found interbedded with volcanic deposits.
Tertiary and Quaternary Igneous Rocks	The most prevalent Tertiary igneous rocks of the southern Great Basin are pyroclastic deposits of rhyolitic to trachytic composition. Eruptions from four calderas at Yucca Mountain between approximately 7 and 16 mega-annum (Ma) produced a complex mixture of pyroclastic flow and fall deposits, epiclastic deposits, and subsidiary lavas approximately 3050 m in thickness at Yucca Mountain. This was followed by scattered, small-volume basaltic or bimodal basaltic-andesitic lava and scoria eruptions, possibly as recently as 16 thousand years ago.
Tertiary and Quaternary Surficial Deposits	Late Tertiary to Quaternary surficial deposits occur throughout the region as unconsolidated alluvial fan, pediment, and basin fill deposits of highly variable thickness and character.

* Deposits largely of sandy and calcareous shales.

represented. Dolomite and limestone dominate, with minor stratigraphic units of clastic rocks (quartzite, sandstone, and siltstone).

Paleozoic rocks intersected at 1,244 to 1,807 m depth in a borehole 2 km east of Yucca Mountain are almost entirely dolomites and have been correlated with the Lone Mountain Dolomite and Roberts Mountains Formation. Seismic reflection data are inconclusive as to the thickness and extent of pre-Cenozoic rocks underlying Yucca Mountain, but the thickness is believed to be substantial.

7.3.2.2 Mid-Tertiary Pyroclastic Rocks

Volcanic rocks ranging in age from about 11.4 to 15.2 Ma form the bulk of the stratigraphic section at Yucca Mountain, including the host rock of the potential repository (Figures 7-19 and 7-20). The sequence consists of welded and nonwelded silicic pyroclastic flow and fallout tephra deposits and volcanic breccias erupted from nearby calderas in the southwestern Nevada volcanic field. The principal stratigraphic units include, in increasing age (adapted from DOE94a):

Unit	Age (Ma)
Younger Pre-caldera Basalts	0.27-3.8
Older Pre-caldera Basalts	8.5-10.5
Shoshone Rhyolite Lava	9
Timber Mountain Group	
Ammonia Tanks Tuff	11.45
Ranier Mesa Tuff	11.6
Post-Tiva/pre-Ranier Rhyolites	12.5
Paintbrush Group	
Tiva Canyon Tuff	12.7
Yucca Mountain Tuff	-
Pah Canyon Tuff	-
Topopah Spring Tuff	12.8
Calico Hills Formation	12.9
Crater Flat Group	
Prow Pass Tuff	13.1
Bullfrog Tuff	13.25
Tram Tuff	13.45
Lithic Ridge Tuff	14.0
Older Tuffs - Pre-Lithic	14-16

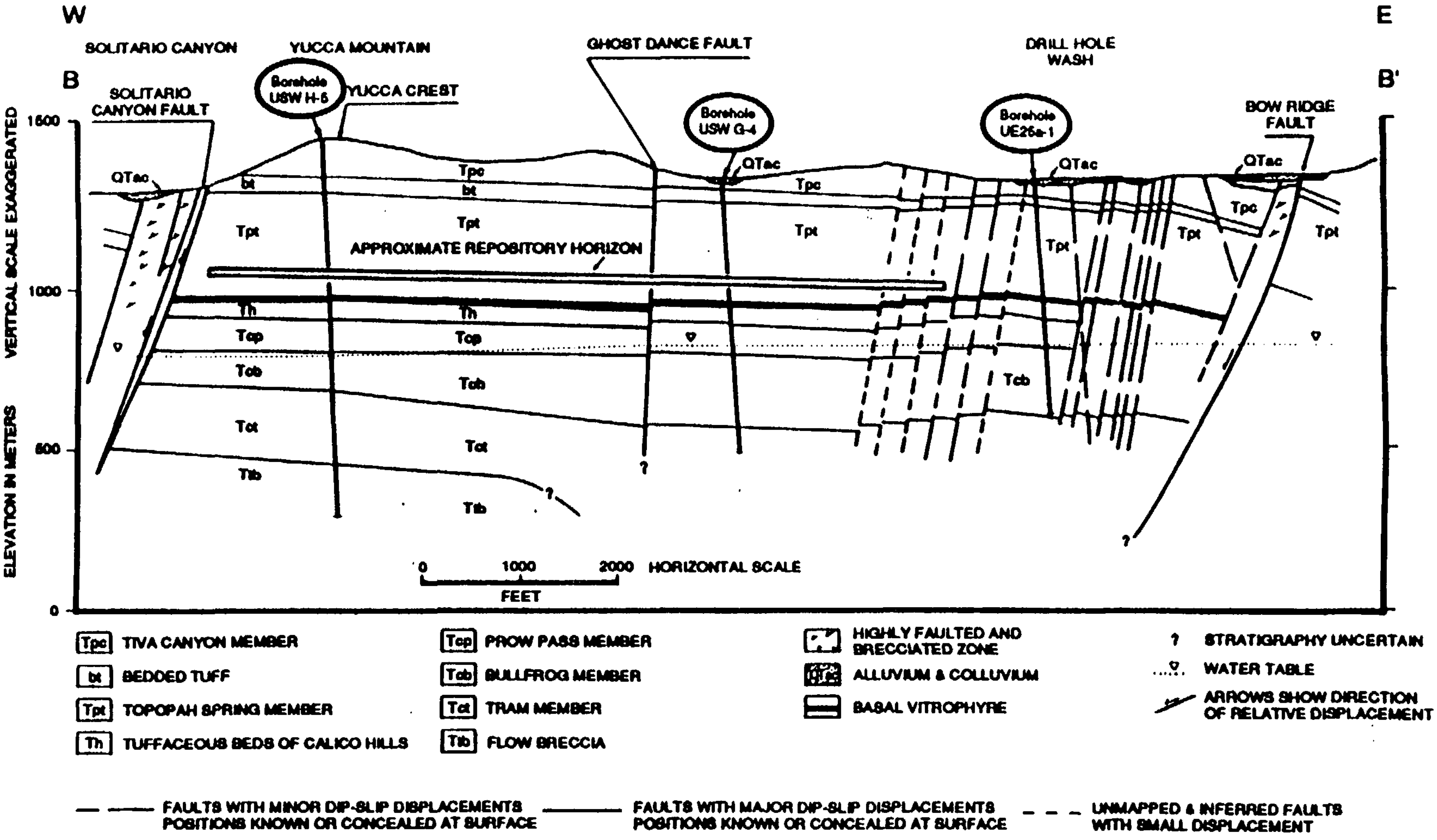

Figure 7-19. East-West Geologic Cross Section for the Yucca Mountain Site. This figure shows the relative positions of various rock units at the site, including the unit proposed for the potential repository and the fault zones that are closest to the site.

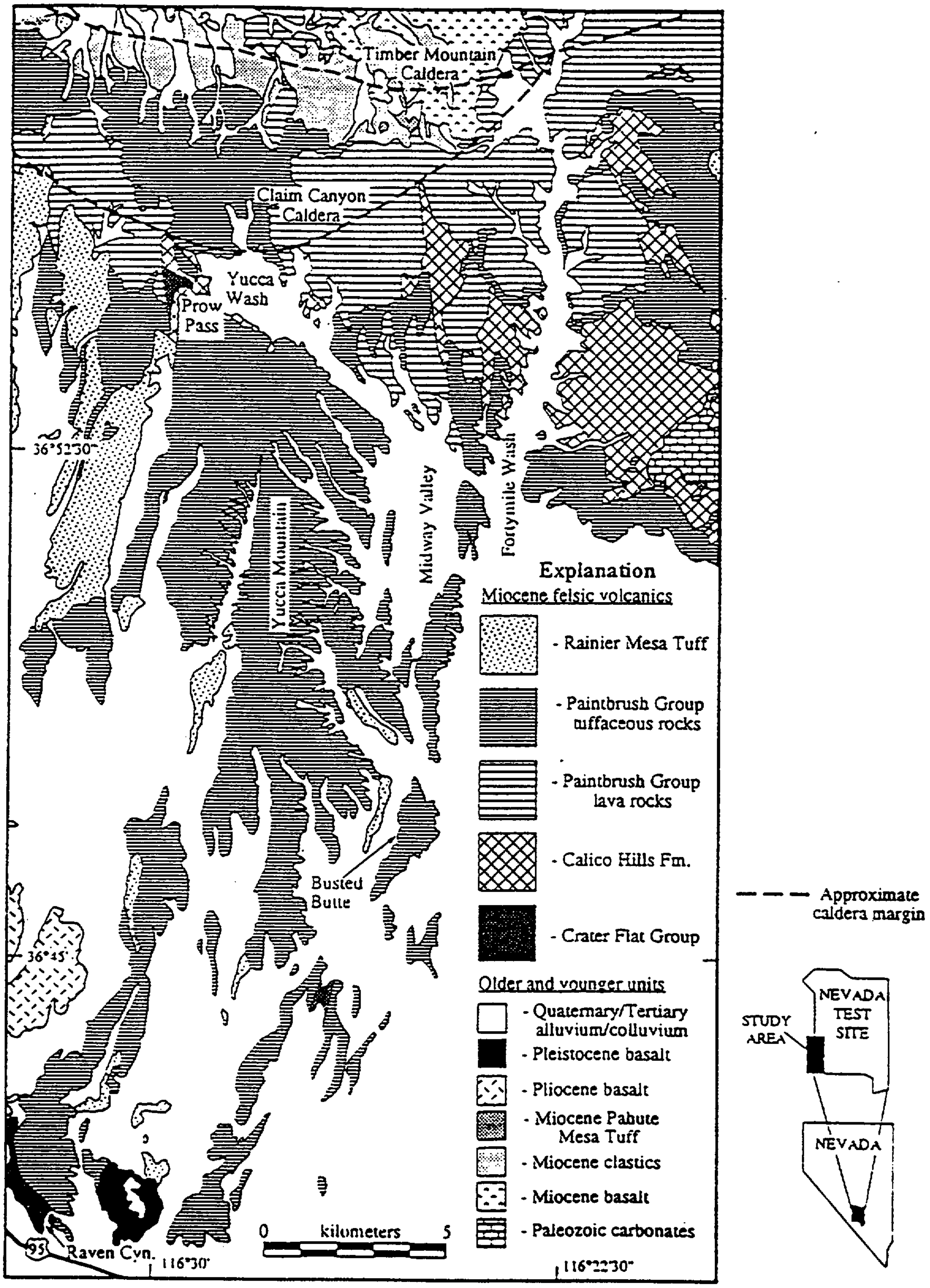

Figure 7-20. Simplified Geologic Map Showing the Distribution of Major Lithostratigraphic Units in the Yucca Mountain Area. (Source: Modified from DOE95a)

Many of these formations, particularly those in the Prow Pass Tuff, Calico Hills Formation, and the Paintbrush Group, are further subdivided into members or units. These are summarized below, from oldest to youngest.

a. Pre-Lithic Ridge Volcanic and Volcanogenic Rocks. The oldest known volcanic rocks in the area were deposited approximately 14 Ma and are represented in site boreholes by 45 to 350 m of bedded tuffaceous deposits, pyroclastic flow deposits, and quartz-latitic to rhyolitic lavas and flow breccia. Correlation of these rocks is difficult because of their heterogeneous character and varying degrees of alteration.

b. Lithic Ridge Tuff. This thick, massive pyroclastic flow deposit overlying the older tuffs appears to represent several eruptive surges and ranges in thickness from 185 m north of the site to 304 m at the south end of the site. This unit is nonwelded to moderately welded and has been extensively altered to smectites and zeolites.

c. Dacitic Lava and Flow Breccia. Dacitic lava and flow breccia overlie the Lithic Ridge Tuff in deep boreholes at the northern and western parts of Yucca Mountain but are absent elsewhere. Observed thicknesses in boreholes range from 22 m to 249 m. Much of the unit has been moderately to intensely altered to smectite clays and zeolites.

d. Crater Flat Group. This group, overlying dacitic lavas and flow breccias in the northern part of Yucca Mountain and the Lithic Ridge Tuff in the southern part, includes three rhyolitic, ash-flow-tuff sheets—the Tram, Bullfrog, and Prow Pass Tuffs, in ascending order. The Crater Flat Group is distinguished from other pyroclastic units at Yucca Mountain by the relative abundance of quartz and biotite phenocrysts.

   - **Tram Tuff**. The Tram Tuff appears to comprise at least 28 separate magmatic pulses and includes two subunits distinguished on the basis of the relative abundance of lithic fragments. The lower subunit is rich in these fragments throughout, while the upper unit is poor in lithic clasts. The upper subunit, 126 to 171 m thick, is partially welded and has a microcrystalline ground mass.

     There are 6 to 22 m of ash-fall and reworked tuff, primarily comprising zeolitic pumice clasts, between the Tram and the overlying Bullfrog Tuff.

   - **Bullfrog Tuff**. The Bullfrog Tuff is 68 to 187 m thick, consisting mostly of pyroclastic flow deposits with thin-bedded tuffaceous deposits. North of borehole USW G-4 this tuff consists of a moderately to densely welded core enclosed by nonwelded to partially welded zones; to the south it is composed of two welded zones separated by a 1-m-thick bed of welded fallout tephra.

- **Prow Pass Tuff**. The Prow Pass Tuff is a sequence of variably welded pyroclastic deposits that erupted from an unidentified source between 13.0 and 13.2 Ma. The formation, 90 to 165 m thick across the repository area, consists of four pyroclastic units overlying a variable sequence of bedded tuffs. These units, designated Unit 1 through 4 in decreasing age, are characterized by orthopyroxene pseudomorphs and the abundance of siltstone and mudstone lithic clasts. Unit contacts are defined by fallout tephra horizons and abrupt changes in sizes and amounts of pumice and lithic clasts.

  A bedded tuff unit at the base of the Prow Pass Tuff consists of unwelded, altered tuffaceous deposits with a total thickness ranging from less than 1 m to 11 m in boreholes.

  Unit 1, a pumiceous pyroclastic flow deposit with an aggregate thickness of 25 to 70 m in cored boreholes, consists of three subunits separated on the basis of their lithic clast content.

  Unit 2 consists of nonwelded to partially welded lithic-rich pyroclastic flow deposits with an aggregate thickness of 3 m to 34 m in cored sections. The unit has not been subdivided since distinguishing characteristics are lacking; however, locally preserved ash horizons and abrupt changes in the amount and size of pumice and lithic clasts suggest at least three flow deposits.

  Unit 3 consists of 40 m to nearly 80 m of multiple welded pyroclastic flow deposits, either separated by thin fallout tephra horizons or defined by abrupt changes in the amount and size of pumice and lithic clasts. Two or three flow deposits have been identified in most core holes but have not been correlated.

  Unit 4 is distinguished by comparatively abundant pseudomorphic pyroxene in pumice clasts and rock matrix and by a comparatively low ratio of felsic to mafic phenocryst minerals. This unit includes three irregularly distributed subunits. The aggregate thickness in cored sections ranges from about 4 m to as much as 20.5 m.

e. <u>Calico Hills Formation</u>. The Calico Hills Formation, a series of rhyolite tuffs and lavas, includes five pyroclastic units overlying a bedded tuff unit and a local basal sandstone unit in the Yucca Mountain area. The formation thins southward across the site area, declining from about 290 m in the north to 43 m in the south. The pyroclastic units are composed of one or more pyroclastic flow deposits separated by pumice- and lithic-fallout tephra deposits included with the unit lying above. The five units, designated Unit 1 thru 5 in decreasing age, are described below.

Basal beds of the Calico Hills Formation include a 9- to 39-m-thick bedded tuff unit containing coarse-grained fallout, primary and reworked pyroclastic-flow, and fallout tephra deposits, and a 0- to 5.5-m-thick laminated to massive volcaniclastic sandstone unit with abundant lithic clasts and swarms of argillitically altered pumice clasts, interbedded with rare pyroclastic-flow deposits.

Unit 1 is a nonwelded, lithic rich, pyroclastic-flow deposit from 0 to 58 m thick in cored sections. Pumice clasts constitute 10 to 15 percent of the unit and lithic clasts increase from 3 to 7 percent at the top to 15 to 20 percent at the base; phenocrysts compose 7 to 12 percent of the rock.

Unit 2, 0 to 54 m thick, is a nonwelded, pumiceous, pyroclastic-flow deposit composed of 20 to 40 percent pumice clasts and up to 5 percent lithic clasts. Fallout deposits at the base are ash-rich, have a porcelaneous appearance, and are less than 1 m thick.

Unit 3 is a nonwelded lithic-rich pyroclastic flow deposit 22 m to 100 m thick in cored sections. The unit is generally composed of 10 to 40 percent pumice clasts and 5 to 10 percent lithic clasts.

Unit 4 is a 0 to 57 m thick nonwelded, pumiceous pyroclastic flow deposit, with pumice clasts and lithic clasts constituting 10 to 30 percent and 1 to 5 percent respectively. Thinly bedded ash-fall deposits, reworked pyroclastic-flow tuffs, and tuffaceous sandstone form a thin basal subunit.

Unit 5 is a nonwelded to partially welded pyroclastic flow deposit from 0 to 20 m thick in cored sections. The unit is characterized by a bimodal distribution of pumice clast sizes; larger, slightly flattened clasts of 20 to 60 mm and smaller equidimensional clasts of 2 to 12 mm. The unit is composed of 20 to 30 percent pumice clasts and 2 to 5 percent lithic clasts.

f. Paintbrush Group. This group—one of the most widespread and voluminous caldera-related assemblages in the southwestern Nevada volcanic field—consists of primary pyroclastic flow and fallout tephra deposits, lava flows, and secondary volcaniclastic deposits from eolian and fluvial processes.

Eruptive centers for the Topopah Spring Pah Canyon Tuffs are uncertain, but the Claim Canyon caldera is identified as the source of the Tiva Canyon and perhaps the Yucca Mountain Tuffs.

- The **Topopah Spring Tuff** (Figure 7-21), the host rock for the proposed Yucca Mountain repository, has a maximum thickness of about 350 m in the vicinity of Yucca Mountain. The formation is divided into two members—an upper crystal-rich member and a lower crystal-poor member—each of which is

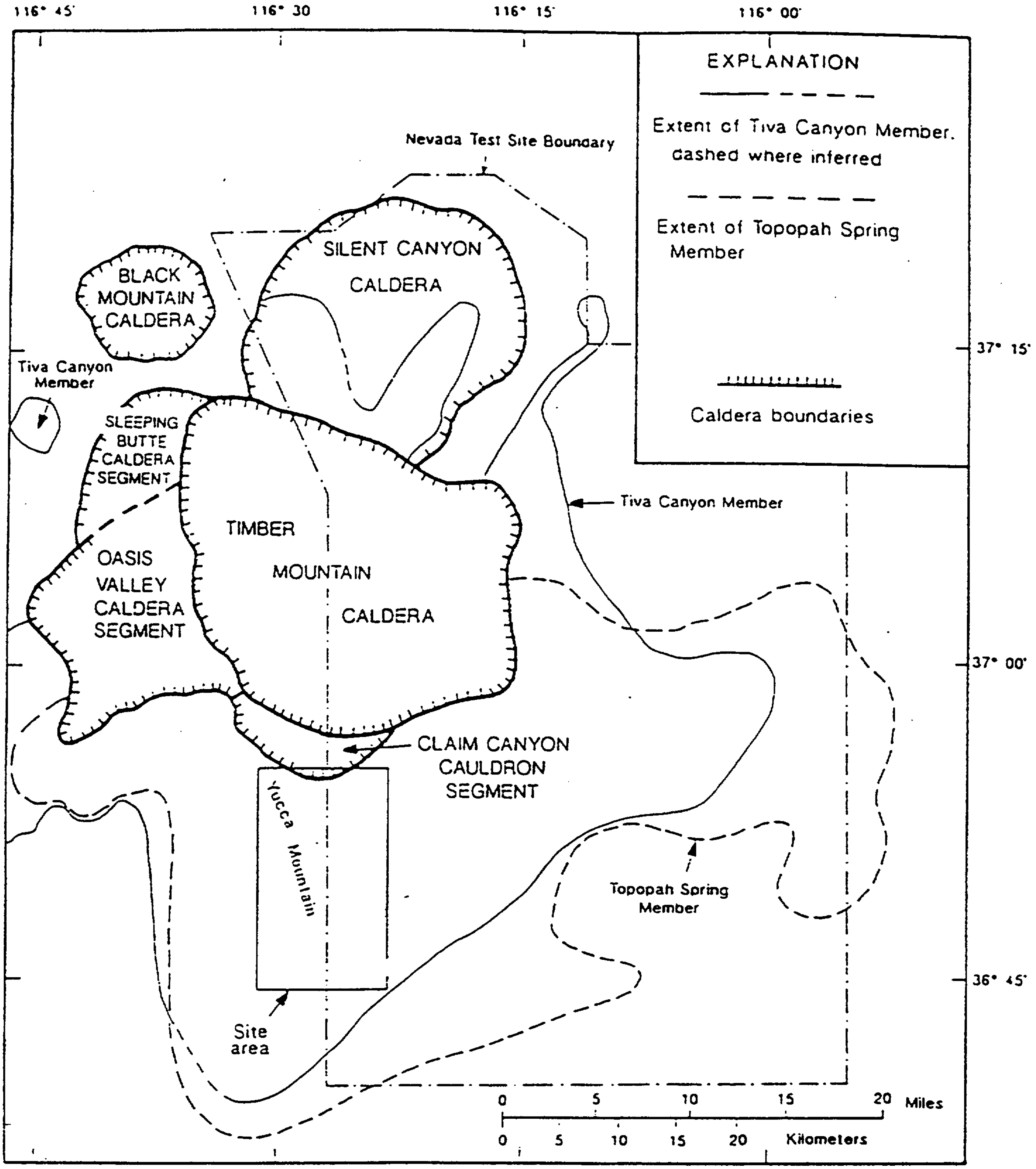

Figure 7-21. Index Map Showing Outlines of Calderas in the Southwestern Nevada Volcanic Field, and Extent of the Tiva Canyon and Topopah Spring Tuffs of the Paintbrush Group (Modified from DOE95a)

subdivided based on variations in crystal content, phenocryst assemblage, pumice composition, distribution of welding and crystallization zones, depositional features, and fracture characteristics.

The upper, crystal-rich member is characterized by greater than 10 percent phenocrysts, with a basal transition zone where the percentage increases from 5 to 10 percent. The member is divided into vitric, nonlithophysal, and, locally, lithophysal zones.

The lower, crystal-poor member, characterized by less than 3 percent phenocrysts, is divided into devitrified rocks of the upper lithophysal, middle nonlithophysal, and lower lithophysal zones and a vitric zone.

- The **Pah Spring Tuff**, a simple cooling unit composed of multiple flow units, reaches its maximum thickness of 70 m in the northern part of Yucca Mountain and thins southward. This tuff varies from nonwelded to moderately welded. Throughout much of the area, vitric pumice clasts are preserved in a sintered or lithified nondeformed matrix.
- The **Yucca Mountain Tuff**, a simple cooling unit in the Yucca Mountain area, varies in thickness form 0 to 30 m. Generally nonwelded, the formation is nonlithophysal throughout Yucca Mountain but contains lithophysae where densely welded in northern Crater Flat.
- The **Tiva Canyon Tuff** (Figure 7-21) is a large-volume, regionally extensive, compositionally zoned (from rhyolite to quartz latite) tuff sequence that forms most of the exposed surface rocks exposed at Yucca Mountain. It ranges in thickness from 100 to 150 m. Separation into crystal-rich and crystal-poor members and into zones within these members is based on similar criteria and characteristics as discussed above for the Topopah Spring Tuff.

g. Post-Tiva Canyon, pre-Rainier Mesa Tuffs. A sequence of pyroclastic flow and fallout tephra deposits occurs between the Tiva Canyon Tuff and the Rainier Mesa Tuff in the vicinity of Yucca Mountain. The sequence is from 0 to 61 m thick and is intermediate in composition between Tiva Canyon and Rainier Mesa Tuffs.

h. Timber Mountain Group. This group includes all of the quartz-bearing pyroclastic flow and fallout tephra deposits that erupted from the Timber Mountain caldera complex about 11.5 Ma. The complex consists of two overlapping, resurgent calderas—one formed by eruption of the Rainier Mesa Tuff, and a younger, nested one formed by eruption of the Ammonia Tanks Tuff.

- The **Rainier Mesa Tuff** is one of the most widespread pyroclastic units of the Yucca Mountain area. It is a compositionally zoned, compound cooling unit consisting of high-silica rhyolite tuff overlain with a partial cooling break by a considerably thinner quartz latite tuff restricted to the vicinity of the Timber Mountain caldera. Exposed thicknesses along the west side of the caldera are as great as 500 m. The formation is absent across much of Yucca Mountain, but appears in down-thrown blocks of large faults in valleys on either side. The tuff is nonwelded at the base, grading upward into partially to moderately welded devitrified tuff.

- The **Ammonia Tanks Tuff** consists of welded to nonwelded rhyolite tuff with a highly variable thickness of up to 215 m. It is absent across Yucca Mountain but is exposed in the southern part of Crater Flat.

7.3.2.3 Younger Basalt

The youngest volcanic rocks in the Yucca Mountain area are the basalts in Crater Flat basin and the basalt dikes along the Solitario Canyon fault near its intersection with the Drill Hole Wash fault on the west side of Yucca Mountain. The basalts in Crater Flat basin, consisting of scoria cones and thin lava flows and flow breccias, were placed during four major eruptions dated at 11 Ma, 3.7 Ma, 1 Ma, and 0.1 Ma.

7.3.2.4 Surficial Deposits

Numerous Quaternary/Tertiary surficial deposits have been defined in the Yucca Mountain area. These include alluvial, colluvial, and eolian deposits.

The alluvial deposits range in age from late Tertiary (probably late Miocene) to late Holocene and generally consist of sandy gravel (granules to boulders), often with interbedded sands. These occur along the washes, drainage channels, and valley slopes.

The colluvial deposits range in age from Quaternary to early Pleistocene and generally consist of a thin mantle of angular gravels on slopes and highlands.

Two deposits of eolian sand ramp are defined, both formed of massive to poorly-bedded sand with 5 to 50 percent fine angular gravel. One (late and middle Pleistocene) forms partially dissected aprons between gullies on lower hill slopes. The other (Holocene and late Pleistocene) forms undissected and poorly exposed sand ramps along Fortymile Wash.

7.3.3 Major Fault Features of Yucca Mountain Region (Adapted from DOE 95j)

Yucca Mountain consists of a series of north-trending, eastwardly tilted structural blocks that were segmented by west-dipping, high-angle normal faults during a period of major extensional deformation. It is situated near the southern end of the northwest trending Walker Lane Belt, a zone of northwest-directed shear about 700 km long and 100 to 300 km wide that absorbs part of the transform motion of the plates and the strain from the extension of the Great Basin. This belt, with associated mountain ranges and faulting, parallels the San Andreas fault and the Sierra Nevada Mountains and is truncated on the south by the east-west Garlock fault (Figure 7-22).

Cenozoic deformation probably took place on preexisting structures and has been characterized by strike-slip faulting, regional folding, and large-scale extension. The current type of deformation in the Walker Lane belt probably began about five million years ago as an overlap between the right-lateral shear caused by the plates and the gravity-driven extension of the Cordillera-wide uplift. In the modern stress field, northwest-striking faults move with left-lateral strike-slip or oblique-slip.

In the Walker Lane belt, right angle-shear totaling 4.27 to 7.35 mm/yr is distributed along three major faults: the Owens Valley, Panamint Valley-Hunter Mountain, and Death Valley-Furnace Creek faults. This, along with lesser amounts of slip on other fault systems to the east, correlates well with the approximate 10 mm/yr estimated from very long baseline interferometry.

The major north-trending faults transecting or close to Yucca Mountain are, from west to east, the Crater Flat, Windy Wash, Fatigue Wash, Solitario Canyon, Stagecoach Road, Ghost Dance, Bow Ridge, Midway Valley, and Paintbrush Canyon faults (Figure 7-23). Bedrock has been displaced down-to-the-west along these faults, which show dominantly dip-slip with varying amounts of left-oblique slip. Estimates of bedrock displacement over the past 12 million years range from less than 100 m to as much as 600 m; displacement increases southward along each fault. The faults are projected up to 25 km, but surface exposures can usually be traced only 1 km or less. Dips of the fault planes are generally 70 to 75 degrees.

Several northwest-trending faults have been identified along valleys, the most prominent being the Yucca Wash, Sever Wash, Pagany Wash, and Drill Hole Wash faults. A northwest-

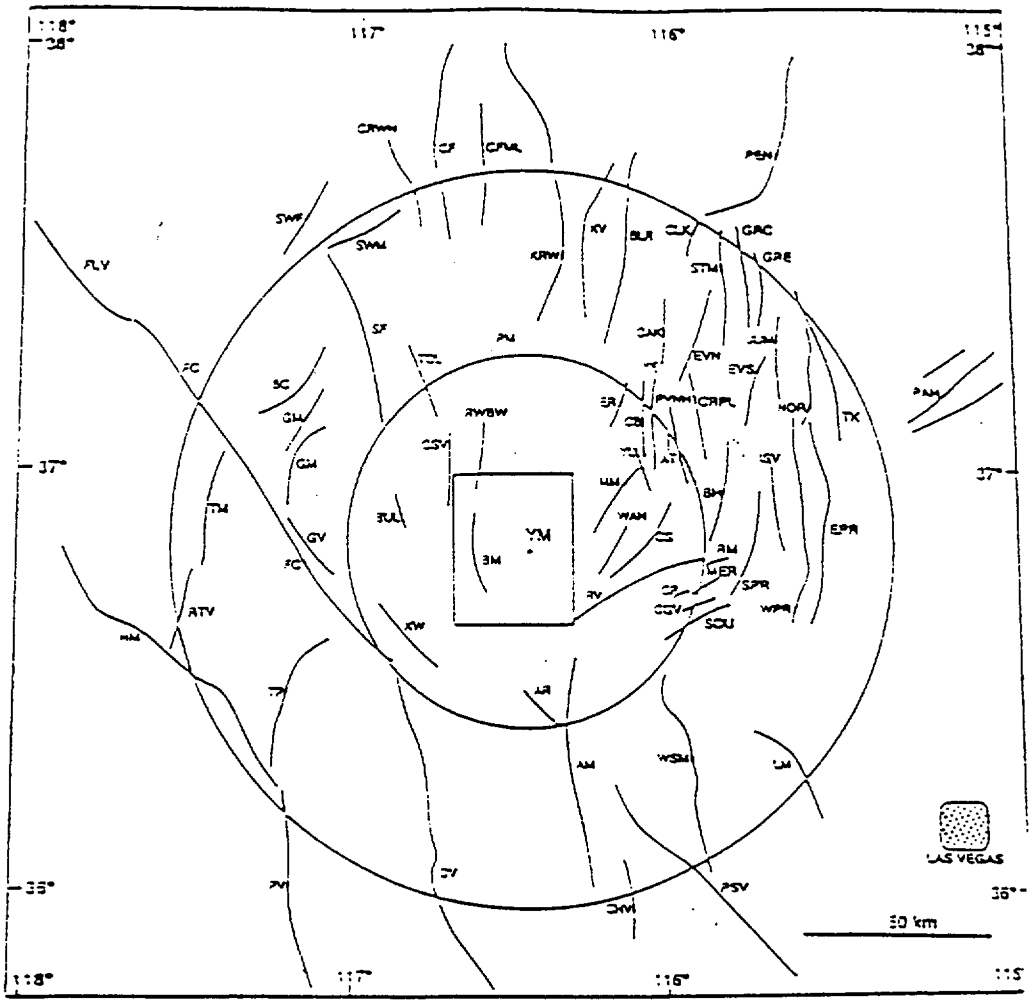

Figure 7-22. Index Map of Known or Suspected Quaternary Faults in the Yucca Mountain Region. Circles are 50 and 100 km radii from Yucca Mountain (YM). Faults are identified as follows:

AM	Ash Meadow
AR	Amargosa River
AT	Area Three
BC	Bonnie Claire
BH	Buried Hills
BLR	Belted Range
BM	Bare Mountain
BUL	Bullfrog Hills
CB	Carpetbag
CF	Cactus Flat
CFML	Cactus Flat-Mellan
CGV	Crossgrain Valley
CHV	Chicago Valley
CLK	Chalk Mountain
CP	Checkpoint Pass
CRPL	Cockeyed Ridge-Papoose Lake
CRWH	Cactus Range-Wellington Hills
CS	Cane Spring
DV	Death Valley
EPR	East Pintwater Range
ER	Eleana Range
EVN	Emigrant Valley North
EVS	Emigrant Valley South
FC	Furnace Creek
FLV	Fish Lake Valley
GM	Grapevine Mountains
GRC	Groom Range Central
GRE	Groom Range East
GV	Grapevine
HM	Hunter Mountain
ISV	Indian Springs Valley
JUM	Jumbled Hills
KRW	Kawich Range West
KV	Kawich Valley
KW	Keane Wonder
LM	La Madre
MER	Mercury Ridge
MM	Mine Mountain
NDR	North Desert Range
OAK	Oak Spring Butte
OSV	Oasis Valley
PAH	Pahranagat
PEN	Penoyer
PM	Pahute Mesa
PSV	Pahrump-Stewart Valley
PV	Panamint Valley
PVNH	Plutonium Valley-North Halfpint Range
RM	Ranger Mountains
RTV	Racetrack Valley
RV	Rock Valley
RWBW	Rocket Wash-Beatty Wash
SF	Sarcobatus Flat
SOU	South Ridge
SPR	Spotted Range
STM	Stumble
SWF	Stonewall Flat
SWM	Stonewall Mountain
TK	Tikaboo Valley
TM	Tin Mountain
TOL	Tolecha Peak
TP	Towne Pass
WAH	Wahmonie
WPR	West Pintwater Range
WSM	West Springs Mountain
YF	Yucca Flat
YL	Yucca Lake

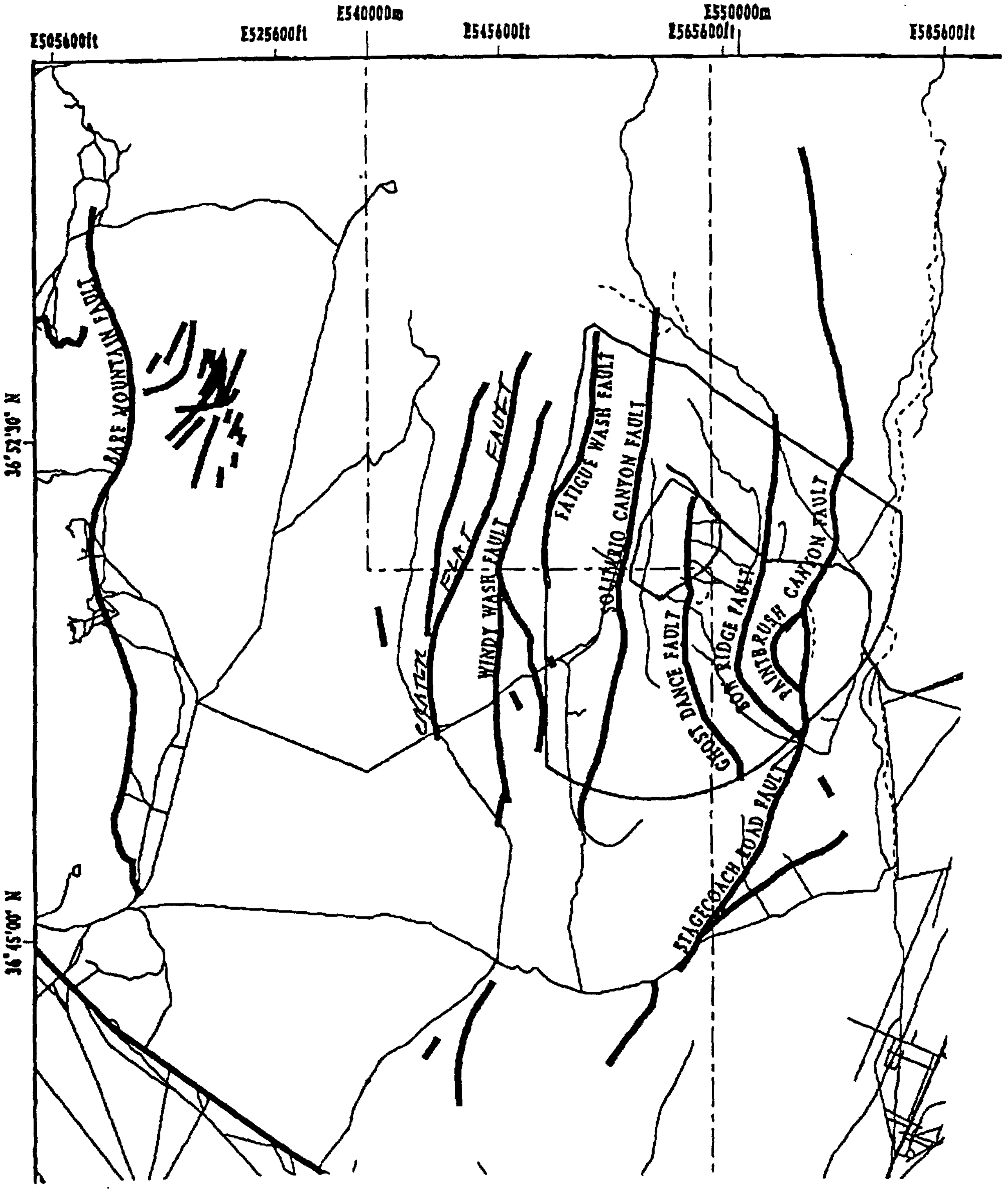

Figure 7-23. Major North-Trending Faults in the Vicinity of Yucca Mountain (DOE95b)

trending shear zone, the Sundance Fault, crosses the potential repository site. These faults are thought to be strike-slip faults with nearly horizontal slickenside lineations and generally less than 5 to 10 m vertical displacements.

7.3.3.1 Quaternary Faulting in Yucca Mountain Vicinity

Of particular concern for the Yucca Mountain site are faults considered to be Type I faults, as classified by the U.S. Nuclear Regulatory Commission. Type I faults or fault zones are those subject to displacement and are sufficiently long or located such that they may affect repository design and(or) performance. Evidence of movement during the Quaternary Period (the past one million years) is the primary criterion for identification of these faults.

Several of the major north-trending faults show evidence of activity during Quaternary time; the total displacement on the most active of these is estimated to be less than 50 m over the past 1.6 Ma. Since the late Quaternary Period (< 128,000 years), displacements have been as much as 6 m but are more commonly in the 1 to 2.5 m range. The northwest-trending faults do not appear to have been active during the Quaternary; Quaternary alluvial terraces in the floors of washes appear undisturbed.

Studies to identify and characterize faults that may be of concern to the Yucca Mountain facility have focused on evaluation of potential Type I faults within 100 km and a few major faults at greater distances. Some 82 known or suspected Quaternary faults and fault rupture combinations have been identified within 100 km of the Yucca Mountain site (Figure 7-22). Current data suggest that 38 of these are capable of generating a peak acceleration of 0.1 g or greater at the proposed repository site; these are classified as relevant earthquake sources. Significant known or suspected Quaternary faults located within 20 km of the Yucca Mountain site are briefly described in Table 7-6. The more distant major fault zones and distances from the Yucca Mountain site include: Garlock Fault (125 km south), Owens Valley Fault (140 km west), Stewart-Monte Cristo Valley Fault (200 km northwest), and Dixie Valley Fault. These are summarized at page 3.1-8 *et seq.* (DOE95a).

Table 7-6. Known or Suspected Quaternary Faults within 20 Km of the Site

Fault Name	Trend	Apparent Length	Dip	Distance from Site	Latest Activity
Bare Mountain	N	20 km	E50-70	15 km W	Most recent surface rupture 16 to ka; 1 to 1.5 m displacement; recurrence interval 100 ka; slip rate 0.01 mm/yr
Crater Flat	NE	14-20 km		5 km W	Quaternary deposits (17 to 30 ka) displaced less than 1 m
Windy Wash	N-NE	25 km	W63	3 km W	At least 4 events in past 300 ka; recurrence interval 75 ka; Pleistocene displacement ~1 m
Fatigue Wash	N	17 km	W73	2 km W	5 late Quaternary events; cumulative displacement 2.2 m
Solitario Canyon	N	20 km	W72	at W boundary	Multiple mid to late Quaternary events; 1.7 to 2.5 m displacement of Quaternary deposits
Stagecoach Road	N-NE	10 km	W	SE corner of area	3 to 7 events during late Quaternary; displacement 1 to 2.3 m; recurrence interval 5 to 70 ka; slip rate 0.01 to 0.06 mm/yr
Ghost Dance	N			center of area	No offset or fracturing of late Pleistocene or Holocene noted except for a single fracture in one trench
Dune Wash	N-NW	8 km		at E side	No evidence of Quaternary activity found
Bow Ridge	N		W65-75	2 km E	Most recent event 48±20 ka; cumulative displacement 0.3 to 0.7 m; recurrence interval 20 to 130 ka; slip rate 0.002 to 0.01 mm/yr
Midway Valley	N	1-4 km			No recognizable ruptures of Quaternary deposits
Paintbrush Canyon	N	25 km	W71	E side of Yucca Mtn.	6 to 8 events evident; Midway Valley excavation: most recent event at 38±6 ka, cumulative displacement 1.7 to 2.7 m; recurrence interval 20 to 80 ka, slip rate 0.007 to 0.02 mm/yr; Busted Butte exposure: Quaternary displacement 4.8 to 7.8 m, recurrence interval 40 to 125 ka, slip rate 0.006 to 0.01 mm/yr

7.3.4 Tectonics and Seismicity (Adapted from DOE95a)

The plate tectonic setting of the southwestern United States is dominated by interaction of the North American and Pacific Plates. In the Yucca Mountain Region, particularly west of Yucca Mountain, this interaction is complicated by the overlap of right-lateral plate boundary stress and extensional stress of the Basin and Range.

7.3.4.1 Plate Tectonic Setting

Based on geologic and geodetic measurements, the Pacific plate appears to be moving northwest at approximately 50 mm/yr relative to the North Atlantic plate. About 35 mm/yr of the motion is absorbed by the San Andreas fault and another 5 mm/yr or so may be absorbed by coastal strike-slip faults parallel to and west of the San Andreas fault. This would leave about 10 mm/yr of plate motion to be accounted for by other structures. Indications are that the eastern edge of the Sierra Nevada microplate (composed of the Sierra Nevada Mountains and the Great Valley of California) is moving northwest at approximately 10 mm/yr. This latter movement, between the eastern edge of the Sierra Nevada Mountains and the western edge of the Colorado Plateau, is the most significant for the Yucca Mountain site. Uncertainties in this analysis include: compression normal to the San Andreas fault induced by Pacific plate motion (N36°W ±2°); the rate of relative motion between plates; and the amount of motion taken up within the Sierra Nevada microplate.

Several mechanisms have been proposed for the extensional tectonic processes that produced the major landforms of the Great Basin. Relatively high-angle, planar, normal faults cutting brittle crust can accommodate up to 10 or 15 percent extension. Normal faults at a high angle at the surface and curving to lower angles at depth (listric faults) may accommodate much greater extension. Modeling of very low angle detachment faults suggests extensive crustal thinning that may accommodate extension of the crust by 200 percent or more.

The typical Basin and Range structures, tilted fault block ranges with relatively large displacement, high-angle normal faults exposed at the surface, bounding one or both sides of each range, were developed by about 11 Ma. It has been suggested (SCO90) that rates of fault movement were highest between 13 and 11.5 Ma, decreasing since then.

This extension varied across the region in time and space; it is thought that rapid Miocene extension migrated westward from Yucca Mountain after about 11.5 Ma and may also have been nonuniform from north to south. Pliocene and later extension, accompanying a postulated Cordillera-wide uplift starting at about 5 MA, is more evenly distributed and is taken up on high-angle normal faults coincident with the Miocene faults at the surface. This is consistent with the faulting to depths of 15 km or more indicated by current seismicity in the region.

7.3.4.2 Structural Features and Seismicity

The relationship between specific structural features, particularly faults, and seismicity in the Basin and Range is not clear. The Central Nevada Seismic Belt (CNSB), for example, is clearly associated with major faults or fault systems evidencing historic surface rupture. However, other zones of seismic activity and areas of diffuse activity show no historic surface faulting. An example is the east-west seismic belt, which includes the Nevada Test Site (NTS).

The apparently poor correlation between earthquakes and faults may be attributable, at least in part, to several factors: the short historical record relative to the long recurrence intervals for earthquakes, the difficulty of accurately locating epicenters in this remote area, and the unknown geometry of faults at depth. Study of the paleoseismic record for the Quaternary suggests that, in the Yucca Mountain Region, recurrence intervals for surface rupture are on the order of thousands to tens of thousands of years.

7.3.4.3 Seismology of the Yucca Mountain Area

In the region around the site there are several zones in which seismicity is concentrated: the Sierra Nevada-Great Basin Boundary Zone (SNGBZ), the CNSB, the Southern Nevada Transverse Zone (SNTZ), the Garlock Fault, and the Mojave Block. All of the zones except the Mojave Block are wholly or partially in the Walker Lane belt, a major tectonic element of southwestern Nevada. In addition, throughout much of the Great Basin there is a broad distribution of seismic activity not associated with any known major tectonic feature.

The **Walker Lane Belt** tectonic element consists of nine structural blocks acting more or less independently and defined by a style of faulting which ranges from northwest-trending right-lateral slip (the Pyramid Lake, Walker Lane, and Inyo-Mono blocks), to northeast-trending left-lateral slip (the Carson, spotted Range-Mine Mountain, and Lake Mead blocks), and to east-west trending left-lateral slip (Excelsior-Coaldale block). Cumulative lateral offset on individual major faults ranges from a few km up to 100 km; faults rarely extend to adjacent blocks.

The Walker Lane belt probably developed in the Mesozoic, however, most of the faults show evidence of Cenozoic movement, and numerous zones exhibit Quaternary and Holocene offset. Although the recurrence interval for the late Quaternary faulting has generally been thousands to tens of thousands of years, in some sections of the seismic zone, e.g., the CNSB, recurrence may be on the order of decades.

Of the four seismic zones identified in the Walker Lane Belt, the SNTZ is most significant to evaluating the Yucca Mountain site because of its proximity. Although the other zones exhibit recent seismic activity, they are further removed from the Yucca Mountain site.

The **Southern Nevada Transverse Zone** is an arcuate belt of seismicity about 150 km wide, extending from the southern region of the Intermountain Seismic Belt (in southwestern Utah) to the Mammoth Lakes area, and includes Yucca Mountain. Historic earthquakes in this zone have been of moderate magnitude with no documented surface rupture. Events include the 1902 Pine Valley, Utah (M_L 6.3), the 1966 Caliente-Clover Mountain, Nevada (M_L 6.0), and the 1992 Little Skull Mountain, Nevada (M_L 5.6) near the proposed site.

7.3.4.4 Seismic Distribution

Studies of the large Great Basin earthquakes suggest faulting on steeply dipping fault planes that penetrate the upper 15 km of crust as the **focal mechanism**; the average dip for large normal faulting Great Basin earthquakes is 44°. In general, mainshock hypocenters for $M > 7$ earthquakes in this region can be located on the down-dip projection of the surface rupture, suggesting that large Great Basin events occur on steeply dipping planar faults at depths less than about 15 km.

Three seismic gaps have been identified in the western Great Basin, with perhaps two more proposed. These gaps occur between the rupture zones of major historic earthquakes and contain structures that show evidence of prehistoric activity. Seismic gaps are generally considered to be significant in plate-boundary regions but their relevance for interplate regions such as the Great Basin is not clear.

7.3.4.5 Significant Historical Earthquakes

Figure 7-24 depicts the epicenters for earthquakes of magnitude 5 and greater (or intensity VI and greater) occurring within 320 km of the proposed site from 1850 through 1992. This shows a clustering of seismicity in the CNSB and the SNGBZ, as well as in the southern Mojave Desert and along the San Andreas fault zone. In addition to those depicted, numerous small magnitude earthquakes have occurred in clusters or as isolated events throughout much of Nevada. The Garlock fault and much of the southern Great Basin appear to show relatively little seismic activity during this period.

Earthquakes occurring since 1850 and believed to be significant in evaluating the repository are summarized in Table 7-7. These either resulted in surface rupturing or represent the largest event in a particular seismic-source zone.

7.3.5 Volcanism (Adapted from DOE95a)

The upper few kilometers of rocks at Yucca Mountain are predominantly Miocene silicic volcanics derived from nearby nested caldera complexes and centers active 7 and 16 Mega-annum (Ma), emplaced during eruptive cycles of the Timber Mountain caldera complex. Yucca Mountain, to the depth of the proposed repository, is comprised of units of the Paintbrush Tuff, a major outflow ignimbrite of the Claim Canyon caldera segment of the Timber Mountain caldera complex. During late Neogene (10 to 2 Ma) and Quaternary (2 to 0 Ma), small-volume, mostly polygenetic, basaltic centers produced lava flows, air falls, and cinder cones.

7.3.5.1 Silicic Volcanism

The silicic volcanism in the Yucca Mountain area is part of an extensive, time transgressive pulse of mid-Cenozoic volcanism that occurred throughout much of the southwestern United States. Yucca Mountain is in the south-central part of the SNVF, a major Cenozoic volcanic

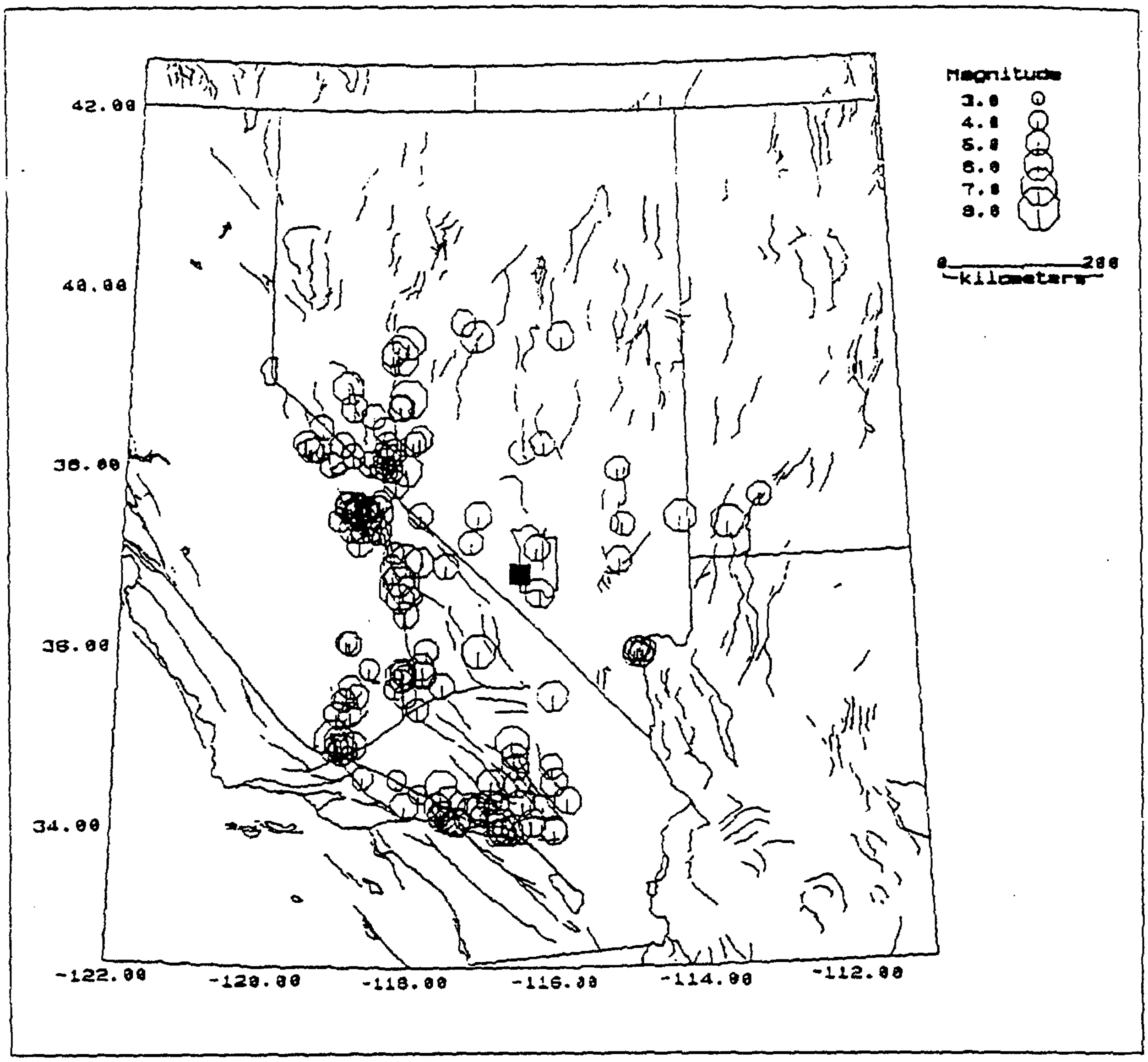

Figure 7-24. Magnitude 5 or Greater Earthquakes within 200 Miles of Yucca Mountain

field covering an area of over 11,000 km^2. Briefly, magmatism in the Cordillera was distributed in linear belts parallel to the continental margin during the Mesozoic. In the southwestern United States, a pause or disruption in the belts at about 80 Ma formed the Laramide magmatic gap or hiatus, which lasted until renewed silicic magmatism in the northeastern part of the Great Basin at about 50 Ma. Sites of eruptive activity migrated south and southwest across parts of Nevada and Utah, progressively in time and space, with eruptive centers distributed along arcuate east-west trending volcanic fronts. The most intensive

Table 7-7. Significant Earthquakes within 320 Km of Yucca Mountain Site Since 1850

Owens Valley, CA, 1872	March 26, 1872; estimated at M_w^{*1} 7.8 to M_s 8.0; considered largest historic event of the Basin and Range; surface ruptures along 90 to 110 km on Owens Valley fault; average net oblique slip of 6.1 ±2.1 m and up to 4 m vertical displacement; liquefaction of unconsolidated sediments.
Wonder, NV, 1903	Fall 1903; estimated magnitude 6.5; rupture of the Gold King fault; ruptures of 5 to 16 km with fissures up to 1.5 m wide and 1.5 m deep in alluvium; in the same area as the 1954 Fairview Peak-Dixie Valley earthquakes.
Cedar Mountain, NV, 1932	December 21, 1932; M_s 7.2; about 61 km of discontinuous faulting in a belt 6 to 14 km wide; displacements up to 1.8 m horizontal and 0.5 m vertical; analysis indicated main shock was two sources occurring about 20 seconds apart; a M_w 6.7 event and the second a M_w 6.6 event; followed by series of seven moderate events in this part of the CNSB from 1932 to 1939.
Excelsior Mountains, NV, 1934	January 30, 1934; M_L 6.3 (MW 6.1); on Excelsior-Coaldale section of the Walker Lane belt; about 60 km west-southwest of the 1932 event; foreshock of M_L 5.6 preceded mainshock by 45 min.; surface rupture 1.4 km in length and less than 13 cm vertical displacement. An M_L 5.5 earthquake occurred on August 9, 1943, approximately 40 km southeast.
Rainbow Mtn.-Stillwater, NV, 1954	July 6, 1954; two events of M 6.6 and M 6.4 in Rainbow Mountain area were followed on August 24 by the Stillwater M 6.8 event initiating a 6-year period of 10 events greater than M 5.5 in the CNSB.
Fairview Peak-Dixie Valley, NV, 1954	December 16, 1954; an M_L 7.3 event on the Fairview fault followed four minutes later by an M_L 6.9 event rupturing the Dixie Valley fault; diffuse fracture zone covering an area 100 km by 30 km from Mount Anna to the northern part of Dixie Valley; displacements 4 m right lateral and 3 m vertical on Fairview Peak fault and over 2 m vertical in Dixie Valley.
Caliente-Clover Valley, NV, 1966	On August 16, 1966; M_L 6.0; near Caliente, Nevada, about 210 km east-northeast of Yucca Mountain. The source depth is estimated at 6 km; with the focal mechanism a strike-slip motion on steeply dipping plates oriented either north-northeast or west-northwest.
Mammoth Lakes, CA, 1978-1988	An M_l 5.8 earthquake midway between Bishop and Mammoth Lake in October, 1978, was followed 18 months later (May, 1980) by a swarm-like sequence of 4 events (M_L 6.5, M_L 6.0, M_L 6.7, M_L 6.3) within two days. This sequence was accompanied by inflation of the resurgent dome in the Long Valley caldera. Activity continued with moderate earthquake swarms in the southern part of the caldera with spasmodic tremor sequences usually associated with magma injection at depth. The Chalfant sequence, discussed below, occurred to the east in 1986.
Chalfant Valley, CA, 1986	On July 21, 1986, an M_L 6.6 earthquake occurred in the Chalfant Valley in eastern California about 15 km north of Bishop with about 10 km of rupture along the White Mountains fault zone. The source-depth was located 11 km below the surface and the focal mechanism indicates right lateral slip on a plane oriented north-northwest dipping 70° southwest.

Table 7-7 (Continued)

Landers, CA, 1992	The Landers sequence began April 23rd with the M_L 6.2 Joshua Tree earthquake, followed by a sequence of 6000 events. On June 28, 1992, an M_s 7.6 earthquake near Landers, California, ruptured sections of several mapped north- to northwest-trending faults and several concealed unmapped north-trending faults in the south-central portion of the Mojave block. An extensive aftershock sequence followed, extending 85 km north of the mainshock and 40 km to the south. The sequence included the M_s 6.7 Big Bear earthquake 3 hours after and 30 km west of the mainshock. Surface rupture extended for 85 km, with displacement averaging 2 to 3 m across the rupture zone, up to 6.7 m on the Emerson fault, and minor rupture of faults within 30 km of either side of the main rupture zone. The Lander event was followed by a sudden increase in seismic activity in the western U.S. up to 1250 km from the mainshock, with an intense cluster of events in the Walker Lane belt. This included the M_L 5.6 Little Skull Mountain earthquake on June 29, 1992.
Eureka Valley, CA, 1993	On May 17, 1993, an M_L 6.1 earthquake occurred 30 km southeast of Bishop, California. The hypocenter was located 9 km below the surface in the southern part of Eureka Valley. Preliminary analysis indicates normal faulting on a northeast striking plane, perhaps paralleling a north-northwest trending inferred Quaternary fault in the area. (Investigations are continuing)

[*1] Terms used for earthquake magnitude in the table above include:

M_L	Local magnitude; this is the original Richter scale, developed in California for earthquakes with epicentral distances less than 600 km and focal depths less than 15 km; uses waves with periods of about 1 s; saturates at $M = 7.25$;
M_s	Surface-wave magnitude; suitable for global distance; uses waves of 20 s; saturates at about $M = 8.6$;
M_w	Moment magnitude; based on seismic moment ($M_0 = \mu AD$), where μ = shear modulus, A = area of fault rupture, and D = fault displacement; $M_w = 2/3 \log M_0\text{-}10.7$; does not saturate;
M	This is assumed to be local magnitude.

eruptions were at the leading edge of the migrating front, with the most voluminous silicic volcanic activity in the Yucca Mountain area occurring between 15 to 11 Ma. The Yucca Mountain area marks the southern limit of time-transgressive volcanic activity.

At about 13 to 10 Ma, there were two significant changes in the regional volcanic and tectonic patterns: the southern migration of volcanism halted and the composition of the volcanic activity changed. Diminished silicic-eruptive activity migrated in less systematic patterns to the southwest and southeast, leaving a conspicuous amagmatic gap from the southern edge of the NTS south to the latitude of Las Vegas. The timing of the transition from predominantly silicic volcanism to bimodal basalt-rhyolite volcanism is uncertain.

7.3.5.2 Basaltic Volcanism

Two episodes of basaltic-volcanic rocks have been defined in the Yucca Mountain area. One, basalt of the silicic episode (BSE), consists of bimodal basalt-rhyolite volcanism postdating most silicic eruptions of the Timber Mountain-Oasis Valley (TM-OV) complex. The second episode is comprised of spatially scattered, small-volume centers marked by scoria cones and lava flows of alkali basalt ranging in age from 8 Ma to Quaternary. These postcaldera basalts of the Yucca Mountain Region are divided into older postcaldera basalts (OPB) and younger postcaldera basalts (YPB).

BSE crops out throughout the Yucca Mountain area and is identified by several characteristics: it is closely associated (in time and space) with activity of the TM-OV complex; the largest volume centers of the BSE are in the ring-fracture zone of the Timber Mountain caldera; all centers of the BSE are large-volume eruptive units ($< 3\text{km}^3$ dense-rock equivalent); and it exhibits a wide range of geochemical composition. BSE occurs in three major groups:

- The **Mafic Lavas of Dome Mountain** (age 10.3 ±0.3 Ma) are exposed in the moat zone of the Timber Mountain caldera and comprise the largest volume of basaltic rocks;
- The **Basaltic Rocks of the Black Mountain Caldera** overlap the peralkaline units of the caldera in age;
- The **Basaltic Volcanic Rocks, Yucca Mountain Area** include the basaltic andesite of Skull Mountain (dated 10.2 ±0.5 Ma), the basalt of Kiwi Mesa, and the basalt of Jackass Flats.

The second episode of basaltic volcanism, marked by the Postcaldera Basalt of the Yucca Mountain Region, occurred at sites either well removed from eruptive centers of the TM-OV complex or are younger than the silicic-magmatic activity. These sites usually consist of small volume ($< 1\ \text{km}^3$) centers marked by clusters of scoria cones and lava flows and are divided into the OPB and YPB groups based on ages and geographic distribution.

The OPB were produced along either north-northwest trending Basin and Range Faults or at the intersection of Basin and Range faults with the ring-fracture zone of older calderas. These range in age from 9 to 6.3 Ma and are represented at four localities:

- At **Rocket Wash** thin, basalt lava flows (8.0 ±0.2 Ma) occur at the edge of ring-fracture zone of the Timber Mountain caldera;

- At **Pahute Mesa** three separate but related basalts (8.8 ±0.1 and 10.4 ±0.4 Ma), occur at intersection of faults with the ring-fracture zone of Silent Canyon caldera;

- At **Paiute Ridge** dissected scoria cones and lava flows (8.5 ±0.3 Ma) are associated with intrusive bodies occurring at the interior of northwest-trending graben; the related Scarp Canyon basalt (8.7 ±0.3 Ma) crops out west of Nye Canyon;

- At **Nye Canyon** there are three surface basalts (6.3 ±0.2 Ma, 6.8 ±0.2 Ma, and 7.2 ±0.2 Ma) and a buried basalt (8.6 Ma).

The second eruptive cycle, resulting in the YPB, usually occurred at clusters of probably coeval, small-volume centers aligned along predominantly northeast structural trends. These eruptions occurred from 4.8 to as recently as 0.350 Ma and are represented at the following localities (in decreasing age):

- At **Thirsty Mesa** a thick accumulation of fluidal lava and local feeder vents erupted onto a pre-existing Thirsty Canyon Group ignimbrite (welded tuff) plateau (ages of 4.6, 4.68 ±0.3, and 4.88 ±0.4 Ma reported);

- At **Amargosa Valley** cuttings from a buried basalt gave ages of 3.85 ±0.05 and 4.4 ±0.07 Ma;

- The **Southeast Crater Flat** basalt lavas (4.27 to 3.64 Ma) are the most areal-extensive of the YPB;

- The **Buckboard Mesa** basaltic andesite (3.07 to 2.79 Ma) erupted from a scoria cone in the northeast part of the ring-fracture zone of the Timber Mountain caldera and from nearby fissures;

- The **Quaternary Basalt of Crater Flat** includes a series of five centers in a north-east trending, slightly arcuate cluster along the axis of Crater Flat: <u>Little Cones</u> (1.1, 1.02, and 0.76 Ma), <u>Red Cone and Black Cone</u> (1.5 to 0.84 Ma and 1.09 to 0.80 Ma, respectively), <u>Makani Cone</u> (1.14 to 1.04 Ma), <u>Sleeping Butte Centers</u> (0.385 to 0.24 Ma), and <u>Lathrop Wells Centers</u>.

7.3.6 Unsaturated Zone Hydrology

The region beneath Yucca Mountain in the vicinity of the proposed repository is characterized by a very thick unsaturated zone, ranging in thickness from about 500 to 750 m. The variable thickness is produced by the combined effects of rugged topography and a sloping water table. The presence of a thick unsaturated zone is desirable for siting an underground waste repository, because ground water and any contaminants it might carry generally travels more slowly through the unsaturated zone than through the saturated zone. The thicker the unsaturated zone, the longer contaminants will take to reach the water table. This relatively long travel time through the unsaturated zone takes advantage of the limited life span of radionuclides, which may undergo decay before reaching the water table. Additionally, rock properties in the unsaturated zone may act as a natural barrier to radionuclide migration. A sequence of nonwelded porous tuffs that overlies the Topopah Spring Member probably act as a natural capillary barrier to retard the entrance of water into the fractured tuffs (Section 7.3.2). A similar sequence of nonwelded tuffs underlies the Topopah Spring Member. These underlying nonwelded tuffs locally contain sorptive zeolites and clays that could be an additional barrier to the downward transport of radionuclides from a repository to the water table.

7.3.6.1 Unsaturated Zone Hydrogeologic Units

In detail, the layered volcanic rock sequence beneath Yucca Mountain is very complex. The various rock units can be separated into a small or large number of units depending upon the scale and aims of a particular study. For the purposes of this document, the unsaturated zone is considered to consist of six hydrogeologic units, based on their physical properties. This grouping is based primarily on USG84a. The description of the six units is also taken largely from USG84a; this publication will not be cited repeatedly in this section. The reader can assume that the descriptions of the six units are taken from this work, except where otherwise referenced. Additional data regarding matrix and fracture properties are presented in the hydrogeologic database developed in DOE95i.

The physical properties within each formation vary considerably, largely due to variation in the degree of welding of the tuffs. In most cases, physical property boundaries do not correspond to rock-stratigraphic boundaries. However, it is the physical properties that largely control water occurrence and flow, and the hydrogeologic subunits into which the volcanic sequence is separated are different than the lithological units outlined in Section 7.3.2. The hydrogeologic units are, in descending order, Quaternary Alluvium (Qal), the Tiva Canyon welded unit (TCw), the Paintbrush nonwelded unit (PTn), the Topopah Spring welded unit (TSw), the Calico Hills nonwelded unit (CHn), and the Crater Flat unit (CFu). See USG84a, pages 10 to 13 for illustrations of the geometry of these hydrogeologic units.

Structural features, although they are not hydrogeologic units in the same sense as stratigraphic units, are mappable, have certain measurable hydraulic characteristics, and may have a significant effect on unsaturated zone flow. Because these structural features are regarded as important components of the unsaturated hydrologic system, they are briefly described later in this section.

Qal. Unconsolidated alluvium underlies the washes that dissect Yucca Mountain and forms the surficial deposit in broad inter-ridge areas and flats nearby. Thickness, lithology, sorting and permeability of the alluvium are quite variable; particles range in size from clay to boulders, and in places the unit is moderately indurated by caliche. Alluvial and colluvial deposits generally have small effective hydraulic conductivity, large specific retention, and large effective porosity as compared to the fractured rocks. Therefore, a large proportion of the water infiltrated into the alluvial and colluvial material is stored in the first few meters of the soils and is lost to evaporation during dry periods. However, the permeability of alluvium generally is substantial compared to the tuff units.

TCw. Lying immediately beneath the Qal is the Tiva Canyon welded unit, consisting of devitrified ash-flow tuffs ranging from 0 to 150 meters in thickness across the site. The TCw is the densely to moderately welded part of the Tiva Canyon Member of the Paintbrush Tuff. This unit is the uppermost stratigraphic layer that underlies much of Yucca Mountain; it dips 5° to 10° eastward within the central block, resulting in a relatively planar eastward-sloping, dissected land surface. The unit is absent in some washes and is about 150 m thick beneath Yucca Crest. This unit has a fracture density of 10 to 20 fractures/m^3 and small matrix permeability. Saturated matrix hydraulic conductivity has been estimated at about 10^{-11} m/s (DOE95j); though effective hydraulic conductivity is considerably less, as saturation is estimated to average about 60 percent.

Ptn. The Paintbrush nonwelded unit is situated below the Tcw unit and consists of the nonwelded and partially welded base of the Tiva Canyon Member, the Yucca Mountain Member, the Pah Canyon Member, the nonwelded and partially welded upper part of the Topopah Spring Member, and associated bedded tuffs. All are part of the Paintbrush Tuff. Within the central block, the unit consists of thin, nonwelded ash-flow sheets and bedded tuffs that thin to the southeast from a maximum thickness of 100 m to a minimum thickness of about 20 m. The unit dips to the east at 5° to 25°; the dip at any location depends on the tilt of the faulted block at that site. In the central block, the dip rarely exceeds 10°. In the vicinity of the central block, this unit crops out only in a narrow band along the steep west-facing scarp along Solitario Canyon.

Tuffs of this unit are vitric, nonwelded, very porous, slightly indurated, and in part, bedded. The unit has a fracture density of 1 fracture/m^3. Saturated hydraulic conductivities of five core samples of the matrix have a geometric mean of about 9.0×10^{-3} m/d. Porosities average about 46 percent, but some porosities are as much as 60 percent. The rocks of this unit are moderately saturated, with an average value of about 61 percent. However, water contents are relatively large; the mean volumetric water content is about 27 percent, and the mean water content by weight is about 19 percent. The maximum values reported are: saturation, 80 percent; volumetric water content, 42 percent; and water content by weight, 36 percent.

TSw. The Topopah Spring welded unit consists of a very thin upper vitrophyre, a thick central zone consisting of several densely welded devitrified ash-flow sheets, and a thin lower vitrophyre of the Topopah Spring Member of the Paintbrush Tuff. The unit, which varies from 290-360 meters in thickness, is densely to moderately welded and devitrified throughout its central part. The TSw contains several lithophysal cavity zones that generally are continuous but vary appreciably in thickness and stratigraphic position, and is intensely fractured.

The Topopah Spring Member is the thickest and most extensive ash-flow tuff of the Paintbrush Tuff. The central and lower densely welded, devitrified parts of the Topopah Spring welded unit are the candidate host rock for a repository. This part of the unit contains distinctive subunits that have abundant lithophysal gas cavities within the central block. The saturated hydraulic conductivity of the matrix of this unit generally is small and has a mean of about 3.0×10^{-6} m/d.

Because of the densely fractured nature of this unit, bulk hydraulic conductivity is substantially greater than matrix hydraulic conductivity. Saturated horizontal hydraulic conductivity of the rock mass is about 1.0 m/d for a 120-m interval of the TSw that was packed off and tested at well J-13, about 6 km east of Yucca Mountain. Because of the marked contrast between the matrix and the bulk hydraulic conductivities in this unit, values of the bulk hydraulic conductivity from well J-13 (USG83) and borehole UE-25a#4 probably represent the hydraulic conductivity of the fractures in this unit. The large bulk hydraulic conductivity of this unit probably promotes rapid drainage of water.

The effect of lithophysal cavities on the hydrologic properties of the TSw is not well understood. Total porosity is much greater where lithophysal cavities are more abundant than in those sections that are free of these cavities. Overall unsaturated hydraulic conductivity probably is decreased by the presence of these cavities. These cavities commonly are several centimeters in diameter, filled with air, and form capillary barriers with the fine grained matrix. In effect, the cavities decrease the transmissive cross-sectional area, decrease effective porosity, and consequently, decrease the effective hydraulic conductivity.

CHn. Beneath the TSw unit is a series of non- to partially-welded ash-flow tuffs called the Calico Hills nonwelded unit. Locally, these may be vitric (CHnv) or zeolitized (CHnz). The CHn includes the following components, in descending order:

1. A nonwelded to partially welded vitric layer, locally zeolitic, that is the lowermost part of the Topopah Spring Member of the Paintbrush Tuff.
2. Tuffaceous beds of Calico Hills.
3. The Prow Pass Member of the Crater Flat Tuff, which is nonwelded to partially welded where it occurs in the unsaturated zone beneath the central block.
4. The nonwelded to partially welded upper part of the Bullfrog Member of the Crater Flat Tuff, where it is above the water table.

The unsaturated thickness of the CHn is determined in part by the position of the water table. At some distance east of the central block, the thickness is zero, because the water table occurs within the overlying Topopah Spring welded unit. In the southeast, this situation occurs east of the Bow Ridge fault. Beneath the northern one-half of the central block, the water table is within the CHn and forms the base of the unit. Thickness in this area ranges from about 140 to 250 m. Beneath the southern one-half of the block, the water table is within the underlying Crater Flat unit, and the full thickness of the CHn is unsaturated. Thickness in this area ranges from about 200 to 250 m.

Both vitric and devitrified facies occur within the CHn. As described below, the permeability of the vitric facies is substantially greater than that of the devitrified facies. Alteration products in the devitrified facies include zeolites (most abundant), clay, and calcite (rare). Because this facies commonly is pervasively zeolitic, this facies of the unit is hereafter referred to as the zeolitic facies. Thickness of the zeolitic facies generally increases from the southwest to the northeast beneath Yucca Mountain. Beneath the northern and northeastern parts of the central block, the entire unit is devitrified and altered.

Both the vitric and zeolitic facies of the CHn are very porous, with a mean porosity of about 37 percent for the vitric facies, and 31 percent for the zeolitic facies. Saturations in this unit generally are greater than 85 percent, with a mean value for the zeolitic facies of about 91 percent.

A significant difference exists in values of vertical hydraulic conductivity of the matrix between the vitric and zeolitic facies of the CHn. The mean vertical hydraulic conductivity of the matrix of the vitric facies is $4.0\text{x}10^{-3}$ m/d. The geometric mean of the vertical hydraulic conductivity of the matrix of the zeolitic facies is about $8.0\text{x}10^{-6}$ m/d. The marked contrast in vertical hydraulic conductivities of the two facies probably is the result of extensive argillization in the zeolitic facies, which tends to decrease permeability.

CFu. In approximately the southern half of the central block, the lowermost unit in the unsaturated zone is the Crater Flat unit. This unit consists of the unsaturated welded and underlying nonwelded parts of the Bullfrog Member of the Crater Flat Tuff. No differentiation is made between the welded and nonwelded components of the Crater Flat unit because of the limited extent of the unit in the unsaturated zone beneath the central block, and therefore, its probable limited effect on the unsaturated flow system. Beneath the central block, thickness of the CFu ranges from 0 to 160 m. Little is known about the unsaturated hydrologic properties of the unit, but it is assumed that the properties are similar to those of the nonwelded and welded counterparts higher in the section.

Structural Features

The central block of Yucca Mountain is bounded on its west side by a major north-striking normal fault with greater than 100 m of offset. West of this fault is a chaotic, brecciated and faulted west dipping zone caused by drag on the fault. A zone of imbricate normal faults forms the eastern boundary of the central block. These faults are west dipping and have vertical offsets of about 2 to 5 m. Northwest striking strike-slip faults also occur in the area, such as the one forming the northern boundary of the central block, beneath Drill Hole Wash.

Because these major faults and fault zones transect the full thickness of the unsaturated zone, they may by hydrologically significant either as flow barriers or as flow pathways. The unsaturated hydraulic properties of these features have not been measured, but some inferences can be made, based on the physical properties of the welded and nonwelded tuff units, and based on observations of cores. The welded units are relatively brittle. Open faults have been observed in cores even from below the water table.

Conversely, the nonwelded units generally are more ductile than the welded units and more readily produce a sealing gouge material. Fault zones are less common in the Calico Hills nonwelded unit. In general, hydraulic conductivity varies greatly along the faults and is greater in welded units than in nonwelded units.

7.3.6.2 Unsaturated Zone Potentials

In the unsaturated zone, water is present both in liquid and vapor phases within the interstitial, fracture, and lithophysal openings. Water flow and storage is complexly three dimensional, and is controlled by the structural, stratigraphic, and climatological setting. Hydrologic evaluation of the proposed repository site constitutes a problem of two-phase, multicomponent, coupled heat (geothermal) and water flow within a layered sequence of tilted, faulted, and fractured, variably saturated, tuffaceous geohydrologic units.

In order to understand the movement and occurrence of water under such complex conditions, a detailed knowledge of the unsaturated zone potentials is imperative. Data from boreholes USW UZ-1, -4, -5, -6, -6s, -7, and -13 have provided much useful information; however, detailed in situ measurements of matric potential have been taken only at the 33 instrumentation stations vertically arrayed within borehole USW UZ-1. This borehole was drilled to a total depth of 387 meters, at which point drilling was discontinued because an apparent perched-water zone was encountered. Presuming conditions within the borehole to have equilibrated with those in the host rock, the data indicate that matric potentials within the PTn unit range from -10 to -1 bars with a mean value of about -5 bars over the thickness of the unit. The matric potentials within the TSw unit range from -10 to -1 bars with an approximately constant mean value of about -3 bars over the thickness of the unit. The Mined Geological Disposal System, 1995 License Application Annotated Outline (DOE95a) states that some researchers have concluded that downward moisture flux within the TSw unit occurs predominantly as liquid-water flow within the fractures. Assuming that the nearly constant matric potential over the depth interval from 122 to 224 meters in borehole USW UZ-1 implies a unit vertical hydraulic gradient, Montazer et al. (MON86) estimates that the downward liquid-water flux within the TSw unit borehole USW UZ-1 probably is within the range of 0.1 to 0.5 mm/yr.

Pneumatic potentials (pore-gas pressures) are being monitored by downhole pressure transducers emplaced in borehole USW UZ-1 and have been measured within a piezometer nest installed in borehole UE-25a#4, about 1.6 km southeast of borehole USW UZ-1. Diurnal and barometrically induced fluctuations of gas pressure of about 0.25 kPa have been observed in these boreholes down to depths of about 30 meters. Such fluctuations are apparently damped out below this depth, although seasonally induced pressure variations are observed to occur at greater depths.

7.3.6.3 Ground Water Flow In The Unsaturated Zone

The dynamics of water flow in the layered, fractured rock unsaturated zone beneath Yucca Mountain are complex and poorly understood. Hydrogeologic features that probably affect flow significantly in the unsaturated zone include the presence of fractured porous media, layered units with contrasting properties, dipping units, bounding major faults, and a deep water table. These features probably result in the occurrence of phenomena such as flow in both fractures and matrix, retardation of flow by capillary barriers, lateral flow, perched ground water zones, and vapor movement.

The first detailed conceptual model of unsaturated zone flow at Yucca Mountain was proposed in USG84a. Most subsequent conceptualizations of unsaturated zone behavior are largely refinements of this model, revised to accommodate newly acquired data. The following presentation of the unsaturated zone flow conceptual model is taken primarily from USG84a. For readability, repeated references to this work are not made. Where appropriate, the published literature is referenced where refinements or revisions have been made to the USG84a model. The following conceptual model is presented as if it were established physical reality. It is important to bear in mind, however, that the proposed model is probably not the only reasonable description that could be made of the system. The end of this section provides a brief summary of critical unknowns, their effects on unsaturated zone flow, and brief discussion of alternative conceptual models.

Infiltration Rates

The ultimate source of water in the unsaturated zone at Yucca Mountain is precipitation on the mountain. The spatial and temporal relationships between infiltration and recharge are complex, because of the hydrogeologic variability of Yucca Mountain. Some water that infiltrates returns to the surface by interflow; another part is returned to the atmosphere by evapotranspiration. A small quantity that is not evaporated or discharged as interflow percolates deep into the unsaturated zone and becomes net infiltration. The quantity of net infiltration that percolates through different paths is quite variable, so the average recharge rate does not represent percolation rates through specific flow paths. At Yucca Mountain, the infiltration rate is both spatially and temporally variable. Spatial variations of infiltration depend mostly on the variations in properties of the surficial units and topography. Temporal variations in infiltration rate are related to the seasonality and relatively infrequent precipitation events in the arid climate of Yucca Mountain.

The actual quantity of net infiltration beneath Yucca Mountain has not been accurately determined. USG84a reports that net infiltration flux probably ranges from 0.5 to 4.5 mm/year. Flint and Flint (FLI94) provide preliminary estimates of spatial infiltration rates that range from 0.02 mm/yr, where the welded Tiva Canyon (Tcw) unit outcrops, to 13.4 mm/yr in areas where the Paintbrush nonwelded (PTn) unit outcrops. The bulk of the area above the repository block is underlain principally by the Tiva Canyon member. The 1995 Total System Performance Assessment (DOE95c) concludes that, if the predominant flow direction is vertical, then the average infiltration through the repository block, using the average infiltration rates of Flint and Flint (FLI94), would be 0.02 mm/yr. If, on the other hand, the predominant flow direction has a significant lateral component due to material property heterogeneity and/or anisotropy and the sloping nature of the hydrostratigraphic unit contacts, then the average net infiltration rate over the repository block could be as high as some weighted average of the infiltration rates inferred from Flint and Flint (FLI94). The 1995 TSPA (DOE95c) also reports that the average spatially integrated infiltration rate is about 1.2 mm/yr; most of this infiltration occurs along the Paintbrush outcrop in the washes north of the repository block.

Knowing the percolation rate is crucial to modeling efforts because of the importance of the relationship of infiltration rate to horizontal and vertical permeabilities of the various units and the effect this has on whether or not significant lateral flow occurs in the unsaturated zone. The higher the actual infiltration rate, the greater the likelihood of significant lateral flow. Lateral flow is important because it could transmit water to structural features which would then move the water downward, possibly acting as a conduit to divert large amounts of water flowing downward through a small area, and possibly into the repository horizon.

Conceptual Model(s)

Percolation of infiltrated water through the exposed fractures of the Tiva Canyon welded unit is relatively rapid, because of the large fracture permeability and small effective porosity of this unit compared to the alluvial material. Therefore, a large proportion of the infiltrated water normally is percolated sufficiently deep within the fractured tuff to be unaffected by the evaporation potential that exists near the surface. Depending on the intensity of the infiltration, percolation downward through the Tiva Canyon welded unit may occur without a significant change in rate. A small proportion of the water percolating through the fractures slowly diffuses into the matrix of the Tiva Canyon welded unit. Downward flow in the matrix is very slow, because of the small effective hydraulic conductivity of the matrix. During dry periods, some of the diffused water flows back into the fractures and probably reaches the land surface by vapor diffusion. The mass of water involved during this process probably is negligible compared to the percolating water.

The densely fractured Tiva Canyon unit, with small matrix porosity and permeability, overlies the very porous, sparsely fractured Paintbrush unit. A marked contrast in material properties exists at the contact between these two units; depending on the magnitude of the infiltration flux, this contrast could impart a significant lateral component of flow. Flow of water through fractures of the Tiva Canyon unit occurs rapidly until it reaches the contact. At this point, the velocity is significantly decreased because of the greater effective porosity and lesser hydraulic conductivity of the Paintbrush unit. As a result, lateral unsaturated flow of water above this contact can occur. Perching of this water may occur above this unit if displacement along faults has created significant permeability contrasts on opposite sides of the fault.

The saturated hydraulic conductivity of the Paintbrush nonwelded unit in the direction of dip is 10 to 100 times greater than saturated hydraulic conductivity in the direction normal to the bedding plane. The combination of dipping beds and differences in directional permeability creates a downdip component of flow. The magnitude of this component depends on the magnitude of the principal hydraulic conductivity ratio. The permeability contrast may be sufficient to decrease vertical percolation into the underlying Topopah Spring welded unit to almost zero. In this case, water would flow laterally downdip until structural features are encountered that create perching conditions or provide pathways for vertical flow.

Some water flows from the matrix of the Paintbrush nonwelded unit into the fractures or matrix of the underlying Topopah Spring welded unit. Owing to the thickness of this unit, water moving through the fractures eventually diffuses into the matrix and moves very slowly downward.

Flow enters the Calico Hills nonwelded unit either from the matrix of the Topopah Spring welded unit or through structural flowpaths. Because little if any flow occurs in the fractures of the lower part of Topopah Spring unit, these fractures do not contribute to flow into the Calico Hills unit.

The nature of flow at the contact between the Topopah Spring welded unit and Calico Hills nonwelded unit depends on whether the vitric or zeolitic facies of the Calico Hills unit is present. The permeability and effective porosity of the vitric facies are much greater than those of the matrix of the Topopah Spring unit, resulting in a capillary barrier where those units are in contact. Conversely, the permeability of the zeolitic facies is about the same as for the matrix of the Topopah Spring unit, resulting in continuity of matrix flux across the contact.

Flux within the Calico Hills unit may occur with some lateral component of downdip flux, because of the existence of layers with contrasting hydraulic conductivity in the unit. A large scale anisotropy probably is caused by intercalation of tuffs with large and small permeability and by compaction.

Water that flows downdip along the top of the Calico Hills unit slowly percolates into this unit and slowly diffuses downward. Fracture flow is known to occur near the uppermost layers of the Calico Hills unit, but diffusion into the matrix probably removes the water from the fractures deeper in the unit and flow becomes limited mostly to within the matrix, except along the structural flowpaths.

Beneath the southern part of the block, the Crater Flat unit occurs between the Calico Hills unit and the water table. Included are the welded part and underlying nonwelded part of the Bullfrog Member of the Crater Flat Tuff. Flow patterns and processes that occur higher in the section are repeated in this unit; therefore, it is an extra buffer to water before it enters the saturated zone.

Fluxes along many structural flowpaths are probably larger than within the units they intersect. The Calico Hills unit is more ductile than the overlying Topopah Spring unit, which may give the Calico Hills unit fracture sealing properties. In addition, because of the lesser shear strength of this unit compared to that of the Topopah Spring, gouge formation along faults and shear zones is more common. These properties probably result in a smaller fracture conductivity in the Calico Hills unit. In the case where the structural flowpaths are hydraulically continuous across the upper contact of the Calico Hills unit, water flows downward without a significant change in its path until it reaches the water table. In the more probable case where the structural flowpaths become hydraulically discontinuous, water either may become perched at the upper contact of the Calico Hills unit or could begin to flow downdip along this boundary, or both. Intermediate conditions between the two extreme cases are also possible.

7.3.6.4 Discussion of Unsaturated Zone Conceptual Flow Model

The net infiltration rate through the unsaturated zone beneath Yucca Mountain is one of the most critical parameters for determining the nature of flow in the unsaturated zone, yet it is one of the least well constrained. Numerous modeling studies, based on varying conceptual models, have been performed to simulate unsaturated flow beneath Yucca Mountain (e.g., DOE94a, DOE95c). Sensitivity analyses performed in these studies indicate that uncertainty in the amount of net infiltration accounts for as much as 90 percent of the variability in the results.

For example, in the 1995 TSPA, ground water travel times from the repository to the saturated zone were found to range from a low of 5.83×10^4 yr, at an infiltration rate of 1.0 mm/yr, to a high of 5.67×10^5 yr at an infiltration rate of 0.1 mm/yr. These ground water travel times can be compared to values presented by the National Academy of Sciences in a 1983 report issued by the Waste Isolation Systems Panel (WISP) (NAS83). In this report the Academy calculated ground water travel times through the unsaturated zone at 2.1×10^4 years. This value, however, assumes an infiltration rate of 3 mm/yr and that flow takes place through the matrix without any interaction with existing fractures. The WISP report suggested that if flow proceeds *via* the fractures, rather than the matrix, these travel times may be several orders of magnitude too long.

The NRC has also performed travel time analyses, which are presented in NUREG-1464 (NRC95b). In these probabilistic analyses travel times are calculated for four distinct cases:

- *Fastest* - the minimum time for the transport of a non-diffusion particle along the fastest combination of possible matrix and fracture pathways.
- *Most Flux* - the travel time through the pathway associated with the greatest flux from the repository to the accessible environment.
- *Flux Averaged* - the average travel time for all paths, weighted by the flux in each path.
- *Averaged* - the average travel time for all paths, irrespective of the flux in each path.

The results for the fastest travel time and most-flux travel time are similar in that travel times controlled by fracture flow are clustered at 1000 years, whereas travel times controlled by matrix flow lie between 10^4 and 10^5 years. Flux-averaged travel times range between 10^4 and 10^5 years, as compared with averaged travel times that fall between 10^5 and 10^6 years.

The magnitude of infiltration flux has a significant bearing on the potential for lateral unsaturated flow beneath Yucca Mountain. In the Paintbrush nonwelded unit, the overall hydraulic conductivity parallel to bedding is 10 to 100 times greater than that in the direction normal to the bedding plane. At higher flux rates, the potential vertical volumetric flow rate of some units is exceeded, thereby inducing a significant lateral component of flow to the infiltration flux. As previously described, lateral flow could direct water to structural flow paths, which may then redirect the flow vertically downward, providing a "fast path" and potentially reducing travel times to the saturated zone.

Also at issue in mathematical modeling of the complex unsaturated flow system beneath Yucca Mountain is the nature of matrix-fracture interactions and their effect on unsaturated flow. Fracture-matrix interactions can be modeled either as equilibrium or non-equilibrium conditions. For instance, LBL95 describes the development of the three-dimensional LBL-USGS site scale model for Yucca Mountain.

The equivalent continuum model (ECM) assumes the existence of an enormous disparity in capillary suction between the matrix and fractures. Thus, during drainage, matrix desaturation does not begin until the fractures are almost completely drained, whereas during imbibition, the fractures remain dry until the matrix is almost completely saturated. In the equilibrium model, fracture flow occurs only when the matrix has reached 100 percent saturation. There is growing evidence to suggest that the equilibrium model is not realistic and that episodic water flow at Yucca Mountain may take place along "fast paths" (LBL95). Such a situation, which may result from non-equilibrium fracture flow, cannot be represented by the ECM. By forcing the fractures to remain dry until the matrix is fully saturated, the ECM formulation artificially inhibits the episodic and rapid movement of water along these fast paths. Some researchers (DOE95k) have proposed an empirical modification to the ECM fracture flow initiation rule to approximate non-equilibrium fracture-matrix flow. This modification involves relaxing the 100 percent matrix saturation requirement to some smaller value, perhaps 90 to 95 percent. In true non-equilibrium flow, matrix saturations would approach 100 percent very near to the fracture, with the rest of the matrix remaining relatively drier (LEH92).

Some authors have examined the possibility of "focused recharge," where surface rainfall runoff is directed to areas where faults intersect the surface. Significant amounts of recharge may infiltrate into these areally restricted zones, which may induce lateral unsaturated flow in the underlying units (LEH92). One obvious area where this may be occurring is the northern extension of Solitario Canyon fault, which bounds Yucca Mountain on the west.

7.3.6.5 Unsaturated Zone Radionuclide Transport

The travel time of radionuclides beneath Yucca Mountain is a function of both physical and chemical processes and interactions between fluid and rock.

Travel time to the accessible environment is a function of the percolation flux distribution in the unsaturated zone and the advective flux distribution in the saturated zone, as well as the conceptual representation of hydrostratigraphy along the likely ground water flow paths between the repository and the accessible environment. The percolation flux distribution within the Topopah Spring hydrostratigraphic unit (and other unsaturated zone units below it) is a function of the infiltration rate and the conceptual model for ground water flow in the unsaturated zone. In particular, the key conceptual uncertainty in the transport of radionuclides through the geosphere at Yucca Mountain is the possible presence of fracture flow and transport which might, if fracture pathways existed and were continuous and interconnected, lead to the formation of so-called "fast paths."

In TSPA-1995, modeling efforts have simulated fluid/rock interactions that can serve to chemically retard the transport of radionuclides with a simple, equilibrium (infinite capacity), distribution coefficient (K_d) model. Generally, values for distribution coefficients are related to both the chemical nature of the individual hydrostratigraphic unit and to the properties of the radionuclide. Chemical retardation processes described by the K_d include ion-exchange, sorption, surface complexation, and precipitation/dissolution. Since distribution coefficients are used to model such a wide variety of phenomenological processes, in TSPA-95 they are modeled as stochastic parameters with a high degree of uncertainty. This process results in a broad range of predicted times it would take radionuclides to travel from the repository to the water table. Radionuclides that are not affected by chemical retardation (e.g., I, Tc) could reach the water table within the same time frame as the ground water (5.83×10^4 to 5.67×10^5 years). Alternatively, K_ds used in TSPA-95 for a number of radionuclides (i.e., Am, Ra, Cs, Sr) result in travel times to the water table that are 50,000 times greater than those for the ground water.

7.3.7 Saturated Zone Hydrology

As previously described, Yucca Mountain is composed of a thick sequence of Tertiary volcanic rocks. Beneath Yucca Mountain the thickness of these rocks is more than 6,000 feet (SPE89). The Tertiary volcanic sequence is underlain by complexly folded and faulted Paleozoic sedimentary rocks, including thick sections of carbonate rocks (SPE89). In the immediate vicinity of Yucca Mountain, only one borehole has been drilled deep enough to penetrate the Paleozoic rocks underlying the Tertiary volcanic sequence. The Paleozoic rocks beneath the volcanic section are water-saturated and capable of transmitting ground water, probably over great distances. Because insufficient data are available concerning the hydrologic properties of the Paleozoic rocks beneath Yucca Mountain, the following discussion of saturated zone hydrology considers only the Tertiary volcanic rocks. For a more complete treatment of the characteristics of the carbonate aquifer as it occurs in other areas, see USG75.

The extent of hydraulic communication between the volcanic and underlying Paleozoic sequence is not well characterized. In the only well (UE-25p#1) at Yucca Mountain which penetrated into the Paleozoic sequence, an upward hydraulic gradient (from Paleozoic to the Tertiary) was measured. Additional evidence, including isotopic composition and temperatures of ground water beneath Yucca Mountain support the concept that ground water may be flowing from the Paleozoic aquifer into the volcanic aquifer (USG88a; STU91).

The stratigraphic sequence of volcanic rocks and related hydrostratigraphy is shown in Figure 7-25 (from USG94a).

The volcanic rock section beneath Yucca Mountain has been divided informally into four hydrogeologic units: the upper volcanic rock aquifer, the upper volcanic rock aquitard, the lower volcanic rock aquifer, and the lower volcanic rock aquitard. These hydrogeologic unit

	VOLCANIC STRATIGRAPHY	HYDROSTRATIGRAPHY
Paintbrush Tuff	Tiva Canyon Member Yucca Mountain Member Pah Canyon Member Topopah Spring Member	Upper Volcanic Aquifer
Tuffaceous beds of Calico Hills		Upper Volcanic Aquitard
Crater Flat Tuff	Prow Pass Member Bullfrog Member Tram Member	Lower Volcanic Aquifer
Lithic Ridge Tuff		Lower Volcanic Aquitard

Figure 7-25. Stratigraphic Sequence of Volcanic Rocks and Related Hydrostratigraphy

designations serve primarily to distinguish between zones in which lateral ground water flow can be expected to dominate over vertical flow ("aquifers") and zones in which vertical flow can be expected to dominate over lateral flow ("aquitards") (DOE95g; USG94a).

The physical properties within each formation vary considerably, which is largely due to variation in the degree of welding of the tuffs. The nonwelded tuffs are characterized by having a relatively large primary porosity, but low permeability. This low permeability results from small pore sizes and the presence in many nonwelded units of secondary alteration minerals (primarily zeolites and clays). The welded tuffs are typically very hard and densely welded. The welded tuffs are commonly more highly fractured than the nonwelded units. The fractures in the welded tuffs endow them with a significant bulk permeability. For this reason, many of the welded tuff units are capable of transmitting greater quantities of water than their nonwelded counterparts (USG84a).

The largely nonwelded and intensely altered lower volcanic section, the Lithic Ridge Tuff and older tuffs, is a confining unit. The variably welded Crater Flat Tuff constitutes an aquifer of moderate yield. The tuffaceous beds of Calico Hills are largely nonwelded and are zeolitized where saturated; however, this unit is significantly less altered than the lower volcanic section. Where saturated, it generally is a confining unit, but locally parts of the formation are permeable. The Topopah Spring Member of the Paintbrush Tuff is predominantly densely welded and has abundant lithophysal horizons. It contains the zones of greatest primary and secondary permeability and constitutes the most productive aquifer in the tuff section, where it is saturated (FRI94).

The occurrence of the water table is not restricted to any one hydrogeologic unit. Directly beneath Yucca Mountain, the water table occurs primarily within the Calico Hills Formation and toward the southern end of Yucca Mountain in the underlying Crater Flat Tuff. To the east of Yucca Mountain, in the vicinity of Fortymile Wash, the water table occurs in the Topopah Spring member of the Paintbrush Tuff. The occurrence of the water table in different hydrostratigraphic units is attributable to three factors: the vertical displacement of hydrostratigraphic units by the numerous faults that dissect the area, the eastward dip (5-10 degrees) of the volcanic units, and the variable elevation of the water table. See USG93 and USG84b for graphical depictions of the relationship of the water table to stratigraphic units and Fridrich et al. (FRI94) for a map of the geology at the water table.

7.3.7.1 Thickness

For the purposes of this report, the thickness of the saturated zone beneath Yucca Mountain is defined as the vertical distance from the water table to the bottom of the Tertiary volcanic section. The thickness of the saturated zone is variable because the surfaces formed by the water table and the bottom of the volcanic sequence are neither flat nor at a constant elevation above sea level. While the elevation of the water table has been relatively well characterized beneath Yucca Mountain, the elevation of the bottom of the volcanic sequence is known at only one location (well UE-25p#1). At this location, the water table is 2,468.6 feet above mean sea level (MSL), and the bottom of the volcanic sequence was encountered at 423.5 feet below mean sea level, giving a saturated thickness of the volcanic rocks of approximately 2,892 feet (USG84c).

The elevation of the water table beneath Yucca Mountain ranges from 3,375 feet MSL at the northern part of Yucca Mountain to 2,392 feet MSL at the southern end of Yucca Mountain, a difference of 983 feet (USG94a). Assuming that the bottom of the volcanic aquifer beneath Yucca Mountain is everywhere at 423 feet below sea level (which it is not), the thickness of the saturated volcanic sequence would range from about 2,900 to 3,900 feet.

7.3.7.2 Hydraulic Conductivity

Rock properties largely control the characteristics of water occurrence and flow in the saturated zone. Rock properties, in turn, are dependent on eruptive history, cooling history, post-depositional mineralogic changes, and structural setting. Permeability of ash-flow tuffs is in part a function of the degree of fracturing, and thus, the degree of welding. Densely welded tuffs fracture readily; airfall tuffs do not. Therefore, the distribution of permeability is affected by irregular distribution of different tuff lithologies and is a function of proximity to various eruptive centers. Permeability is also a function of proximity to faults and fracture zones (USG82a).

USG91a reports that, for well USW H-6, water production during pumping tests was coincident with fractured, partially, and partially to moderately welded tuff units. The reverse was not necessarily true; that is, not all fractured partially welded tuff units produced water. These authors also state that for this well "porosity and permeability of these rocks is generally inversely related. Porosity is greatest near the top and bottom of ash flow tuff units and is the least near the center. Permeability, as indicated by water production, is greatest near the center of units, where the degree of welding is greatest."

Welded units have matrix hydraulic conductivities with geometric means ranging from 6.5×10^{-6} to 9.8×10^{-6} feet per day, and bulk hydraulic conductivities of 0.32 to 33 feet per day. The nonwelded units have variable hydraulic conductivities, with geometric means ranging from 8.59×10^{-5} to 9.76×10^{-2} feet per day (USG84a).

Hydraulic conductivity of the Topopah Spring Member, as determined from aquifer testing of the 394-foot interval of well J-13, located about 5 miles east of the crest of Yucca Mountain, is about 3.25 feet per day. Below the Topopah Spring Tuff Member, tuff units are confining beds. Hydraulic conductivities of units tested below the Topopah Spring Member at well J-13 range from 0.0085 to 0.5 feet per day (USG83).

Beneath Yucca Mountain, the Topopah Spring Member is above the water table. Wells installed in Yucca Mountain are open to the upper volcanic aquitard (Calico Hills Formation) and the lower volcanic aquifer (Crater Flat Tuff). Pumping tests conducted in these wells derived water primarily from the Bullfrog and Tram Members of the Crater Flat Tuff (e.g., USG91a). Hydraulic conductivities calculated from pumping test data are shown in Table 7-8.

Table 7-8. Hydraulic Conductivities Calculated from Pumping Test Data

Well	K (Feet per Day)	Source
UE-25b#1	1.5	USG84d
USW H-4	1.0 - 3.6	USG85a
USW H-6	2.8	USG91a
USW G-4	4.4	USG86

It is important to recognize that the values of hydraulic conductivity presented here are average values for the entire pumped interval in the well. Borehole flow surveys, in conjunction with acoustic televiewer logging, indicate that the volcanic rocks are highly inhomogeneous in the vertical direction and that the majority of water yielded from the wells derives from a few highly fractured water-bearing zones of limited thickness. The hydraulic conductivities shown above are likely to significantly underestimate the actual horizontal hydraulic conductivity of the water-bearing zones and to overestimate the hydraulic conductivity of the less transmissive zones. USG91a estimates hydraulic conductivities for specific intervals within the volcanic section. The authors calculated a hydraulic conductivity of about 30 feet/day for a 50-foot section of the Bullfrog Member, and 22 feet/day for a 34-foot section of the Tram Member.

7.3.7.3 Porosity

In terms of bulk porosity, the volcanic sequence may be considered to consist of two different types: welded and nonwelded (or bedded) tuffs. The welding process generally reduces the matrix porosity. Therefore, the welded tuffs typically have a lower porosity than the non-welded tuffs (USG75, USG84a). The welded tuffs are also more highly fractured than their nonwelded counterparts. USG84a reports that welded units have a mean fracture density of 8 to 40 fractures per cubic meter and mean matrix porosities of 12 to 23 percent. The nonwelded units have a mean fracture density of 1 to 3 fractures per cubic meter, and mean matrix porosities of 31 to 46 percent. In both rock types, however, matrix porosity probably comprises the majority of bulk porosity because fracture porosities, even in the more highly fractured units, are reportedly quite small (USG85b). USG85b, using a theoretical model to calculate fracture porosity, reports a fracture porosity of tuffs penetrated by well USW H-4 ranging from 0.01 to 0.1 percent. Spengler and Fox (SPE89) state that matrix porosities probably decrease with depth due primarily to lithostatic loading and formation of secondary minerals.

7.3.7.4 Effective Porosity

Effective porosity is that portion of the total porosity that contributes to saturated flow. Many of the volcanic rocks are characterized by relatively small pore sizes and lack of interconnectedness of pores; thus the effective porosity is normally significantly less than the total porosity. USG84a, p. 18 reports that preliminary laboratory studies of the vitric facies of the Calico Hills unit show that only about 5 percent of the pore space is large enough to contribute significantly to flow under saturated conditions. USG85b, p. 28 considers that fracture porosity is a reasonable estimate of effective porosity. USG83, p.13 reports that effective porosities in samples of welded tuff, vitrophyre, and zeolitized clayey pumiceous tuff range from 2.7 to 8.7 percent.

7.3.7.5 Storage Properties

Numerous pumping tests have been conducted in water wells completed in the volcanic rocks under Yucca Mountain, from which storage properties may be estimated. However, most calculations of storage coefficients for the volcanic rocks are based on single well pumping tests, which generally do not produce reliable estimates of storage properties. The ground water storage characteristics of the fractured tuffs at Yucca Mountain are complex (USG85b). Estimates of storage properties of the volcanic rocks vary widely, depending partly upon the lithology and the degree of hydraulic confinement of the unit being tested. A particular hydrostratigraphic unit may be under unconfined conditions at one location and under confined conditions at another. USG91a calculates a storage coefficient of about 0.2. USG93, p. 78 calculated storage coefficients for the more densely welded units that ranged from 1×10^{-5} to 6×10^{-5}; for nonwelded to partially welded ash flow tuff zones storage coefficients were estimated to range from 4×10^{-5} to 2×10^{-4}.

7.3.7.6 Ground Water Chemistry

Chemistry of water flowing through the volcanic aquifers exhibits complex dependency upon rock chemistry, residence time in the aquifer, and its position along a flow line (USG75).

Ground water chemistry in a volcanic rock is controlled by primary glass, pumice fragments, and the diagenetic minerals (NAN89). Water samples from wells drilled in Yucca Mountain indicate that the water is predominantly a sodium bicarbonate water containing small concentrations of silica, calcium, magnesium, and sulfate (USG83). Sodium levels are generally elevated in these rock types due to the presence of volcanic glass, which is not stable in the presence of water and contains appreciable sodium. Two water wells, J-12 and J-13, currently supply water for site characterization activities at Yucca Mountain and have been pumped extensively for decades with no signs of deterioration of water quality (USG83; USG94b). Additional sources of information regarding ground water chemistry are found in USG86, USG84d, USG91a, USG91b, and USG93.

Apparent ages of ground water in the volcanic rocks as dated by carbon-14 methods are on the order of 9,000 to 15,000 years (USG93; USG83).

With the exception of substances deliberately introduced into wells during drilling and testing, such as drilling fluids (including diesel fuel at well J-13 (USG83)) and radioactive tracers (Iodine-131; USG93), no anthropogenic effects on water quality are observed in the volcanic rocks. This is attributed to the relatively low levels of human activity and the presence of a thick unsaturated zone with long travel times for infiltration to reach the saturated volcanic rocks.

7.3.7.7 Recharge

Average annual rainfall at Yucca Mountain is approximately 6 inches per year (USG84a). Various methods have been employed to estimate the amount of this precipitation which recharges the saturated zone beneath Yucca Mountain (NDC70; USG84e; USG82b). NDC70 estimated that the maximum recharge for Crater Flat and Jackass Flats is 3 percent of precipitation, or about 0.18 inches per year. USG84a considers that this is the upper bound for the recharge rate that may be occurring in certain parts of the saturated zone beneath Yucca Mountain; these authors estimate that recharge ranges from approximately 0.5 to 4.5 mm/year.

Hydrogen and oxygen isotope analyses of ground water samples taken from the volcanic rocks indicate that the primary source of the water is from precipitation (USG86; USG83). An upward hydraulic gradient from the underlying Paleozoic carbonate unit to the volcanic units (measured in well UE-25p#1) indicates the potential for flow in the carbonate rocks to move into the overlying volcanic units. Additional evidence of upwelling flow from the carbonates includes zones of elevated ground water temperature and carbon isotopic relationships (FRI94). Stuckless et al. (STU91) used the relationship of the $^{13}C/^{12}C$ ratio to the $\delta^{14}C$ of the ground water to argue for at least three sources of water under the mountain. They tentatively identified the three sources as: lateral flow from the tuff aquifer to the north; local recharge, probably introduced dominantly by flow in flash-flood watercourses on the eastern side of Yucca Mountain; and water that upwells from the deep carbonate aquifer into the tuff aquifer. Savard (SAV94) has documented recharge to the volcanic aquifer from intermittent streamflow in Fortymile Wash, east of Yucca Mountain.

7.3.7.8 Ground Water Flow

On a regional and subregional scale, ground water flow in the volcanic aquifer is generally considered to be to the south (USG75). At the scale of Yucca Mountain, there are significant variations from the regional flow pattern, resulting in local ground water flow with a strong easterly component. Potentiometric surface maps are presented in USG95a, USG94a, and USG84f, among others. The potentiometric surface can be divided into three regions: 1) a small-gradient area (0.0001) to the southeast of Yucca Mountain; 2) an area of moderate-gradient (of about 0.015), on the western side of Yucca Mountain, where the water level altitude ranges from 775 to 780 meters and appears to be impeded by the Solitario Canyon Fault and a splay of that fault; and 3) a large-gradient area (0.15 or more), to the north-northeast of Yucca Mountain, where water level altitudes range from 738 to 1,035 meters (USG94a). Numerous theories have been proposed to explain the presence of the three domains and especially the cause of the large gradient area, where water levels decline by more than 900 feet over a distance of slightly greater than one mile. The position of the large gradient area does not correlate well with any observed geologic feature in the upper 1,500 feet of the mountain (FRI91). The large gradient area is defined by water levels in four test wells: USW G-1, USW H-1, UE-25 WT#6, and UE-25 WT#16. The area where the gradient has been defined is about 1.7 miles upgradient from the design repository. Lithologic units in the saturated zone in the area of the gradient include the tuffaceous beds of the Calico Hills, the Crater Flat Tuff, the tuff of Lithic Ridge, and older unnamed ash-flow, ash-fall and lava flow rocks. If the gradient is caused by a barrier to ground water flow, it could be of particular importance to the design and performance of the repository; an increase in the permeability of such a barrier could cause a substantial rise in water table altitude in the area of the proposed repository. A rise in the water table would decrease the thickness of the unsaturated zone beneath the repository and decrease ground water travel time from the repository to the accessible environment (SIN89).

Other possible causes of the large gradient include, but are not limited to, a fault or fault zone, an intrusive dike, a change in lithologic facies or a pinch-out, a change in fracture orientation, density, aperture, or fracture fillings; perched water zones, or some combination of the above phenomena. Fridrich et al. (FRI94) have proposed two models for the large gradient zone, integrating geologic, geophysical and geochemical evidence to support their analysis. These and other authors interpret a northeast trending low gravity and drill hole data to indicate the presence of a buried northeast striking graben immediately south of the water table decline. The large gradient zone is coincident with the northern bounding fault of the proposed graben. The presence of the northern bounding graben fault, which is not exposed at the surface and is not known to have been encountered in any drill holes in Yucca Mountain, is central to both models proposed by Fridrich et al. (FRI94). Briefly, the first conceptual model proposes that the buried fault zone provides a permeable pathway through the volcanic section into the underlying deep carbonate aquifer. The second model has the buried fault acting as the northern boundary for a much thicker and more transmissive volcanic section south of the buried fault. These authors also suggest that rapid draining of water in the large gradient zone may cause the low gradient area to the south and southeast. In this model, the small gradient zone may result partly from a reduced ground water flux in the volcanic rocks owing to the capture of flow by the underlying deep carbonate aquifer.

7.4 GROUND WATER FLOW AND RESOURCES NEAR THE SITE

7.4.1 Regional Hydrologic Characteristics

In the study area, three principal hydrogeologic units are recognized: unconsolidated alluvial deposits (variously referred to as valley-fill or basin-fill deposits), Tertiary volcanic rocks, and Paleozoic carbonates (USG75). The volcanic rocks, where present, generally overlie Paleozoic carbonates, although the nature of the contact between these units is not well known due to the paucity of wells drilled completely through the volcanic rocks. The alluvial deposits overlie the Tertiary volcanic sequence over much of the area. Geophysical evidence has been interpreted to indicate that in the southern portion of the area, the volcanic units thin and may pinch out (USG85c). Data from DOE94d indicate that the volcanic unit is not present immediately south of the town of Amargosa Valley. Where the volcanic unit is not present, alluvial deposits presumably directly overlie Paleozoic sedimentary rocks. The principal hydrogeologic units are described in detail in the following sections.

7.4.1.1 Alluvial Aquifer

Valleys, topographic basins, and other topographic and structural lows are filled with variable thicknesses of unconsolidated, often poorly-sorted sand and gravel deposits. Basin-fill deposits are generally 2,000 to 5,000 feet thick, but, in some basins, exceed 10,000 feet in thickness. Basin-fill ground water reservoirs are restricted in areal extent, generally being bounded on all sides by mountain ranges. Alluvium beneath the valley floors is commonly saturated only at great depth (NDC70).

In the Yucca Mountain area, several basin fill aquifers exist. These are: Crater Flats, west of Yucca Mountain; Jackass Flats, east of Yucca Mountain, and Amargosa Valley, located south of Yucca Mountain. The Amargosa Valley aquifer is substantially larger than the Crater Flat and Jackass Flats basins (USG91c). Farther to the south, across the Funeral Mountains, lies the Death Valley alluvial aquifer. Beneath the central parts of the deeper valleys, the water table is encountered in the alluvium. At and near the valley margins, the alluvium is relatively thin, and the water table occurs in the underlying consolidated rocks.

Aquifer Geometry

The intermontane alluvial basins tend to be elongated in a north-south direction and are of roughly the same dimensions as the mountain ranges which separate them (FIE86). The alluvial fill tapers and thins away from the center of the basins. The Crater Flat and Jackass Flats alluvial basins are bounded on their northern sides by mountainous areas at approximately the latitude of the north end of Yucca Mountain. The Jackass Flats basin does not have a well-defined southern terminus; it merges into the larger, northwest trending Amargosa Desert Basin. Crater Flat is bounded at its southern end by a small, southeast trending ridge of rock outcrops. Topographic map patterns and satellite photographs

(DOE95l) suggest that the Crater Flat Basin is closed. The Amargosa Basin is bounded on its northwest end by the Bullfrog Hills, and on its southwestern boundary by the Paleozoic carbonate sequences of the Funeral Mountains. Both the Crater Flat and Jackass Flats alluvial basins are bounded below by their contact with Tertiary volcanic rocks (USG88b; USG83). Beneath Amargosa Valley, the volcanic sequence thins and probably pinches out (USG85c). If so, there are areas in the southern part of the basin where alluvium rests directly on top of the Paleozoic carbonate sequence.

Thicknesses of the alluvial deposits in the three alluvial basins in the study area are not well known due to the scarcity of drill holes which penetrate the entire alluvial sequence. Two drill holes in Crater Flat (USW VH-1 and USW VH-2) penetrate through the alluvial cover into volcanic rocks. Thickness of the alluvium in drill hole USW VH-2 is approximately 1,000 feet. In Jackass Flats, well J-13 penetrated approximately 450 feet of alluvium prior to entering Tertiary volcanic rocks (USG83). Most of the wells drilled in the Amargosa Valley are water wells for irrigation and water supply. Since most of these wells encountered sufficient water in the alluvium, drilling was not carried through to the underlying units, and thus direct evidence for the thickness of the Amargosa Basin alluvial deposits is lacking. Indirect evidence (geophysical methods) indicates that the thickness of the alluvial cover in the southern Amargosa Desert may be as much as 5,200 feet (USG89). In USG89, the authors also state that the unsaturated zone is always less than 15 percent of the allvuial thickness.

Hydraulic Conductivity

USG75 reports the results of several single well pumping tests in alluvial aquifers at the Nevada Test Site. These wells are located outside of the current study area, but the formations tested are broadly similar, and the results are generally applicable to alluvial deposits within the immediate area of concern. These authors found the hydraulic conductivity of the alluvial deposits to range from 0.067 to 9.33 feet per day. Due to the discontinuous nature of individual lenses or units within alluvial fill, both hydraulic conductivities are expected to show wide variations in magnitude.

Porosity

USG75 reports that the total interstitial porosity of 42 samples of valley fill range from 16 to 42 percent and averaged 31 percent.

Effective Porosity

Poorly sorted sediments often have values of effective porosity that are substantially less than their total porosity. Given the grain size and poorly sorted nature of the alluvium, effective porosity values may range from a few percent to perhaps as much as 25 to 30 percent. Caliche may increase the disparity between total and effective porosity of the valley-fill deposits. USG75, p. 37 reports that caliche is a common cementing material at all depths in a shaft sunk in alluvium to a depth of 550 feet.

Storage Properties

NDC63 estimated specific yield for the alluvial deposits in the Amargosa Basin using grain size distribution methods. Their estimate of specific yield for this basin is 17.34 percent, and they suggest that the actual values range from not less than 10 percent to not greater than 20 percent.

7.4.1.2 Volcanic Aquifer

A detailed description of the hydrologic properties of the volcanic aquifer has been provided in Sections 7.3.6 and 7.3.7.

Aquifer Geometry

The subsurface extent of the volcanic units south of Yucca Mountain is not reliably known. See USG85d for an illustration of the generalized extent of the volcanic rocks in southern Nevada. Aeromagnetic maps suggest that the volcanic rocks pinch out at about the latitude of Lathrop Wells and therefore that alluvial deposits constitute most or all of the cover in the Amargosa Desert (USG85c). USG85c, p. 12 notes that the "southward thinning of the volcanic rocks has been placed in question by recent north-south unreversed seismic refraction measurements. Preliminary profiles suggest that some highly magnetized volcanic rocks may indeed thin as proposed but that an underlying rock sequence of less magnetized volcanic rocks may continue southward far beyond Lathrop Wells." Well data in DOE94e indicate that the volcanic units are missing from the section just south of the town of Amargosa Valley. USG91c notes the presence of rhyolitic volcanic units within the Amargosa Basin, although the relationship of these units to the volcanic rocks that comprise Yucca Mountain is not clear.

Bare Mountain, located approximately 9 km to the west of Yucca Mountain across Crater Flat, consists of Paleozoic rocks. Tertiary volcanic rocks are known to lie beneath the alluvium in Crater Flat, so the western limit of the volcanic sequence in this area may be located at the eastern bounding fault of Bare Mountain. To the east of Yucca Mountain, the volcanic sequence continues for several to several tens of kilometers.

The thickness of the volcanic units is greatest to the north of Yucca Mountain toward the eruptive centers of the Timber Mountain Caldera Complex (USG85c; USG90). In the vicinity of Yucca Mountain, the only direct measurement of the thickness of the volcanic sequence has been at well UE-25p#1, where the thickness was measured to be 1,244 meters. Drill hole USW H-1, located immediately north of the proposed repository drift perimeter outline boundary, was drilled to a depth of 1,829 meters entirely in volcanic rocks. Thus the thickness of the volcanic sequence at the north end of Yucca Mountain could exceed 2,000 meters.

7.4.1.3 Paleozoic Carbonate Aquifer

Thick sequences of carbonate rock form a complex regional aquifer system or systems that are largely undeveloped and not yet fully understood. Secondary permeability in limestone and dolomite beds within this sequence has developed as a result of fracturing and enlargement of existing fractures by solution. The area underlain by carbonate rocks is characterized by relatively low volumes of runoff. Flow can be complex and may include substantial interaction with basin fill reservoirs (USG75).

The sequence of sedimentary rocks underlying eastern Nevada has been divided into four general hydrogeologic units; the lower clastic aquitard, the lower carbonate aquifer, the upper clastic aquitard, and the upper carbonate aquifer.

Aquifer Geometry

Evidence suggests that the Paleozoic sedimentary sequence underlies the entire area. Exposures of Paleozoic rocks at the perimeter of the study area include Bare Mountain to the west of Yucca Mountain, the Funeral Mountains south of the Amargosa Desert, and the Specter Range to the east and southeast. Further compelling evidence comes from drill hole

UE-25p#1 on the eastern flank of Yucca Mountain, which penetrated through Tertiary volcanic rocks into the underlying carbonate sequence.

USG75 indicates that water circulates freely to depths of at least 1,500 feet beneath the top of the aquifer and up to 4,200 feet below land surface.

Hydraulic Conductivity

Interstitial permeability of the carbonate rocks is negligible; essentially all of the flow transmitted through these rocks is through fractures. Estimates of fracture transmissivity range from 1,000 to 900,000 gallons per day per foot (USG75). USG75 reports the results of six pumping tests in the lower carbonate aquifer. The average calculated transmissivity was 13,000 gallons per day per foot.

Storage Properties

USG75 reported that, based on examination of rock cores, the effective fracture porosity of the lower carbonate aquifer is probably a fraction of 1 percent; accordingly, the storage coefficient under unconfined conditions is not likely to exceed 0.01. Because of the extremely low effective porosity of the carbonate rocks, the specific storage under confined conditions probably ranges between 10^{-5} and 10^{-6} per foot. Where the aquifer is several thousand feet thick the storage coefficient may be as large as 10^{-3}.

Porosity

USG75 reports that total porosity determinations were made for 16 samples of the lower carbonate rocks. Total porosities ranged from 0.4 to 12.4 percent with an average of 5.4 percent. Fracture porosity of the rock is estimated to range from 0 to 12 percent of rock volume.

Effective Porosity

Owing to the extremely low matrix permeability of the carbonate rocks, effective porosity can be approximated as the effective porosity of the fractures. Many of the fractures in the carbonate units are partially filled with clay or other materials which reduce both fracture permeability and effective porosity. USG75 reports that effective porosity determinations were made for 25 samples of the lower carbonate rocks. Values ranged from 0.0 to 9.0 percent, with an average of 2.3 percent.

7.4.2 Regional Flow Characteristics

7.4.2.1 Alluvial Aquifer

Ground Water Flow Directions

Available data indicate that the overall direction of ground water flow is to the south and southwest. Within the alluvial aquifers, there are certain to be localized variations in the direction of flow. Regional potentiometric maps are presented in USG91c, USG75, and KOL94. Potentiometric maps derived from numerical simulations are shown in USG82a and USG84e.

Ground Water Velocities

No estimates of advective velocity in the alluvial aquifers have been made downgradient of the potential repository.

Recharge and Discharge

There are several potential sources of recharge for alluvial aquifers. One source is direct recharge from precipitation falling on the alluvial areas. A second is infiltration of runoff from mountainous areas (SAV94). A third source of recharge to alluvial aquifers is infiltration or leakage from underlying bedrock aquifers. Human activity may also provide a source of recharge to the aquifers, chiefly by return infiltration of irrigation and percolation of sewage or wastewater. NDC63 and USG85e provide methods of calculation and estimates of recharge for the Amargosa Basin.

The are several potential modes for natural discharge from alluvial basins, including interbasin flow to other alluvial basins; leakage to the underlying units, either volcanic or carbonate; or evapotranspiration (NDC63). Discharge from the alluvial aquifers also occurs in the form of ground water withdrawals by pumping. In the Amargosa Valley alluvial basin, ground water is pumped for domestic and irrigation purposes (USG91c). Further details of ground water recharge and discharge are given in Section 7.4.3.4. See NDC71 for quantitative estimates of ground water recharge and discharge in the alluvial aquifers.

7.4.2.2 Volcanic Rock Aquifer

Ground Water Flow Directions

Ground water flow directions in the volcanic aquifer are discussed in detail in Section 7.3.7.

Ground Water Velocities

A range of ground water travel times in the Tertiary volcanic aquifer have been developed in support of TSPA-93. The range in predicted advective velocities is between 5.5 and 12.5 m/yr. These velocities represent average velocities in the Tertiary volcanic aquifer between the footprint of the potential repository and a 5-kilometer "accessible environment" located to the south and east of the potential repository (DOE95m).

Recharge and Discharge

Precipitation is the primary source of recharge to the volcanic aquifer(s). Snowmelt in the Timber Mountain area to the north of Yucca Mountain, as well as on Yucca Mountain itself, provides some of the precipitation-derived recharge. The occasional intense rainstorms experienced in the area also provide a source of recharge to ground water (SAV94). Total recharge to the volcanic aquifers beneath Yucca Mountain from precipitation is estimated to be on the order of 0.5 to 4 millimeters per year (USG84a). Isotopic evidence (see Section 7.3.7) indicates that some of the ground water flowing in the volcanic aquifer beneath Yucca Mountain may have been derived from the underlying Paleozoic carbonate sequence. Insufficient data exist to estimate reliably the amount of recharge to the volcanic aquifer from below.

There are not enough data available to make even first order approximations of the locations and amounts of discharge from the volcanic units. The Department of Energy states that the "current conceptual model for the regional ground water flow system considers that ground water in the volcanic rocks beneath Yucca Mountain moves generally southward and discharges in the subsurface into the valley fill alluvium as the volcanic section thins and ultimately pinches out south of Yucca Mountain" (DOE95m).

Water is also discharged from the volcanic aquifer in the study area by pumping from water wells. Two wells, J-12 and J-13, supply water for part of the Nevada Test site as well as for all site characterization activities at Yucca Mountain.

7.4.2.3 Paleozoic Carbonate Aquifer

Ground Water Flow Directions

Because the carbonate aquifer in the study area is overlain by thick deposits of volcanic rocks or alluvium, flow directions and gradients are not well defined. Regional ground water flow through the lower Paleozoic aquifer is considered to be generally southward. Small-scale potentiometric surface maps are presented in USG75.

Ground Water Velocities

Velocities through the lower carbonate aquifer range from an estimated 0.02 to 200 feet per day, depending upon geographic position within the flow system (USG75). It should be borne in mind that the figures given above are for an area of carbonate rocks outside and much larger than the study area. No data are available regarding actual ground water flow velocities in the study area. Carbonate rocks with solution widened fractures, cavities and caves typically exhibit an extremely large variation in ground water velocities.

Ground water velocities are dependent, in part, upon hydraulic gradients. The gradient of the potentiometric surface is commonly only a few feet per mile; however, across major faults, water level altitudes may differ by as much as 500 feet within a valley and 2,000 feet between valleys (NDC70).

Recharge and Discharge

Direct areal recharge to the carbonate aquifer occurs where these rocks are exposed at the surface. The highest amounts of areal recharge are expected to occur in highland areas where precipitation levels are highest and where the highly fractured rocks are exposed at the surface. Recharge to the carbonate units may also derive from downward infiltration through overlying volcanic or alluvial deposits. The relationship of flow potential in the carbonate aquifer to that in the overlying units is not well known, but is expected to be downward in parts of some basins (USG75).

One major discharge location for flow in the regional carbonate aquifer is at Ash Meadows, located southeast of Yucca Mountain. Additionally, Death Valley, located about 60 kilometers south-southwest of Yucca Mountain is regarded by many researchers as the base level or terminus for the entire regional system, and as such accommodates discharge from the carbonate aquifer (USG88c). There are also numerous small, relatively low flow springs located throughout eastern Nevada, though to a lesser extent in the study area, which represent discharge points for the carbonate aquifer(s) (USG75).

The Planning Department of Inyo County, CA has expressed concern that residents and visitors to Furnace Creek, in Death Valley National Park, might receive radiation doses as a result of transport of radioactivity from the repository via the carbonate aquifer that flows under Yucca Mountain. The County commissioned an independent evaluation of the regional hydrology (INY96).

The Inyo County study did an independent assessment of the hydrologic data base. They noted that 11 boreholes have been drilled into the deep unsaturated zone, and one borehole, UE25p1, has penetrated the deep lower carbonate aquifer in the vicinity of Yucca Mountain. The UE25p1 borehole data indicated that the potential for ground water movement in the lower carbonate aquifer is upward. If this result is valid and maintained in the future, it indicates that the potential for contamination transport to Furnace Creek is limited. The data base to address this issue is, however, quite limited at present.

7.4.3 Ground Water Resources

Many of the studies performed in the Yucca Mountain characterization process have focused narrowly on the immediate area in and around the proposed repository. Few studies have attempted to present a regional ground water resources picture for the areas downgradient from Yucca Mountain, where potential receptors may be located. This section presents a summary description of water resources in the area downgradient (generally south) of Yucca Mountain in order to provide more detailed information as to potential numbers and locations of people who might be exposed to contaminants migrating from Yucca Mountain in ground water.

7.4.3.1 Water Quality

Alluvial Aquifer

The chemical quality of the ground water in the study area varies from place to place. In general, ground water in wells closer to the ultimate discharge areas of the system, such as the southern Amargosa Desert and Death Valley, contains higher concentrations of dissolved constituents, and is less suitable for most purposes (NDC63). NDC63 states that, for ground water in the Amargosa Desert, "although the chemical quality of ground water may be suitable generally for irrigation, water of median salinity is common and water of high salinity occurs locally." Ground water in the alluvial aquifers in many cases contains excessive concentrations of fluoride; a dental examination of school children in Beatty found that 19 out of 20 children who lived in Beatty since birth were affected with dental fluorosis (NDC63). See USG94b and USG91d for additional ground water chemical quality data for the alluvial aquifer.

Volcanic Aquifer

A description of ground water chemical quality for the volcanic aquifer is presented in Section 7.3.7. Additional data may be found in USG94b, USG86, USG84d, USG91a, USG91b, and USG93.

Carbonate Aquifer

In general, water occurring in the carbonate rocks is a calcium and magnesium carbonate water. Where water in the carbonate aquifer has moved through the overlying volcanic rocks, analyses show increased levels of sodium and potassium (USG75). See USG84c for chemical analyses of water from well UE-25p#1, completed in the carbonate aquifer beneath Yucca Mountain.

7.4.3.2 Water Use

Nevada is an "appropriative" state, meaning that water rights are strictly controlled by the state and appropriated to users on a case-by-case basis. Ground water use in Nevada is regulated by the Department of Conservation and Natural Resources through the State Engineer's Office. For purposes of water resources administration, Nevada is divided into 253 hydrographic basins (see USG88c for details). The state limits the amount of water that may be withdrawn from any hydrographic basin to the perennial yield for that basin. Perennial yield is the amount of water that may be withdrawn annually without depleting the amount of available ground water.

The five hydrographic basins listed in Table 7-9 are of principal concern to water resource issues. These basins are shown in context with related basins in Figure 7-26. The hydrologic boreholes associated with the Yucca Mountain project that are used to characterize the hydrologic regime are shown in the Figure 7-27 map.

Table 7-9. Hydrographic Basins in the Vicinity of Yucca Mountain

Hydrographic Basin Name	Basin Number
Mercury Valley	225
Rock Valley	226
Fortymile Canyon - Jackass Flats	227-A
Crater Flats	229
Amargosa Desert	230

The regional aquifers in the five hydrographic basins that are used for human activities are the valley fill aquifer and the lower carbonate aquifer. The welded tuff aquifer is locally

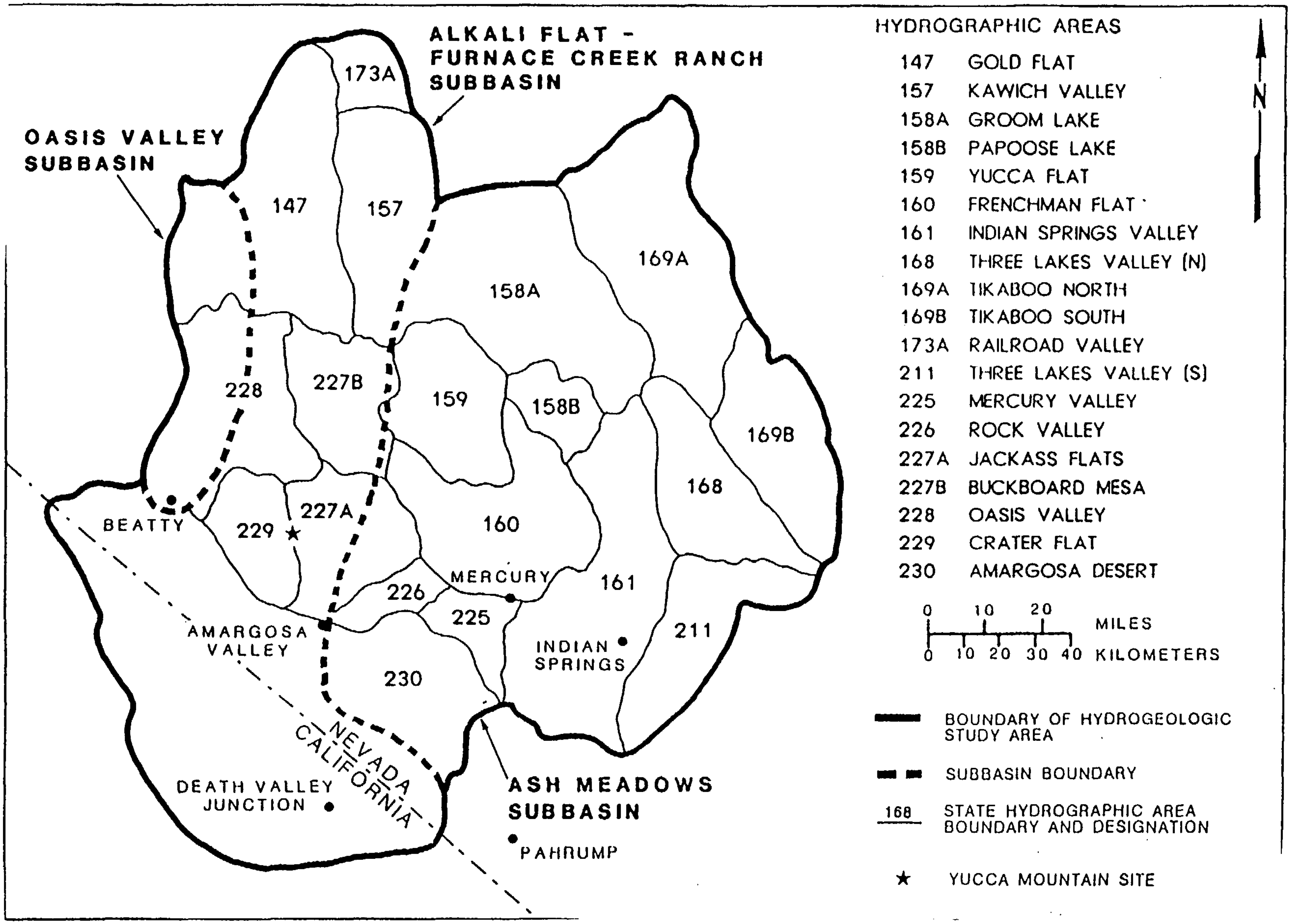

Figure 7-26. State Hydrographic Areas Within the Hydrogeologic Study Areas (Source: DOE95a)

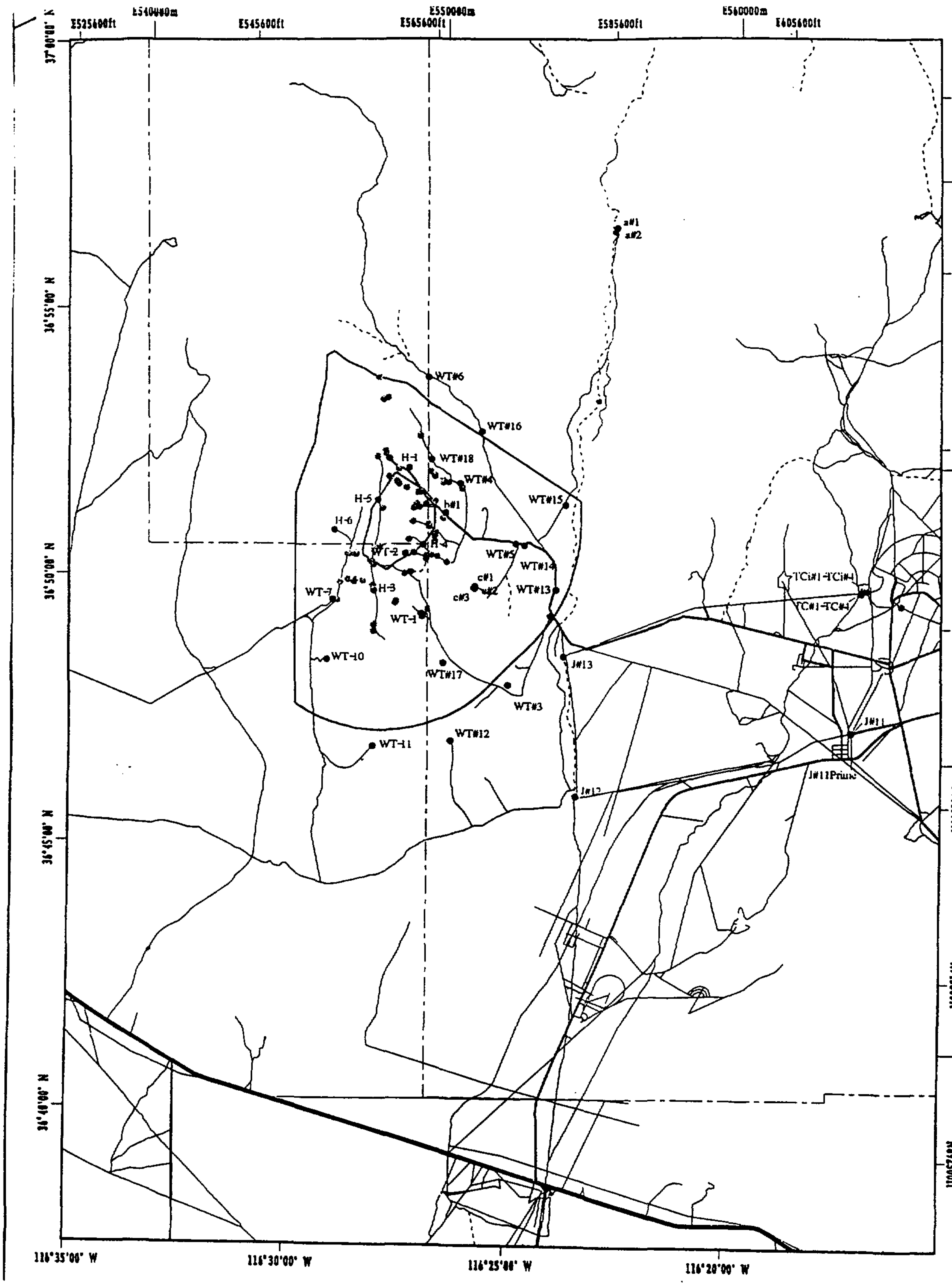

Figure 7-27. Hydrologic Boreholes in the Yucca Mountain Vicinity (Source: DOE95l)

important; it is developed only in the southwestern areas of the Nevada Test Site. This water is withdrawn from two water wells (J-12 and J-13) located in Basin 227-A, and is used for all site characterization activities at Yucca Mountain, including human consumption. A significant percentage of this water is used to wet local unpaved roads for dust suppression (STE95). Well J-13 currently is being pumped at a rate of 550 gallons per minute, and well J-12 is being pumped at a rate of 800 gallons per minute, depending on demand (DOE95m).

The other principal locus of ground water withdrawals is in the vicinity of the town of Amargosa Valley, located south of Yucca Mountain. This is the only substantial human settlement within the hydrographic area. Ground water in the Amargosa Valley is used for a variety of purposes, including withdrawals for domestic, agricultural, mining, recreational (golf course), and industrial uses. All of the wells in Amargosa Valley derive water from saturated alluvial deposits. The Nevada State Engineer's office has determined that the maximum perennial yield for Basin #230 is 24,000 acre-feet per year. Amounts of perennial yield and quantities currently appropriated to users are shown in the Table 7-10.

Table 7-10. Water Appropriations by Hydrographic Basin in the Study Area

Hydrographic Basin	Perennial Yield	Total Appropriated	Approved Use				
			Irrigation	Community	Municipal	Stock/ Other	Mining
225		0	0	0	0	0	0
226		0	0	0	0	0	0
227-A		56	0	39	0	17	0
229		2,995	0	0	0	61	2,934
230	24,000*	41,093	28,600	85	2,486	4,255	5,667

* The perennial yield is a combined total for all of the above basins and Basin # 228.

Source: Hydrographic area summaries, State Engineer's Office Nevada Department of Conservation and Natural Resources.

An examination of the above table shows that ground water usage rights are over-allocated in Basin No. 230 by approximately 17,000 acre-feet per year. Actual usage in Basin No. 230 has thus far not exceeded the estimated perennial yield. Available usage figures indicate that annual basin-wide withdrawals have not been in excess of approximately 12,000 acre-feet per year. A 1993 pumpage inventory (State of Nevada, Division of Water Resources, Las Vegas Office) for the Amargosa Desert Basin (#230) shows that the actual ground water usage for 1993 amounted to 11,300 acre-feet per year. Table 7-11 provides a breakdown of 1993 ground water usage in Basin 230 by category.

Table 7-11. 1993 Ground Water Pumpage Inventory for Basin No. 230

Ground Water User/Use	1993 Pumpage (acre-feet)
Irrigation	8,559
Irrigation (no permits or certificates)	150
American Borate (314 acre-feet pumped from CA side)	512
Industrial-Mineral Ventures	495
St. Joe Bull Frog	1,474
Commercial	10
Quasi-Municipal & Domestic	100

The 1993 ground water withdrawals for the Jackass Flats Basin (No 227-A) and Mercury Valley Basin (No. 225) were 205 and 338 acre-feet, respectively (USG95b). The pumpage from Basin 227-A reflects withdrawals from the J-12 and J-13 wells described earlier.

Most of the water pumped from the Ash Meadows subbasin is pumped from the lower carbonate aquifer (USG76). In 1971, ground water withdrawals associated with the planned development of a large agricultural enterprise caused a decline in the water level of the pool at Devil's Hole. This natural pool, formed from the collapse of the limestone bedrock, is the only habitat of the Devil's Hole pupfish, an endangered species. As a consequence of court action, ground water withdrawals in this area are now restricted to a degree that is sufficient to maintain the water level in Devil's Hole (USG76).

The Nevada Test Site receives its water from wells drilled on the NTS. The NTS accommodates a worker population of approximately 5,000 individuals, most of whom reside in Las Vegas and other nearby communities; a very small percentage of this workforce resides in Mercury on an intermittent basis. There are 12 NTS wells that currently withdraw water from the Ash Meadows subbasin for construction, drilling, fire protection, and consumption uses. Some of the water requires treatment before distribution (DOE95j).

The major ground water users in Basin No. 230 are the town of Amargosa Valley and small rural communities of the northeastern Amargosa Desert. Most of the water to these users is supplied by wells; however, there has been some spring development. Most residences rely on individual wells, while some trailer parks, public facilities, and commercial establishments are served by small, private water companies. All wells are completed in and produce from the valley-fill aquifer (DOE95j).

Two mineral production operations are located in the Amargosa Desert. One operation, owned by the American Borate Corporation and located between Amargosa Valley, Nevada and Death Valley Junction, California, was decommissioned in July 1986. The facility consisted of a large mineral processing plant and a housing development for its employees. Water for the community was pumped from a shallow well and was treated by a reverse osmosis process to reduce total dissolved solids before distribution. The other operation is owned by IMV Division of Floridin, Inc. and is also located between Amargosa Valley, Nevada and Death Valley Junction, California. As of 1995, the operation employed approximately 53 people to mine specialty clays (DOE95j).

In addition to well production, a number of springs supply water to the region. The main concentration of springs is in Death Valley in the vicinity of Furnace Creek Ranch, approximately 60 km southwest of Yucca Mountain. The water supply for the National Park Service facilities is derived principally from three groups of springs: Travertine Springs, Texas Springs, and Nevares Springs. The population served by this water supply varies during the year. From October through April, approximately 800 persons live in the area on a semipermanent basis, and an additional 2,000 persons live in the area as visitors. From May through September, the number of semipermanent residents decreases, and there are few visitors (DOE95j).

Three resorts are located within the boundaries of the Death Valley National Monument; the Stovepipe Wells Hotel, Furnace Creek Inn, and Furnace Creek Ranch. Water for the Stovepipe Wells hotel is trucked in from Nevares Spring. Water for Furnace Creek Inn and Furnace Creek Ranch is reportedly conveyed from an excavated sump lined with drainage tile in the Furnace Creek Wash (DOE95j).

Crater Flat (Basin No. 228) is currently overdrawn because of an appropriation made to Saga Exploration, Inc. for development of the Panama-Sterling Mine, located on the east side of Bare Mountain. The mine uses its own well for its heap-leach operation and relies on municipal water for its potable water. The mine employs approximately 40 individuals and is expected to be in operation until 1997 or 1998 (DOE95j).

7.4.3.3 Water Availability/Perennial Yields

In order to estimate the population that may be supported at some time in the future by the water resources available in the Yucca Mountain area, the following analysis was performed.

Perennial yield is defined as the maximum amount of water that can be withdrawn from the ground water system for an indefinite period of time without causing a permanent depletion of the stored water or causing a deterioration in the quality of the water (NDC63). It is ultimately limited by the amount of water annually recharged to or discharged from the ground water system through natural processes plus that which might become available by artificial recharge and water returned to the ground water system by infiltration of irrigation or waste water.

In estimating perennial yields, the effects that ground water development may have on the natural circulation in the ground water system should be considered. The location of the withdrawal centers in the ground water system may permit optimum utilization of available supply or, at the other extreme, may be ineffective in the utilization of available water supply. The location of the wells may favor improving the initial quality with time or may result in deterioration of quality under continued withdrawals. Development by wells may or may not induce recharge in addition to that received under natural conditions. Part of the water discharged by wells may re-enter the ground water reservoir by infiltration of excess irrigation or waste water and thus be available for re-use. Ground water discharged by wells eventually reduces the natural discharge. In practice, decreasing natural discharge by pumping is difficult, except when the wells are located where the water table can be lowered to a level that eliminates evapotranspiration in the natural area of discharge or underflow from the basin.

There are a number of means by which the perennial yield can be calculated. The State of Nevada accepts the method proposed by NDC63, which estimates the perennial yield of hydrologic basins by assuming that perennial yield is equal to the volume of water that would naturally discharge through evapotranspiration and lateral outflow (underflow). In other words, perennial yield is considered equal to total natural basin discharge.

An alternative method to that presented by NDC63 for the determination of perennial or safe aquifer yields is presented by Linsley et al. (LIN82), in which the perennial yield is expressed as a function of the quantity of water available. This hydraulic limitation is often expressed by the equation

$$G = P - Q_s - ET + Q_g - S_g - S_s \qquad (1)$$

where G is safe yield (i.e., perennial yield), P is precipitation on the area tributary to the aquifer, Q_s is surface streamflow from the same area, ET is evapotranspiration, Q_g is net ground water inflow to the area, S_g is the change in ground water storage, and S_s is the change in surface storage. If the equation is evaluated on a mean annual basis, S_s will usually be zero.

All terms in Eq. (1) are subject to artificial change, and G can be computed only by assuming the specific conditions for each item. For example, artificial recharge operations can reduce Q_s. Irrigation diversion from influent streams may increase evapotranspiration. Lowering the water table by pumping may increase ground water inflow (or reduce ground water outflow) and may make otherwise gaining streams into losing streams.

The factors that control the assumptions on which Eq. (1) is evaluated are primarily economic. The feasibility of artificial recharge or surface diversion is usually determined by economics. If water levels in the aquifer are lowered, pumping costs are increased. Theoretically, there is a water-table elevation at which pumping costs equal the value of the water pumped and below which water levels should not be lowered. Practically, the increased cost is often passed on to the ultimate consumer, and the minimum level is only attained after excessive lowering of the water table results in contamination of the ground water by upcoming and inflow of undesirable waters.

The permanent withdrawal of ground water is called mining. If the storage in the aquifer is small, excessive mining may be disastrous to any economy dependent on the aquifer for water. On the other hand, many ground water basins contain vast reserves of water, and planned withdrawal of this water at a rate that can be sustained over a long period may be a practical use of this resource. The annual increment of mined water, S_g in Eq. (1), increases the yield. Thus, Eq. (1) cannot properly be considered an equilibrium equation or evaluated in terms of mean annual values. It can be evaluated correctly only on the basis of specified assumptions for a stated period of years. The following discussion presents a methodology by which the various parameters in Eq. (1) were determined for the Yucca Mountain area.

The hydrographic areas (HA) that are most relevant to the determination of perennial yields downgradient of Yucca Mountain are basin numbers 225, 226, 227-A, 229 and 230. Table 7-12 presents the water budget information for these hydrographic areas, obtained from State of Nevada's water planning report (NDC71). Each of the column entries are discussed below.

Table 7-12. Ground Water Budget for Hydrographic Basins in Study Area

Hydrographic Basin Number	Hydrographic Basin	Ground Water Recharge From Precipitation (ac.-ft/yr.)	Ground Water Inflow		Ground Water Discharge to the surface (ac.-ft./yr.)	Ground Water Outflow	
			Acre Feet Per Year	From Hydrographic Area		Acre Feet Per Year	To Hydrographic Area
225	Mercury V.	250	16,000	160	0	17,000	230
226	Rock V.	30	17,000	160, 227-A	0	17,000	230
227-A	Jackass Flats	900	7,200	227B	0	8,100	230
229	Crater Flat	220	1,500	228	0	1,700	230
230	Amargosa Desert	600	44,000[1]	225, 226, 227-A, 229	24,000	19,000	Death Valley

[1] This value of 44,000 is inconsistent in context to the other data presented in the table, in that it should total the ground water inflow from all of the contributing basins (i.e., 43,800 acre-feet/year), as well as total basin discharge (i.e., 43,000 acre-feet/year).

Column 1 - Hydrographic Basin Number

The State of Nevada has been subdivided into 253 hydrographic basins. The boundaries for each basin are generally coincident with surface-water divides defined by topography.

Column 2 - Hydrographic Basin Name

Almost all of the current ground water use downgradient of Yucca Mountain is derived from the Amargosa Desert Hydrographic Basin (ADHB). Those basins within the study area that are hydraulically connected to the ADHB via ground water include Mercury Valley (HB 225), Rock Valley (HB 226), Jackass Flats (HB 227-A), and Crater Flat (HB 229).

Column 3 - Ground Water Recharge from Precipitation

Ground water recharge from precipitation represents the volume of precipitation that moves vertically through the unsaturated zone (region above the water table) and becomes available for pumping. Other sources of recharge (e.g., irrigation return flow) are thought to be insignificant and are not included in this column.

Column 4 - Ground Water Inflow

The ground water inflow is the volume of ground water that enters the hydrologic area from other hydrologic basins. In the case of the Amargosa Desert Hydrologic Area, ground water enters from Hydrologic Basins 225, 226, 227-A and 229. The volumes derived from each of these basins are presented in acre-feet/year and total 43,800 acre-feet/year. As noted in the table, these values should total ground water inflow into ADHB (44,000 acre-feet/year). Apparently either an error was made in the data entries or the value was rounded to 44,000.

Column 5 - Ground Water Surface Discharge

The ground water surface discharge volumes represent the volume of water that is discharged to the surface via streams and seeps in addition to water that is removed from the aquifer by evaporation and the transpiration of plants. In the perennial yield calculations performed below, this discharge is actually treated as ground water outflow and is assumed to be available for consumption. The rationale for this assumption, presented by Walker and Eakin (WAL63), is that once the water table has sufficiently dropped below some point, significant transpiration and surface discharges will no longer occur.

All of the 24,000 acre-feet/yr that is discharged to the surface in the Amargosa Desert Hydrographic basin is removed from the system by evapotranspiration. Furthermore, almost all of this water is attributed to spring discharges at Ash Meadows.

Column 6 - Ground Water Outflow

The ground water outflow is the volume of ground water that flows out of the hydrologic basin into adjacent basins. The table indicates that the outflow from the Amargosa Desert Hydrologic Basin of 19,000 acre-feet/yr flows into the Death Valley Hydrographic Basin. Note that ground water outflow (19,000 acre-feet/yr) added to evapotranspiration (24,000 acre-feet/yr) should be equal to ground water inflow (43,000 acre-feet/yr) for Basin # 230. However, there is a discrepancy of 1,000 acre-feet/year and it is unclear why this discrepancy exists. In any case it will not significantly effect perennial yield estimates.

We now determine the size of the potentially affected population that could be sustained by the ground water available in the Yucca Mountain region. The available ground water has been defined as that ground water which is contained within Hydrographic Basins 225, 226, 227-A, 229 and 230. These hydrologic basins are considered to be the most relevant to the analysis because they are located downgradient of Yucca Mountain to a distance of approximately 50 miles. Since Basins 225, 226, 227-A, and 229 discharge into Hydrographic Basin 230, this basin (HB 230) will be the focus of our calculations.

In Eq. (1), the precipitation (P) minus evapotranspiration (ET) is assumed to equal ground water recharge. Table 7-12 indicates that Basin #230 receives 600 acre-feet/yr of recharge from precipitation. There is no significant surface streamflow (Q_s) or change in surface storage (S_s). As mentioned previously, S_g represents the annual increment of mined water and should be set to zero for perennial yield determinations. This suggests that Eq. (1) may be written as:

$$G = 600 + Q_g \tag{2}$$

Table 7-12 indicates that 44,000[11] acre-feet/yr enters Basin #230 as lateral ground water inflow (Q_g) from other hydrographic areas (225, 226, 227-A, 229). Therefore, based on Eq. (2), the total volume of yearly sustainable water under current conditions would be 44,600 acre-feet/yr. A ground water modeling study performed in USG95c, and an alternative analysis (NDC63), indicate sustainable yields may be closer to 24,000 acre-feet/yr. Furthermore, the State of Nevada assumes a perennial yield of 24,000 acre-feet/yr for Basin #230. The State's estimates are based on work by in NDC63 in which the authors estimated that discharge via evapotranspiration is 23,500 acre-feet/yr. and ground water outflow is 500 acre-feet/yr for a total perennial yield of 24,000 acre-feet/yr. However, USG88c indicates that ground water outflow could be as high as 19,000 acre-feet/yr, which, by NDC63's method of determining perennial yields (i.e., evapotranspiration plus lateral ground water discharge), would result in a perennial yield of 42,500 acre-feet/yr. Therefore, the estimate of 44,600 acre-feet/yr appears to represent a reasonable upper bound maximum for the water available. This value would also tend to maximize the estimates of the potentially affected population size.

In 1993, there were approximately 1,100 people residing within Basin #230. The water withdrawal for the same year from the underlying aquifer was 11,300 acre-feet (Table 7-11). This translates to a yearly per capita withdrawal rate of 10.27 acre-feet. If it is assumed that future growth in the area is proportional to current water-use practices, the total population that could be sustained by a perennial yield of 44,600 acre-feet/yr is 4,342 people.

A second scenario that provides a reasonable upper bound on the number of people that could be supported by the ground water in this area can be made by assuming that all water use in the basin would be consumed entirely by domestic use, possibly exported to Las Vegas. Van der Leeden et al. (VAN90) indicate that the average person in the United States utilizes 86.5 gal/day of water for domestic use (0.097 acre-feet/yr). Van der Leeden et al. (VAN90) also indicate that the average individual in the State of Nevada utilizes 141 gal/day (.16 acre-feet/yr), which is somewhat higher than the national average. In order to maximize the size of the potentially affected population, the lower value for domestic water use (0.097 acre-feet/yr) is used in conjunction with an assumed sustainable yield of 44,600 acre-feet/yr. This results in a potentially affected population size of 459,794; this value is expected to be a reasonable maximum.

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[11] Although there may be a slight error in the reported value, it is used, rather than the corrected value, because its use will provide higher population estimates.

The calculations above assume that no "mining" of the water occurs. To evaluate the additional size of the population that could be affected if the hydrographic areas were mined of their resources, the following calculation is made.

USG88c presents the volume of water that can be derived from each of the hydrographic areas for each foot of aquifer dewatered (Table 7-13). The total volume available for all five hydrographic basins per foot of drawdown is 43,900 acre-feet. For the purpose of these calculations, it was assumed that the maximum drawdown that could be achieved without significant water-quality deterioration is 1,000 feet. NDC63 indicates that in many instances the ground water is already of relatively poor quality and the assumption that the water would remain potable after the water table is drawn down 1,000 feet may be overly optimistic. The assumed drawdown of 1,000 feet would yield 43,900,000 acre-feet from storage. Since this mined water represents a one-time occurrence, its use must be integrated over some time period to determine how many additional people could be supported. If this time frame is set to 10,000 years, the additional volume that would be available is 4,390 acre-feet/yr, whereas if the time frame is one million years, the additional volume derived from mining the aquifer would be 44 acre-feet/yr. Therefore, under an assumed time frame of 10,000 years, an additional 427 (4390/10.27) people per year could be served at current usage rates, and 45,257 (4390/.097) people at lower usage rates. Similarly, for a one million-year time frame, the increase in population that could be sustained from additional water due to mining would range from 4 to 440 people per year.

Table 7-13. Ground Water Storage Values for Relevant Hydrographic Basins

Hydrographic Basin Number	Hydrographic Basin	Basin Area	Ground Water Storage in Upper 1 ft Saturation (AF)
225	Mercury V.	110	minor
226	Rock V.	82	1,500
227-A	Jackass Flats	519	7,400
229	Crater Flat	—	—
230	Amargosa Desert	896	35,000

7.5 GEOLOGIC STABILITY

The NAS Panel report (NAS95) asserts that the Yucca Mountain site exhibits long-term geologic stability on the order of one million years. The principal implication of this assertion is that peak dose can be reliably assessed during this time frame. The panel therefore concludes that there is no need to select arbitrarily a shorter compliance evaluation period such as 10,000 years. The panel recommends "...that compliance assessment be conducted for the time when the greatest risk occurs, within the limits imposed by long-term stability of the geologic environment."

This section examines the panel's assertion of long-term stability and related issues. Factors addressed include characteristics of the geologic and hydrologic systems implied by the panel's concepts of "stable" and "boundable;" validity of the assertion of stability; and the significance of stability to the occurrence, magnitude, and evaluation of peak dose.

7.5.1 Characterization of Geologic Stability by the NAS Panel

The NAS report (NAS95) does not specifically define geologic stability. The existence of stability is asserted six times in the report, in different ways:

> "The geologic record suggests that [the time frame during which the geologic system is relatively stable or varies in a boundable manner] is on the order of one million years." (Executive Summary, page 9)
>
> "...the long-term stability of the fundamental geologic regime [is] on the order of one million years at Yucca Mountain." (page 55)
>
> "The long-term stability of the geologic environment at Yucca Mountain ... is on the order of one million years." (page 67)
>
> "The time scales of long term geologic processes at Yucca Mountain are on the order of one million years." (page 69)
>
> "The time scale for long-term geologic processes at Yucca Mountain is on the order of approximately one million years." (page 72)
>
> "The geologic record suggests that [the time frame over which the geologic system is relatively stable or varies in a boundable manner] is on the order of about one million years." (page 85)

These characterizations of geologic stability are quite similar, although some are expressed in terms of the geologic regime itself and others are described in terms of the processes that operate on or within that regime. These two assertions are not necessarily the same. For example, characteristics of the geologic regime that are important to peak dose evaluation might remain stable while tectonic and other natural processes and events continue in the future, even varying from past characteristics. Alternatively, natural processes and events may continue in the future as they have occurred in the past (i.e., the processes and events exhibit stability), while the effects they produce may change the features of the geologic regime that are important to peak dose evaluation. Conditions in which past and continuing tectonic movement produces differential movement of deep geologic structures might cause changes in the hydrologic regime important to peak dose occurrence. The various expressions of stability used in the panel's report imply no significant change in either the geologic regime or in the processes and events that affect the characteristics of that regime.

The panel's report does not explicitly justify the assertion of million-year stability by providing a synopsis and interpretation of the geologic record. Some of the references cited in the report contain information about the geologic record (e.g., DOE's Site Characterization Plan for the Yucca Mountain site (DOE88)), but none of the cited references interprets the record to indicate a million-year stability of the geologic regime or the processes associated with it.

7.5.2 Existing Documentation Related to Stability

Existing documentation has not directly addressed long-term stability of the natural features of Yucca Mountain and its environs. Until quite recently, the DOE documents containing information about geologic features of the Yucca Mountain site anticipated that evaluations of site suitability would be made in accord with DOE's 10 CFR Part 960 Site Suitability Regulations, and safety performance of a repository at the site would be evaluated in terms of EPA's 40 CFR Part 191 regulations and NRC's 10 CFR Part 60 regulations. Under this regulatory framework, the time period of concern was 10,000 years, not 1,000,000 years.

The 10,000-year time frame for the regulatory period was previously selected by EPA because it was short compared to long-term factors such as tectonic motion that might affect, and change in ways that could not be characterized, the natural environment conditions important to regulatory compliance evaluations. The time period was also long enough to bring into consideration, at least in principle, factors such as seismicity that are important in geologic time scales and might affect repository safety performance.

DOE has, in many Yucca Mountain project documents, implied geologic stability or the equivalent for time periods of 10,000 years. The State of Nevada believes, however, that the record does not justify such an assertion. For example, the State's comments (NEV85) on DOE's draft Environmental Assessment (DOE84) for the Yucca Mountain site refer to DOE's statement, "Neither major tectonic activity nor the resumption of large-scale silicic volcanic activity in the area near Yucca Mountain is likely in the next 10,000 years." The State asserts, "Based on existing evidence, this conclusion is premature." The State also asserts that other DOE reports "... suggest that possible hydrovolcanic activity at Yucca Mountain has not been sufficiently evaluated" (NEV85, Volume II, page 125).

In general, the documents of record show controversy over the stability of the geologic regime and associated natural processes and events at the Yucca Mountain site. Much of the controversy stems from opposing interpretations of available data by DOE and by the State of Nevada. The opposing viewpoints reflect in part the institutional positions of the parties involved, but nonetheless the uncertainties in the data permit such alternative interpretations to be made.

7.5.3 Interpretation of the Geologic Record Related to Stability

The geologic history of the area provides the basis for assertions concerning the stability of the geologic regime for Yucca Mountain and its vicinity. Site characterization activities for DOE's Yucca Mountain project and other activities not related to the Yucca Mountain project (e.g., commercial characterization of natural resource potential) have yielded an extensive data base concerning geologic features and the geologic record of the region.

Such data do not, however, definitively resolve the question of the long-term stability of the geologic regime, as indicated by the differences between DOE's and the State of Nevada's interpretation of data. Such issues can be resolved only in context, through the exercise of judgment by involved parties. The NAS panel's assertion of long-term geologic stability at Yucca Mountain for the next million years is an example of such judgment.

The basis for the panel's judgment of million-year geologic stability is indicated by its use of the phrase "fundamental geologic regime" and its discussion (NAS95, page 68) of the calculations enabled by stability. The panel asserts that the properties and processes of the geologic regime that are important to repository performance "... are sufficiently understood and stable over the long time scales of interest to make calculations [of repository performance] possible and meaningful."

The relevant properties and processes include, in addition to the radionuclide inventory of the waste, the influx of water to the repository, migration of the water and its contained waste materials from the repository to the ground water, and subsequent dispersion and migration of contaminated ground water in the regional biosphere. The panel considers, for example, that it will be possible to estimate, with acceptable uncertainty, concentrations of wastes in ground water at various locations and times.

These assertions imply a judgment that the basic features of the geologic regime that affect waste release and transport will remain as they are, or change in a limited and reasonably predictable fashion, for the next million years. In other words, phenomena that would substantially and unpredictably change the current, relevant geohydrologic regime are not expected. Such phenomena would include tectonic motion, seismicity, and volcanism of scale and location sufficient to change the features of the geologic regime that govern radionuclide release and transport.

The panel's assertions also imply that the geologic and hydrologic features of the site and region can and will be characterized so that repository performance can be reliably projected on the basis of current conditions. Two of the parameters cited by the panel as important to performance predictions—water influx to the repository and dispersion and migration of ground water in the biosphere—have been shown by DOE modeling studies (DOE95c, herein also termed TSPA-95) to be highly important to potential health effects from the repository. These two parameters are currently, however, among the least well-known of the repository performance-related parameters.

The DOE performance assessment reports (e.g., DOE95c) indicate that these hydrologic parameters will be extremely difficult to evaluate reliably. Bounding values, consistent with the NAS Panel's concept of bounding, can be established, but the bounds may have to be narrowed considerably from present ranges. Current estimates of values for infiltration rates through the geologic strata and to the repository, for example, span several orders of magnitude. The range is such that, with all other factors constant, projections of repository performance against potential standards range from non-compliance to compliance by orders of magnitude (DOE95c). As DOE notes in TSPA-95, direct observation of infiltration rates is not possible. The TSPA-95 evaluations treat infiltration rate to the repository as an uncertain parameter.

This situation raises an issue not addressed directly by the NAS Panel: can key performance-related parameters be adequately characterized? The Yucca Mountain site may indeed exhibit the long-term geologic stability seen by the NAS panel, but the inherent variability of geohydrologic parameters for the site and region may be a much greater obstacle to predicting long-term risks to the critical group.

The most comprehensive data available for judging the geologic stability of the Yucca Mountain site are presented in DOE's Site Characterization Plan (SCP) (DOE88). A discussion of SCP information and recent data and analyses concerning the site's geologic and hydrologic features is presented in the predecisional preliminary draft Chapter 3 of DOE's License Application for a repository at the site (DOE95a). A brief summary of the stability-related data and interpretations provided by DOE in the draft license application is presented below. More detailed descriptions of the site geologic and hydrologic features are provided in Sections 7.3 and 7.4 of this Background Information Document.

7.5.3.1 Overall Characteristics of the Geologic Regime

Yucca Mountain is located in the Southern Great Basin, about 150 kilometers northwest of Las Vegas, Nevada. The Great Basin is a broad, arid to semi-arid region characterized topographically by linear, usually north-trending mountain ranges separated by deep alluvium-filled valleys. The mountain ranges are typically tilted fault blocks that are delineated by mountain-front normal faults on which considerable vertical displacement has occurred.

Older rocks have been faulted and folded in response to compressional tectonic events. The entire stratigraphic section has been deformed by extensional basin and range tectonics. The crest of Yucca Mountain ranges between 1,500 and 1,930 meters above sea level and more than 300 meters above the adjacent valley floors. Yucca Crest is a relatively undisturbed structural block that is tilted 5 to 8 degrees to the east.

7.5.3.2 Volcanism

Ages of the volcanic rocks of the Yucca Mountain stratigraphic section range from about 11.4 to 15.2 million years. As previously noted, silicic volcanism (i.e., the volcanism that created the Yucca Mountain geologic materials) ended about 7.5 million years ago.

Basalts in the Crater Flats basin erupted during episodes dated 11, 3.7, 1.0 and 0.1 million years ago. The most recent balsaltic eruption may have been as recent as 20,000 years ago. The DOE Yucca Mountain project has used available data to estimate (DOE94a) the probability of magmatic events which might have heat and aggressive volatile effects on the repository without direct waste-magma contact. The DOE TSPA-95 analyses indicate that the probabilities of zero, one, and two such events over the next million years, for a repository with waste package spacing to produce thermal loads of 57 kW/acre, are 0.98, 0.023, and 0.00028, respectively. In other words, it is estimated to be almost certain that no such events will occur, and there is less than one chance in a thousand that two such events will occur.

7.5.3.3 Seismicity and Faulting

The Yucca Mountain area is one of three in the region (the others are Death Valley and extreme southern Nevada) with little or no seismic activity. A zone of quiescence centered on Yucca Mountain is apparent in all studies describing seismicity in the southern Great Basin; this is a real feature and not an artifact of seismic sensor network design.

The largest seismic event in the area since 1978 was an M_L 2.1 event on November 18, 1988, centered 12 kilometers northwest of the proposed repository location. An earthquake of magnitude M_W 5.7 occurred on June 29, 1992, beneath Little Skull Mountain approximately 20 kilometers southeast of Yucca Mountain. This small earthquake is the largest ever recorded (in about 100 years of records) in the vicinity of the site. It caused minor structural damage to the Yucca Mountain project field office near Yucca Mountain but had no apparent effect on geologic features near the mountain.

In general, DOE does not consider seismicity to be a significant factor in repository safety performance. Seismic effects are not considered in the total system performance assessments (DOE94a, DOE95c), because seismic events have virtually no effects underground.

There are numerous major geologic faults in the Yucca Mountain region, and one of them, the Ghost Dance fault, passes through the proposed repository location. The faults are evidence of past deformation due to tectonic and seismic activity. Faults could be pathways for ground water transport, and movement on the faults could affect repository stability and the hydrologic regime.

The three major faults in the immediate region of Yucca Mountain are the Ghost Dance, which passes through the mountain and the proposed repository; the Bow Ridge, just to the east of the mountain; and the Solitario Canyon fault, just to the west of the mountain. According to DOE's interpretation of available data (see Section 7.3.3 of this BID), the Solitario Canyon fault has shown no significant movement over the last 40,000 to 110,000 years, and no movement at all during the last 10,000 years. The most recent surface-rupturing motion on the Bow Ridge fault is estimated to have occurred 48,000 plus-or-minus 20,000 years ago, with a recurrence interval most likely in the range of 60,000 to 100,000 years. There has been no offset or fracture on the Ghost Dance fault for the past 20,000 years.

7.5.3.4 Climate

The Yucca Mountain area is basically arid; precipitation currently averages about 170 millimeters (6 inches) per year. Precipitation amounts vary during the year. Average precipitation during the fall and winter months is about 18 mm/month; average precipitation during the spring and summer months is about 9 mm/month.

The climate in the Yucca Mountain area has apparently cycled between wet and dry in the past. During the past few thousand years (i.e., since the last ice age), the climate has been essentially unchanged. The DOE Total System Performance Assessments to date (DOE95c) have assumed that the climate will in the future cycle between wet and dry, with cycle times of 100,000 years. In preparing to review a License Application for a repository at the Yucca Mountain site, the NRC conducted an expert elicitation concerning potential future climate conditions at Yucca Mountain (DeW93). According to the experts' consensus forecast, temperatures in the area would cycle by about 2-3 degrees centigrade, and precipitation would also vary, not necessarily in direct correlation with temperature, with maximum precipitation being about twice the present annual rates.

The key issue for repository performance assessments is the quantity of water that infiltrates the repository and causes release and transport of radioactive waste materials. DOE's performance assessments have established that the rate of water infiltration to the repository is a parameter of critical importance to performance, and quite possibly the single most important parameter. The relationship between infiltration rate and precipitation rate has not been established, however.

A simple, direct relationship between precipitation rate and infiltration rate may not exist. Even if it does, the relationship may be difficult to establish reliably because of the complexity of the Yucca Mountain stratigraphy and the high evapotranspiration rate, which has been estimated to be about 1,000 mm/year (i.e., much greater than the present precipitation rate). DOE currently estimates an infiltration rate to the repository of 0.5 mm/year, with an uncertainty band that spans several orders of magnitude. The NAS panel observed in its report (NAS95) that changes in climate at the surface would probably have little effect on repository performance deep below ground.

7.5.3.5 Summary of Evidence for Stability

The information presented above, and in more detail in the cited reports, generally supports the NAS Panel's assertion that the long-term stability of the fundamental geologic regime at Yucca Mountain is on the order of one million years. The overall picture that emerges from the data and their interpretations is that the site and region had a highly dynamic period of volcanism and geology-creation long in the past, but these processes and events have matured into a system in which there are ongoing natural processes and events, but the magnitudes, frequencies, locations, and consequences of these phenomena are not significant to the long-term safety performance of a radioactive waste repository at the Yucca Mountain site.

The possible exception to this finding is the chance that on-going processes and events are producing differential changes to the geologic and hydrologic regimes that are currently unrecognized but could affect repository performance and potential radiation risks for affected populations. For example, on-going tectonic processes and movements could potentially have different effects on the geologic and hydrologic regimes near the surface and at depth.

At present, tectonic movement in the area varies by location but falls generally within the range of 4 to 10 mm/year (DOE95a). Over one million years, an annual tectonic movement of 10 mm/year will produce a total translation of location of about 5 miles. If all of the elements of the geologic and hydrologic regime important to repository performance and dose estimation do not move in space and time as an intact block, differential movement could invalidate results of performance and exposure assessments. The effects of differential movements and their consequences might, however, be insignificant in comparison with uncertainties associated with other factors involved in repository performance and dose assessments. The potential for differential movement and its consequences have not yet been addressed.

7.5.4 Perspective on the Significance of Stability of the Geologic Regime

A judgment that the geologic regime at Yucca Mountain will be stable for a period of about one million years enables confidence in the results of model-based assessments of the effects of natural processes and events over that time frame on repository performance and potential human exposure. Long-term natural phenomena may not, however, control repository performance or uncertainties in performance assessment results. Uncertainties in other factors involved in performance projections may ultimately control the reliability of the projections. A finding, with a high degree of certainty, that geologic stability can be expected for periods up to one million years shifts the burden of characterization of uncertainty in input data and assessment results from repository conditions to biosphere conditions.

The existence of long-term geologic stability can assure reliable estimation of long-term peak doses only if stability-related issues are confirmed to dominate repository performance and dose potential, and numerical values of relevant parameters have been established with confidence. As discussed in Section 7.5.3, DOE's total system performance assessments to date indicate that the rate of infiltration of water to the repository, and the dilution and dispersion characteristics of ground water containing radioactive contamination released from the repository, are the dominant factors in repository performance and dose assessment.

These findings suggest that geologic stability is not significant unless it affects these water-flow parameters (e.g., through climate change or differential tectonic displacement). However, the scope of DOE's performance assessments for a million-year period is to date highly limited. In addition, DOE carefully notes (Chapters 9 and 10, DOE95c) that the approach to dose estimation used in its TSPA evaluations does not correspond to that considered by the NAS Panel. Overall, DOE's performance assessments to date have not attempted to establish a perspective on geologic stability and other factors that might affect repository performance and radiation doses for a million-year time frame.

In accord with EPA's previously applicable 40 CFR Part 191 regulations, DOE's performance assessments to date for Yucca Mountain have emphasized release of nuclides from the repository over a 10,000-year time frame. EPA had selected this time frame because it brought into consideration long-term natural processes and events such as seismicity, but avoided the large uncertainties believed to be associated with longer time frames. The generic 40 CFR 191 rule had to accommodate the possibility that candidate disposal sites that do not exhibit long-term geologic stability would be considered.

Experience in characterizing repository performance for the 10,000-year time frame (DOE94a, DOE95c) has now shown that this time period requires characterization of repository conditions at, or near, the maximum of transient conditions evolving from the elevated temperatures and temperature gradients associated with the heat emission from the waste and the engineered design features of the repository. The 10,000-year time period essentially corresponds to the time of maximum uncertainty concerning the repository design factors important to waste isolation and safety performance. Under these circumstances, adverse natural processes and events would have to be highly potent (e.g., high intensity and frequency of seismic events) to have a significant or dominant role in repository performance and regulatory compliance.

The 10,000-year time frame is just one percent of the million-year time period for geologic stability. Within the initial 10,000 years, the thermal pulse stemming from heat emissions from the waste will have peaked and disappeared, as will have the thermal and chemical gradients that drive corrosion of the waste canisters and mobilization of waste radioactivity. Technical factors associated with repository design features that dominate performance issues for periods up to about 10,000 years therefore can be expected to become unimportant to regulatory compliance if it is to be evaluated at the time of peak dose, which occurs after 10,000 years but before 1,000,000 years. The initial (up to 10,000 years) repository conditions are important to long-term performance only if it can be shown that the long-term "ore-body" conditions important to nuclide release and far-future human exposure depend significantly on detailed design and thermal pulse conditions in the first 10,000 years. The DOE performance assessment report by Intera, Inc. (DOE94d) states that variations in assumptions and conditions for package degradation produce less than 20 percent variation in results for a 10,000-year assessment period and less than 10 percent variation in results for a 100,000 year period. Supplemental calculations in DOE94f show that peak doses and releases at the accessible environment boundary over a million-year period are generally unaffected by waste package lifetimes up to 100,000 years.

If the actual rate of infiltration of water to the repository is 0.5 mm/year, as estimated by DOE (DOE95c), the total quantity of water entering the repository each year is sufficient, under principles of mass action, to oxidize by corrosion the entire mass of metal in all waste canister walls under DOE's multipurpose canister emplacement concept. Actual rates and duration of degradation by corrosion will, of course, depend on many factors, as discussed in DOE's TSPA-95 assessments (DOE95c). It is apparent, however, that completion of oxidation of canister and spent fuel materials will not be constrained by water infiltration quantities. The actual duration over which corrosion will occur will depend on the interception of canisters by the infiltrating waters and other rate processes under evaluation by DOE.

In approximately 10,000 years, the repository will become essentially an isothermal system, at ambient temperature, of oxides of the canister and waste materials originally emplaced in the repository. In other words, after only about one percent of the elapsed time for a regulatory period of 1,000,000 years, the transitional processes associated with engineered features and heat-emission characteristics of the repository will be essentially complete. For purposes of performance assessment, the physical state of the repository at that time will serve as the initial condition for assessment of repository performance and dose assessment under conditions of geologic stability for a period of 1,000,000 years.

As indicated above, three periods of repository conditions in the future can be characterized, with geologic stability being maintained throughout all three. In the first, short-term period, lasting about 1,000 to 10,000 years, the repository is characterized by intact waste canisters, high temperatures and temperature gradients serving as driving forces for change such as chemical reaction, and the presence of short-lived and long-lived radioactivity retained in the canisters. Infiltrating water may or may not contact the canisters.

In the transition period, with a duration between 1,000 and 10,000 years, gradients are diminishing or gone and the engineered features of the repository are degrading. Canisters are corroding, and radioactivity is long-lived; some of the waste radioactivity is released from the canisters, but most is retained within the repository. Infiltrating water contacts and transports radioactive waste. Repository markers are in place, but human intrusion could occur.

In the long-term period, from 10,000 to 1,000,000 years, the repository is an isothermal ore body of the oxides of waste-package materials at ambient conditions. Infiltrating water seeps through the bed of oxides and transports long-lived radioactivity to the environment, where the radioactive contamination is diluted and dispersed by ground water flow processes. There is no human control, and there is potential for inadvertent human intrusion of the repository. If intrusion does occur, it might not be noticed because all engineered features of the repository are degraded.

Climatic cycles will occur throughout all three periods. Average temperatures will cycle by about 2 to 3 degrees centigrade, and there will be periods when precipitation will be twice present levels. The variations in climate and precipitation will not affect repository performance, and the potential for anthropogenic impacts on climate will be highest in the near term. If ground water is mined, human populations in the area will peak in the near term and diminish in the long term. If a policy of sustainable development is maintained, human populations will be maintained at levels consistent with acceptable ground water extraction rates. Anticipation of long-term stability of the geologic regime provides a basis for predicting human activities in the area.

7.6 CURRENT PERFORMANCE ASSESSMENTS

7.6.1 Overview of DOE Total System Performance Assessment (TSPA)

The Department of Energy's Total System Performance Assessment (TSPA) process is an iterative, comprehensive evaluation of factors relevant to the waste isolation performance of a possible Yucca Mountain radioactive waste repository. These factors include processes, parameters, events, and conditions that may affect the potential magnitude of the release of radionuclides to the accessible environment and the temporal or spatial variations of the radionuclide release.

The DOE has to date reported TSPA evaluations performed in 1991, 1993, and 1995. The results of these evaluations were compiled in comprehensive technical documents made available to the scientific and regulatory communities, as well as to the public at large. These documents discuss a wide range of technical issues associated with performance assessment for a potential Yucca Mountain repository. Topics involved in the assessments include site features (geological, meteorological, biological, etc.), repository design features, dynamics of repository performance within a complex environmental system, the workings of each identified important sub-system, and the uncertainties associated with repository performance evaluations both on the individual-component and the overall-system level.

The ultimate objective of the TSPA process is to identify, quantify, and document factors affecting repository performance relevant to compliance with regulatory requirements. Beyond the regulatory aspects of the performance assessments, TSPA is a means to guide and document progress in selecting the repository system design. The process of iterative total system analysis helps to focus subsequent research and evaluation on the key aspects of the repository system that are both significant to the release of radionuclides to the accessible environment and sources of uncertainty in the TSPA results.

The DOE TSPA efforts reported to date are documented in DOE92, DOE93, DOE94a, DOE94d and DOE95c. The basis for the TSPA process was provided by the PACE-90 project (DOE91), a hydrologic flow and transport modeling exercise for Yucca Mountain that built upon previous fluid flow modeling results such as COVE2A (SNL92) and HYDROCOIN (SNL90). PACE-90 initiated the TSPA process and established the basis for future activities.

PACE-90 was a numerical modeling project used to simulate ground water flow and aqueous transport at Yucca Mountain. Specifically, the goal of PACE-90 was to calculate the expected performance of the possible repository over a period of 100,000 years. A team including Sandia National Laboratories, Pacific Northwest Laboratory, and Los Alamos National Laboratory was assembled to establish one- and two-dimensional flow models to make performance evaluations. Modeled parameters included: 0.01 mm/year percolation flux, two different water/waste contact models, four different radionuclides, and a 19-layer unsaturated zone stratigraphic column. Ground truth data were sparse at the time, and several factors used in subsequent studies were not modeled (e.g., saturated zone flow, fracture-induced permeability, waste package cathodic protection, gaseous release, and thermal effects). In spite of its limited scope, this study established a benchmark for future analyses. The study concentrated on the role of hydrology in repository performance. The work evaluated the amount, location, and movement of water in the repository system and identified the hydrologic regime as critically important to the performance of the repository.

In 1991, the DOE used the flow simulation results of PACE-90 as a basis for the first of three total system performance assessments, TSPA-1991 (DOE92). In this iteration, the goal was to improve upon the techniques used to evaluate repository performance. Several new ideas were integrated into TSPA-91 that extended its scope beyond that of PACE-90. New modeling concepts established in TSPA-91 included:

External Disturbances: Both human intrusion and volcanic activity were modeled in terms of their impact on hydrothermal conditions within the repository.

Stochastic Modeling: Hydrogeologic parameters were represented as probability distributions developed from site and laboratory data.

Additional Radionuclides: A greater number of nuclides were considered in the analysis, in terms of potential contributing effects to final dose and cumulative release estimations.

Analysis of Gaseous Carbon-14: This non-aqueous mode of contamination release was first modeled.

Saturated Zone Modeling: Radionuclide transport through the saturated repository sub-strata was modeled.

Accessible Environment Modeling: The concept of defining the accessible environment five kilometers from the repository boundary was established.

Climatic Variations: Changes in climatic conditions were simulated by modeling a range of percolation flux rates.

Fracture Flow: Aqueous flow was analyzed using two different conceptual models in order to evaluate the relative effects of fracture and matrix flow.

The 1993 TSPA continued to increase the realism of the numerical models used in the performance evaluations. The stated objective of TSPA-93 was to assess the repository's ability to perform in compliance with applicable regulations, and to support a license application for construction and maintenance of the repository. Within this framework, three goals were established. The first goal was to provide participants in the Yucca Mountain Project with feedback regarding the significance of design and site characterization information important to regulatory compliance. The second goal was to promote more complete and accurate technical assessments of repository performance in relation to the repository design. Assessments were performed involving development, refinement, and testing of mathematical models for physical processes, designs, and events that could affect the performance of the repository. The third goal was to involve a broad technical community in the preparation of the TSPA.

TSPA-93 differed from the 1991 TSPA iteration in several important respects. New detailed modeling was included (three-dimensional geostatistical models of stratigraphy, three-dimensional modeling of the saturated zone, and repository thermal effects); a phenomenological source term was developed; climate changes extrapolated from the paleoclimatic record were used; various repository designs with different containers were considered; and different thermal loading cases were implemented.

Two different conceptual models of the unsaturated zone are considered in TSPA-93: the Composite-Porosity model, which assumes that flow is shared between the rock matrix and the fractures because of capillary forces (pressure equilibrium between the matrix and fracture flows), and the Weeps model, which assumes that water flows in locally saturated fractures with no matrix/fracture interaction. Both of these conceptualizations are idealized, and reality is probably somewhere in between.

The basic difference between the two conceptual models is that the Composite model assumes that ground water percolates slowly and uniformly through the mountain, whereas the Weeps model allows ground water to flow in episodic pulses through the fractures. In the Weeps model, no interaction is allowed between the tuff matrix and the flow in the fractures. In the Composite-Porosity model, flow is allowed in both the matrix and the fractures and the flow is completely coupled. The Weeps model also assumes that the only contact of water with the wastes is at discrete fracture interception points. If a fracture passes through the repository without contacting a container, there is no effect. It is assumed that containers not contacted by fractures do not corrode. If containers do fail, releases are assumed to be controlled by diffusion of dissolved radionuclides unless the container is subsequently contacted by a fracture.

Significance to Performance Assessment

- If the Weeps model conceptualization is correct, Yucca Mountain acts as a sieve, offering waste containers little or no protection from fast moving streams of water.
- The Weeps conceptual model places emphasis on a knowledge of fracture properties. The fracture parameters required for the TSPA-93 flow and transport models are frequency, spacing, porosity, air entry parameter, aperture, angle/orientation and hydraulic conductivity. Of these parameters, only two—fracture frequency and orientation—are available from actual measurements of down-hole properties. The remaining parameters are derived by assuming that the fractures can be modeled as parallel plates.
- The potential for lateral flow within the repository also becomes a greater concern with the Weeps conceptualization, in that if lateral flow is shown to occur, the current conceptualization of discrete contact points will no longer be valid, as significant releases could occur with only a few fractures intercepting the repository.
- The Weeps model results in an assumption of nearly negligible ground water travel time in the unsaturated zone, thereby placing a significant emphasis on flow and transport parameters in the saturated zone.

In the 1993 TSPA the term "infiltration" is used to describe the ground water inflow near the surface. "Percolation" is used when discussing ground water flow at depth through the unsaturated zone. The amount of "recharge" is the percolation rate over a given time period and region. The percolation rate is one of the most critical parameters in the TSPA, it also has the highest uncertainty. Because of the lack of quantitative data and the inherent uncertainty in determining future ground water flow rates, the strategy used in the 1993 TSPA is to determine likely past climates and percolation rates and extrapolate them into the future.

Sensitivity analyses showed that the Composite-Porosity aqueous-release results were extremely sensitive to percolation flux, with flux accounting for 90 percent of the variability in the results.

Significance to Performance Assessment

- One conceptual model of flow through the unsaturated zone (i.e., Weeps model) assumes that all infiltration flows through the fractures. This assumption has 3 major effects: 1) faster transit times through the unsaturated zone; 2) greater volumes of water reaching the waste package and therefore greater releases; 3) larger radionuclide mass-fluxes will reach the saturated zone and therefore dilution in the saturated zone will be less effective.

- The amount of percolation will also affect the waste container release scenarios, since higher percolation rates will increase waste-container contact times enhancing corrosion and also promote advective releases over the slower diffusive releases.

- The transit time through the unsaturated zone will dictate which radionuclides drive the risk. Slow transit times allow radioactive decay to decrease concentrations of relatively short-lived radionuclides.

- Higher percolation rates will be more conducive to allowing lateral flow in the unsaturated zone as well as in the repository drifts. If lateral flow occurs in the repository, water will be in contact with more waste and will release greater quantities of radionuclides.

Two performance measures were considered as part of TSPA-93: a normalized cumulative radioactive release, as defined by the EPA in 40 CFR 191.13, and radiation dose to a maximally exposed person. These measures were applied to time periods of both 10,000 and 1,000,000 years.

In the 1995 TSPA, the DOE continued its efforts to use representative data-based parameter values as current information allowed, and to predict the performance of the natural and engineered components of the system. An explicit TSPA-95 goal was to place bounding or conservative assumptions on processes that were not well understood in order to make the overall predicted performance assessment results more extreme than would be the case if better and more complete information were available. TSPA-95 also emphasized identifying and quantifying the components of the repository system that contributed significantly to the uncertainty of the assessment and were important to the performance of the repository.

Four specific goals were applied in TSPA-95: 1) utilization of the best models available; 2) incorporation of recent design information; 3) utilization of recent site information; and 4) evaluation of release and dose performance measure. As in 1993, alternative measures of system performance (cumulative release and maximum dose) were used. Similarly, the 10,000-year and 1,000,000-year time frames defined in 1993 also were adopted.

Compared with the TSPA-93 studies, the TSPA-95 evaluations used improved and more realistic models in the following areas:

> *Drift scale thermal-hydrologic environment:* More comprehensive and representative models of the humidity and temperature in the drift-scale (near-field) environment were employed.
>
> *Waste Package Degradation:* Analyses of various mechanisms associated with the corrosion of the waste packages were performed.
>
> *Near-field Unsaturated Zone Flux:* Models describing the transport of water in the near-field engineered barrier system were incorporated.
>
> *Unsaturated Zone Flow:* The strata in which the proposed repository is to be built are not saturated with ground water. The flow of water in these strata was modeled.

As a result of previous TSPA results, two topics were omitted from TSPA-95 work because DOE deemed them statistically insignificant in terms of final performance assessment evaluation. These topics were the disruptive events such as human intervention and volcanism and the gaseous-phase transport of radionuclides in the unsaturated zone.

7.6.2 TSPA Models Used by DOE

The TSPA iterative process involves the development, use, and interpretation of mathematical models to predict various performance features of the proposed repository. This section of the BID describes the models and modeling strategy used by DOE to date.

The goal of TSPA modeling is to predict the state of the repository and surrounding environment 10,000 years and 1,000,000 years in the future. The TSPA uses various techniques to predict the future conditions of Yucca Mountain. The predictive models attempt to embody the state of knowledge of the physical processes active at the site and to provide a rigorous methodology for characterizing future conditions.

To predict the long-term behavior of the possible repository at Yucca Mountain, numerical simulation techniques are used to model the processes identified as significant to the performance of the repository. These simulations were developed using computer programs designed to represent the current understanding of the physical dynamics of all active, relevant, and significant phenomena controlling the performance of the repository and surrounding media. Numerical models are mathematical constructs, designed by scientists and engineers, which reflect the state of knowledge of the various constituent processes.

The accuracy and usefulness of computational models used to evaluate complex systems depend on two essential factors: first, how well the system's physical dynamics are understood; and second, the use of simulation techniques that can accurately and appropriately predict the future.

The means by which predictive computational models are constructed, implemented, and utilized also affect their accuracy and usefulness. These factors include: the overall strategy used to quantify conditions controlled by imperfectly understood processes; the techniques used to define the uncertainty of the predictions; the effects of the computational techniques themselves on the predictive process; the specific methods by which computational numerical techniques are implemented; and the degree and effect of possible errors in the construction of the computational component of the predictive model. DOE's TSPA process has addressed all of these issues.

7.6.2.1 Information Flow in DOE's TSPA Methodology

In general, the flow of information within TSPA-95 followed four steps:

1. Laboratory and field tests were developed and executed. Data from these tests were gathered, interpreted, and documented.

2. Process-level models were constructed, tested, and refined to create a working numerical simulation of an individual process. The results were theoretical models, grounded in observation, able to predict specific aspects of waste isolation.

3. Process-level models, describing how various aspects of the containment system work, were *abstracted* to yield response surfaces, tabulated parametric relationships, or functional relationships. Abstractions, or generalizations, were required to facilitate integration of interconnected process level phenomena, and to construct the overall complex repository system comprised of individual process-level components.

4. Once overall numerical constructs were established, the total system performance was evaluated.

Not all aspects of the TSPA require the process-level modeling and abstraction steps. In some cases, information developed from laboratory or field testing is used directly in performance analyses. Examples include the alteration/dissolution rate of the waste form, radionuclide solubility values, and radionuclide sorption rates. In general, processes and parameters that are not static, but vary over time, require predictive models. These include saturated zone flow and waste package corrosion.

In cases where process-level models are needed, the results of the model simulations are used to define relationships between the parameters used in the construction of the process-level model and the required input to the total system performance assessment calculations. The abstraction step is needed to simplify the process, with minimal loss of predictive capability.

This step facilitates the inclusion of the process model within the final assessment where repetitive simulations, or stochastic realizations, of each constituent process are required.

Repository performance assessments are calculated through the use of probabilistic modeling techniques. In these techniques, abstracted models and parameter probability distributions derived from process-level models are input into the total system performance assessment model. The Repository Integration Program (RIP), developed by Golder Associates (GOL94) for the evaluation of the repository at Yucca Mountain, was used in TSPA-95. RIP (discussed in more detail below) is a stochastic prediction computer program used to predict the total repository system or subsystem performance based on the uncertainty and variability of input distributions (derived from process-level analysis).

The process by which the repository performance is evaluated has evolved during the history of the TSPA process. In 1993, the risk assessment technology followed a five-step procedure: 1) develop repository design scenarios (thermal loading, canister configuration, un-saturate flow models, etc.); 2) develop models for the important constituent components of the studied scenario; 3) estimate all important parameters and associated probability distributions; 4) make performance calculations for models, parameter values, and uncertainty estimates; and 5) interpret the results.

In 1993, the performance evaluations were still in the early stages as described in the Yucca Mountain Site Characterization Plan (SCP), published five years earlier (DOE88). The features, events, and processes (FEPs) characterizing the scenarios required to initiate the assessment process were still being formulated. TSPA-93 concluded that significant additional work was still required.

Several groups contributed to the 1993 TSPA, as well as conducting evaluations of their own. The Electric Power Research Institute (EPRI) (EPR90, EPR92), the NRC (NRC92), Golder Associates (MLR92), Pacific Northwest Laboratory (PNL) (DOE93), and INTERA (DOE93) all produced preliminary design performance assessments. A consensus regarding the best approach to site evaluation did not exist, and different organizations applied different methods with different emphases. EPRI, Golder, and INTERA employed techniques using highly abstracted system performance models. PNL utilized multi-dimensional models and concentrated on flow and transport issues, and the NRC used techniques similar to those applied to the Waste Isolation Pilot Plant.

One striking characteristic of these earlier studies was the degree to which the sub-process models were abstracted in order to facilitate statistical analysis of system performance uncertainty. At the earliest stages of the evaluation process, it was recognized that important parameters that control the performance of the repository were imperfectly understood. It was also known that the required performance uncertainty bounds, associated with performance analyses in which ill-defined parametric values are used, can be determined if a stochastic approach is taken. This cause-and-effect relationship between imperfectly known key performance assessment parameters and the use of stochastic modeling procedures necessitated the use of probability distributions for describing input parameters.

7.6.2.2 Uncertainty in the DOE TSPA Methodology

As shown in TSPA-93, uncertainty in DOE's TSPA methods and results originate from a variety of sources.

Measurement Errors: All measurements made in laboratory or field experiments contain some form of error, introducing uncertainty into the actual values of the measured quantity. In natural systems, the uncertainty is even greater because measurements are generally made at spatial and temporal scales much smaller than the distance and times of interest.

Spatial Variability: The physical systems that need to be modeled to produce useful performance evaluations range from the mm level (corrosion pitting) to the km level (aqueous saturated flow). Since the media in which a process is active cannot be completely and accurately characterized, spatial variability is not completely accounted for in numerical models, and uncertainty is introduced.

Temporal Variability: A significant goal of the TSPA performance process is to predict the future conditions of the planned repository. However, the predictive process requires knowledge of the future conditions of important driving parameters. For example, future climatic variations in Nevada may directly influence repository performance. Imperfect knowledge of these temporal aspects of the TSPA process introduces uncertainty.

Model Errors: The accuracy with which computer models simulate a physical process depends on several factors, including the state-of-knowledge of the process, the methods used to transfer that knowledge into a tractable computer program, the nature of incorporated simplifying assumptions, and possible human error in the creation and use of the model. Because of these factors, the use of models introduces uncertainty into the TSPA process.

Uncertainties in sub-process performance, associated with the factors listed above, may not be significant in the overall assessment of repository performance. Extreme uncertainty may exist in areas that are not significant in the performance of the repository. Large variations in the estimates of parameters controlling such processes may not influence the final results of the evaluation. In contrast, some factors are critical to accurate assessment of overall repository performance assessment, and small uncertainties in these areas can significantly affect performance predictions. This variation in importance of models is due to the variation in the sensitivity of the overall performance to specific factors.

The methods for addressing uncertainty were of major concern to the TSPAs, particularly in 1993 and 1995. In 1993, three techniques were utilized.

PDF Development: Simple scalar parameter estimates were replaced with probability distribution functions (PDF) that explicitly describe the expected range, and associated likelihood, of each parameter value. The PDFs were developed either through experimentation, sampling, or through quantification of expert opinion.

Alternative Models: If the conceptual framework of a model is in question, then highly accurate input parameters may still yield highly uncertain results. In these cases, alternative conceptual models are required. For example, TSPA-93 considered two conceptual models for unsaturated ground water flow, the Composite-Porosity model and the Weeps model. Each model has its own uncertainty characteristics. The uncertainties in conceptual models at the process-level compounds the uncertainty of total system performance.

Repetitive Realizations: When several sub-processes are linked together mathematically to form a numerical simulation for the entire repository system, the propagation of various types and degrees of uncertainty associated with the overall performance assessment must be quantified. The DOE's TSPA has taken a stochastic approach to this problem. Stochastic techniques utilize observations from PDFs. They are used in the TSPA for repeated sampling of each PDF associated with a performance scenario. Each set of parameters sampled from the PDFs constitutes a statistical realization and is used to determine an individual estimate of the system performance. Repetitive realizations of the PDFs are used to establish the statistical likelihood of overall system performance in a rigorous way that incorporates the uncertainties of each PDF.

7.6.2.3 The Repository Integration Program (RIP) Code

In TSPA-95, the DOE used the Repository Integration Program (RIP) to perform site performance evaluations in conjunction with detailed process-level models. The RIP was specifically developed by Golder Associates in order to evaluate the proposed Yucca Mountain repository (DOE95c; MLR92, GOL94). The program has subsequently been applied to proposed radioactive waste disposal sites in the United States and abroad. RIP was used for evaluation at the WIPP site in New Mexico and for low-level waste disposal in New York.

RIP contains four major components: 1) a waste package behavior and radioactive release component; 2) a radionuclide transport pathways component; 3) a disruptive events component; and 4) a biosphere dose/risk component.

Waste Package Behavior: Used to quantify waste package behavior and radionuclide release, this component requires a description of the radionuclide inventory, the near-field environmental conditions, and estimates of parameters controlling simulated container failure. Package failure rates are used in conjunction with matrix alteration and dissolution rates to compute the rate at which radionuclides are exposed. RIP then calculates the rate of mass transfer out and away from the waste packages, producing time histories of each modeled radionuclide.

Radionuclide Transport: Radionuclide transport is modeled in RIP, for both the near and far field, in a probabilistic mode. RIP utilizes a phenomenological approach to describe, rather than explain, the transport system. The algorithm is based on user-defined pathways that reflect the major features of the hydrologic system. The pathways are used to represent large-scale heterogeneity in the system, such as geologic structures and formation-scale hydrostratigraphy. The concept of the breakthrough curve is established in the transport mode, which refers to a cumulative probability distribution that describes the radionuclide travel time along a defined pathway.

Disruptive Events: Disruptive events are defined as discrete occurrences that have some quantifiable effect on the processes described by the other two component models. Examples include volcanism, faulting, and human intrusion. The user defines the occurrence rate of such events and establishes parameters describing the characteristics of the event. User-defined probability distributions describing the event consequences are subsequently incorporated into RIP analysis.

Biosphere Dose/Risk: The biosphere dose/risk module of RIP allows the user to define dose receptors in the biosphere. Receptors receive radiation doses from specified geosphere or biosphere pathways. Concentrations in these pathways are converted into radiation doses based on user-defined factors.

Stochastic methods are used both to facilitate the performance evaluations and to quantify the sources of uncertainty within the performance analysis results. Linear regression techniques are used to determine which parameters used in the assessments are most influential. By determining the amount of variability in the results explained by each parameter, the factors most important in the modeling process are identified.

7.6.2.4 DOE Process-Level Codes

The DOE TSPA effort has developed a number of process-level codes that are used to provide input to the RIP code.

TOSPAC: Aqueous flow in the unsaturated rocks of the repository and surrounding environment is modeled by two different computer programs, depending on which conceptual model is used. For the composite porosity conceptual model, TOSPAC is utilized. This program numerically simulates the pressure equilibrium between matrix and fracture flows and solves the flow and transport equations in one spatial dimension.

Within the saturated zone, radionuclide transport parameters are calculated by TOSPAC under the composite porosity conceptual model. Aqueous release predictions are made under the one-dimension model assumption. In this model, water velocity is not calculated, but rather entered as a user-defined input parameter.

STAFF3D: This program provides TOSPAC with an estimate of the expected range of values for water velocity and dispersivity within the saturated zone. Three-dimensional simulations of saturated zone and transport are made and subsequently abstracted for input into TOSPAC.

WEEPTSA: This program models unsaturated flow under a weeps conceptual model, where the assumption is that water flows in locally saturated fractures with no matrix/fracture interaction. The program is probabilistic in nature; its basic calculation is the probability that a flowing fracture contacts a waste container. Also, WEEPTSA performs preliminary analysis of the corrosion rates of the waste containers before passing its results to YMIN.

YMIN: Both programs that model fluid flow in the unsaturated zone (TOSPAC and WEEPTSA) are coupled with the radionuclide source program, YMIN. This program calculates container failure, waste form alteration, and radionuclide dissolution and release. YMIN is actually incorporated within both TOSPAC and WEEPTSA as a subroutine because the flow, source release, and transport routines share much information.

COYOTE: COYOTE constructs and solves a three-dimensional implementation of the analytical heat conduction equations utilizing a finite-element approach for the solving of non-linear partial differential equations. Container temperatures and fuel temperatures are determined with this procedure. The COYOTE container temperatures are thought to be more accurate at early times and the analytical temperature estimates more accurate at later times, so the estimates are merged to provide a composite container-temperature history. This curve is used by YMIN for modeling the temperature-dependent corrosion process.

TOPAZ: TOPAZ is used to estimate internal temperatures for waste containers described in the Site Characterization Plan. These temperature estimates are made through modeling of the detailed geometry of the fuel assemblies in the waste packages. The results are utilized by YMIN for modeling the temperature-dependent fuel-rod-cladding failure, spent fuel alteration, and radionuclide alteration.

ANSYS: For the multipurpose type containers, ANSYS is used to estimate internal temperatures. This commercial program also models the detailed geometry of the fuel assemblies in the waste packages.

V-TROUGH: V-TROUGH estimates the dryout fraction and dryout volume through calculation of the multiphase fluid flow coupled with heat transfer. These results, once abstracted, are used only in the weeps conceptual model calculations.

GASTSA: This program predicts the gaseous carbon-14 transport from the repository to the ground surface. This calculation is required under both the composite-porosity and the weeps conceptual models. GASTSA makes these estimates through a convolution integral which combines the source release of carbon-14 with the distribution of transport times. The calculations involve the two-dimensional simulation of coupled gas and heat flow, assuming 100 percent humidity to eliminate the need for simulating water flow as well. The carbon-14 source term comes from the results of the WEEPTSA/YMIN combination in the weeps conceptual model, and from the SRCTSA/YMIN combination in the composite porosity conceptual model.

SRCTSA: This program is a subset of TOSPAC used to generate the carbon-14 source term for the composite porosity conceptual model. SRCTSA contains only the components of TOSPAC necessary for the source term calculation and excludes the flow and transport calculations.

SISIMPDF: This program is a geostatistical routine used to generate the stratigraphic realization from known borehole data. Geostatistical methods are used to incorporate uncertainty regarding the unit-interface locations into the performance calculations. These calculations are required in the composite porosity conceptual model but not in the weeps model, which does not utilize stratigraphic information.

DRILL: In addition to release calculations under a "nominal" case, the TSPA process calculates the potential effects of intrusive activity. DRILL is used to calculate release estimates from human intrusion through exploratory drilling. This computer program calculates the probability that a drill will intercept a waste container or contaminated rock outside a waste container, and then uses a probabilistic calculation to determine the amount of radioactive waste brought to the surface by the drilling. DRILL also contains its own sampling routines and can calculate releases for a large number of drill realizations efficiently. DRILL contains its own random-sampling routine and also uses LHS (described below), converted into a callable subroutine, for sampling.

VOLCAN: Like DRILL, VOLCAN is used in predicting the effects of a disturbed repository. This basaltic-volcanism simulation program is used to modify the estimates of waste container degradation and radionuclide release resulting from high temperatures and corrosive gases near a dike intrusion, assumed not to come in actual contact with the waste containers. VOLCAN calculates the number of waste containers affected by a dike intrusion. The effect modeled is a modification of the radionuclide release source term for a period of time after the intrusion. The actual release calculation is a composite-porosity, aqueous-release calculation that is not performed with VOLCAN. Neither gaseous releases nor weeps model releases are calculated. VOLCAN also calculates the releases from direct magma/waste-container interactions, as was done in TSPA-91. The code also performs Monte Carlo simulations of direct release due to volcanism.

ROCKTEMP: Used in tandem with VOLCAN, ROCKTEMP is used to model the temperature excursions caused by the volcanic intrusion of magma near the repository.

TSA: In order to introduce a probabilistic approach to uncertainty estimation, multiple realizations of various stand-alone programs are executed. Monte Carlo simulation of nominal releases is done with the total-system analyzer, TSA. TSA, developed to facilitate total system evaluation, is a script written in the UNIX C-shell language. As such, it provides a highly flexible framework used to establish a sequence of programs and routines to be executed for repository performance evaluation. TSA can control the execution of TOSPAC, WEEPTSA, SRCTSA, and/or GASTSA.

LHS: LHS is an implementation of the Latin-hypercube sampling method, which is used to generate the specific input realizations from input probability distributions (PDFs).

7.6.3 System Options Considered by DOE

DOE has considered a wide scope of issues and options throughout its TSPA process. This section summarizes the key modeling and design options considered to date.

7.6.3.1 Waste Inventory and Nuclide Content

The TSPA-93 evaluations used the assumption that a total of 70,000 MT of radioactive waste will be emplaced in the repository, consisting of 63,000 MT of spent fuel from commercial nuclear reactors and 7,000 MT of defense high-level waste. The spent fuel was considered to be aged 25 years, with burnups of 30,000 MWd/MTU for boiling-water reactor fuel and 40,000 MWd/MTU for pressurized-water reactor fuel. The defense waste was assumed to be in a vitrified (glass) form with many of its original heavy metal products removed.

The assumptions concerning waste inventories were refined and revised for TSPA-95. These evaluations assumed that the 63,000 MTHM inventory of commercial spent fuel would be made up of 40,785 MTHM of PWR spent fuel, with a burnup of 39,651 MWd/MTHM, and

22,210 MTHM of BWR spent fuel with a burnup of 31,186 MWd/MTHM. Thirty-year-old fuel was assumed. The burnup value for the defense high-level waste was assumed to be 10,000 MWd/MTHM, and the 7,000 MTHM-equivalent of this material was assumed to be emplaced in 14,000 containers.

The way this material is placed in the repository affects the temperature and temperature gradients in the engineered and natural features of the repository. This is referred to as the thermal loading of the facility. Thermal loading can be expressed in terms of areal mass-loading of the repository by the waste containers. Thermal loading conditions range from "low" (between 20 and 40 metric tonnes of uranium (MTU) per acre) to “high” (between 80 and 100 MTU per acre). TSPA-95 used the values of 25 and 83 MTU per acre.

7.6.3.2 Hydrologic Regime

The hydrologic regime plays a critical role in TSPA evaluations. The amount, distribution, and flow of water controls many aspects of radionuclide release. Important aspects of the hydrologic regime include: the infiltration of water through the stratigraphic column; the effects of water on the corrosion of waste containers; the aqueous transport of radionuclides through the engineered-barrier system; the unsaturated zone and the saturated zone aqueous transport of released radioactivity; and the concentrations of nuclides in the ground water of the biosphere.

Two water infiltration scenarios were considered, based on the work of Flint and Flint (FLI94). For the "low" rate scenario, the surficial infiltration over the extent of the potential repository was established at 0.02 mm/year, and assumed invariant with depth due to the predominant one-dimensional vertical flow. The "high" infiltration rate was set at 1.2 mm/year over the site and assumed to be uniformly distributed over the potential repository horizon. Within the infiltration range, which reflects possible precipitation variations in the future, six discrete cases were considered: 0.01, 0.02, 0.05, 0.5, 1.0, and 2.0 mm/yr.

Once the amount of water entering the system has been established, the modeling of aqueous flow next requires a model of stratigraphic transport parameters. The hydrogeologic data base in DOE95i was used as the source of matrix and fracture properties. Ten random sets of porosity and saturated conductivity and two van Genuchten parameters were established. The properties of the welded Tiva Canyon unit and non-welded Paintbrush unit, located above the potential repository, were set to their expected values, and the fracture properties were assumed constant in all units.

To facilitate the modeling of fluid flow through fractures, two conceptual models were used. The first is an equivalent-continuum model where liquid flow is initiated only after the bulk-liquid saturation exceeds a threshold value corresponding to full saturation of the matrix. The second model relaxes this assumption and initiates flow when the matrix liquid saturation is equal to or greater than a "satiated" matrix saturation.

Modeling based on the site-scale considerations discussed above results in estimates of the amount and distribution of water at the repository horizon. The drift-scale infiltration flux, the "average" percolation flux at the repository horizon, is distributed among waste packages to simulate the local drift-scale percolation flux conditions. The local percolation flux is then partitioned into a "dripping" fracture intercepting the drift and the rock matrix surrounding the drift. Variation in the partitioning of these factors was considered.

Within a specific site design consideration, and with the site-scale and drift-scale thermal and hydrologic considerations established, local (near-field) environmental conditions can be estimated. Near-field parameters characterized included: waste package surface temperature; average liquid saturation in the drift; and relative humidity at the surface of the drift.

7.6.3.3 Waste Package Degradation

The near-field conditions of water, chemistry, and temperature affect how the containers will perform over time. The packages must fail in order for water to enter and mobilize radionuclides prior to the dissolution of the waste. Thus, the TSPA requires a prediction of when, and how, the waste packages will degrade. Degradation is based on four factors: 1) waste package design (composition and construction); 2) repository design (thermal load, backfill, drift size); 3) near-field thermohydrologic regime; and 4) degradation characteristics of the waste package materials.

For the performance evaluation, it has been assumed that waste materials will be emplaced in robust waste containers. The waste container design concept has two or three layers of different metals, depending on the assumed thermal load. For spent fuel emplaced in the high thermal load configuration, a corrosion-allowance material (CAM), such as mild steel, has been proposed for the outer containment barrier and a corrosion-resistant material (CRM) such as Inconel 825 has been proposed and tested as an inner containment barrier. For the low thermal load, a moderately corrosion resistant material (MCRM), such as Monel 400, had been proposed as a containment barrier, added to the two-layer containment barrier used for the high thermal loading case. However, since adequate performance prediction data on MCRM were not available, TSPA-95 work proceeded under the assumption that no MCRM was used in the waste containment package. It was assumed in 1995 that spent fuel and DHLW packages were of the same design, with a corrosion-resistant inner barrier made of Alloy 825 and a corrosion-allowance outer barrier made of carbon steel. The containment barriers are separate from the stainless steel shells of the multi-purpose canisters (MPC) and the DHLW pour canisters, neither of which were considered as part of the waste package degradation study in the 1995 TSPA.

A large MPC, which would be about 5.7 m long and 1.8 m in diameter, contains either 21 pressurized water reactor (PWR) or 40 boiling water reactor (BWR) radioactive fuel assemblies. With the MPC concept, in-drift emplacement of the containers is necessary, due to requirements associated with operational factors, ease of retrieval, safety, and flexibility.

Three models of waste package degradation were considered:

> *Humid-Air Corrosion Model for the Corrosion-Allowance Outer Barrier:* Based on 16 years of analysis and 166 corrosion data points on a suite of cast iron and carbon steel containers, a corrosion and pitting model was developed. Pitting corrosion is represented by a pitting factor defined as the ratio of the maximum pit depth to the general corrosion depth for a specified exposure time. This is important because both general corrosion of the containers and pitting are expected to occur.
>
> *Aqueous Corrosion for the Corrosion-Allowance Outer Barrier:* A general model for aqueous corrosion was developed as a function of exposure time and temperature. The corrosion reaction and time-dependent terms were based on collected data. The aqueous pitting corrosion was modeled using the same approach as in the humid-air pitting corrosion model.

> *Aqueous Pitting Corrosion Model for the Corrosion-Resistant Inner Barrier:* The same stochastic waste package degradation model developed for TSPA-93 was used in TSPA-95. In this model, the corrosion-resistant inner barrier (Alloy 825) provides time-independent pit growth rates in aqueous conditions at 70 and 100 degrees C. Extrapolation techniques are used to find pit-growth-rates at other temperatures.

To determine when a waste package is breached, temperature and humidity profiles, developed using the flow models discussed above, are input into the waste-package degradation simulation sub-process. The simulation module provides a failure time for each waste package, which corresponds to the initiation time of radionuclide mobilization. Failure of a waste package is defined as having at least one pit penetrating through all barriers. The total number of pit penetrations at a given time yields the area available for transport of mobilized radionuclide through the waste package. The waste package failure time and pitting history are used as input to the EBS transport model.

7.6.3.4 Release from the Engineered-Barrier System (EBS)

Near-field environmental conditions affect processes and parameters such as waste-form dissolution, the solubility of the radionuclides in the aqueous phase in contact with the waste form, and the magnitude of both the advective and diffusive components of transport from the waste-form surface through the degraded waste package and the in-drift materials into the host rock. Waste-form dissolution rates have been empirically derived from laboratory experiments under a range of thermal and chemical conditions. The advective flux component of the transport was derived from the distribution of local percolation flux in excess of the saturated hydraulic conductivity of the host rock, and the diffusive flux component was derived from the hydrologic conditions within the drift materials.

Three conceptual models for EBS release were considered:

> *Drips-on-Waste-Form Model:* After initial pit-penetration has occurred, it is assumed that the near-field environmental conditions are immediately transferred inside the package (temperature, humidity, saturation, and the presence of drips). These conditions are used to predict radionuclide release from the EBS to the host rock. Advective release occurs at a rate proportional to the flow of dripping water in the drift, and diffusive release occurs at a rate proportional to the number of pits penetrating the container.

Drips-on-Waste-Container-Model: This more realistic model allows for partially intact waste containers to act as a release barrier. The model assumes only diffusive releases through the waste container, while simulating both advective and diffusive release from the EBS.

Capillary Barrier Model: This model was constructed to simulate the effects of emplacement of backfill materials to form a capillary barrier that would be designed to channel drips away from the waste package and underlying materials. If established, only diffusive releases from the failed waste package would occur.

Eight radionuclides were selected for consideration in the EBS release evaluations: carbon-14, cesium-135, nickel-59, neptunium-237, lead-210, radium-226, selenium-79, and technetium-99. These were selected on the basis of the criterion of a potential maximum release rate that exceeds 0.1 percent of the NRC total-release-rate limit. EBS release rates were calculated as a function of time for simulations utilizing the expected values of the stochastic distribution of various model parameters.

7.6.3.5 Near-Field Thermal Hydrology Models

Numerous processes affect the onset of corrosion, alteration/dissolution of the waste form, and release of radionuclides to the accessible environment. These processes can alter the distribution and movement of heat and/or water in the vicinity of the emplaced HLW; thus, the near-field thermal hydrology is most important in design choices.

In TSPA-95, drift scale thermohydrologic choices were evaluated to determine 1) waste package surface temperatures, and 2) water content in the drift and relative humidity (RH) at or near the waste package. Eight simulations were performed for the two areal mass loading cases of 83 and 25 MTU/Ac and for high and low water infiltration rates into the near field. Figures 7-28 through 7-30 depict the 25 MTU/Ac case, and Figures 7-31 through 7-33 show the 83 MTU/Ac case.

For the 83 MTU/Ac case, peak waste package surface temperatures occur within 10 to 20 years after waste emplacement and then decrease to about 60 degrees centigrade at 10,000 years. The relative humidity at the WP surface is inversely proportional to its surface temperature and directly proportional to the temperature in the host rock or backfill where a "dry-out front" exists; thus, at later time periods relative humidity in the near-field increases since WP temperatures are decreasing and the "dry-out front" nears the WP.

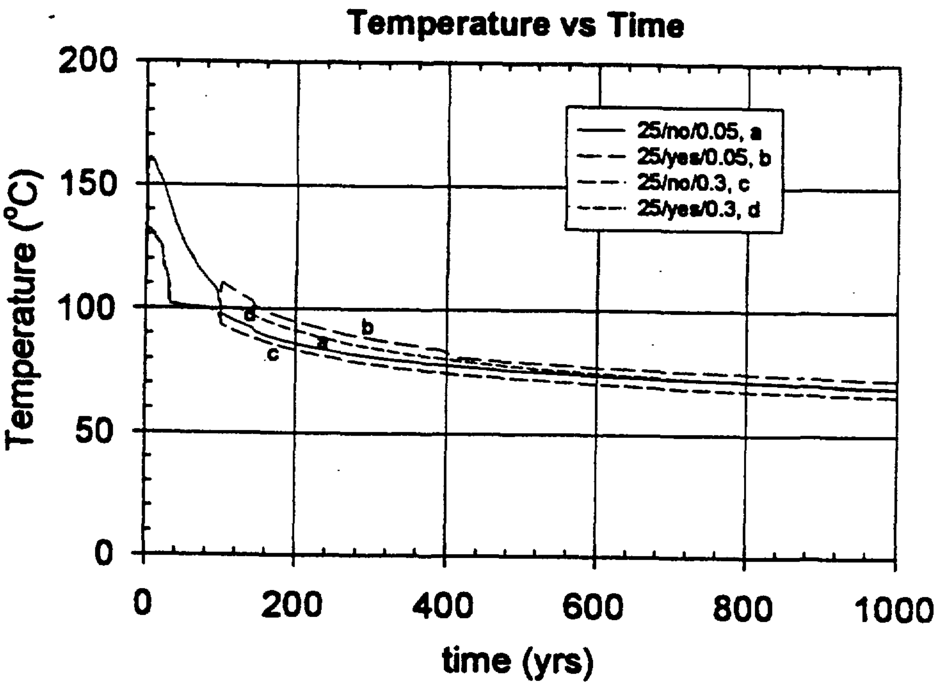


Figure 7-28. Waste Package Surface Temperature Predictions for 25 MTU/acre Case (Source: DOE95c)

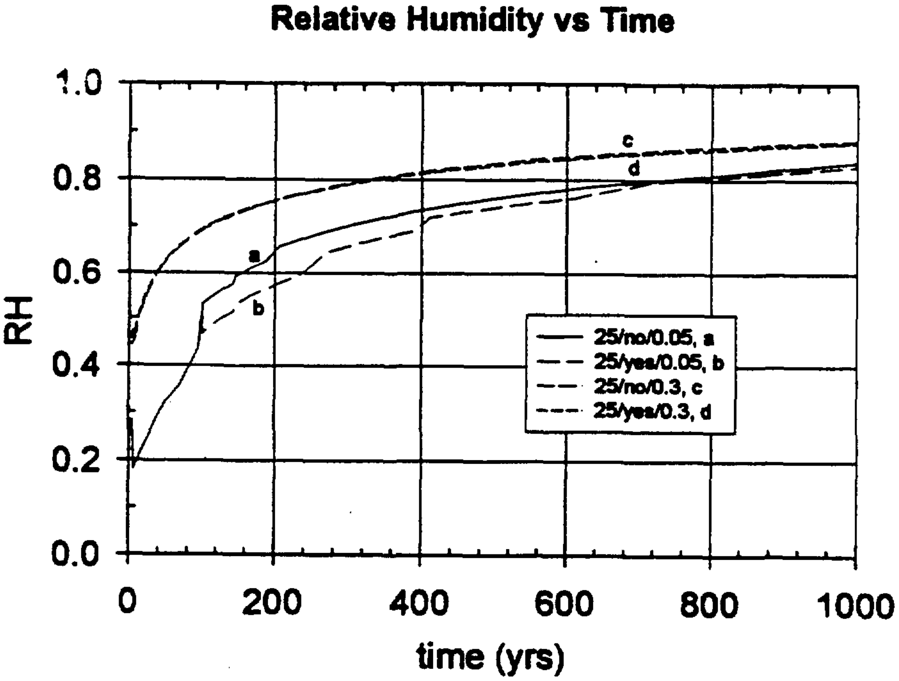


Figure 7-29. Relative Humidity Predictions for 25 MTU/acre Case (Source: DOE95c)

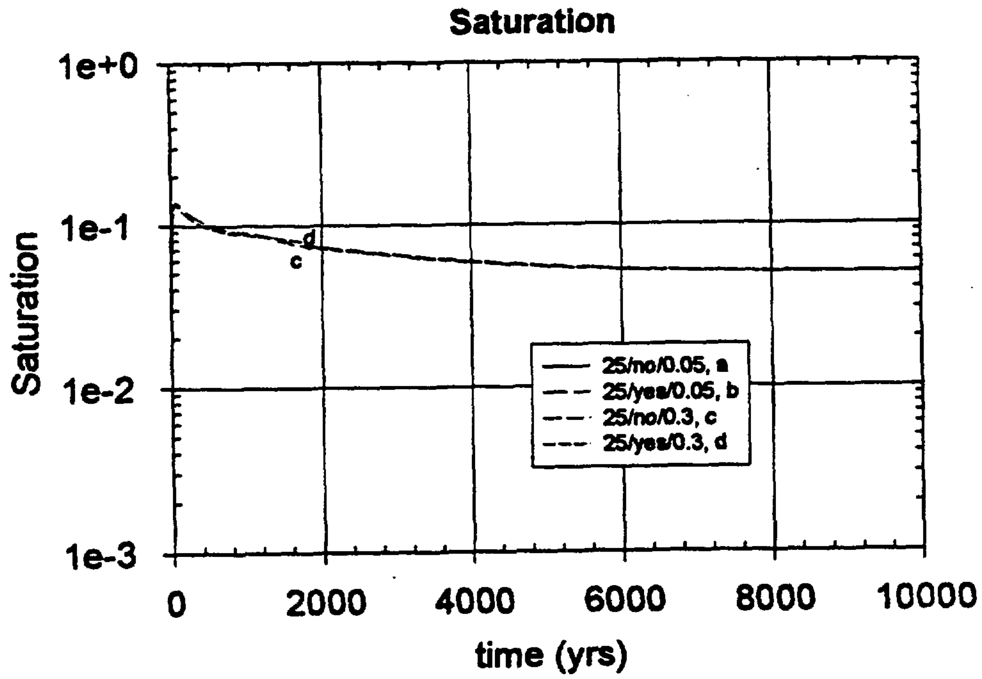


Figure 7-30. Predictions of Liquid Saturation within Backfill for 25 MTU/acre Case (Source: DOE95c)

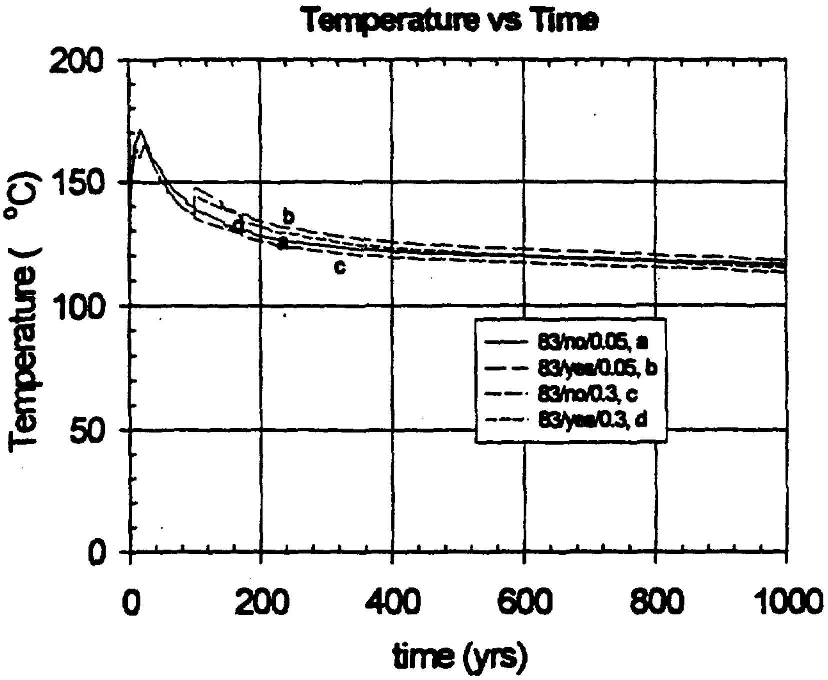


Figure 7-31. Waste Package Surface Temperature Predictions for 83 MTU/acre Case (Source: DOE95c)

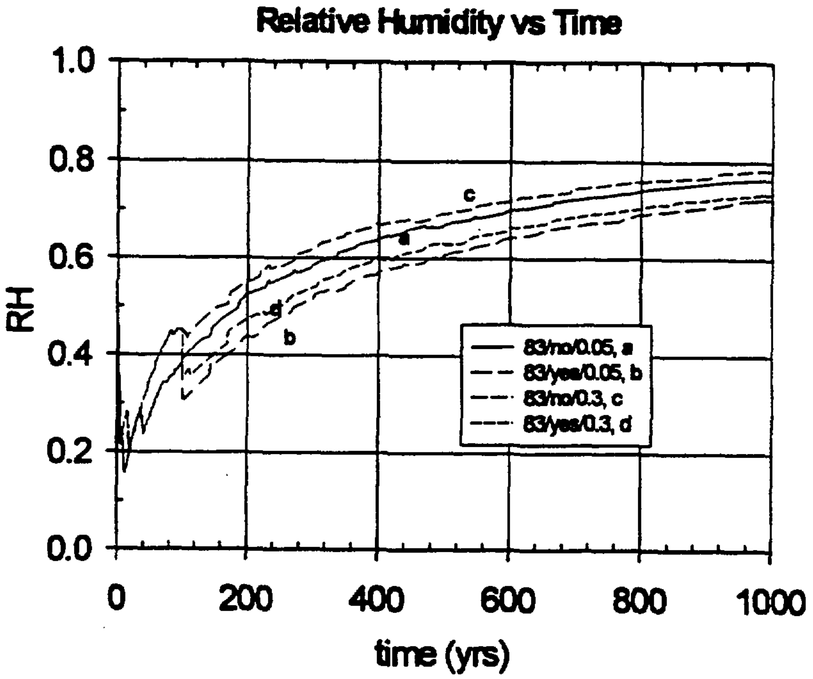


Figure 7-32. Relative Humidity Predictions for 83 MTU/acre Case
(Source: DOE95c)

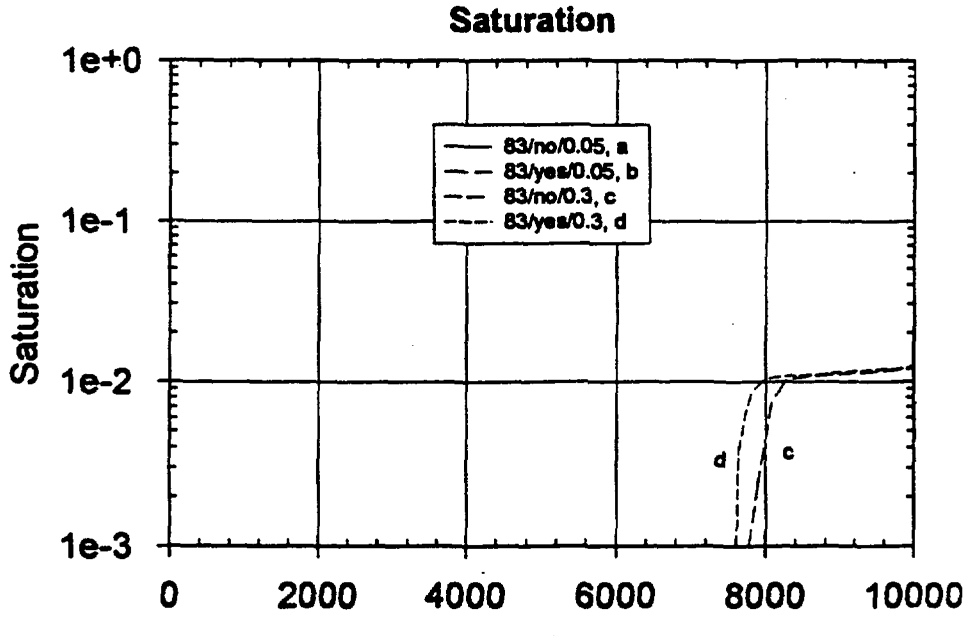


Figure 7-33. Predictions of Liquid Saturation within Drift for 83 MTU/acre Case
(Source: DOE95c)

The rate of water infiltration directly affects the waste packages, as does the backfill. Backfill reduces the relative humidity near the waste package by increasing WP temperature and acts as a capillary barrier to delay the rewetting process of the WP (see TSPA-95 Section 4.2.9). At high infiltration rates for the 83 MTU/Ac case, rewetting is predicted by DOE to occur within about 8,000 years (see DOE95c Figure 7-34).

This projection of repository conditions has been questioned by the state of Nevada (FRS96). The State alleges that there is a significant discrepancy between DOE models that predict return to high-humidity conditions in a few thousand years and others that predict such conditions only after tens of thousands of years. The State believes that the alleged discrepancies will not be adequately resolved by planned in-situ tests.

For the 25 MTU/Ac case, a wider drift spacing can be chosen so that the waste package spacing is smaller than for the 83 MTU/Ac case. Predictions in TSPA-95 are for a drift spacing of 45 m and a waste package spacing of 32 m. The results in this case show relative humidities in the near-field increasing faster than in the 83 MTU/Ac case. Maximum relative humidity within a 10,000 year period still is in the range of 0.90 to 0.95. With a low thermal loading, the amount of dry-out is small and rewetting occurs very early (see Figure 7-35).

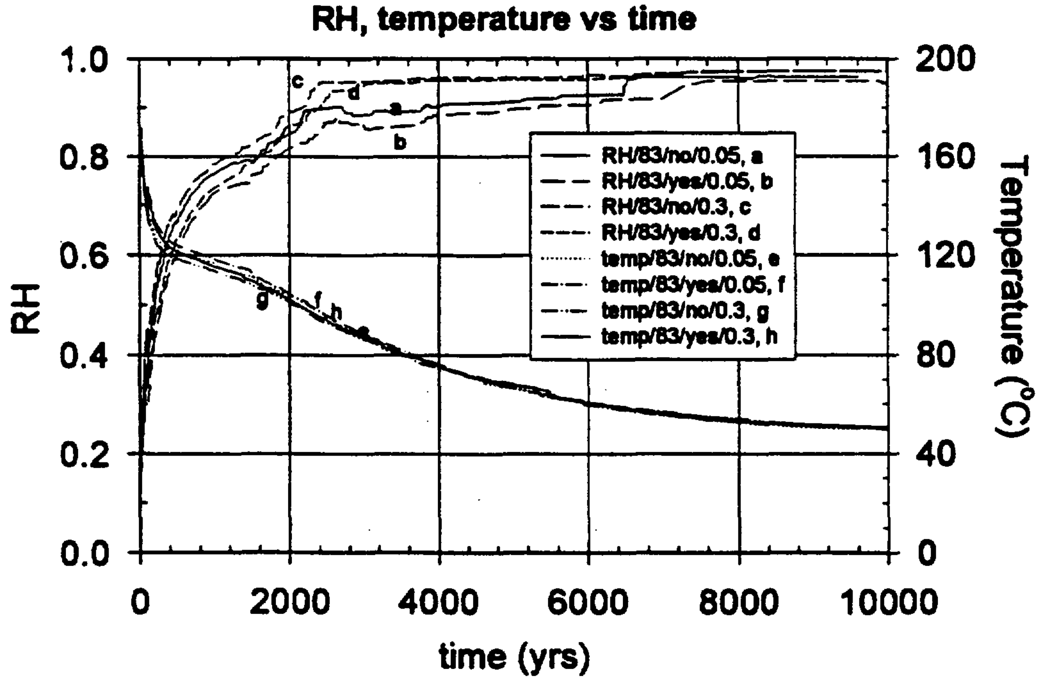


Figure 7-34. Abstractions of Temperatures and Relative Humidities for 10,000 Years (83 MTU/acre Case) (Source: DOE95c)

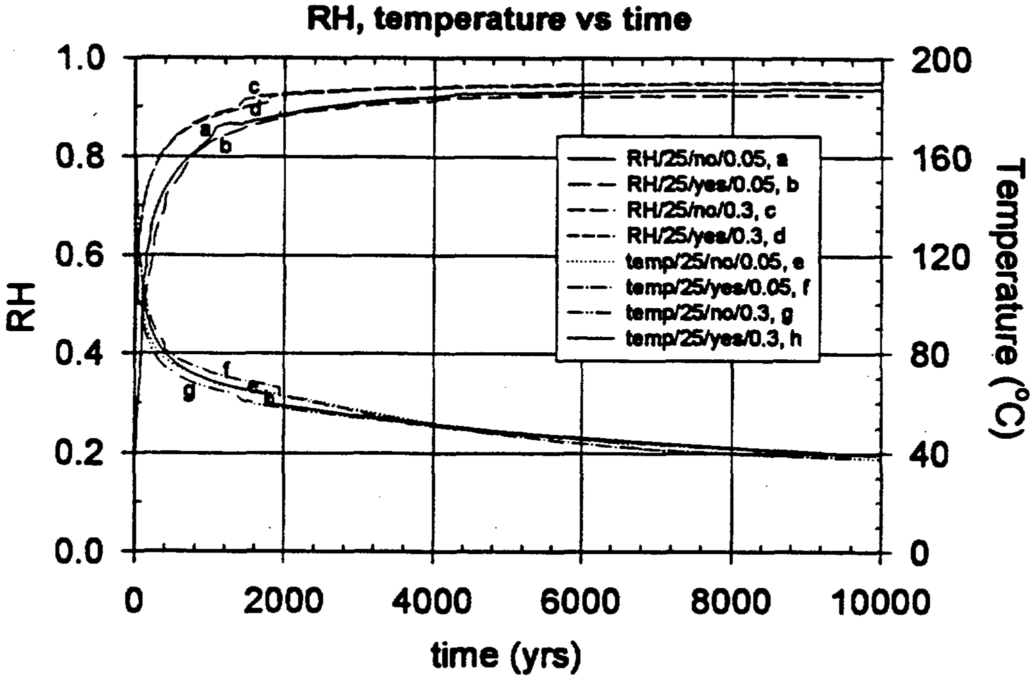


Figure 7-35. Abstractions of Temperatures and Relative Humidities for 10,000 Years (25 MTU/acre Case) (Source: DOE95c)

The drift temperature distributions for both cases modeled show high temperatures at the waste package at 10 years, without appreciable thermal affects progressing far into the host rock. With the passage of time, heat released by the waste package diminishes, but the host rock experiences increasing temperature, with near-equilibrium occurring after 1,000 years and gradual cooling thereafter.

The results of near-field thermohydrology modeling, reported in TSPA-95 Section 4.2.9, are based upon a two-dimensional X-Y model which is best applied at the center of the repository. The performance at the repository edge, where heat transfer to the host rock occurs, has not yet been modeled.

The temperatures in the emplaced drifts will not only affect near-field hydrology but will also create responses in the rock mass, such as fracture creation, closure, or widening. Studies to date show a 111 MTU/Ac thermal load may cause closure of vertical joint apertures and horizontal joint apertures to open up to 0.2 mm (see TSPA-95, Section 4.4). This suggests that at this high repository thermal load, continuous joint slip may be induced in the near-field rock mass.

The temperature changes caused by the emplaced waste packages will affect the near-field geochemical environment. Temperature changes may affect WP corrosion, waste dissolution, and radionuclide solubilities and transport characteristics. DOE is continuing to study repository natural and engineered parameters and their effect on repository performance.

7.6.3.6 Geosphere Transport Models

Radionuclides released from the EBS are available for transport through the unsaturated and saturated rocks of Yucca Mountain on their way to the biosphere. The travel times of the radionuclides depend on the physical and chemical interaction between the fluid and the rock; they are a function of the percolation flux distribution in the unsaturated zone, the advective flux distribution in the saturated zone, and conceptual representation of the hydrostratigraphy along the likely ground water flow paths between the repository and the accessible environment.

> *Unsaturated Zone Transport:* The percolation flux distribution within the Topopah Spring hydrostratigraphic unit, and other unsaturated zones below it, is a function of the infiltration rate and the conceptual model for ground water flow through unsaturated media. The key conceptual uncertainty arises from the possibility that "fast" pathways exist through the unsaturated rocks within connected fractures.
>
> Since no appropriate process-level model exists for the simulation of fracture and matrix interactions, a statistical algorithm that randomly transitions particles between fractures and matrix transport modes was implemented. The magnitude of the transition is modulated through a parameter that controls the strength of the fracture/matrix coupling. Unsaturated zone transport results based on various parameter coupling levels were considered.
>
> *Saturated Zone Transport:* The saturated zone flux affects the arrival time of the radionuclides to the accessible environment as well as the degree of mixing and dilution of the radionuclide contamination in the ground water of the tuff aquifer prior to its extraction and use. The entire flux distribution, dependent on the hydrogeologic parameterization of the aquifer, incorporates the effects of large-scale heterogeneity of aquifer properties. Small-scale heterogeneity is included through the use of dispersion in the solution on the one-dimensional advection- dispersion equation. A log-normal distribution of the Darcy velocity in the saturated zone is used. Various realizations of the saturated zone velocity transport distribution were considered. A detailed description of saturated zone transport modeling is provided in Section 7.4 of this BID.

7.6.3.7 Evaluation of Radiation Doses

Evaluating radiation dose to humans requires definition of the potentially exposed population and the potential biosphere pathways by which individuals may be exposed to any radionuclides released. In the TSPA, it was assumed that the peak individual dose corresponds to an individual taking drinking water from the tuff aquifer at a rate of 2 liters/day. Further, it was assumed that the maximally exposed individual was located at the accessible environment boundary, where the radionuclide concentration within the tuff aquifer is at its peak. Mixing volumes were based on a fixed cross-section of the saturated zone with the horizontal mixing dimension defined by the width of the potential repository; the vertical mixing dimension was defined by a well with a 50-m saturated-zone interval. Dose conversion factors were derived from those published by the EPA (EPA88).

Assumed negligible, gaseous release of carbon-14 to the biosphere was considered in only one case. The TSPA modeled the carbon-14 released from the inventory as dissolved in the aqueous phase once it reached the geosphere, where it was then transported to the biosphere via aqueous transport.

7.6.3.8 Climate Change

Given that the regulatory period of concern may extend to 1,000,000 years in the future, it is highly likely that the climate in the region of Yucca Mountain will change significantly over the life of the possible repository. Of primary concern is the potential for change in atmospheric conditions that may result in changes in precipitation and net evapotranspiration (evaporation from soil plus transpiration from vegetation). Increased precipitation, potentially initiated by episodic glaciation, may cause an increase in percolation flux and a rise in the water table, both of which were considered in the DOE TSPA evaluations. For the million-year time frame, the evaluations considered 100,000-year cycles of precipitation rates in the future.

7.6.3.9 Disruptive Processes and Events

In the 1993 TSPA, the DOE defined a "nominal" case in which the post-closure facility was undisturbed by natural or human activities. This case consisted of a heat-generating repository subjected to climate-dependent ground water flow. Two alternative conceptual models were used to define the flow in the unsaturated zones. The nominal case took into account the effects of eight radionuclides chosen for their expected transport characteristics or expected dose contribution.

A "disturbed" case was also defined. It considered the effects on repository performance of performance-disrupting scenarios, which included volcanic activity and inadvertent human intrusion through drilling. The effects of drilling would come from environmental exposure of drill cores and drilling fluids contaminated with radionuclides. In TSPA-93 the effects of volcanism that were considered were limited to indirect factors associated with the magmatic-induced corrosion of the waste containers. The human intrusion analysis included the study of a broad suite of 43 radionuclides. The indirect volcanic intrusion case evaluated eight radionuclides, chosen for their expected transport characteristics or expected dose contribution.

In 1995, the TSPA excluded disruptive events such as volcanism and human intrusion because the results of TSPA-93 showed that these processes and events were statistically insignificant with regard to post-closure repository performance.

7.6.4 Qualitative Results and Conclusions of DOE TSPA Evaluations to Date

DOE's performance assessments to date have been designed primarily to support selection of program strategy and activities. As a result, the summaries and interpretations of results in DOE's TSPA documents are presented primarily as qualitative descriptions of the influence of parameters and assumptions on repository system performance. These qualitative DOE findings are summarized in this section. Examples of DOE's quantitative results are discussed in Section 7.7.

A general conclusion derived from the DOE TSPA studies completed and reported to date is that the level of sophistication of the analyses has increased significantly from the precursors of TSPA-91 (e.g., PACE-90 (DOE91)) through TSPA-95. Knowledge of the features, events, and processes controlling the performance of the repository has increased significantly. This knowledge is incorporated into the TSPA in the form of more representative and comprehensive models describing the physical, chemical, and biological phenomena defining each constituent sub-process.

The results of TSPA-95, which are the most recent available, reflect the results of the entire iterative evaluation process to date and can be summarized in two categories based on the time scale of the performance evaluation.

10,000 Year Performance Results

The total peak dose, due mainly to technetium-99 and iodine-129, is most sensitive to the matrix velocity in the Calico Hills nonwelded unit below the repository and the percolation flux in the unsaturated zone.

No release to the accessible environment occurs when: (a) the infiltration range is low (0.01 to 0.05 mm/year); (b) cathodic protection is applied to the waste package; (c) the Buscheck 80 MTU/acre thermal load with and without backfill is assumed; (d) the Buscheck 24 MTU/acre thermal load with backfill is assumed; or (e) matrix flow, and no fracture flow, in the unsaturated zone is assumed.

Depending on the conceptual model of intra-unit fracture connectivity, fracture/matrix interaction can significantly affect peak dose and cumulative release.

Depending on the conceptual model for radionuclide transport across the EBS, a capillary barrier preventing drips (advective flow) from contacting the waste packages can reduce the peak dose at the accessible environment by at least a factor of 20, and up to many orders of magnitude if only diffusive releases are possible through the EBS.

1,000,000 Year Performance Results

The peak dose, due primarily to iodine-129 over the infiltration range of 0.01 to 0.05 mm/year, and due to neptunium-237 at high infiltration ranges between 0.5 and 2.0 mm/year, is most sensitive to dilution in the saturated zone and percolation flux in the unsaturated zone.

The total peak dose may be greatly reduced by a barrier that intercepts water dripping on the waste packages.

Low intra-unit fracture connectivity in the unsaturated zone can significantly delay the breakthrough of peak doses to the accessible environment, but reduces only slightly the ultimate peak dose that occurs during the entire 1,000,000 year time frame.

Alternative thermal loads, thermohydrological models, and corrosion-initiation models (including cathodic protection) do not have a very large impact on the total peak dose.

> Climatic change with water table rise can increase the peak dose at the accessible environment by a factor of about 2 to 10 compared to no change in climate; climate changes with no rise in water table (varying infiltration rate only) falls between these extremes.
>
> DOE believes that the use of conservative assumptions when definitive information was not available, a specific goal of each TSPA, has resulted in estimates of repository performance that are, overall, conservative. Relaxation of some specific conservative assumptions is expected to increase the realism of performance assessment results. For example, in cases where neptunium release dominates the peak dose, the solubility of neptunium plays a key role. It has been suggested that the TSPA neptunium solubilities represent metastable equilibrium concentrations and that actual equilibrium conditions of the repository may result in concentrations several orders of magnitude lower. The relaxation of this conservative assumption would lower the predicted peak dose by about one order of magnitude.

The 1995 TSPA concludes that several factors control the results of the 10,000-year assessments. Under some assumptions, the EBS itself can completely contain the waste for the entire period. Similarly, when the percolation flux is at the low end of its range of estimated values, the natural barrier system can completely contain the waste for the entire period.

When the most conservative estimates of both the EBS and natural barrier are assumed, some release is predicted over 10,000 years. The predominant factor in this case is the percolation flux distribution, which controls the possibility of dripping in the drift (advective flux) and the magnitude of the advective release from the EBS after the waste package fails. Thus, the percolation flux distribution controls the 10,000-year cumulative release (a result obtained in 1991, 1993, and 1995).

On the million-year time scale, fewer parameters significantly influence the predicted peak dose than for the 10,000-year case. Over much longer time periods, factors that delay the arrival of the peak concentration are less significant because of the extremely long time period considered and the long half-lives of key radionuclides (technetium-99 at 200,000 years, neptunium-237 at 2 million years, and iodine-129 at 20 million years). Even under optimistic conditions, these nuclides are either only slightly sorbed or not sorbed at all, and generally break through to the accessible environment in one million years. Thus, given the analyses of TSPA-95, it is unlikely that the barriers considered can be reasonably shown to delay the above nuclides sufficiently to preclude their release over one million years.

Although it was concluded that long time periods do not significantly affect long-term doses, the TSPAs also found that radionuclide dispersion and dilution play a highly significant role. Dispersion, from both the EBS and the geosphere, spreads the release effect over time and reduces peak concentrations and peak doses. These beneficial factors can be amplified by increasing the assumed distance between the repository and the assumed user of drinking water from the tuff aquifer. If the critical group is located 25 km down gradient from the "accessible environment," the peak concentration and peak dose are reduced by more than an order of magnitude.

Within the EBS, if release is diffusion-dominated, with only a small percent due to advective release, then ultimate release rates are significantly reduced. Diffusion-dominated release is controlled by either the percolation flux distribution or an effective capillary barrier used as backfill. The finding that diffusive release from the EBS is the dominant transport mechanism is consistent with results from other repository analyses around the world (Neall et al., 1995).

Concurring with each prior study, TSPA-95 concluded that characterization of the hydrologic regime is the key issue. The amount of water present in the natural and engineered systems and the magnitude of the aqueous flux through these systems control the overall performance of the repository.

7.6.5 DOE Waste Isolation and Containment Strategy

In 1995, DOE described a waste isolation and containment strategy for a potential repository system at Yucca Mountain (DOE95d). The strategy is based in large measure on the total-system performance assessment results to date.

The strategy has two principal objectives:

1. To limit annual dose to any member of the public for a period on the order of a million years.
2. To provide total containment of the waste within emplaced waste packages for thousands of years.

These objectives have been selected to be consistent with the recommendations and considerations of the NAS panel concerning potential Yucca Mountain standards (NAS 95). The first objective is intended to assure that peak annual dose to individuals is low whenever it occurs. The second objective is intended to prevent any contact of radionuclides with natural elements of the repository system and the environment for thousands of years.

The DOE strategy identifies three characteristic time periods for repository performance:

1. The containment period, lasting on the order of hundreds to thousands of years, during which the waste is completely contained and isolated by the waste containers. Important performance parameters for this period include the humidity in the repository and the corrosion rate of the waste containers.

2. A transition period, lasting on the order of thousands to tens of thousands of years, during which the waste packages will degrade and waste radioactivity will be mobilized and transport to the environment. Factors important to performance during this period include those that affect the dissolution rate of the waste form and those that affect radionuclide transport, such as rock permeability and chemical holdup.

3. The peak dose period, which extends beyond the transition period and during which the peak doses due to long-lived radioactive isotopes such as Np-237 occur. The magnitude of the peak dose will depend on the rate at which water enters the repository and mobilizes radioactivity, the duration of the containment period, the transport characteristics of the barrier system, and the amount of dilution and dispersion that occurs in the ground water system.

Implementation of the strategy and capability to fulfill its objectives can be represented in terms of five hypotheses that are amenable to evaluation by testing and performance assessments. The five hypotheses are:

1. The flow of water into the repository will be low, so that there will be little water available for mobilizing and transporting radionuclides.

2. Heat emission from the waste as a result of radioactive decay will cause low humidity in the repository, which in turn will inhibit corrosion of the waste containers for thousands of years. Engineered features of the repository, such as use of a two-barrier container in which the outer barrier will provide cathodic protection to the inner barrier, can help assure that the containers remain intact for very long time periods.

3. After waste packages corrode so that water can contact the waste, mobilization of waste radioactivity in the water will be low because of factors such as low dissolution rates of the waste form and low solubility of radionuclides in the water.

4. The engineered barriers will inhibit migration of waste radioactivity into the host rock and the ground water regime. The inhibiting effect of the engineered barriers can be augmented by use of selected backfill materials.

5. Concentrations of radionuclides in ground water that moves to and through the biosphere will be greatly diluted by dispersion processes and by mixing of radioactively-contaminated and uncontaminated waters in the hydrologic regime.

DOE expects the investigation of these hypotheses to involve extension of data acquisition and performance assessment work already underway. DOE believes the technical issues associated with testing of these hypotheses are well understood as a result of data now available and performance assessments already completed. Use of this containment and isolation strategy as a basis for future work is expected to focus and alter previous plans for data acquisition and analysis. Plans for implementation of the strategy are currently being developed.

7.6.6 Overview of NRC's TSPA Effort to Date

7.6.6.1 NRC's Overall TSPA Program Approach

The Nuclear Regulatory Commission believes that performance assessment of a geologic repository, like any other systematic safety assessment, benefits from being conducted in an iterative manner. The lessons learned about modeling improvements, data needs, and methodology can be used to improve subsequent iterations. NRC has planned a sequence of iterative performance assessments (IPA) in preparation for its review of a potential DOE license application to construct and operate a geologic repository for spent nuclear fuel and high-level radioactive waste. Through these repeated iterations, NRC staff will improve both its capability for reviewing DOE's demonstration of repository performance and its understanding of the key factors in repository performance.

During the pre-licensing phase, the specific objectives of NRC's IPA program include:

- evaluating the ongoing DOE site characterization program (including field studies, laboratory studies, and analyses, and interim performance assessments generated by DOE and its contractors)

- evaluating ways to implement the 10 CFR Part 60 performance objectives

- providing input to the ongoing evolution of the radiation protection standard for the geologic repository, set forth by the EPA in 40 CFR Part 191, which is incorporated by reference in 10 CFR Part 60

- providing input to regulatory guidance and other regulatory products related to performance assessment, especially the Draft License Application Review Plan

- Assisting in the definition of the Office of Nuclear Material Safety and Safeguards (NMSS) technical assistance and research programs in the area of high-level waste (HLW)

During the licensing phase, additional specific objectives of the IPA program include:

- providing an independent calculation of key aspects of DOE's total-system performance assessments (TSPA) submitted as part of a license application

- Probing DOE's assessment for potential weaknesses, based on a familiarity with the methods, data, and assumptions used in the performance assessments

The following six steps define NRC's general approach to developing and analyzing a TSPA:

1. System Description

   For the purposes of modeling (both conceptual and computational), the repository is broken into its component parts. These components include the waste form, the mined geologic repository system (including the engineered barriers such as the waste package), and the portion of the geosphere surrounding the geologic repository through which the radionuclides, in time, may migrate. The system description includes information that supports the development of models describing repository performance and identifies data and parameters for the models used to support the next step, Scenario Analysis.

2. Scenario Analysis

   Alternative possibilities for the future environment can be represented by different scenarios reflecting the repository. These scenarios are identified and screened. For this analysis, scenarios are formulated based on classes of events and processes external to the repository system. (Events and processes internal to the repository system are treated in the third step, Consequence Analysis.) Probabilities are also estimated for the selected scenarios.

3. Consequence Analysis

   The performance of the subsystems of the geologic repository are described by release, transport, and dose models. These models are linked to describe overall performance. Overall repository performance, in terms of cumulative releases of radionuclides to the accessible environment, over a specified time period (e.g., 10,000 years), is calculated for each scenario, using numerous simulations of possible ranges of parameter values. In addition to the complementary cumulative distribution function (CCDF) for cumulative releases, other types of system performance measures, such as maximum doses to individuals, can also be considered.

4. Probabilistic Performance Measure Calculation

   The consequences, in terms of normalized cumulative releases of radionuclides to the environment over a specified period of time, are calculated for each scenario identified. The results are then displayed in a plot of total releases versus the probability that such consequences are exceeded (i.e., the CCDF of total releases to the accessible environment for 10,000 years, normalized by the EPA release limit for each radionuclide and summed over all contributing pathways). The total results incorporating scenario probability are compared with release limits established by the EPA standard.

5. Sensitivity and Uncertainty Analysis

   The fractional change in calculated results caused by incremental changes in the values of input parameters and data is evaluated by conducting a sensitivity analysis. Then, an uncertainty analysis is also conducted to quantify the uncertainty in performance estimates in terms of the major sources of uncertainty in input parameters. Uncertainty in modeling, including conceptual model uncertainty and uncertainty regarding the probability of future conditions, is quantified to evaluate the representativeness and completeness of conceptual models used in the analysis.

6. Documentation

The assumptions used in the analysis, their bases, and the implications of their uses are documented. Auxiliary analyses, which evaluate the adequacy of the consequence modules and the assumptions underlying them, synthesize data into parameters, and provide other insights, must also be documented.

7.6.6.2 Progress of NRC'S IPA Effort to Date

Over the past four years, NRC has completed two major and one intermediate phase of its IPA activity. The first phase involved an evaluation of the adequacy of existing NRC Performance Assessment (PA) methodology and the staff education regarding the conduct of TSPAs for the Yucca Mountain repository license application. Phase 1 showed that, although the PA methodology needs improvement, NRC is capable of conducting a full TSPA for Yucca Mountain. The second phase of IPA implemented the major recommendations of the Phase 1 study concerning methodology development needs, modeling of additional scenarios, and conduct of new auxiliary analyses. An intermediate study phase was added to conduct an expert judgment elicitation exercise for future climate scenarios in the Yucca Mountain region.

Iterative Performance Assessment Phase 1

Phase 1 of the IPA was undertaken primarily to demonstrate the NRC's ability to conduct a PA. The analysis depended on the availability of limited site-related data, numerous simplifying assumptions, and a small number of scenarios. The analysis had two major components:

1. Quantitative estimation of total system performance using available mathematical models and computer codes.

2. Documentation of the rationale and auxiliary analyses supporting the assumptions made in the total system calculations.

The calculations centered on the total system performance measure, stipulated in the containment requirements in 40 CFR 191.13. Uncertainty analysis was used to quantify the uncertainty in the performance measure due to both the uncertainty in the input parameters and the uncertainty concerning future conditions of the repository. The results from the uncertainty analyses were used to construct the CCDF of total radionuclide discharges to the accessible environment over the 10,000-year regulatory period, as prescribed in 40 CFR 191.13. A sensitivity analysis was performed to identify which parameters had the greatest influence on the uncertainty of the estimated performance measure. The sensitivity analysis provided some insights into data needs and priorities.

Other objectives satisfied in NRC's IPA Phase 1 include:

- Evaluation of the adequacy of existing tools, both in terms of methodologies and approaches, and of calculational tools (e.g., computer codes).
- Identification of improvements needed in these tools.
- Limited evaluation of data needs and their associated priorities for the DOE site characterization program.

Iterative Performance Assessment Phase 2

Phase 2 of the IPA used the same basic approach as Phase 1, but with significant enhancements. These enhancements were the result of implementing many of the recommendations from Phase 1, including:

- creation of a largely automated total-system computer code
- addition of a dose assessment capability
- evaluation of the SNL scenario-selection methodology
- addition of two disruptive processes (volcanism and seismicity, in addition to climate change and human intrusion)
- refinement of modeling of ground water flow and radionuclide transport processes in unsaturated fractured rock
- addition of gaseous transport pathway
- addition of radionuclide transport in the saturated zone

- improved treatment of the radionuclide source term (including mechanistic waste package failure, improved waste dissolution and transport model, and gaseous-phase source term)
- addition of new methods for uncertainty analysis and sensitivity analysis
- improvement of spatial resolution of the repository layout, source term, and hydrostratigraphy

The ability to incorporate more scenario classes and other potentially important radionuclide transport pathways (e.g., gas-phase transport) increases NRC's flexibility to explore more alternatives. Therefore, NRC is now able to calculate more meaningful CCDFs than after Phase 1. Specifically, NRC has the capability to examine different approaches to construct the CCDF, as well as different uncertainty analysis and sensitivity analysis methods, which enhances its ability to evaluate the DOE PA analyses.

The main conclusions of Phase 2 are:

- Infiltration rate has the highest correlation with overall system performance.
- The time of liquid water contact with waste packages has a high correlation with overall system performance.
- Pitting and crevice corrosion potentials, as well as solubilities and alteration rates, have high correlations with overall system performance.
- Repository heat load has the potential to affect performance significantly.
- Carbon-14 dominates a release-based performance measure but is relatively insignificant for a dose-based performance measure.
- Both seismicity and volcanism can lead to large releases; however, their probabilities are low enough so that they have no significant effect on the overall system performance in the context of a release standard.
- Dominant radionuclides for population dose include niobium-94, lead-210, americium-243, and neptunium-237.
- The subsystem performance measures of ground water travel time and substantially complete containment show a strong positive correlation with the

overall release-based performance measure; the fractional release rate performance measure shows a milder, yet still positive, correlation.

- The scenario methodology is sound and suitable for conducting and evaluating a TSPA.

Iterative Performance Assessment Phase 2.5

NRC undertook this intermediate phase primarily to conduct the expert judgment elicitation exercise for future climate scenarios in the Yucca Mountain region. This exercise addressed an issue of high priority to PA: the use of expert judgments in the regulatory process. The four basic objectives of Phase 2.5 are:

- Acquire experience in the expert judgments elicitation process that will aid in reviews of the use of expert judgments and the development of NRC guidance on expert judgment elicitation.
- Examine the formal and informal application of expert judgments and investigate aggregation and consensus-building techniques for use with panels of multiple experts.
- Form an expert panel and apply techniques to elicit expert judgments on characteristics and probabilities of future climate scenarios in the Yucca Mountain region.
- Provide estimates of future precipitation, cloud cover, and temperature that can be used in IPA Phase 3.

As a result of Phase 2.5, NRC now will be able to comment on specific aspects of the DOE's use of expert judgments, develop guidance on the use of expert judgments, if necessary, and better evaluate the effect of climate changes on the performance of the proposed repository.

Iterative Performance Assessment Phase 3

Phases 1, 2, and 2.5 have familiarized the NRC staff with the issues concerning the Yucca Mountain site. With the knowledge they have gained, the staff has a basis for the detailed review and evaluation of the DOE TSPAs, including assumptions, logic, and methods used to estimate performance. This will be one aspect of IPA Phase 3. Also in Phase 3, the NRC will enhance its capability to evaluate the DOE Technical Site Suitability (TSS) Determination for Yucca Mountain, expected in FY98. NRC will also provide staff guidance to the DOE and participate in DOE/NRC Technical Exchanges and Technical Meetings and in the evaluation of key technical uncertainties (KTUs).

Phase 3 of the IPA is expected to contribute to the evaluation of the TSS determination in the 1) evaluation of the DOE waste isolation strategy; 2) evaluation of DOE High-Level Findings (HLFs) supporting the TSS determination; and 3) review of DOE TSPAs expected in FY96 and FY98.

NRC is currently preparing a detailed plan for Phase 3 to identify specific DOE KTUs or HLFs to be addressed by IPAs. DOE is expected to issue a number of reports between now and FY99, and IPA Phase 3 results will be used in the staff's review and evaluation of these reports.

Phase 3 will support the Systematic Regulatory Analysis (SRA) through the evaluation of selected NRC KTUs with respect to their likely impact on repository performance and their relative importance. This will help focus NRC technical assessment and research activities on those areas of greatest significance. It will also provide a basis for the development of compliance determination methods (CDMs). Phase 3, like its predecessors, will support NRC HLW reactive activities and the participation of the staff in National Academy of Sciences (NAS) deliberations on the development of new EPA standards for the Yucca Mountain site. Finally, because DOE has yet to finalize its repository design (including thermal loading and waste emplacement options), the IPA will explore different design alternatives and assumptions and their potential impact on performance. This exploration will assist in the identification of important processes, phenomena, and parameters and will aid in the model abstraction process.

7.6.7 NRC's TSPA Conceptual and Computational Models

7.6.7.1 NRC's TSPA Conceptual Model

The first step in a TSPA is to develop a system description of the geologic repository. This description should include information to support development of models describing repository performance and to determine assumptions and parameters on which the models depend. In this manner, the geologic repository is broken into its components for the purposes of modeling. These components include the waste form, the mined geologic repository system, and the portion of the geosphere surrounding the geologic repository through which the radionuclides, in time, migrate. NRC used the descriptions of the Yucca Mountain site and the geologic repository (including waste package) from DOE's 1988 Site Characterization Plan (SCP) (DOE88).

In the IPA Phase 2, NRC used a base case system model, comprised of subsystem and process models. The major subsystems modeled were the waste package and engineered barrier system (EBS), the local hydrosphere, and a postulated biosphere. Major processes modeled separately were water percolation, gas transport, and ground water transport. The disruptive events considered were pluvial climate change, seismicity, human intrusion, and volcanism.

The Yucca Mountain site was conceptualized as a layered stratigraphy for the liquid source term and transport. The geologic repository and water pathways were divided into seven distinct columns. These columns helped account for the variation in stratigraphic sequences and thickness, differences in unsaturated and saturated pathway distances, and temperatures within the repository. Auxiliary analyses conducted with a two-dimensional dual continuum representation of the repository cross-section were used to determine how percolation from rainfall should be distributed among the seven sub-areas (columns), as well as to determine the distribution of flow between rock fractures and matrices in the unsaturated portion of the pathways. The source term module considered the environment of the waste package and near-field, including the EBS. When disruptive events were not a factor, the source term module used repository zone temperature as an indicator of whether each particular zone is wet or dry. After the initial dry-out period in each zone, corrosion was calculated as a function of environmental conditions. When the wall thickness from corrosion was thin enough to result in failure of the waste package canister, water was assumed to enter the canister, and the waste dissolution process was assumed to begin. Also, a small quantity of packages were assumed to have initial defects so that dissolution and release can take place without corrosion. Transport out of the waste package canister by advection and diffusion was calculated as a function of the fracture flow rate into the zone, and the results were passed on to the liquid flow and transport module. In the transport model, in-situ matrix and fracture velocities and matrix geochemical retardation were used to determine the time-varying amounts of radionuclides reaching the biosphere.

For the gas source term and transport, the Yucca Mountain site was again conceptualized as a layered stratigraphy. The site itself and geologic repository were not divided into sub-areas but were modeled as a two-dimensional cross-section with a time-varying temperature distribution. The temperature distribution was calculated based on conductive heat transfer, taking into account the repository thermal loading and the heat transfer properties of the rock. The gas source term that results from initial defects and corrosion (in the absence of disruptive events) was assumed to be evenly distributed throughout the repository. A model employing many simplified assumptions was developed to determine the velocity vectors throughout the performance assessment period. The gas source term releases were tracked through the repository, using these time-varying velocity vectors. They were reduced for radioactive decay, in accordance with their travel time (including geochemical retardation) to the surface.

Mutually exclusive scenario classes were developed to simulate the performance of the repository system in the case of several credible external events. For calculation of dose and release within the various scenario classes, the base case system model parameters and logic were changed to account for the disruptive event or combination of events being modeled. For scenario classes involving climate change, the water table was raised and infiltration was sampled from a different distribution than was used for the normal climate. Scenario classes involving drilling allowed damage to emplaced waste packages and also added a direct pathway (the borehole) to the accessible environment. Scenario classes involving seismicity required the interaction of a seismic canister failure module with the source term module. The source term module calculated waste package thicknesses, based on corrosion processes, and combined this information with seismic acceleration probability data from the seismic module to determine when and if a waste package should fail. Seismicity caused no changes in the existing pathways or the addition of new pathways. For classes involving volcanism, the repository was assumed to be in the possible path of intrusive and extrusive volcanic events. The intrusive events were assumed to be underground magma intrusions that damaged waste packages but did not provide an additional pathway to the surface. Extrusive events were assumed to entrain a portion of the repository waste and carry it directly to the surface, resulting in an airborne release.

After the transport of radionuclides to the biosphere was determined as a function of time, the radionuclides from the various pathways were accumulated to determine the cumulative release, for comparison with the EPA standard and to calculate the cumulative population dose. The biosphere used for the dose calculation was assumed to be a 2,700-acre farm, with 3 people maintaining a year-round residence and 177 people off-site eating beef cattle, which grazed on the farm, for the waterborne dose; and 22,200 persons in the region for the airborne dose. The dose was calculated in 70-year (lifetime) intervals and accumulated. After completion of the runs and construction of the CCDFs (for both release and dose), sensitivity and uncertainty analyses were performed.

For each modeled scenario class, the system code was run 400 times. Over 200 parameters were sampled for some of the classes. The sampling was performed using the LHS routine. Because of the large uncertainty in both site parameters and process parameters, probability distributions were determined for hydrologic characteristics of the individual geologic strata, corrosion parameters, percolation distribution parameters, and scenario-related parameters, as well as other site- and process-related parameters.

7.6.7.2 NRC's TSPA Computational Model

The diversity of the physical processes in the natural system being simulated requires that theories from many disciplines be integrated into an overall system model. The TSPA program is designed to simulate the behavior of a geologic repository located in a partially saturated medium; both the natural system and the engineered barriers are accounted for in the program design. The evolutionary change in the natural system is described in terms of disruptive scenarios which, in addition to a parametric description of the changed state, also have a probability of occurrence attached to them. Consequently, the TSPA computer code is designed as a set of consequence modules or largely independent computational units, with their execution controlled by a system manager or executive module.

Almost all the concepts necessary to model a repository system are included in modules that are controlled by the executive module. However, the implementation of these modules is kept flexible so that various scenarios may be simulated. In other words, no specific conceptual module is embedded in the executive module, except for the adoption of scenario analysis as a general approach. In the scenario approach, the future state of the repository system is defined by a set of parameters whose values are chosen from specified probability distributions. This set of parameters is assumed to be independent of time for a particular scenario, although this is not a strict requirement of the scenario approach. A different scenario is defined, if parameter values change within the time span of interest (e.g., 10,000 years). In the analyses conducted so far, disruptions occur at a specified time and the disturbed state then remains constant, which is probably reasonable for every scenario except those involving pluvial climate.

Automated features included in the system facilitate the unattended running of a set of multiple scenarios with associated output.

The TSPA computer code consists of four basic parts:

- The executive module
- Algorithm(s) to sample from statistical distributions
- Algorithm(s) to model future states or scenarios
- Algorithm(s) to model internal repository system processes such as source term, transport, and consequences.

Consequences are quantified in terms of cumulative releases and dose-to-man. In addition, algorithms to compute sensitivities and perform uncertainty analyses are executed separately as an auxiliary process.

Descriptions of Computational Modules

Executive Module. The TSPA executive module acts as the controller for the overall computer code. It executes the consequence modules in the desired sequence and ensures that appropriate values of the common parameters are passed to the appropriate consequence modules. It controls the sequence of execution of various modules, transfers data to other modules, and controls data transfer from one module to another.

AIRCOM. This mainly utilitarian module does not perform any calculations related to the physics of the geologic repository system, except for the introduction of fractions of contaminated soil that become airborne and respirable in the drilling and volcanic scenarios. Its main purpose is to merge the various airborne release data files (*VOLCANO*, *DRILLO*, and *C14*) into one file in the proper format for use by the dose module, *DITTY*.

In IPA Phase 2, contaminated soil or gaseous $^{14}CO_2$ was assumed to be transported to the ground surface above the repository as a result of disruption of the geologic repository itself, either by human intrusion or by an extrusive volcanic event. In this analysis, only a fraction of this surface radioactivity was assumed to become available for transport by the air pathway to members of the public beyond the controlled area (10 CFR60.2) of the geologic repository. The fractions of the radioactivity that were assumed to become airborne were stored in the *AIRCOM* module. All the airborne radioactivity was assumed to be respirable (whether in the solid, liquid, or gaseous state). Any radioactivity that did not become airborne was assumed to remain undisturbed at the point of release to the above-ground surface.

CLIMATO. This module is a place holder for a future climate-related constituent of a disruptive event. In IPA Phase 2, climate change is treated by specifying climate-dependent infiltration rate and water table position for use in the *FLOWMOD* transport module.

CANT2. In this module, the time-dependent temperature of the surface of a waste package is calculated. The *CANT2* module is based on an analytic solution of the linear heat conduction equation, by the principle of superposition, assuming a finite number of heat sources. In IPA Phase 2, the repository is assumed to consist of seven regions or sub-areas, and each repository sub-area comprises several waste emplacement panels. The main purpose of *CANT2* is to predict the temperature of a representative waste package in each of the seven repository sub-areas needed for the source term module, *SOTEC*, to determine the time at which liquid water can come into contact with the waste packages.

C14. In IPA Phase 2, $^{14}CO_2$ is considered to be the only radionuclide that can be transported in the gaseous phase. The *C14* module calculates the travel time and decay of $^{14}CO_2$ releases from the source term module. This module uses an independently calculated time-varying far-field temperature field, to determine time-dependent gas velocities. *C14* uses the equations of flow, hydrologic parameters from the LHS sampling module, and the time- varying temperature field induced by the spent nuclear fuel, to calculate a time-varying gas velocity field from the water table to the atmosphere. Releases from the source term are tracked through this field and reduced by radioactive decay, allowing for retardation of $^{14}CO_2$, because of the interaction of the host rock and water. The amount of $^{14}CO_2$ released from the repository, as calculated by the source term module, *SOTEC*, is provided to *C14* as an input.

DITTY. The transport of radioactivity to the biosphere is modeled in the *DITTY* (Dose Integrated for Ten Thousand Years) module. This module estimates the time integral of collective dose over a 10,000-year duration for releases (or concentrations) of radionuclides to the accessible environment. In IPA Phase 2, the exposure pathways of interest included: the atmosphere, land surfaces, the top 15 cm of surface soil, vegetation, animal products (milk, beef), and drinking water. (Aquatic pathways were not considered.) The annual releases to the air or water pathways over the 10,000-year period of interest were provided as input to *DITTY* by average annual concentrations. The values for these concentration-time pairs were obtained as outputs directly from *NEFTRAN* or indirectly from *C14*, *DRILL02*, and *VOLCANO*, via the *AIRCOM* module. This module considers both air and liquid transport pathways and calculates both the individual and population doses. It is designed to deal with both acute and chronic releases, and annual, committed, or accumulated doses can be calculated.

DRILL01. In IPA Phase 2, the human intrusion disruptive event is stipulated to consist of drilling above the geologic repository. The location of boreholes and the timing of drilling are assumed to be random. Although a random spatial distribution may not be physically realizable, it is used here for simplicity. The drill bit can either hit a waste package directly or it may penetrate only the contaminated rock. Radioactive material may be brought to the surface, in either case.

DRILL02. Consequences from drilling (identified in *DRILL01*) are calculated in *DRILL02*. A drill bit hitting a waste package directly or penetrating contaminated rock is assumed to lift a certain portion of the radionuclide inventory to the ground surface. The inventory in a waste package and in the rock surrounding waste packages is a function of time. It is used by *DRILL02* to determine consequences. A small percentage of the radioactive material brought to the surface is assumed to be particulate material that becomes airborne. This information is then provided to the *AIRCOM* module, for calculation of the respirable fraction of the human dose in *DITTY.*

FLOWMOD. This module determines the hydrologic flow regime that provides ground water flux for use in the source term module (*SOTEC*) and transport pathways and properties for use in the transport module, *NEFTRAN.* The primary functions within *FLOWMOD* are the determination of: (a) the spatial distribution of ground water flux; (b) the quantity of flow in the matrix and the fracture; (c) fluid velocities; and (d) saturation-dependent retardation coefficients. The computational demands of solving partially saturated flow in fractured tuff precluded a direct solution of the flow equation; therefore, a table interpolation scheme was used to determine spatial distribution of flow and the quantity of matrix versus fracture flow. The tables used for these interpolations were based on the results of a dual-continuum approach set forth in the *DCM3D* computer program for simulating Yucca Mountain. The interpolation scheme made use of sampled data for the infiltration rate and the hydraulic properties of the matrix and fractures.

FLOWMOD uses these relationships to determine the mass fluxes and particle travel times for each of the stratigraphic units comprising a certain number of vertical columns corresponding to each of the geologic repository sub-areas.

LHS. The TSPA user can specify various parameters pertaining to any number of consequence modules to be sampled where statistical distributions represent uncertainty. The *LHS* (Latin Hypercube Sampling) module creates equally likely parameter vectors. Although only uncorrelated parameters were used in IPA Phase 2 calculations, the *LHS* module is designed to sample from correlated parameters also. Two special aspects of the *LHS* module are: (a) all sampled parameters, irrespective of which consequence module they belong to, are sampled at one time; and (b) for the analysis of any one scenario, a single call to the *LHS* module provides all the vectors or realizations.

NEFTRAN. The far-field transport of radionuclides is treated in the *NEFTRAN* (NEtwork Flow and TRANsport) module. This module simulates the transport of radionuclides in the aqueous phase away from the geologic repository and calculates the integrated discharge of radionuclides, over 10,000 years, at the boundary of the accessible environment and the time-varying concentration of radionuclides at this boundary. In the simulation of radionuclide transport, NEFTRAN accounts for the following two primary factors: (a) element-specific retardation of radionuclides, based on the geologic unit and the degree of saturation; and (b) multiple flow paths, to represent the possible dual flow paths caused by fracture and matrix flow.

SEISMO. This module calculates the probabilities of failures of waste packages as a result of a seismic event. The probability of occurrence of an event of certain magnitude is considered to be time-dependent. To simplify the analysis, a seismic hazard curve representing time-dependence of earthquake magnitudes (peak accelerations) at a certain probability level (e.g., 95 percent) is first obtained. This curve is obtained from a family of postulated plots between the occurrence probability versus earthquake magnitude for a set of fixed time periods. Based on the structural properties of the container material, a fragility curve representing a relation between peak acceleration and the critical container wall thickness is derived. The actual container wall thickness, as affected by corrosion processes, is obtained from *SOTEC* as a function of time, which produces a time history of nominal wall thickness, when considering the undisturbed case. Any time the critical wall thickness obtained from the fragility curve is greater than the actual thickness produced by the *SOTEC*, failure occurs. The number of such failures is fed back to *SOTEC* for calculation of the source term.

SOTEC. This module calculates aqueous and gaseous radionuclide time- and space-dependent source terms for the geologic repository. It does so by considering the variations in those physical processes expected to be important for the release of radionuclides from the EBS. Carbon-14 is the only radionuclide that is treated in the gaseous phase in IPA Phase 2. However, all radionuclides, including carbon-14, are considered in the aqueous phases. SOTEC makes three primary calculations: (1) failure of waste containers because of a combination of corrosion processes and mechanical stresses; (2) the leaching of spent fuel; and (3) the release of $^{14}CO_2$ gas from the oxidation of UO_2 and other components in the spent nuclear fuel and hardware.

VOLCANO. This module calculates consequences caused by magmatic events. In the geometric approach followed in *VOLCANO*, Monte Carlo sampling is used to generate a volcanic event randomly in a rectangular region surrounding the repository horizon. Random sampling is used to specify: (a) the location of the sampled volcanic eruption; (b) the nature of the volcanic event (intrusive, leading to dike formation, and extrusive, leading to dike and/or cone formation); (c) the dimensions of the dike or cone; and (d) the orientation of the dike. Based on the area of the geologic repository intercepted by dikes and cones, VOLCANO determines the numbers of waste packages failed by the magmatic event assuming all intercepted waste packages have failed. This information is used in *SOTEC*. When a volcanic event is extrusive, the contents of the failed waste packages are assumed to be released to the accessible environment (direct release), and a fraction of this is assumed to be ejected to the atmosphere, which is then used in *AIRCOM* to calculate human dose, and in the executive module, to calculate the total release.

Overall Performance of the Computation Code

The NRC's TSPA system manager, *Executive Module*, is being modified to investigate the feasibility of converting it to a coarse grain parallel system. This effort is motivated by the exceedingly long run times for the Phase 2 TSPA code (run times were between 1.5 and 2.0 min/vector on the Idaho National Engineering Laboratory Cray, depending on the scenario). For a full complement of 16 scenarios of 400 vectors run, run times range from 6.6 to 8.8 days. Long run times like these greatly reduce the iterative nature of the PA process and impede its investigations. It may also be necessary to provide many more vectors per scenario for proper analysis of the probabilistic nature of the consequence modules. The parallelizing process is intended to reduce the run times to permit more frequent iterations of the system with more vectors in each scenario.

It is anticipated that several SUN platforms at the CNWRA and NRC, plus one or two Cray sites, can be used in a heterogenous collection of host machines for the Parallel Virtual Machine (PVM) implementation of the TSPA code. These platforms will be under the control of the TSPA *Executive Module,* which assigns each machine a single unique vector to process. The *Executive Module* makes sure that all machines are busy and that a given host is operating on only one vector at a time. This could potentially reduce the run time by a factor of 5 or 6 and produce a complete set of CCDFs in about one day, greatly increasing the iterative capability and responsiveness of the TSPA system.

7.6.8 NRC Evaluation of DOE Options

NRC has not evaluated different DOE options in their TSPA effort to-date.

7.6.9 Key Findings and Action Items of NRC TSPA Efforts

7.6.9.1 Regulatory Focus of NRC's Iterative Performance Assessment

Iterative Performance Assessment is a key component of the License Application (LA) review strategy in the NRC Overall Review Strategy. In the pre-LA phase, the IPA technical analysis capabilities are being developed and interactively applied to identify and address potentially contentious technical issues before DOE submits the LA. The technical capabilities of IPA are focused on determining which site and repository components and features most strongly influence overall system performance. In the LA phase, NRC anticipates using an audit-based approach in which only selected areas of the LA are investigated in detail. Predictive tools and methodologies developed in IPA will be used for selective probing into the assumptions, models, and data used in the DOE TSPA for Yucca Mountain.

7.6.9.2 Benefits of the IPA Process

The IPA process has demonstrated both direct and indirect benefits to the HLW regulatory program. The process has been useful in:

- establishing a broad understanding of the behavior of the repository and the surrounding geologic medium
- providing the technical basis for commenting on DOE repository design, site characterization, and PA programs
- providing a basis for raising and addressing technical concerns during the pre-LA period
- establishing the framework for CDMs and assisting in the development of the NRC License Application Review Plan
- determining the feasibility of implementing regulatory requirements and identifying modifications, if necessary
- developing relevant information needed to identify and prioritize NRC-sponsored research activities

Specific results of the IPA process include:

- Dominant radionuclides for population dose include niobium-94, lead-210, americium-243, neptunium-237.
- The subsystem performance measures of GWTT and substantially complete containment show a strong positive correlation with the overall release-based performance measure, and the fractional release rate performance measure shows a milder, yet still positive, correlation.

- The scenario methodology employed by the NRC is sound and suitable for conducting and evaluating a TSPA.

7.6.9.3 License Application Review Plan

Staff experience gained through IPA has benefitted the preparation of the License Application Review Plan (LARP). This experience has been applied in the consolidation and integration of the LARP Key Technical Uncertainties (KTUs). In the area of total system performance, five major KTUs were identified pertaining to performance assessments including: 1) conceptual model; 2) computational model; 3) model validation; 4) model parameters; and 5) future states.

Conceptual Model

Staff experience gained in the IPA has contributed to identifying and understanding uncertainty in the conceptual model. For regulatory purposes, a conceptual model describes the conditions or processes believed to exist in the system under consideration, the geometry and dimensionality of the system, the temporal and spatial scales of the conditions or processes, the parameters governing the conditions or processes, and the initial or boundary conditions. The main challenge is to develop an acceptable procedure to evaluate the representativeness and completeness of conceptual models used by DOE in its TSPAs. It has been proposed that when alternative conceptual models exist, independent calculations be performed using each model. The results of independent calculations would then be compared. It is not suggested that alternative models be blended into a single model, for the blending process appears problematic. IPA has been identified as a useful tool for evaluating alternative models.

Computational Model

To predict the performance of the system, conceptual models need to be translated into mathematical or computational equations. The set of one or more equations that represents a given conceptual model is the associated mathematical model. The need for solution tractability in general requires that simplifying assumptions be involved to reduce the qualitative description in the conceptual model to mathematical equations. Because the validity of the assumptions used in the formulation of the mathematical models cannot be ascertained a priori, there is uncertainty associated with these models. The models typically used to simulate the behavior of the repository system, or any of its subsystems, are quite complex and generally do not allow for analytical solutions. Instead, the solution of these models often requires the use of numerical techniques (such as finite elements and finite differences) that need to be implemented in computer codes.

There is no single approach to the development and application of mathematical models and computer codes for total system and subsystem PAs. Therefore, the staff will determine the appropriate approach to establish the representativeness and adequacy of mathematical models and computer codes used by the DOE in its PAs. Through the conduct of IPA and other detailed analyses, the NRC staff will be able to examine the effect of assumptions embedded in the DOE models and codes on the PA results. This will allow the NRC staff to identify those assumptions having the largest impact on the PA for the Yucca Mountain repository.

Model Validation

During the LA phase, the NRC staff will need to evaluate the models and codes used by DOE to demonstrated the compliance of the proposed repository. The staff will need to ascertain whether the models adequately represent the conditions and processes existing in the system. Due to the extended spatial and temporal scales, prediction of the systems and subsystems will be made with minimal scientific observations. As a result, large uncertainties in model predictions will persist.

Staff experience gained in IPA has contributed to the development of a preliminary model validation strategy applicable to performance assessment. The proposed method has been developed as part of international model validation activities in a joint effort with the Swedish Radiation Protection Inspectorate.

Model Parameters

The numerical values of the parameters needed to run the models and codes are obtained through estimation. This involves interpretation or analysis (e.g., parameter fitting, interpolation, scaling, etc.) of typically sparse field or laboratory data. Limits in the ability to characterize fully a repository system and its inherent temporal and spatial variability lead to uncertainty in the numerical values of model parameters.

The NRC advocates the use of IPA to increase the likelihood that data collection generates the information most critical to the estimation of performance. The basic tenet behind such a philosophy is that by propagating the uncertainty in input parameters to the uncertainty in the estimate of the performance measures (i.e., uncertainty analysis), and by identifying those parameters that have the largest influence on the performance measure (i.e., sensitivity analysis), data collection can be focused on the most important parameters.

The IPA exercises can identify the important parameters, and through pre-LA reviews of the DOE total system and subsystem performance assessments, the NRC staff is able to provide guidance to DOE about those parameters. In this manner, NRC will provide guidance and recommendations on site characterization activities. Finally, as part of the IPA effort, the NRC staff will examine different procedures in order to construct the distributions needed to represent the uncertainty in the numerical values of parameters.

Future States

The performance of the proposed repository must be estimated over 10,000 years. Such estimation will require consideration of the possible future states of the repository system over that timeframe. The complexity of the system and the length of the regulatory period introduce uncertainty. To account for this uncertainty, the NRC and the DOE develop scenarios believed to represent possible future states of the system. Each scenario is assigned a probability of occurrence representing the degree of belief that the scenario will occur. 40 CFR Part 191 assumes that performance assessments need not consider categories of events or processes that are estimated to have less than one chance in 10,000 of occurring over 10,000 years; that is, scenarios with a probability of less than 10^{-8}/yr need not be considered. Therefore, some accuracy in the estimation of probability of occurrence is required. Through the IPA effort, NRC is applying and evaluating different approaches or methods for developing scenarios and for estimating their probability of occurrence. The insights gained from the IPA exercises will be a major source of information for formulating guidance on scenario development and on estimating their probability of occurrence.

In addition to the five KTUs pertaining to the TSPA, the IPA has identified three other key parameters or factors important to the overall performance of the geologic repository. These three key parameters and their relative importance in the TSPA are described below.

Infiltration

The major TSPAs conducted to date by the NRC, DOE, and the Electric Power Research Institute (EPRI) have consistently found that infiltration rate is a key parameter that strongly influences postclosure performance. This finding has motivated a number of subtasks at the Center for Nuclear Waste Regulatory Analyses (CNWRA), including an elicitation of future climate, a study of climate-linked infiltration, shallow near-surface infiltration, and deep percolation (recently started). Each of these activities is directed at obtaining better estimates of the water flux through the repository horizon. All of these efforts are integrated when the climated-linked model uses elicitation data for climatic conditions, such as cloud cover, atmospheric temperature, and precipitation, along with established models for evapotranspiration. Parameter uncertainties were also evaluated and evapotranspiration was found to be an important parameter. The results suggest site characterization include field lysimeter studies to better quantify evapotranspiration at the site.

Following the climate-linked study, more detailed modeling studies were initiated to better quantify the movement of water in the near-surface (top 10m) to contribute to percolation. This study has found that seasonal variations penetrate deeper than daily variations, vapor diffusion is an important mechanism in the top half meter, and in regions where there is shallow cover, the Paintbrush Tuff (PTn) layer is strongly influenced by atmospheric conditions. This suggests infiltration is spatially correlated with alluvial conditions that vary with the topography.

In the deep infiltration study, the spatial and temporal focusing of deep infiltration (i.e., percolation through the repository horizon) is being characterized for Yucca Mountain. This effort uses estimates of shallow infiltration based on alluvium cover. An initial set of uniform calculations was performed, then more realistic spatially focused cases were considered. Findings indicate the PTn unit exhibits significant lateral flow, a juxtaposed geologic strata provides the opportunity for perching of water to occur, and preferential spatial focusing along the Solitario Canyon fault has a large effect on the deep percolation through the repository.

Faulting

DOE has identified several postclosure qualifying and disqualifying conditions as part of the Technical Site Suitability Determination to be made by the Secretary of Energy. Seven High-Level Findings (HLF) are scheduled to be completed between July 1996 and June 1998 to support the TSS determination expected in September 1998. Three of the HLFs are based on an assessment of postclosure issues; the first postclosure HLF deals with tectonics as a disqualifying condition of the site. DOE describes the disqualifying condition as fault movements that are expected to cause loss of waste isolation. The purpose of this IPA study is to design a new FAULTING module for the NRC TSPA code, in a collaborative effort of NRC/CNWRA geoscientists, to address this HLF.

It is assumed that DOE will take precautionary measures by siting the repository away from known faults; hence, only undetected faults are expected to pose future risks to waste isolation. For the Yucca Mountain region, the probability of one (or more) currently undetected fault(s) becoming active (having a large discrete slip event) is being quantified and used as the probability of the scenario. The new TSPA module determines the location, strike orientation, dip angle, length (assuming negligible width), recurrence interval, slip magnitude (for a large discrete event), and continuous slip rate. Currently, only direct disruptive effects of fault movement are modeled, although it is recognized that indirect effects through hydrology are potentially important. Having described the fault, the magnitude of discrete slip is compared to the threshold displacement required to induce waste package failure. Currently, the threshold displacement is based on the structural integrity of the package and the potential mechanical load induced by a fault-degraded drift. If a disruption occurs, then the module transfers waste package disruption information to the source term model of the TSPA code.

At this time, the technical basis for the module has been established and is being reviewed before producing the software. This module is being developed for NRC's use in independent assessments of the probabilities and consequences of faulting at the Yucca Mountain site. Hence, it will be used to evaluate the postclosure tectonics HLF that will support the TSS determination.

Seals

Site characterization activities at Yucca Mountain will result in numerous boreholes being drilled at various locations and depths to gather data for characterizing the hydrologic, geochemical, structural, and seismic properties of the underground environment. There are approximately 191 existing and 322 proposed boreholes within the region, varying in depth from very shallow (1.5 m) to very deep (1,830 m), and with diameters from 0.15 to 10.45 m.

The NRC regulations for seal performance require that "Seals for shafts and boreholes shall be designed so that following permanent closure they do not become pathways that compromise the geologic repository's ability to meet the performance objectives" (10 CFR 60.134). In developing its LARP, two KTUs were identified for seals and seal performance. Because seals were not explicitly included in IPA Phase 2, NRC decided to conduct an independent IPA auxiliary analysis in Phase 3 to investigate the importance of degraded seals. In doing so, IPA would provide quantitative input to LARP development on this issue. This activity has been completed and documented.

The analyses are based on consideration of both gaseous and aqueous phases. Gas streamlines will have the characteristics of large-scale buoyant flow; hence, the flow will be upward through the repository and directed towards the ground surface. The liquid streamlines will be dominated by downward gravity-driven percolation; hence, aqueous phase flow will be predominately downward through the repository horizon and directed towards the water table. The fluid particle travel time was selected as the performance measure used to assess the effects of a degraded seal on the flow system. For gaseous flow, fluid particle travel time over the distance from the repository horizon to the ground surface is a key factor. For aqueous flow, the fluid particle travel time over the distance from the repository horizon to the water table is important. A degraded seal (i.e., more porous and permeable) will lead to shorter travel times, resulting in less effective isolation of the waste.

For the assumptions and conceptual model employed, the calculations support the conclusion that postclosure performance of the seals may not be a significant factor in the TSPA. Based on this analysis, the current review type for the seals KTU can be re-evaluated and possibly reduced.

7.7 PERFORMANCE ASSESSMENT RESULTS, POTENTIAL DOSES, AND KEY TECHNICAL ISSUES

Performance Assessment Results

Assessments of the waste isolation performance of deep geologic repositories containing spent fuel and HLW have been performed by DOE, NRC, the National Academy of Sciences (NAS 83), EPA, and the Electric Power Research Institute (EPRI). None of the results obtained to date purports to be representative of a repository at Yucca Mountain; the reports of performance assessment (PA) results in fact carefully and clearly state that they are not intended to demonstrate performance of an actual Yucca Mountain repository.

The DOE performance assessments to date have been directed at exploring the relative importance of parameters important to waste isolation, characterizing the performance of alternative engineered designs, and providing guidance for data acquisition programs. The purpose of the NRC studies was to develop and demonstrate capability to perform reviews of a license application for construction and operation of a repository. The EPA studies were generic evaluations to support the 40 CFR Part 191 regulations, and the EPRI studies were performed to support alternative regulations proposed by EPRI (EPR 94).

Although the DOE and NRC PA results are stated not to be representative of a repository at Yucca Mountain, the studies to date have been based on concepts and characteristics of that site because of the Nuclear Waste Policy Act directive to evaluate only that site for spent fuel and HLW disposal. A principal reason for stating that the results to date are not representative of a Yucca Mountain repository is the fact that the assumptions and uncertainties that underlie the available results are extensive. For example, there is significant uncertainty concerning flow characteristics of the hydrologic regime, and DOE has not yet selected a repository design. A particular combination of parameters used by DOE in its extensive PA studies to date may eventually prove to be appropriate for a repository at the site, but at present there is no way to know what that combination is. Similarly, NRC states as caveats on its Iterative Performance Assessment (IPA) Phase 2 results (NRC 95b) that its models are based on limited data and review; results cannot be confirmed; models are simplified; there are large input data uncertainties; coupled effects have not been fully modeled; and the dose calculation is only illustrative.

The results of most of the previous performance assessments by both the NRC and DOE have been expressed in terms of cumulative release of radionuclides from the repository to the accessible environment. These results were appropriate for testing compliance with the EPA 40 CFR Part 191 regulations that applied to the Yucca Mountain site until 1992 (EPA85 and EnPA92). The 40 CFR Part 191 regulations required use of Cumulative Complementary Distribution Functions (CCDF) that displayed exceedence probabilities and the normalized cumulative release of nuclides from the repository to the accessible environment. Figure 7-36 illustrates the NRC IPA Phase 2 results of this type; details of the basis for these results are provided in the text of NRC95b. The NRC also performed extensive related studies, such as evaluating of the sensitivity of releases to waste-package failure time and liquid travel times.

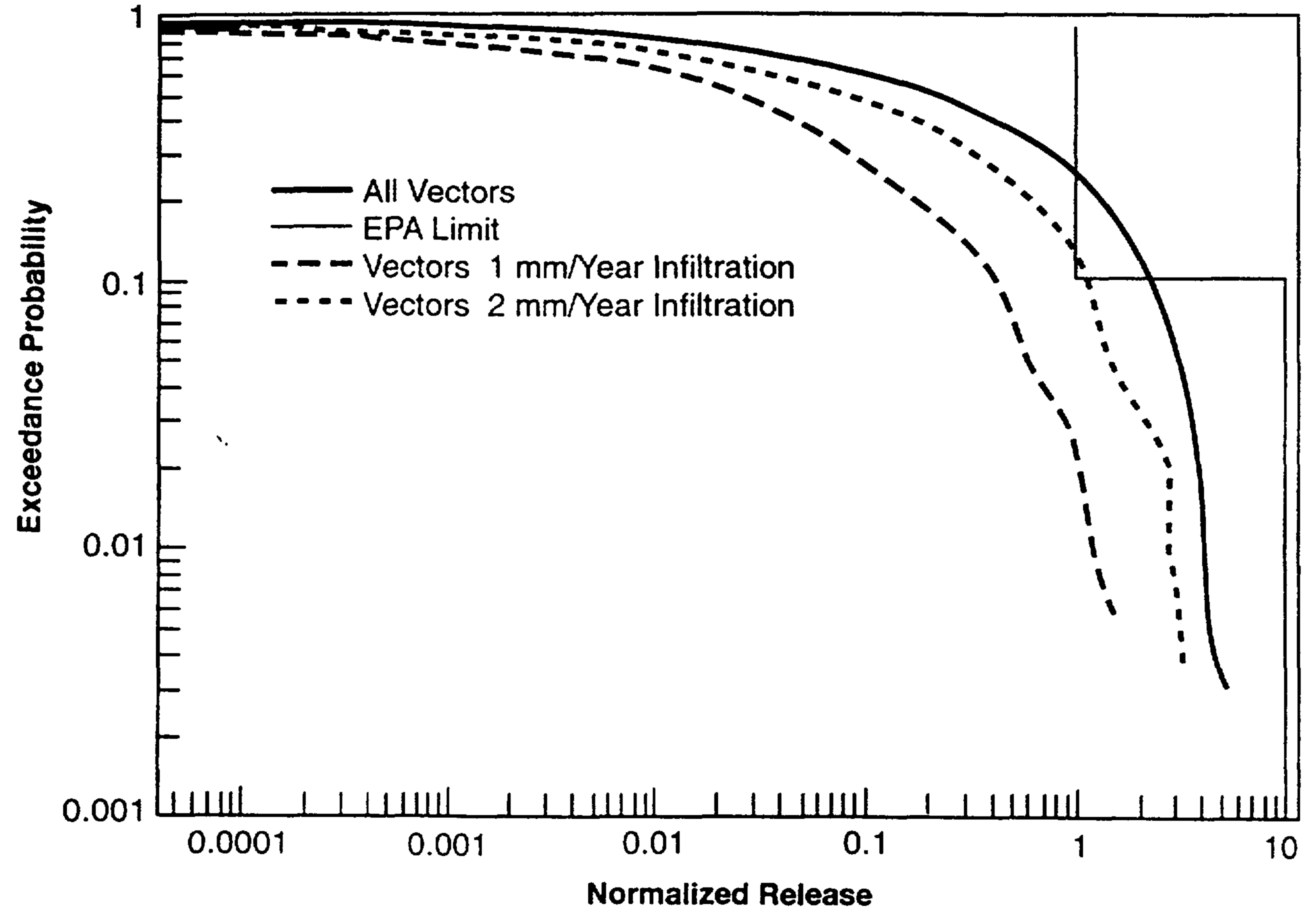


Figure 7-36. CCDF for Dissolved Radionuclides, Base Case Scenario (Vectors screened for less than 1 or 2 mm/yr infiltration) (Source: NRC95b)

Results of DOE performance assessments are illustrated by Figure 7-37, which was derived from TSPA-93 (DOE94a). Figure 7-37 was chosen to illustrate the DOE results because it indicates the cumulative releases to the accessible environment for periods of 10,000, 100,000 and 1,000,000 years under the same assumed conditions. Figure 7-38, also taken from TSPA-93, illustrates the effect of repository design alternatives under consideration on performance. DOE has done numerous PA studies for a wide range of assumed conditions, as reported in TSPA-91, TSPA-93, and TSPA-95 (DOE 92, 94a, and 95c, respectively).

Potential Radiation Doses

Evaluations by the NRC and DOE of potential radiation doses to humans in the biosphere outside the repository have been comparatively limited due to the previous focus on integrated release of radionuclides as the compliance measure. Dose estimations require assumptions

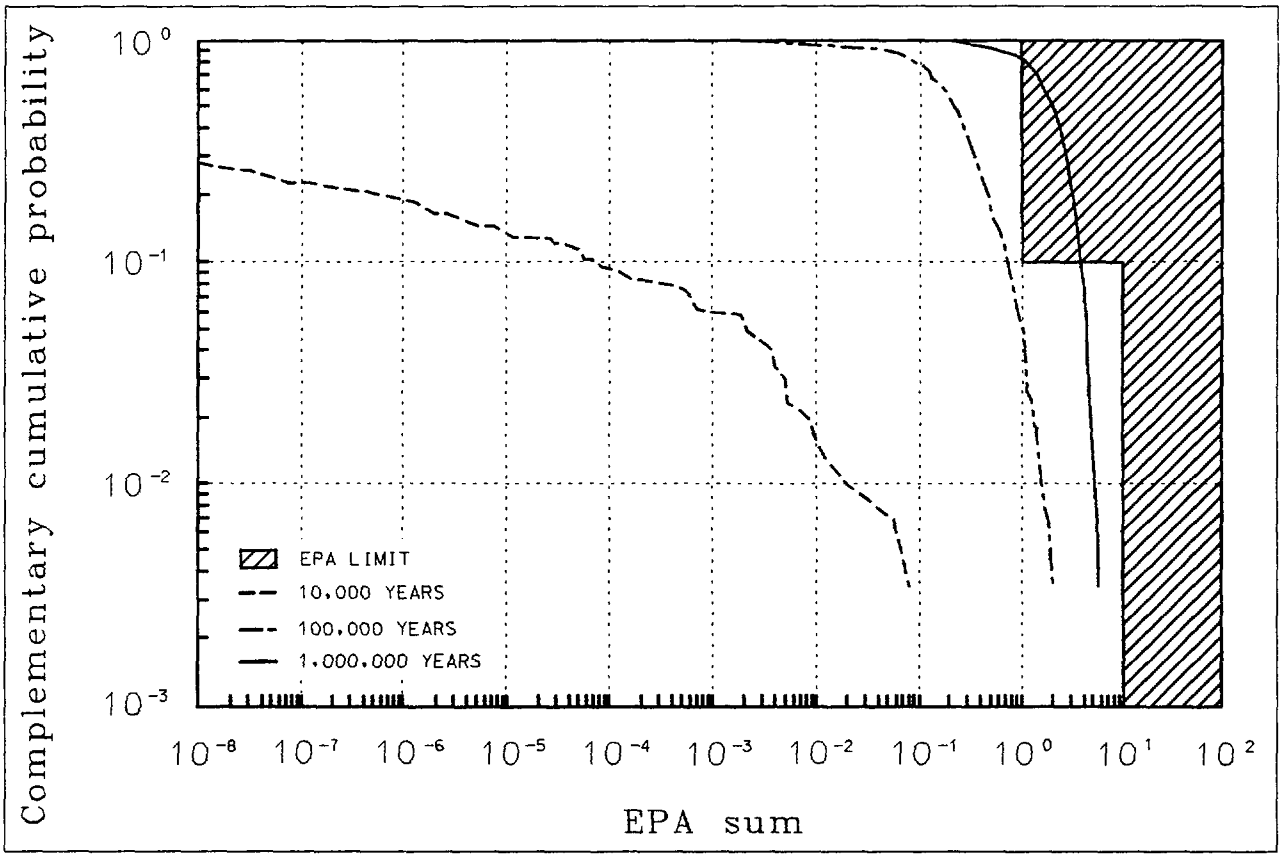


Figure 7-37. CCDFs for Normalized Cumulative Aqueous Release to the Accessible Environment over 10,000, 100,000 and 1,000,000 Years (57 KW per acre, vertical emplacement) (Adapted from DOE94a)

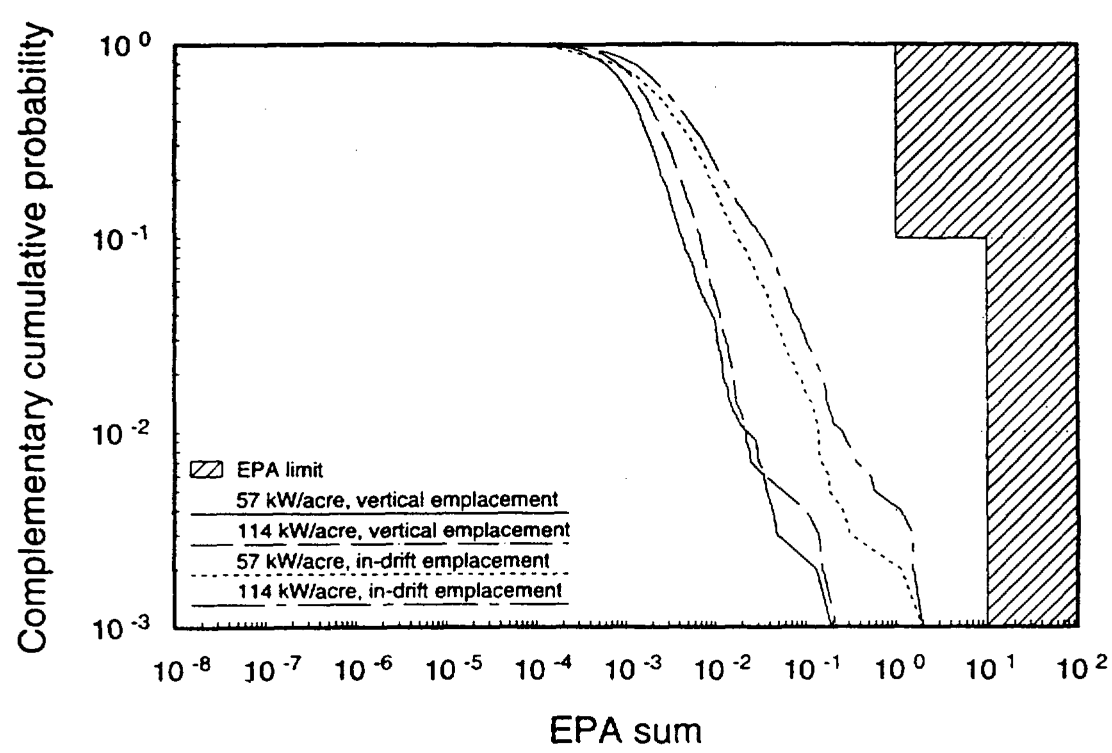
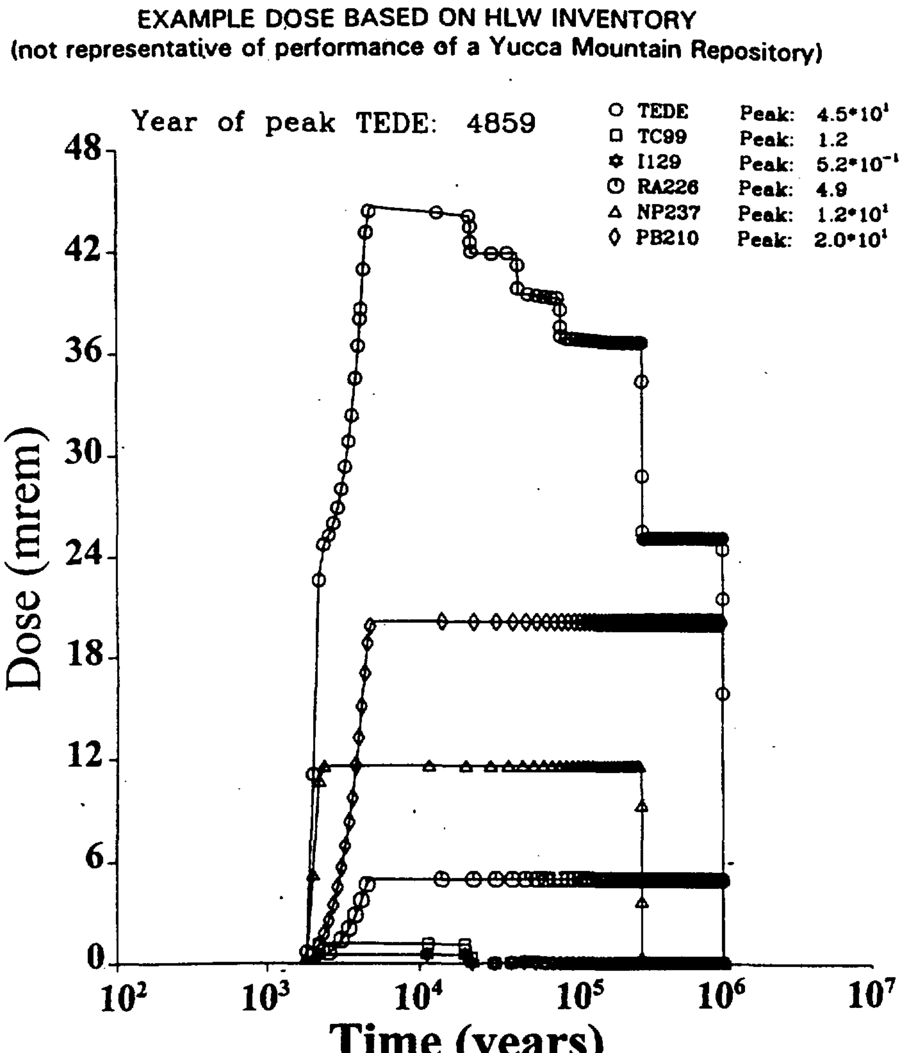


Figure 7-38. Comparison of the Combined Conditional CCDFs of Cumulative Aqueous and Gaseous Releases to the Accessible Environment, at 10,000 Years, Normalized by the EPA Limits, for the Four Repository Cases (Cross-hatching denotes area of noncompliance with the EPA standard) (Source: DOE94a)

concerning characteristics of the critical group of dose receptors and data or assumptions concerning characteristics of the hydraulic regime outside the repository. Available data on the deep hydraulic regime in the vicinity of Yucca Mountain are at present extremely limited.

The NRC made what they described as "crude estimates" of radiation dose for limited conditions as part of their IPA Phase 2 effort (NRC95b). These studies were done principally to illustrate some of the statistical techniques available to NRC for use in future performance assessments. The evaluations were done for a) an individual member of a farm family whose only source of drinking water was a well pumping water contaminated with radioactivity released from the repository, and b) for 177 individuals who reside within 100 kilometers of Yucca Mountain and eat contaminated beef. Results were obtained as histograms of frequency of annual effective dose equivalents in ranges up to 6 rem/year for the waterborne release and up to one rem/year for the contaminated beef. The histograms showed that more than 80 percent of the doses would occur at very low levels for the assumed conditions.

As part of an illustration to the Advisory Committee on Nuclear Waste the NRC staff also recently developed some dose estimates using factors that might affect the choice of the regulatory compliance period (NRC96). Repository-like inventories of long-lived, mobile radionuclides (Tc-99, I-129, Np-237, Se-79, and U-234) were assumed, and dose levels, peak times, and durations were calculated. For the scenario conditions assumed, which were stated not to be representative of a Yucca Mountain repository, the peak annual dose of 45 mrem occurred at 4,859 years, and the dose dropped gradually to a level of 24 mrem at one million years. These NRC results are shown in Figure 7-39. These studies illustrate the potential for a non-negligible dose beyond a regulatory period of 10,000 years. They also show the relative contributions of the various radionuclides to potential dose.



EXAMPLE DOSE BASED ON HLW INVENTORY
(not representative of performance of a Yucca Mountain Repository)

Figure 7-39. Dose Estimates for Repository-like Inventories

The timing of the peak dose, the nuclides contributing to dose, and the magnitude of the peak dose will depend on many factors. Key factors include time and rate of release from the repository; travel time to the saturated zone; dilution and dispersion in the saturated zone; distance and rate of travel in the saturated zone to the critical group; and critical group lifestyle. There are currently many uncertainties in the technical factors that affect peak dose. For example, the time and rate of release from the repository will depend on the water infiltration rate, which cannot be measured directly, and on engineered design of the repository, which has not been selected. Travel time to the saturated zone will depend on how the flow distributes between matrix and fracture flow, and travel times for fracture flow may be much faster than matrix flow. Information on hydrologic properties of the saturated zone, such as flow paths and rates, and potential for dilution and dispersion, is highly limited because of the prior focus, under EPA's 40 CFR Part 191 regulations, on evaluation of cumulative releases at the accessible environment boundary.

The NRC investigated the effect of matrix and fracture flow on travel time as part of their IPA Phase 2 studies (NRC95b). These studies showed that travel times for fracture flow could be on the order of two orders of magnitude shorter than travel times for matrix flow. Depending on factors such as nuclide half-lives and path lengths, travel time differences of this magnitude (i.e., fracture flow versus matrix flow) could have a significant effect on the nuclides contributing to dose and the time at which peak doses occur. DOE studies of travel time have also shown the potential for widely different travel times in the Yucca Mountain geohydrologic regime (DOE95c).

DOE evaluations of dose have been concerned with persons located at the boundary of the Accessible Environment as defined in 40 CFR Part 191, i.e., 5 kilometers from the boundary of the repository. In TSPA-95, DOE reported results of peak dose evaluations at the accessible environment boundary for a large number of assumed repository conditions, such as design characteristics and infiltration rates. Examples of DOE's dose evaluation results, taken from TSPA-95 (DOE95c), are shown in Figures 7-40 and 7-41, which are reproduced from DOE95b. These figures illustrate the effect of infiltration rate on the time and magnitude of peak dose at the boundary for two nuclides under the same assumed repository conditions. Results are presented for a million-year dose history.

Alternate Infiltration Rates (0.03 vs. 1.25 mm/yr)

(83 MTU/acre, gravel backfill, climatic variation of q_{Inf})

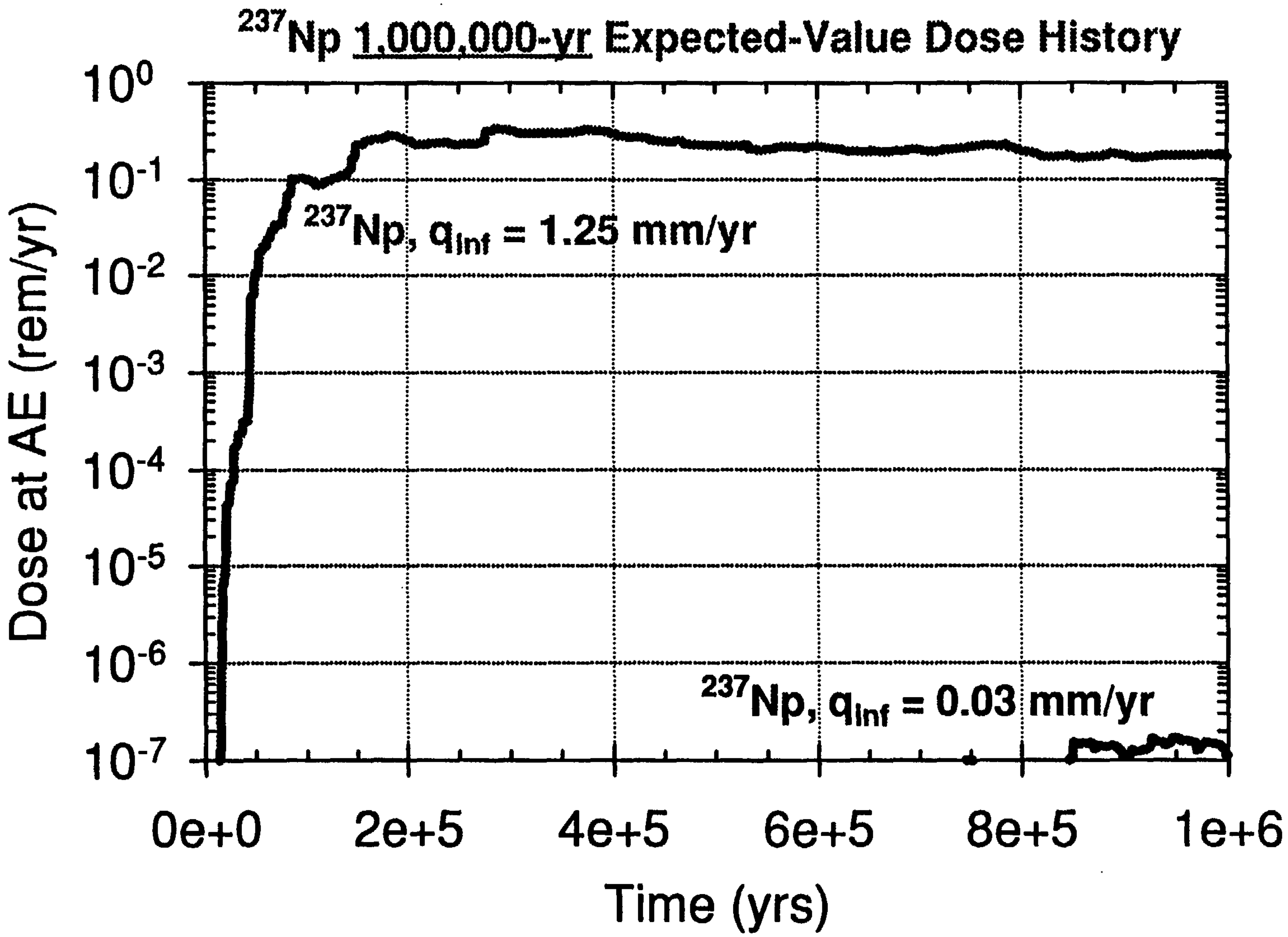
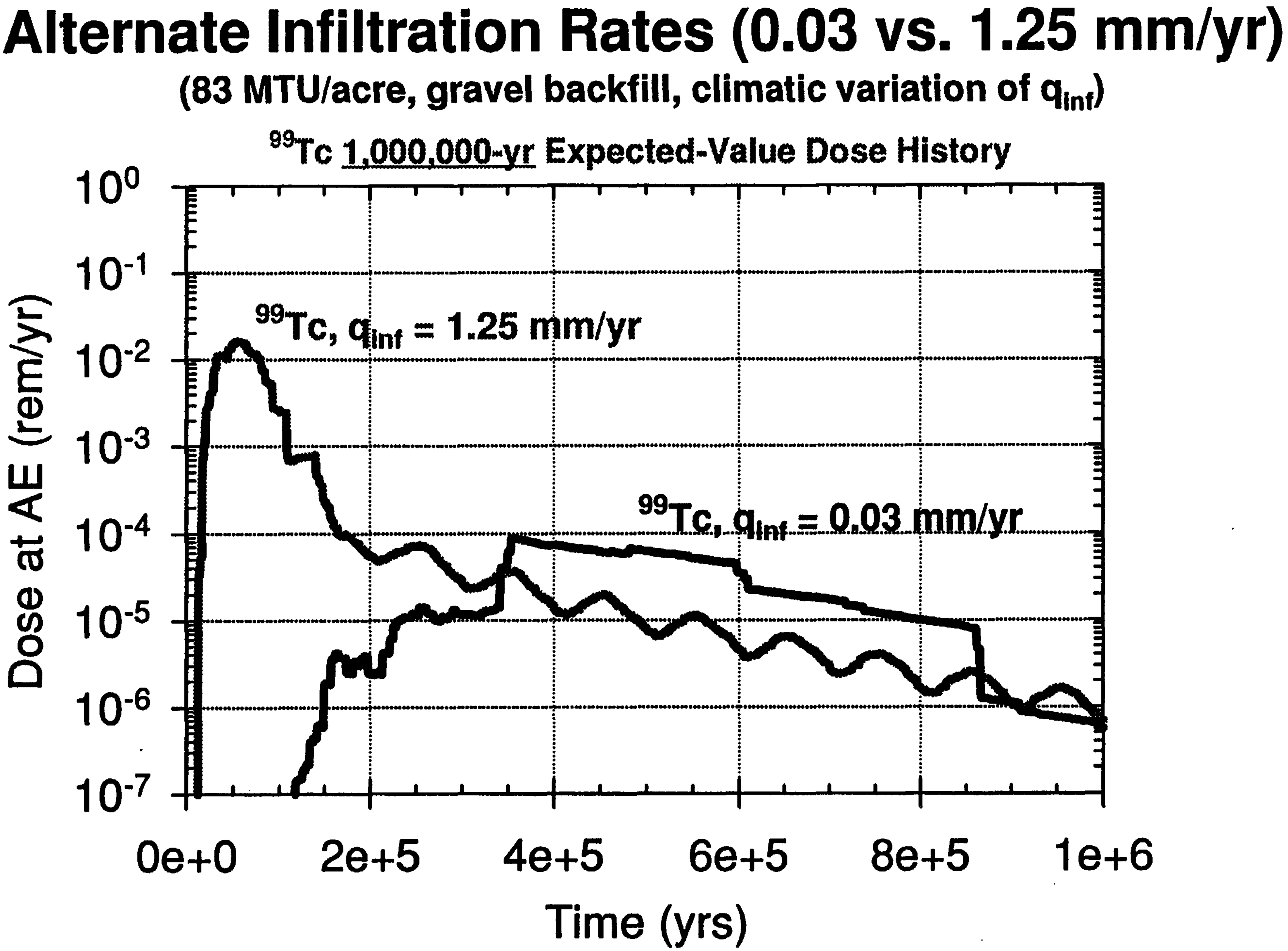


Figure 7-40. Effect of Infiltration Rate on Radiation Dose from Np-237 (Source: DOE95c)



Alternate Infiltration Rates (0.03 vs. 1.25 mm/yr)

(83 MTU/acre, gravel backfill, climatic variation of q_{inf})

^{99}Tc <u>1,000,000-yr</u> Expected-Value Dose History

Figure 7-41. Effect of Infiltration Rate on Radiation Dose from Tc-99 (Source: DOE95c)

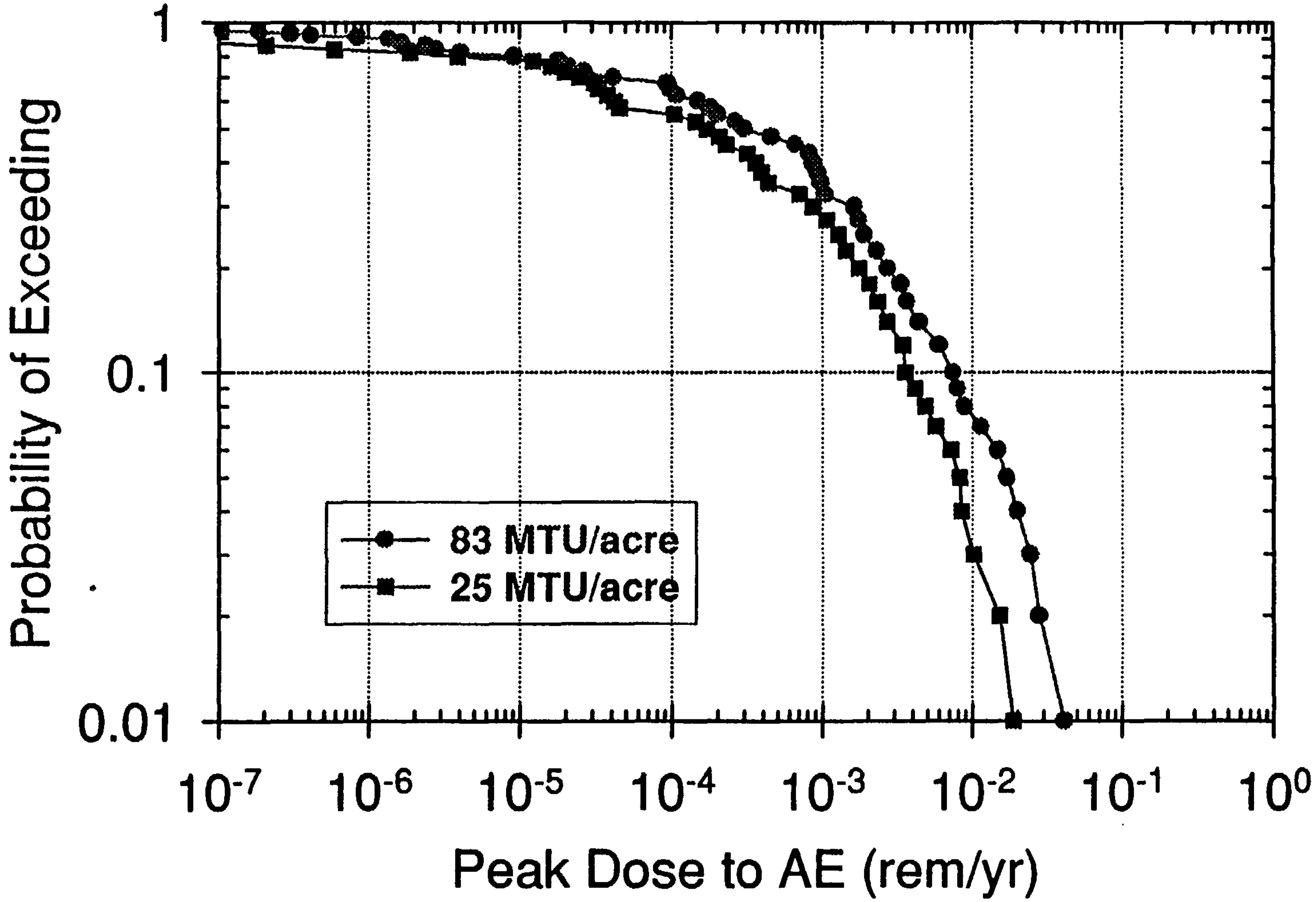
As can be seen in Figure 7-40, the time and magnitude of peak dose for Np-237 is dramatically affected by the infiltration rate. Figure 7-41 shows that the effect of infiltration rate on peak dose from Tc-99 is relatively small. Comparison of Figures 7-40 and 7-41 shows how the contributions of different nuclides to peak dose can differ under the same conditions as a result of different nuclide properties. At an infiltration rate of 1.25 mm/year, the peak Np-237 dose during the period 200,000 to one million years is relatively high and remains essentially constant. During the same period, the Tc-99 peak dose is initially about four orders of magnitude less than Np-237 and diminishes over the period by two orders of magnitude.

An example of DOE peak dose results at the accessible environment for the 10,000-year time frame is shown in Figure 7-42. These results show that the peak dose is virtually independent of thermal loading for the conditions considered, and there is a 1 percent or less chance of achieving a dose on the order of 0.01 to 0.1 rem/year, which is the dose level achieved with Np-237 at a time on the order of 100,000 years with a high infiltration rate (Figure 7-40). Also of interest in Figure 7-42 is the fact that these results correspond to relatively poor repository conditions, i.e., no backfill and high infiltration rates. In general, the DOE has found in their TSPA 93 and 95 analyses that releases and doses at 10,000 years can be prevented through use of engineering strategies such as cathodic protection or capillary control.

As previously noted, the DOE dose evaluations, such as those shown in Figures 7-40, 7-41, and 7-42, were performed at the boundary of the accessible environment. The peak dose for a critical group some distance from the repository may be significantly less than that at the accessible environment boundary because of dilution and dispersion processes in the saturated zone. DOE states, in TSPA-95, that "If the critical group is located in the Amargosa Valley, some 25 km down gradient from the present 'accessible environment' as defined in 40 CFR Part 191, the increased geosphere dispersion may be expected to reduce the peak concentration and peak dose by several orders of magnitude." Dilution and dispersion in the saturated zone are two of the key features of the hydrologic regime that can affect doses to the critical group.

Peak Dose for Alternate Thermal Loads

(no backfill, high q_{inf} range)



10,000-year Total Peak Dose

Figure 7-42. Example of Peak Dose at 10,000 Years (Source: DOE95c)

Key Technical Issues

As a result of performance assessments and other analyses of available information concerning Yucca Mountain and a potential repository at the site, NRC and DOE have identified a number of key technical issues concerning waste isolation performance. As part of development of its License Application Review Plan, the NRC identified about 100 detailed Key Technical Uncertainties. NRC and DOE interactions have determined the most significant of these issues, which are summarized below.

1. Rate of infiltration of water into the repository. DOE and NRC studies both show that the water infiltration rate is key to repository performance. Current estimates range over two orders of magnitude (0.01 to 2 mm/year); DOE's TSPA-93 studies indicate that the infiltration rate could approach zero under current conditions and could be as great as 10 mm/year in a wet climate. Figure 7-41 illustrates that the infiltration rate can have a very large effect on timing and magnitude of peak dose.

2. Mechanism(s) of flow in the unsaturated zone. The basic options for unsaturated-zone flow mechanisms that have been modeled are fracture flow and a combination of matrix and fracture flow. The modeling studies show that the actual flow mechanism for infiltration flow to the repository will strongly affect how much water can intercept emplaced canisters and lead to canister degradation and waste transport. For example, the Executive Summary of TSPA-93 shows, in Figure ES-10, that the peak drinking water doses within one million years are two orders of magnitude higher for the composite-porosity model, which describes matrix/fracture flow, than for the Weeps model, which describes fracture flow.

   Mechanisms of unsaturated-zone flow are important both for infiltration flow to the repository and percolation flow from the repository to the saturated zone, which lies several hundred meters below the proposed repository horizon. Flow mechanisms for infiltration and percolation flow may not be the same because of differing structural characteristics, such as extent of fracturing of the various geologic strata above and below the repository. Flow characteristics in the unsaturated zone cannot be directly measured.

3. Effect of design options on engineered barrier system (EBS) performance. Key design parameters are selection of canister materials, thermal loading imposed on the hydrogeologic regime, waste emplacement mode (horizontal or vertical), and use of strategies such as humidity control, cathodic protection, and capillary barriers provided by backfill to enhance maintenance of waste package integrity and EBS performance. These factors interact in highly complex ways to affect the time at which

waste package failure starts, the rate of failure, and the time and rate of release of radioactivity from the EBS to the environment. Within the range of options considered, DOE has shown (TSPA-95) that waste package failure might be initiated within a few hundred years after repository closure or might be deferred for tens or hundreds of thousands of years, by using design strategies such as capillary control.

4. Flow phenomena in the saturated zone. These include directions, rates, and quantities of flow, and occurrence of dilution and dispersion of contaminated water from the repository. These factors will affect the timing, location, and quantities of radioactivity that can produce doses to humans. Data that provide the basis for characterizing of the saturated zone are at present highly limited. DOE has estimated (TSPA-95) that dilution during transport in the saturated regime from the repository to the critical group could reduce peak doses by as much as six orders of magnitude.

5. Potential for processes and events that can disturb and affect performance of the repository. Key processes and events of this type include seismic/tectonic activity, climate change, volcanism, and human intrusion. Selection and characterization of the occurrence and effects of these processes and events is largely a matter of judgment. The NRC and DOE agree that these are the performance-disturbing factors of importance, but complete agreement on details has not been established. For example, DOE and NRC agree that magmatic volcanism is potentially of importance, and they agree that the likelihood of volcanic activity is small. Current DOE and NRC views of the consequences of magmatic activity differ, however, because of differing views of the characteristics of the activity. DOE anticipates relatively quiet phenomena, such as exhibited by the volcanoes of Hawaii; NRC anticipates somewhat more violent phenomena. DOE's viewpoint indicates no significant effect of magmatic action on repository performance; NRC's viewpoint indicates important effects that cannot yet be neglected.

6. Verification of models of phenomena such as ground water flow, waste package degradation, and near-field chemical processes that affect and control waste isolation performance of the repository. Data that could verify models may not be obtainable (e.g., water infiltration rates cannot be measured directly), or residual uncertainties may inevitably remain large because of natural variability of the performance-affecting factor. Consensus expert judgment will be needed where data cannot resolve key issues.

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CHAPTER 8

PATHWAYS THROUGH THE BIOSPHERE

8.1 CURRENT BIOSPHERE CONDITIONS IN THE YUCCA MOUNTAIN REGION

This brief introductory section is included for completeness and to set the stage for the discussion of release pathways in Section 8.5. Chapter 7 provides more detail on the topics summarized here. The discussion here provides cross-references to the appropriate portions of Chapter 7 or Chapter 8, where more detailed information can be found.

8.1.1 Geology

8.1.1.1 Principal Geologic Features

Yucca Mountain is located about 150 km northwest of Las Vegas, Nevada, in the Southern Great Basin near the southwest border of the Nevada Test Site. The Great Basin extends over portions of Nevada, Utah, and parts of adjacent states. The topography of the Great Basin is characterized by relatively straight, parallel, mountain ranges, usually tending north to northwest, separated by broad and relatively flat basins and valleys. The structure of Yucca Mountain fits this pattern. The proposed site for the repository is within Yucca Crest, a major ridge that rises to more than 1,500 meters above sea level and more than 300 meters above the surrounding valley floors. (More detail is found in Section 7.1.1.1 and 7.1.1.3.)

Yucca Crest is delimited by the Solitario Canyon Fault to the west. To the east and north, the Crest is bounded by a more generalized area of several normal faults and possibly a few strike-slip faults. The Ghost Dance Fault transects the Yucca Crest block, but since only about 25 meters of vertical displacement has occurred, Yucca Crest is considered to be a relatively undisturbed block. (Further detail is provided in Section 7.3.3.)

The rock units in the Yucca Mountain region range in age from the Precambrian (primarily sedimentary) through recent alluvial deposits. During the Mesozoic era, the Paleozoic "basement" rocks were subjected to at least one major tectonic event that compressed them, causing folding and thrusting. The rock units of most interest to the repository are those of the unsaturated zone beneath Yucca Crest; they consist primarily of silicic ash-flows and air-fall tuffs ejected from the Timber Mountain-Oasis Valley caldera complex immediately north of the proposed site. These emissions formed a thick sequence of volcanic rocks ranging in age from 9.5 to 16 million years. The various layers correspond to major volcanic eruptions, and differ in how compacted ("welded") they are. Non-welded ash is quite porous, while welded ash-flow and ash-fall rocks are less porous. (More details can be found in Section 7.3.2.)

8.1.1.2 Seismic Activity

The relationship between faults and seismic activity in the Basin and Range area is not clear, due to several factors: the relative brevity of historical information in comparison to long earthquake recurrence intervals, the difficulty of locating epicenters in remote areas, and unknown fault geometry at great depths. Some seismic areas can be clearly linked with faults or fault systems that show evidence of surface disturbance, while other seismic areas show no surface faulting.

The region around Yucca Mountain contains five major seismic zones and also exhibits seismic activity that is not linked to any known major tectonic structural feature. Three of the five major zones have been the source of earthquakes that resulted in surface rupture within the last century, but earthquakes of magnitude five or greater that occurred between 1850 and 1992 within 320 kilometers of the Yucca Ridge site have been concentrated in two of the five major zones (the Central Nevada Seismic Belt and the Sierra-Nevada Great Basin Zone) as well as in the southern Mojave Desert and along the San Andreas fault zone.

Yucca Mountain itself is intersected by or near to nine major north-trending faults and five northwest-trending faults. In general, while the area is clearly seismically active, the geological seismic record suggests that recurrence intervals for surface rupture are very long—from thousands to tens of thousands of years. (More detailed information is found in Sections 7.3.3, 7.3.3.1, 7.3.4.2, 7.3.4.3, and 7.3.4.5.)

8.1.1.3 Natural Resources

Previous studies strongly indicated that the Yucca Mountain region is not rich in natural resources that would encourage development, and further work has supported this conclusion. In general, only shallow mining of industrial materials now takes place in the vicinity, no resources have been identified that would be likely to cause increased mining, and there are no ongoing or expected future activities to recover currently valuable minerals outside the controlled area that could lead to loss of isolation.

Mining for precious minerals has taken place in the past or is currently underway in the Bullfrog Hills/Bare Mountain area, Shoshone Mountain on the Nevada Test Site, Timber Mountain Caldera, and a few other locations on the Nevada Test Site. Some mineral claims were filed in the 1980s on the Yucca Mountain site, but these appeared to be nuisance claims and were purchased by DOE. A study published in 1989 evaluated the resources of the Yucca Mountain Addition (on BLM land) to support a land withdrawal request by DOE. The report identified no surface or near-surface mineral resources within the Yucca Mountain Addition and found the potential for base and precious metals to be very low. However, more evaluation is likely to be needed, since new types of deposits (e.g., disseminated gold and gold-silver) have been made in geologic formations and areas of the Basin and Range province that have historically produced base metals.

Earlier information indicated a very low potential for coal, oil, and gas in the Yucca Mountain region. A state-wide evaluation of petroleum potential concluded that southern Nye County has a low potential for petroleum resources, and none of the areas found to have high to moderate potential are within 100 to 130 kilometers of Yucca Mountain. In its assessment of the Yucca Mountain area for purposes of land withdrawal, the Nevada Bureau of Mines and Geology also determined that the potential for oil and gas was low. Still, some debate over various geologic models of oil needs to be resolved to assess their applicability to the Yucca Mountain area. Other studies have shown that the Yucca Mountain area has a very low potential for geothermal resources, uranium, and industrial minerals and other industrial materials.

Finally, a review of mining claims, together with the literature on resources, shows that the probability is extremely low that past mining or exploration activities produced any significant pathways to the accessible environment from the proposed repository site. There is also no evidence, judging from the extensive literature base, that future drilling, mining, or exploratory work in the region outside the controlled area would result in an inadvertent loss of isolation. The issue of whether future exploitation of resources could lead to releases greater than allowed by the guidelines is not completely clear, but recent studies support the interim conclusion that they would not. (Section 8.4.1 contains more detail.)

8.1.2 Hydrology

8.1.2.1 Principal Hydrologic Features

Surface Hydrology

Nevada is the most arid state in the nation, with mean annual precipitation of about 238 millimeters (9.4 inches). Precipitation is strongly influenced by topography, with the greater precipitation in the mountains producing localized excess moisture that provides most of the state's surface runoff and recharge. An average of about 54 million acre-feet of rain and snow falls annually, but most evaporates where it falls. Annual runoff from the mountains is only about 3.2 million acre-feet, and total annual recharge to ground-water reservoirs is about 2.2 million acre-feet. Thus, runoff and recharge represent only 10 percent of total precipitation. In the Yucca Mountain area, precipitation is about 150 to 170 millimeters per year, but evapotranspiration can be as much as 1,000 millimeters per year. This leaves very little water to infiltrate into the unsaturated zone.

The main drainage system for the Yucca Mountain area is the Amargosa Valley, which carries runoff from the region south through the Tecopa basin and into the southern part of Death Valley. The Amargosa carries runoff only after extraordinarily large events. There are no perennial streams or other bodies of surface water on or near Yucca Mountain. Major drainages to the east and west, and their tributaries, flow only briefly immediately after rainstorms. There are permanent springs southeast of Yucca Mountain, in Ash Meadows, but most of the spring discharge evaporates, with a smaller portion running off and eventually either evaporating or infiltrating into the subsurface. (See Section 7.1.1.3 for more details.)

Sub-Surface Hydrology

Unsaturated Zone. The unsaturated zone beneath Yucca Mountain consists of stratified units of welded and nonwelded tuffs of contrasting hydrologic properties. The degree of fracturing may influence the flow of ground water, as may the faults bounding and intersecting the Yucca Mountain block. The small amount of precipitation that does infiltrate primarily enters the Tiva Canyon welded unit, but also may enter the Paintbrush nonwelded and the Topopah Spring welded units where they are exposed at the surface. Percolation through the matrix occurs primarily vertically in the welded units and both laterally and vertically in the nonwelded units. The overall ground-water flow from Yucca Mountain is generally to the south. (More detail can be found in Section 7.3.6.)

Saturated Zone. Three main hydrogeologic units are found in the saturated zone: unconsolidated alluvial deposits (valley- or basin-fill deposits), Tertiary volcanic rocks underlying the alluvium, and Paleozoic carbonates generally underlying the volcanic layer. Relatively little is known about the interface between volcanic and carbonate rocks, as only one well has been drilled completely through the volcanic stratum. However, the isotopic composition and the temperature of ground water suggest that some ground water may be moving from the Paleozoic aquifer into the volcanic aquifer. The ground-water table lies from 600 to 800 meters below the surface and is relatively flat beneath Yucca Mountain itself, though it rises to the north and northwest. (More information is found in Section 7.3.7 and in DOE94a.)

Alluvial Aquifer. Basins and valleys in the area are filled with unconsolidated sand and gravel deposits ranging in thickness from 600 to 1,500 meters, though some areas may be up to 3,000 meters thick. The major basin-fill aquifers in the area are Crater Flats to the west, Jackass Flats to the east, Amargosa Valley to the south, and the Death Valley aquifer even further to the south. The water table appears in the alluvium in the deeper central parts of the basins or flats but is found in the underlying rocks at the edges of the flats, where the alluvium thins out. (See Section 7.4.1.1 for more detail.)

Volcanic Aquifer. This section of rock is informally considered to be composed of two aquifers (upper and lower) and two aquitards (again, upper and lower), generally distinguished by the degree of welding in the tuffs and the resulting capability for fracturing. The thickness of the saturated volcanic zone varies, because both the water table and the bottom of the volcanic sequence are neither level nor at a constant elevation above sea level. Assuming that the bottom of the volcanic sequence is constant (based on data from the one well to completely penetrate this layer), the thickness of the saturated volcanic zone ranges from about 884 to 1,189 meters. (Sections 7.3.7 and 7.4.1.2 provide more detail.)

Paleozoic Carbonate Aquifer. The thick sequences of carbonate rock that appear to underlie the entire area, and the aquifer systems they contain, are not well understood. Runoff is low, and some areas may be in direct contact with the alluvial reservoir system rather than the volcanic aquifer. The aquifer is at least 460 meters thick, with the top of the aquifer about 820 meters below the land surface. Essentially all of the flow in this aquifer is through fractures. (More information can be found in Section 7.4.1.3.)

8.1.2.2 Hydrologic Flow Characteristics

Data indicate that the overall ground-water flow direction in the alluvial aquifer is to the south and southwest, with local variations. Currently, there are no known data on velocities but, due to the bounding bedrock surrounding the basins, it is reasonable to assume that water would flow very slowly due to low hydraulic gradients. These aquifers are recharged directly by precipitation; by runoff from the surrounding mountains; by infiltration from the underlying bedrock formations; and possibly by returns of irrigation water and percolation of wastewaters. Water leaves the alluvial aquifers by flowing to other basins, percolation to the volcanic or carbonate aquifers, evapotranspiration, and pumping for domestic and irrigation uses. (See Section 7.4.2 for more information.)

Ground water flow in the volcanic aquifer is generally to the south, with a strong tendency to the east in some areas. In one area about three kilometers upgradient from the proposed repository site, water levels drop over 275 meters in slightly less than two kilometers. The precise cause of this large gradient is not known, although it could possibly be due to perched water being mistaken for water from a deeper zone. Outside of this large-gradient zone, hydraulic gradients measured in the volcanic units are quite low, around 0.0003.

The volcanic aquifer is recharged primarily by snowfall on Timber and Yucca Mountains, with occasional intense rainstorms adding to the infiltration, which is estimated to be about 2 to 4 millimeters per year. There is also some unquantified recharge from the underlying carbonate aquifer. The location and amount of the volcanic aquifer discharges are not currently known, but some water very likely moves to the south to enter the alluvium as the volcanic layer pinches out. A few wells account for some discharge, supplying water for the Nevada Test site and Yucca Mountain characterization activities. (Section 7.4.2.2 contains more detail.)

Flow direction and gradients in the Paleozoic carbonate aquifer are not well defined because very few wells have penetrated this layer. However, regional flows are generally thought to be southward. The velocities in an area of similar rock outside the study area have been estimated at .006 to 60 meters per day. The carbonate aquifer can be recharged directly where highly fractured rocks are exposed at the surface at higher elevations, where precipitation is greatest. Recharge also occurs by infiltration from the overlying volcanic and alluvial deposits. The carbonate aquifer is known to discharge at Ash Meadows, southeast of Yucca Mountain, and probably in Death Valley, about 100 kilometers south-southwest of Yucca Mountain. Other discharge points may include small, low-flowing springs, though most of these are not in the study area. (More information is available in Section 7.4.2.3.)

8.1.2.3 Hydrologic Resources and Water Utilization

The chemical quality of ground water in the area varies considerably. Generally, ground water in wells closest to the discharge area of the system (mainly Death Valley and the Amargosa Desert) contains higher concentrations of dissolved minerals and is less suitable for most uses, though it is generally useable for irrigation. The alluvial aquifers tend to have high concentrations of fluoride; water from the volcanic aquifer is dominated by bicarbonates of sodium and also contains small amounts of silica, calcium, magnesium, and sulfate. The wells that supply water for the Nevada Test Site and for characterization activities at Yucca Mountain draw from the volcanic aquifer and have shown no deterioration in water quality over decades of pumping. Water from the carbonate aquifer shows elevated levels of calcium and magnesium carbonates, and also increased sodium and potassium if it has percolated through the volcanic formation. (See Sections 7.3.7.6, 7.4.3.1, and 7.4.3.2 for more detail.)

In terms of water rights, Nevada is an appropriative state and limits the amount of water that may be withdrawn from any hydrographic basin to the perennial yield for that basin. Ground-water "mining" is not allowed. Water for almost all human activities (consumption, irrigation, ranching) is drawn from the alluvial and the lower carbonate aquifers; only at the Nevada Test Site and for characterization at Yucca Mountain is water taken from the volcanic aquifer.

The major users of ground water in the area are the town of Amargosa Valley and small rural communities in the northeast Amargosa Desert. In Amargosa Valley, water is supplied by wells into the alluvial aquifer. Primary uses are domestic, agricultural, mining (specialty clays), recreation (such as golf courses), and industrial. Most residences are supplied by individual wells, though some trailer parks, public facilities, and commercial establishments are served by small, private water companies. A number of springs also supply water, primarily to the resort area in Death Valley.

The hydrographic basin in which Amargosa Valley is located (Basin 230) currently is rated at a perennial yield of 24,000 acre-feet per year. The 1993 population of Basin 230 was about 1,100. While currently allocated usage rights for Basin 230 stand at a little more than 41,000 acre-feet, the actual usage has not yet exceeded the estimated perennial yield - the amount pumped in 1993, for example, was about 11,300 acre-feet per year. In 1993, water actually pumped and used for irrigation was 8,709 acre-feet, or about 30 percent of the amount allocated for that purpose; water for mining operations stood at 44 percent of the allocated amount; and water for community and municipal uses was a little more than 4 percent of allocation. (See Section 7.4.3.2 for more detail.)

8.1.3 Demography

Land Use

The environment in and around Yucca Mountain (YM) is characterized by desert valley and Great Basin mountain terrain and topography. Its climate, flora, and fauna are typical of the southern Great Basin deserts. Access is restricted to the Yucca Mountain area due to its remoteness and its location near the Nevada Test Site (NTS) and adjacent U.S. Air Force lands. The predominant land use surrounding Yucca Mountain is open range for livestock grazing, with scattered mining, farming, and recreational areas.

Figure 8-1 delineates the variety of land uses within 180 miles (300 km) of NTS/YM. The area southeast of Yucca (shown on the southwest border of the NTS) is relatively uniform, since the Mojave Desert ecosystem comprises most of this part of Nevada and California. The area directly south is the Amargosa Valley, which has limited but locally intensive farming and ranching activity. In the relatively barren area north of Yucca Mountain, the major agricultural activity is the grazing of cattle and sheep.

Population

As shown in Figure 8-2, eight counties in Nevada and one county in California border Nye County, Nevada. The county population levels shown in this figure are from the 1990 census and are shown for consistency. However, estimates for Nye County and its communities were updated in 1994 and will be used in the remainder of this section. Excluding Clark County, the major population center in Nevada (about 750,000 persons), the population density of counties adjacent to Yucca Mountain is about 0.7 people per square mile (0.4 per square km).

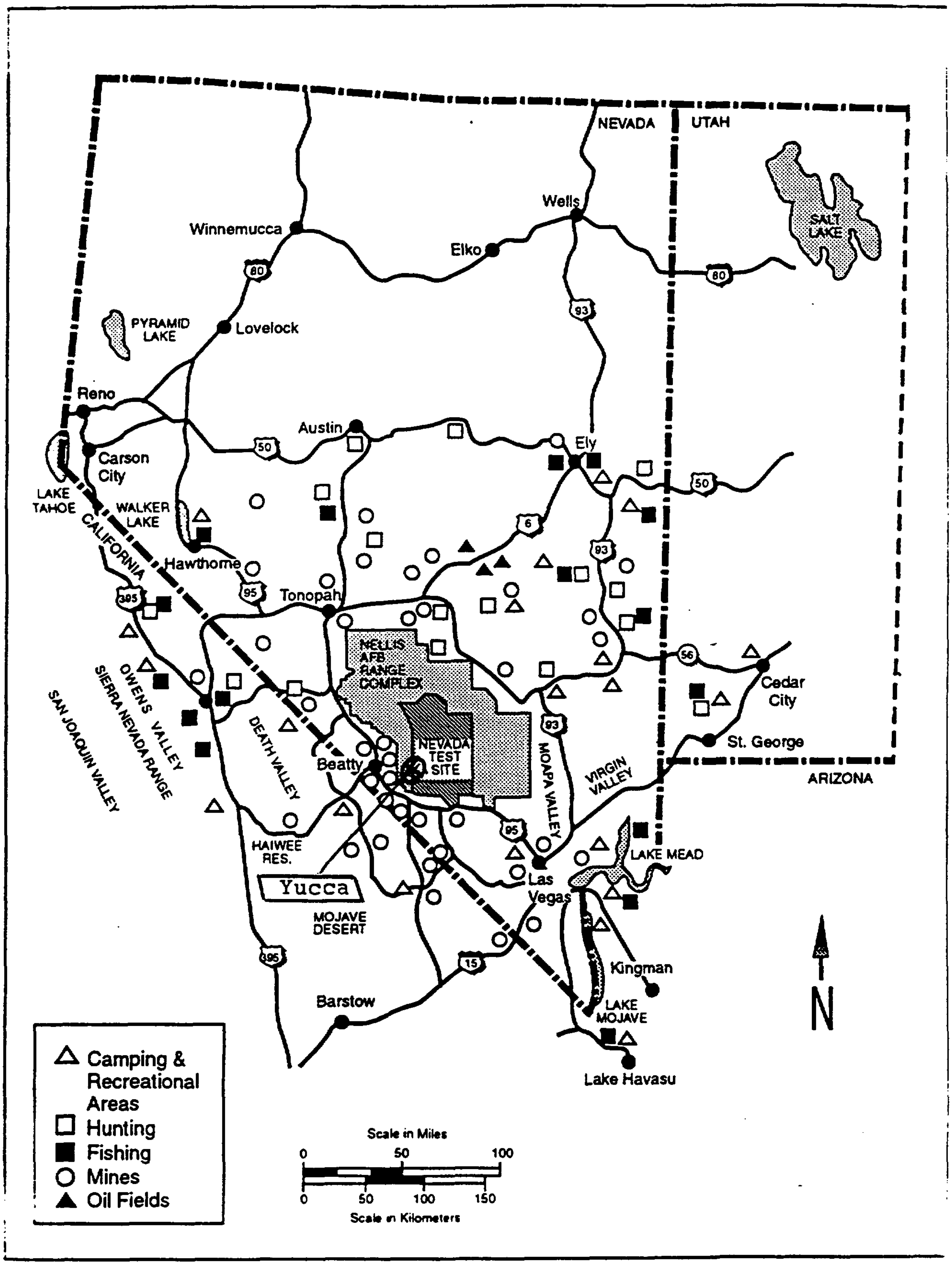


Figure 8-1. General Land Use Within 180 Miles (300 km) of the Nevada Test Site (EPA93a)

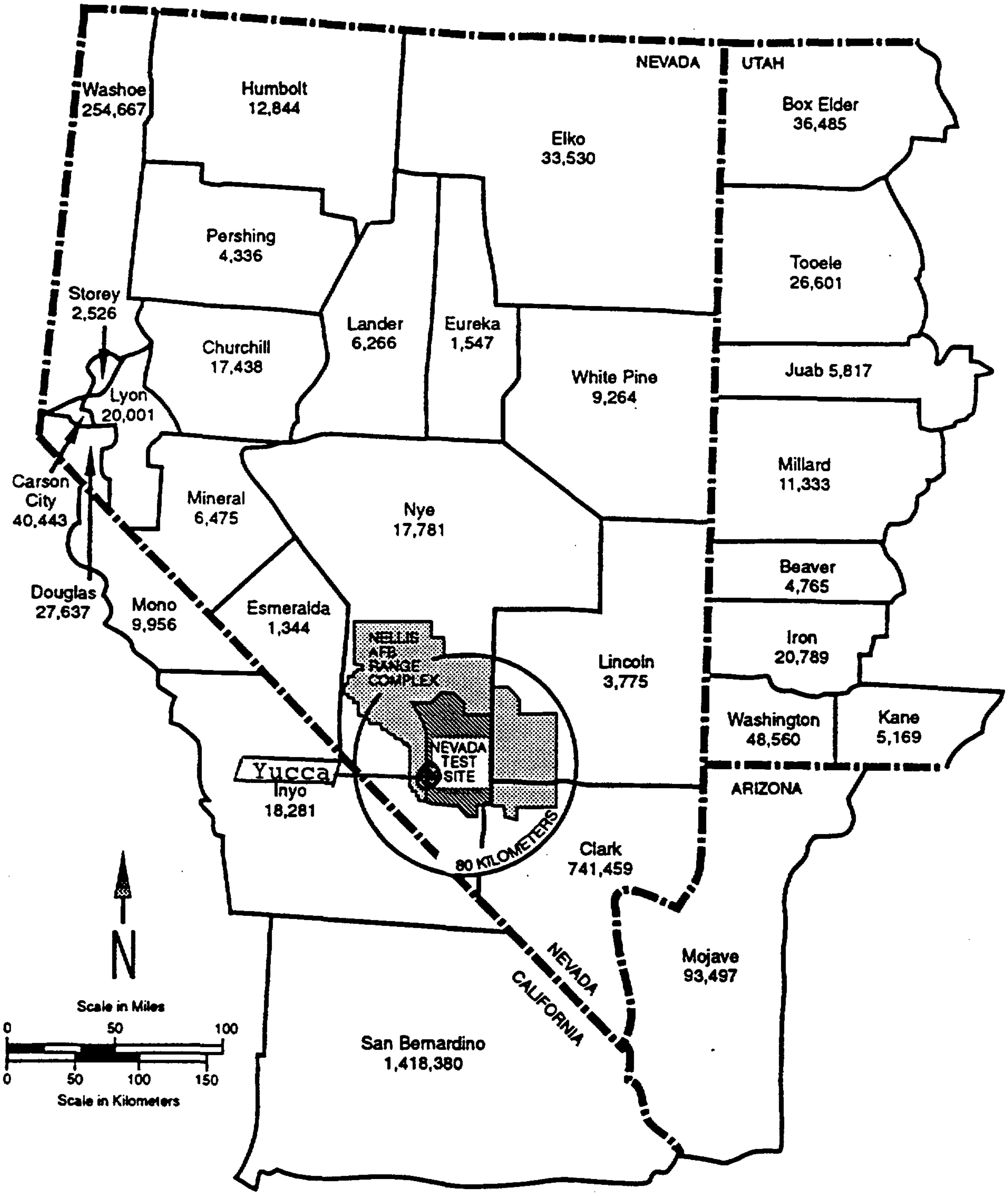


Figure 8-2. Population of Arizona, California, Nevada, and Utah Counties Near the Nevada Test Site (EPA93a)

For comparison, the population density of the 48 contiguous states is 70.3 persons per square mile (27 per square km). The average population density of Nevada is 10.9 persons per square mile, or 3.1 per square km. The only region in Nye County with a density greater than three people per mile is in the extreme southern portion, in and around the community of Pahrump, which is 60 miles west of Las Vegas.

The primary area of interest (based on DOE95a) is within an 80-km radius of Yucca Mountain, shown approximately in Figure 8-2. This region also includes several small communities that are not shown here, but are generally located southeast to west of the site. The largest of these communities, Pahrump, is a growing rural community with a 1994 estimated population of 10,892, located about 80 km southeast of Yucca Mountain. Other communities in the immediate area are Beatty (25 km west) and Amargosa Valley (20 km south) in Nye County, Nevada; Indian Springs (70 km east) in Clark County, Nevada; as well as Death Valley Junction in Inyo County, California (55 km south). Also contained in this area are portions of Death Valley National Park (DVNP). The socioeconomic characteristics of the Nevada communities are summarized in Table 8-1 and in NYE94.

The Mojave Desert of California, which includes DVNP, lies along the southwest border of Nevada. Population within the park ranges from a minimum of 200 residents in the summer to 5,000 tourists per day in winter (excluding major holidays, when as many as 30,000 could be present). The largest populated area in the region is Ridgecrest, California (160 km southwest), with a population of 28,000. The Owens Valley, beginning 50 km west of Death Valley, contains many small towns, the largest of which is Bishop, California, with a 1990 population of 3,475. As shown in Figure 8-2, based on the population levels, the area of southwestern Utah, due east from Yucca Mountain, is more developed than the adjacent parts of Nevada. St. George (200 km east) is the largest community, with a 1990 population of 28,500. The extreme northwestern part of Arizona (Mojave County) is mostly range land except for the portion containing Lake Meade and other small communities along the Colorado River.

Employment

The NTS, which is adjacent to Yucca Mountain, accounts for a high concentration of employment in southern Nye County. Many of the workers live in group quarters during the week. In December 1994, NTS employment was reported to be 3,000, down significantly

Table 8-1. Summary of Socioeconomic Characteristics Compiled by Community for the First Quarter of 1994 (DOE95a)

Socioeconomic Characteristics	Pahrump	Beatty	Amargosa Valley	Indian Springs
Square Miles	298	692.5	499	18
Acreage (1)	190,720	443,200	319,360	11,520
Total Occupied Housing Units:	4,879	788	352	492
Single Family (2)	4,692	719	344	492
Multi Family	187	69	8	
Group Quarters (3)	2	4		
Total Estimated Population:	10,892	1,947	909	1,200
Single Family (2)	10,463	1,747	888	1,200
Multi Family	417	168	21	
Group Quarters (3)	12	32		
Establishments by Standard Industrial Classification Group:	660	134	60	37
Ag/For/Fishing (4)	20	6	7	
Mining/Construction	98	15	6	
Manufacturing	21	2	1	
TCEGSS (5)	42	8	7	2
Wholesale & Retail Trade	178	29	15	5
FIRE (6)	67	17		7
Services	209	47	20	13
Government	25	10	4	10

* Tax boundaries specified by the Nye County Board of Commissioners are used to delineate the boundries for Pahrump, Beatty, and Amargosa Valley. For Indian Springs, the legal description specified by the Clark County Commissioners for the unincorporated town is used.

Please note: Community boundries encompass many whole, as well as some partial, cells. Therefore, information within this table is not directly comparable to the information presented in the Appendix. For Pahrump, the information included in this table is for the entire community both inside and outside of the RadMP grid.

(1) Acreages for the communities in Nye County were supplied by the Nye County Assessor's Office, and are the best estimate of the actual acreages encompassed within the taxation boundries (Nye County Assessor's Office, 1988)

(2) This category was refined to include all single-family dwellings and mobile homes, due to the new method of data collection. Units housing persons visiting or residing in the area on a "short-term" temporary basis, such as in RV parks, are not included.

(3) This category includes the group quarters in Pahrump and the employee housing in Beatty, reported as the number of facilities in the housing section (not included in total) and number of residents in the population section (included in total and not used to calculate the PPH).

(4) Agriculture/Forestry/Fishing

(5) TCEGSS refers to Transportation, Communications, Electric, Gas, and Sanitary Services.

(6) FIRE refers to Finance, Insurance, Real Estate

from more than 5,000 workers in the mid-1980s. At the same time, the employment at Yucca Mountain increased twelve-fold, from 38 workers in 1988 to 478 by December 1994. Table 8-1 shows establishments in the Nye County communities by Standard Industrial Classification (SIC) Groups and demonstrates potential employment concentrations in the region.

Agriculture

Within the 80 km radius around Yucca Mountain, agricultural activity appears to be holding steady, with increases in alfalfa being offset by decreases in acreage planted in barley and oats. The Amargosa Valley primarily grows sod/turf, alfalfa, barley, and oats, and relatively small amounts of fruits. The area west of Pahrump grows primarily alfalfa. The majority of livestock in the region consists of bee colonies in Pahrump (honey production), catfish farming in Amargosa Valley, dairy cows in Pahrump and Amargosa that produce milk shipped to southern California, pigs raised for commercial consumption locally, and range cattle. Recent openings of new dairies in Amargosa represent additional local demand for these products.

Mining and Construction

Within the 80-km radius, the areas west and south of Yucca Mountain near the communities of Beatty, Amargosa Valley, and Pahrump contain 12 mining and open pit operations and 71 construction and drilling operations. The activities associated with these businesses include mining, sand and gravel operations, construction, drilling, and landfills. Other active mines and oil and gas wells are widely dispersed throughout the state.

8.1.4 Atmospheric Conditions

South central Nevada is arid to semiarid; dry air masses dominate the upper air flow 70 percent of the year. Moist maritime air (both tropical and polar) occurs only about 28 percent of the year. Outbreaks of cold and dry arctic air occur during the winter and account for 5 percent of yearly airmass dominance. The area around Yucca Mountain receives moisture as a consequence of two storm types, caused by (1) winter cyclonic activity and (2) intense summer convection.

Late autumn through early spring is the cold season, when southward migration of the subtropical high-pressure zone brings depressions to the region. Winter precipitation results from Pacific-type storms or Great Basin lows. Pacific-type storms are caused by fronts from eastward-moving lows imbedded in the prevailing westerlies. These are generally north of Yucca Mountain; however, they are sometimes diverted far enough south to affect the area. Great Basin lows are non-frontal cyclonic disturbances. They are the chief source of winter precipitation in central and eastern Nevada. This low develops near the extremity of a front moving inland from the Pacific and is caused by thermal contrast between cold air behind the front and warmer air ahead. Great Basin lows may remain stationary over Nevada for a few days but eventually move eastward. Moisture-bearing winds are drawn into the area from the northwest. Great Basin lows are most common from April to June, while Pacific-type storms are more frequent between October and April.

When the subtropical high-pressure belt shifts northward during the summer, the lows no longer affect the region. Summer precipitation occurs as localized thunderstorms caused by vertical air currents over heated terrain. In the Yucca Mountain region, about 25 percent of the annual precipitation falls during the summer.

These conditions, combined with the area's wide range in altitude and rugged terrain, makes the climate highly variable. This climate, characterized as mid-latitude steppe, provides insufficient water to support the growth of food crops without irrigation. The average annual precipitation ranges from about 4 inches at the lower elevations to around 10 inches at the higher levels. During the winter months, the plateaus may be snow-covered for a period of several days or weeks, but snow is uncommon on the flats.

As with precipitation, temperatures vary considerably with elevation, slope, and local air currents. The average daily temperature ranges at the lower altitudes are around 25 to 50°F in January and 55 to 95°F in July, with extremes of -15°F and 120°F. Corresponding temperatures at the higher elevations are 25 to 35° (January) and 65 to 80° in July, with extremes of -30° and 115°. The wind direction, as measured several kilometers northeast of Yucca Mountain in the NTS, is predominantly northerly in the fall and winter. During May through August, winds from the south-southwest predominate. Table 8-2 summarizes the characteristics of climate types for Nevada.

Table 8-2. Characteristics of Climatic Types in Nevada (EPA93a)

	Temperature °F (°C)		Annual Precipitation inches (cm)			
Climate Type	Winter	Summer	Total*	Snowfall	Dominant Vegetation	Percent of Area
Alpine tundra	0 to 15 (-18 to -9)	40 to 50 (4 to 10)	15 to 45 (38 to 114)	Medium to heavy	Alpine meadows	--
Humid continental	10 to 30 (-12 to -1)	50 to 70 (10 to 21)	25 to 45 (64 to 114)	Heavy	Pine-fir forest	1
Subhumid continental	10 to 30 (-12 to -1)	50 to 70 (10 to 21)	12 to 25 (30 to 64)	Moderate	Pine or scrub woodland	15
Mid-latitude steppe	20 to 40 (-7 to 4)	65 to 80 (18 to 27)	16 to 15 (15 to 38)	Light to moderate	Sagebrush, grass, scrub	57
Mid-latitude desert	20 to 40 (-7 to 4)	65 to 80 (18 to 27)	3 to 8 (8 to 20)	Light	Greasewood, shadscale	20
Low-latitude desert	40 to 50 (-4 to 10)	80 to 90 (27 to 32)	2 to 10 (5 to 25)	Negligible	Creosote bush	7

* Limits of annual precipitation overlap because of variations in temperature which affect the water balance.

8.1.5 Ecosystems

As described in previous sections, the diverse topography, geology, and climates of the southern Nevada desert create a complex variety of plantlife. Vegetation ranges from sparse desert scrub in the lowest valleys to well-developed woodland on highlands above 2,000 m. Only sheer cliffs and playa floors are devoid of plants. Even the apparently barren hills of the Amargosa Desert support widely spaced shrubs and succulents.

Table 8-3 shows plant types and associations found in the regions in and around Yucca Mountain. As described in the table, plant associations classified as Great Basin conifer woodlands are distinguished by dominance of single-needle pinyon pine and Utah juniper. In south-central Nevada, these pygmy conifer communities are restricted to elevations above 1,800 meters. Dominant plant taxa in Great Basin desert scrub communities are flowering plants such as shadscale. These desert scrub associations usually occur below the tree line, but

Table 8-3. Principal Plant-Community Types and Examples of Representative Plant Associations on Rock Slopes (SPA85)

Representative Plant Association	Distribution and Common Associates
Great Basin Conifer Woodland	
Pinus monophylla-Quercus gambelii-Juniperus osteosperma	Volcanic highlands in the northern test site; generally at elevations above 1,950 meters (such as the eastern Pahute Mesa, Timber Mountain). Associates include Artemisia tridentata Symphoricarpos longiflorus, Purshia tridentata, and Lupinus argenteus.
Pinus monophylla-Artemisia tridenta-Juniperus osteosperma	Highlands at elevations above 1,770 meters; restricted to xeric habitats at elevations above 2,100 meters. Common associates include Artemisia nova. Cowania mexicana. Haplopappus nanus. Brickellia microphylla.
Great Basin Desertscrub	
Atriplex canescens-mixed scrub	The flanks of hills and rocky mesas, usually of volcanic substrate; at elevations from about 1,500 to 2,000
Atriplex confertifolia-mixed scrub	On limestone and dolomite slopes; at elevations from about 850 to 1,700 meters. Common associates are usually Mojave Desert shrubs such as Amphipappus fremintii; Ephedra torreyana, Larrea divaricata. Gutierrezia microcephala.
Mojave Desertscrub	
Lepidium fremontii-mixed scrub	On the talus slopes and ridges of calcareous mountains: at elevations from about 1,050 to 1,700 meters. Associates include a diverse complement to upper elevation Mojave desertscrub species such as Coleogyne ramosissma, Ephedra torreyana, Buddleja utahensis, and Lycium andersonii.
Gutierrezia microcephala-mixed scrub	Talus slopes, cliff bases, and ridges; generally at elevations below 1,400 meters on calcareous substrates. Common associates include Larrea divaricata. Ambrosia dumosa, Ephedra spp., Amphipappus fremontii, Lycium pallidum,
Ambrosia dumosa-Larrea divaricata	On talus slopes, ridges, and mesas; generally at elevations below 1,200 meters. Normally occurring with lower-elevation Mojave Desert species such as Peucephyllum schottii. Eucnide urens, Gutierrezia microcephala, Echinocactus polycephalus. Atriplex confertifolia is common at some sites.

above Mojave Desert vegetation. The plant species typical of lower elevation Mojave desert scrub vegetation, like creosote bush and white bursage, have their center of distribution south of Yucca Mountain. Exceptions include plants dominated by species endemic to the northern Mojave, such as box thorn and greasewood. The vegetation classifications at Yucca Mountain are dominated by representatives of Mojave desert scrub, but many Mojave desert scrub species are at the northern limits of their distribution in southwest Nevada, and most are restricted to elevations below 1,800 meters.

According to the Nye County Overall Economic Development Plan (NYE93a), the county, like most of the state, hosts a number of threatened and endangered species, as shown in Table 8-4. According to the Nye/Esmeralda Economic Development Authority, however, only two of these have affected growth and development.

The Devil's Hole Pupfish has limited development in the Ash Meadows area of Amargosa Valley. Protection of the Pupfish and several other threatened and endangered species resulted in the creation of the Ash Meadows National Wildlife Refuge. Thus, casual use of this region is restricted to existing roads, trails, and washes. The ground water level is protected in spring flows in Ash Meadows, which is managed by the Fish and Wildlife Habitat Management Program of the BLM.

Also, like much of southern Nevada, certain areas in Pahrump and Amargosa Valley have been classified as desert tortoise habitat. Land within this classification requires a biological assessment before it can be developed. If necessary, the U.S. Fish and Wildlife Service may require a second environmental assessment and a site-specific habitat conservation plan.

8.1.6 Sources of Human Radiation Exposure

All members of the public are exposed to ionizing radiation from a variety of sources. Exposure to some sources is not only inevitable but life-long, while exposure to others may be episodic and influenced by numerous factors. For convenience, sources of public exposures are commonly categorized as (1) of natural origin and unperturbed by human activities, (2) of natural origin but affected by human activities (termed enhanced natural sources), and (3) man-made sources.

Table 8-4. Threatened and Endangered Species in Southern Nevada (NYE93b)

Common Name	**Scientific Name**
Endangered	
American Bald Eagle	Haliaeetus leucocephalus
Ash Meadow speckled dace	Thinichthys osculus nevadensis
Ash Meadow speckled pupfish	Cyprinodon nevadensis mionectes
Cui-ui	Chasmistes cujus
Devil's Hole pupfish	Cyprinodon diabolis
Hiko White River springfish	Chrenichthys baileyi grandis
Moapa dace	Moapa coraicea
Pahrump poolfish	Empetrichthys latos
Peregrine falcon	Falco peregrinus
Warm Springs pupfish	Cyprinodon nevadensis pectoralis
White River spinedace	Lepidomeda albivallis
White River springfish	Chrenichthys baileyi baileyi
Threatened	
Ash Meadows naucorid	Ambrysus amargosus
Big Spring spinedace	Lepidomeda mollisinis pratensis
Desert dace	Eremichthys acros
Desert tortoise	Xerobates agassizzi
Lahontan cutthroat trout	Oncorhynchus clarki henshawi
Railroad Valley springfish	Crenichthys nevadae

Natural radiation and naturally occurring radioactivity in the environment are by far the major sources of human radiation exposure. Consequently, these sources have been extensively studied and are commonly compared with various man-made sources of ionizing radiation.

Natural sources include cosmic radiation from outer space; terrestrial radiation from radionuclides in soil, rocks, and other materials; and radionuclides within our bodies. Each of these natural sources has specific characteristics that cause variations in individual exposures, which are influenced by geographic location, dietary habits, lifestyles, and other factors.

Enhanced natural sources include those for which human exposures have increased due to deliberate or inadvertent behavior. For example, extensive air travel may significantly increase exposure to cosmic radiation, and tailings from phosphate mining, when used for construction fill, can increase terrestrial exposure. Another example involves the combustion of fossil fuels (coal, oil, natural gas) by industry and electric utilities, which results in the localized release of naturally occurring radionuclides in stack gases released into the air. Even indoor exposure to radon might be considered "enhanced," because air concentrations of radon and radon daughters are significantly affected by the design, construction, and use of a home or building.

In addition to natural sources, most individuals are also exposed to radiation from numerous man-made sources, materials, and devices. The largest category among man-made sources is classified as medical and refers to a variety of diagnostic and therapeutic procedures (e.g., x-rays, fluoroscopic examinations, CAT-scans, radioactive pharmaceuticals and implants, exposure to teletherapy units). The public is also exposed to a variety of consumer products, such as televisions, smoke detectors, nuclear weapons production and testing, and nuclear power and the associated fuel cycle.

The scientific literature abounds with information on public exposure to natural and man-made sources. Among the most comprehensive reports are those issued by the EPA (EPA72a, EPA77) and several prominent scientific committees, including the United Nations Scientific Committee on the Effects of Atomic Radiation (UNS88, UNS93, UNS94), the Committee on the Biological Effects of Ionizing Radiation of the National Academy of Sciences (NAS80, NAS88, NAS90), and the National Council on Radiation Protection and Measurements (NCR87a, NCR87b, NCR89).

This section summarizes average exposures received by the general public in the United States, as well as estimates of exposures to individuals currently residing in the vicinity of Yucca Mountain.

8.1.6.1 Natural Radiation Sources and Exposures

Cosmic Radiation

Cosmic radiation refers to primary energetic particles originating from the sun and from outside the solar system, and to secondary radiation generated by their interaction with the earth's atmosphere. In the absence of the earth's atmosphere, the biosphere's dose of cosmic radiation would be about 1,000 times greater than current levels. The intensity of cosmic radiation is also affected by the earth's geomagnetic field, which varies with latitude, and by the sun's activity, which follows a cycle of about 11 years. However, at ground level, variations in cosmic radiation intensity within the continental United States, due to the geomagnetic field effect, are less than 2 percent (CAR69), while the 11-year variations due to solar activity are less than 10 percent of the mean level (ERD77).

At sea level, the average cosmic-ray annual dose equivalent is about 26 mrem. This annual dose rate essentially doubles with each 2,000 meter increase in altitude in the lower atmosphere. Accordingly, inhabitants of Denver at 1,600 meters and Leadville, Colorado, at 3,200 meters have estimated annual external exposures from cosmic radiation of 50 mrem and 125 mrem, respectively (NCR87b). Considering the distribution of altitudes for the U.S. population, an average annual cosmic-ray dose of 27 mrem is generally assumed, although the dose to a specific individual is affected by the amount of time spent outdoors and the shielding provided by indoor environments.

The cosmic-ray exposure rates in aircraft are considerably higher. At normal commercial jet aircraft altitudes of about 11-12 km, average dose rates of 0.5 mrem/hour have been estimated (NAS86). A single transcontinental flight from New York to Los Angeles would be expected to result in an average dose of 2 to 3 mrem, so crew members working ordinary schedules on high-altitude, long-distance routes would likely receive average doses in excess of 500 mrem per year (BAR95). Solar flares, although infrequent, can yield dose rates in excess of 1,000 mrem/hour at these altitudes (UNS82).

Naturally Occurring Radionuclides

Several dozen naturally occurring radionuclides exist with half-lives of the same order of magnitude as the estimated age of the earth (about 4.5 billion years). These include potassium-

40, rubidium-87, and radionuclides belonging to the decay chain series of uranium-238, uranium-235, and thorium-232 (such as radium). These naturally occurring radionuclides contribute to exposure that is external to the body, internal to the body, or (when inhaled) to tissues of the lungs and respiratory tract.

External Terrestrial Radiation

Potassium-40 and several decay-chain members in each of the uranium and thorium series emit penetrating gamma radiation. These radionuclides exist in low, but varying, concentrations in virtually all types of rocks and soil. Since many building products, such as cut stone, brick, cement, and gypsum are derived from natural stone, they also contribute to external radiation. For most individuals, external exposure indoors to radionuclides derived from the terrestrial environment is nearly equivalent to terrestrial exposure outdoors (Table 8-5).

Table 8-5. Comparison of External Indoor to Outdoor Radiation Dose

Building Materials (exterior walls)	Percent of Outdoor Terrestrial Radiation
1. Mostly Woodframe (single homes)	70 - 82
2. Brick (apartment building)	96
3. Stone (apartments & houses)	80 - 100
4. Steel and Concrete (office building)	87 - 106

Source: EPA72b

In EPA72b, two distinct major regions of the United States were observed that differed in average terrestrial dose rates by a factor of about two; the lower dose rate of 16 mrem/year corresponded to the Atlantic and Gulf Coastal Plain and the remaining major portion of the country (referred to as Non-Coastal Plain) yielded average dose rates of about 30 mrem/year (Figure 8-3). The Denver area showed average terrestrial dose rates of 63 mrem/year. (It is assumed that other areas on the eastern slopes of the Rocky Mountains would show similar levels.) It should be further noted that for each average value cited above, the range in values between the 10th and 90th percentile differed by at least a factor of two.

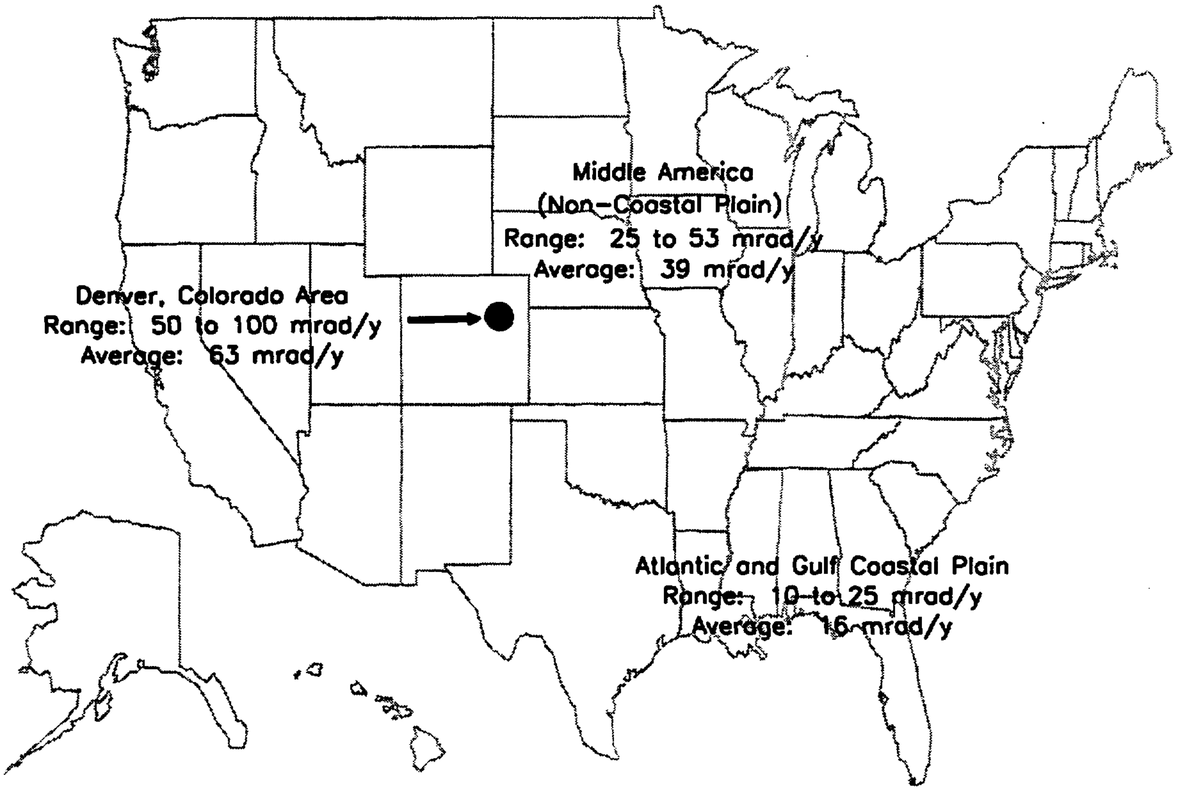


Figure 8-3. Terrestrial Dose Rates in the United States

Radionuclides in the Body

Naturally occurring radionuclides in rocks and soil also give rise to internal radiation exposure because they are present in drinking water and foods that are consumed by humans. Upon ingestion, their distribution in the body is complex and is governed by their chemical properties and the physiological regulatory mechanisms of the body. For example, radioactive potassium-40 exists in a fixed ratio to its non-radioactive form of potassium. Potassium is an essential dietary element to most living systems, and it is distributed nearly uniformly in lean soft tissue.

The concentrations of other radionuclides associated with the decay chains of uranium and thorium are potentially highly variable. For humans, internal exposure to these naturally occurring radionuclides is not only determined by dietary composition but also by the total quantities of foods consumed and the geographic location/origin of the food products. Maximum body burdens would, therefore, be expected for individuals who consume foods (and water) 1) that are derived from areas of high terrestrial background radioactivity, 2) that are higher in radionuclide content than foods of normal diets, and 3) in large quantities.

Internal Doses. Reports of unusual exposure in the United States are contained in the literature. High exposures have been documented for Eskimo and select Indian tribes whose diet includes reindeer caribou, moose, elk, and deer that subsist largely on lichen during winter months. Here the lichen-animal-human food chain leads to high concentrations of natural Pb-210 and Po-210 (as well as Cs-137 from fallout) in the diet (HOL66, ECK86, MAT75).

In general, however, there are insufficient data at this time to define the extent of variability of internal exposures among individuals. Factors cited above that would potentially yield significant differences in exposure are largely eliminated by a food distribution system that is nationwide and offers a wide range of foods.

Dose estimates from internally deposited naturally occurring radionuclides are principally based on limited post-mortem measurements of the nuclide content of various organs. Estimates of average doses to specific tissues of the body are given in Table 8-6. By multiplying annual tissue doses by their corresponding weighting factors, it is estimated that the average individual in the United States receives an internal dose of 39 mrem per year from naturally occurring radionuclides.

Inhaled Radionuclides

Except for radon and its short-lived decay products, the inhalation pathway yields doses that are small relative to the ingestion pathway. For this reason, reference to inhaled radionuclides

Table 8-6. Average Annual Dose from Internal Exposure to Naturally Occurring Radionuclides

Tissue	Tissue Dose (mrem)	Tissue Weighting Factor	Effective Dose Equivalents (mrem)
Lung	33	0.12	4
Gonads	36	0.25	9
Bone Surfaces	100	0.03	3
Bone Marrow	50	0.12	6
Other Tissues	35	0.48	17
Total Body		1.00	39

is limited to gaseous radon and its solid radioactive daughter products. This group of radionuclides belongs to the uranium-238 and thorium-232 decay chain series and is, therefore, present in all soils and rocks, as previously discussed.

As a gas, radon is able to diffuse through soil pore spaces and escape into surrounding media. In outdoor air, radon quickly diffuses and results in relatively small exposures. Significant exposures, however, result when radon gas enters homes and other buildings where it is concentrated and contained for periods of time that allow the buildup of its solid (i.e., non-gaseous) radioactive daughters. When inhaled, particulate radon daughters deposit onto the mucous layer of the airways, some of which emit energetic alpha particles and impart substantial doses to cells that line the upper respiratory tract and lungs.

Estimates of Radon Doses. The magnitude of human exposure to radon and radon daughters is not only influenced by the amount of radon formed in the soil, but also by numerous other factors that affect its 1) migration and movement through soil, 2) rate of entry into a home or building, and 3) containment and daughter buildup in indoor air. For example, factors affecting exposure include soil porosity and moisture content, the design of the structure, and building operating variables related to heating, air conditioning, and ventilation.

A large number of localized survey data exist of indoor radon or decay-product concentrations. However, most of these measurements have been made in single-family houses, and little data exist for high rise apartments, commercial and industrial buildings, and other structures in which various population subgroups spend significant amounts of time.

Further complicating estimates of average exposure and the distribution of individual exposures is the fact that existing data exhibit very large variations. For example, clusters of unusually high indoor radon concentrations have been found in parts of northeastern Pennsylvania, New Jersey, and southeastern New York. This area, known as the Reading Prong, is characterized by soil concentrations of uranium series radionuclides that are about 100 times greater than the national average.

Best estimates suggest an average indoor radon concentration of about 1.25 pCi/L, which yields an average individual dose (i.e., effective dose equivalent) of about 200 mrem per year (NCR87a, NCR87b).

Summary of Natural Background Exposures

Table 8-7 summarizes the average individual exposures in the United States to natural background radiation. On average, cosmic radiation, external terrestrial gamma radiation, and radionuclides within the body contribute nearly equally, and yield a total dose of about 100 mrem per year. However, this is only about one-half of the annual inhalation exposure resulting from indoor radon and its decay products, estimated to be 200 mrem. It is important to note, however, that the dose resulting from exposure to indoor radon has a very large amount of variability, with a significant number of people experiencing doses that are several times that of the average value.

Table 8-7. Estimated Average Annual Dose* To Members of the Public in the United States from Natural Background Radiation

Source	Dose* (mrem/yr)
Cosmic	28
Terrestrial	28
In the Body	39
Inhaled (Radon)	200
Rounded Total	300

* All doses are expressed as effective dose equivalents.

8.1.6.2 Public Exposures to Artificial and Other Sources of Radiation

Members of the public are also exposed to a variety of radiation sources categorized as "artificial," those that involve human technological activities. Among artificial sources, the largest exposures involve medical x-rays and radioactive pharmaceuticals used for diagnostic purposes or the treatment of various diseases. There are also a number of consumer products that contain radioactivity or emit radiation.

It is estimated that on average, artificial sources contribute an annual dose of 60 mrem. An important aspect of human exposure to most of these sources is that the exposure is 1) episodic, 2) voluntary, or 3) has an associated benefit to the individual or society at large.

Medical Sources

Radiation has broad application in the diagnosis and treatment of various diseases that affect humans and is widely employed by physicians, dentists, podiatrists, chiropractors, and other health-care professionals. Diagnostic radiation principally involves x-rays, fluoroscopic examinations, and specialized medical imaging procedures such as computerized tomography and scans involving radiopharmaceuticals. Therapeutic uses of radiation, such as in cancer therapy, involve similar sources of radiation but deliver larger individual doses.

Table 8-8 provides summary data on medical radiation exposures in the United States (NCR89). It is estimated that on average, medical radiation contributes an annual dose of about 54 mrem to individuals living in the United States. While such an average value is an important and useful statistic, it must be pointed out that the distribution of medical exposures among individuals is highly variable.

Consumer Products

Many commercial and consumer products either emit radiation or contain radioactive materials. In 1977, the NCRP issued a comprehensive report, NCRP Report No. 56 (NCR77), that estimated population exposures from consumer products. Because of revised Federal regulations and newly introduced technologies, the NCRP updated its earlier estimates in a revised report issued in 1987 (NCR87c). Scaling the population estimates contained in the 1987 report to the current 260 million, it is estimated that the average annual exposure in the United States to consumer products is in the range of 6 to 13 mrem.

Table 8-8. Estimated Total Number and Frequencies of Diagnostic Procedures in the United States and Associated Radiation Doses

Group	Total No. of Exams	No. of Exams per 1,000 Population	Range of Doses per Exam (mrem)[1]	Avg. Dose Per Exam (mrem)[1]	Avg. Annual Dose to a Member of the Public (mrem/yr)[1]
Diagnostic X-Rays[2]	181,000,000	1,230	6 > 400	50	40
• Physicians/Osteopaths	164,000,000	724	---	---	---
• Chiropractors	9,800,000	43	---	---	---
• Podiatrists	5,900,000	23	---	---	---
• Dentists	101,000,000	440	---	---	---
Radiopharmaceuticals[3]	7,400,000	32	150-1,200	430	14
Total					54

[1] Doses are expressed in effective dose equivalents.
[2] Data reflect procedures performed in 1980.
[3] Data reflect procedures performed in 1982.

This estimate does not include the contribution of tobacco, which contains Po-210. The deposition of Po-210 on the bronchial epithelium of smokers is estimated to result in a localized tissue dose of 16 rem/year, which corresponds to an effective dose equivalent of 1,300 mrem/year. Current data compiled by the American Cancer Society identified 26 percent of persons aged 18 and over as smokers (ACS95). Therefore, the average annual dose to adults from tobacco is estimated to be 340 mrem.

Thus, radiation exposure from the use of tobacco not only represents the greatest contributor to the effective dose equivalent of all consumer products, but also exceeds the average combined contributions from indoor radon and all other natural sources of radiation.

Miscellaneous Sources of Public Exposures

Nominal exposure to members of the public may also result from past and ongoing activities related to nuclear power generation and associated fuel cycle facilities, other NRC-licensed facilities (e.g., radiopharmaceutical manufacturers, hospitals, research facilities), DOE facilities associated with nuclear weapons, and the transportation of radioactive materials. For all but transportation, exposure results primarily from airborne releases.

Public dose estimates for these source categories rely on computer models. The discrete nature of these facilities/sources suggests that public exposures are highly variable and primarily affect near-field residents. Doses as high as 50 mrem/year, for example, have been assigned to near-field residents of the Oak Ridge Reservation (NCR87c). The total estimated population dose of 16,000 person-rem from all such sources yields an average annual dose of about 0.6 mrem to individuals in the near-field. Since the exposed near-field population represents only about 10 percent of the total U.S. population, these miscellaneous sources are thought to contribute about 0.06 mrem/year on average to members of the total population.

Between 1945 and 1962, atmospheric tests conducted by the United States involved nuclear weapons with an explosive yield of about 140 megatons of TNT equivalence, which represents about 27 percent of the world's total, estimated to be 510 megatons (UNS82). The radiation dose commitment from fallout has changed significantly over the years due to natural decay and depletion/removal mechanisms from environmental media, which limit further biological uptake. The current dose rate to members of the general public is estimated to be 1 mrem per year (NCR87b), derived largely from residual radionuclides contained in our bodies.

8.1.6.3 Exposures to Persons in the Vicinity of the Nevada Test Site/Yucca Mountain Site

For populations living near the Nevada Test Site (NTS), exposure to normal background levels may be modified by NTS activities. Past NTS activities have introduced radioactivity into air, soil, and ground water. Atmospherically released radioactivity has contributed to local and global exposure in the past and will continue to do so in the future. Radioactivity introduced on-site into soil and proximally to ground water have the potential for exposure to future persons living in the vicinity of NTS. For purposes of this discussion, past NTS activities relevant to human exposure may be grouped into several discrete categories that are discussed below. In addition, risks and consequences from these activities are summarized.

Atmospheric Weapons Testing. Over 100 events on the Nevada Test Site have resulted in the release and deposition of radionuclides on the soil surface outside the test-site boundary (HIC90). Atmospheric weapons testing at NTS began in 1951 and continued into 1962. Atmospheric testing included weapons that were dropped by airplanes, those detonated from towers at heights ranging from 30 to 213 m, tests conducted on land surfaces, and tests in which helium balloons lofted weapons 137 to 457 m above ground.

It is estimated that for near-surface detonations, about 12 percent of the fission products are distributed locally, with the remaining 10, 76, and 2 percent introduced into the troposphere, stratosphere, and very high altitudes, respectively (UNS82). In addition, the large number of neutrons released at the time of the detonation results in significant quantities of activation products in the bomb's structural components as well as ambient surface materials.

The primary radionuclides deposited locally are americium, plutonium, cobalt, cesium, strontium, and europium. Based on the most recent estimates, about 20 Ci remain in surface soils at or near the original testing area(s) (MCA91).

Safety Tests. Between 1954 and 1963, more than 30 tests were conducted to investigate safety issues regarding nuclear weapons in accident scenarios. The safety tests used mixtures of plutonium and uranium that were detonated using conventional explosives. These tests also assessed the disposal and transport of these isotopes in the environment, including plant and animal uptake. In the 3,500 acres originally contaminated, the inventory of radionuclides is estimated to be between 34 and 39 Ci.

The primary isotopes at test locations are plutonium, uranium, and americium, with lesser amounts of cesium, strontium, and europium. Currently, these long-lived radionuclides are contained in surficial soils and are relatively immobile. They are, however, potentially available to be transported off-site by wind erosion.

Nuclear Rocket and Related Tests. Between 1959 and 1973, the Nuclear Rocket Development Station area was used for a series of open air nuclear reactor, nuclear engine, and nuclear furnace tests, and for the High Energy Neutron Reactions Experiment. The total estimated inventory of soil contaminants that include strontium, cesium, cobalt, and europium has been estimated to be 1 Ci (MCA91).

Waste Disposal Activities. Since the early to mid-1960s, NTS Areas 3 and 5 were established for disposal of low-level waste from on-site and off-site DOE waste generators and include landfill cells (pits and trenches) and greater confinement disposal (GCD) boreholes. Approximately one-half of the buried waste represents atmospheric testing debris generated during cleanup activities of above-ground nuclear test areas, with the remaining half from other defense-related facilities.

Currently, NTS operates Areas 3 and 5 as a LLW repository and receives waste from on-site activities and off-site defense generators. Approximately 500,000 Ci of low-level waste are disposed in shallow pits and trenches and approximately 9.3 million Ci of high-specific-activity waste containing primarily tritium have been disposed in GCDs. In addition, both areas also contain smaller inventories of mixed waste.

Underground Testing. In August of 1963, the United States and the former Soviet Union signed the Limited Test Ban Treaty (LTBT), which effectively banned weapons testing in the atmosphere. Approximately 800 underground nuclear tests have been conducted that include shallow borehole tests (< 60 m) and deep underground tests (about 600 m). Of the total inventory estimated to be 300 million Ci, about 112 million Ci are considered a potential hydrological source term. About 90 percent (or 100 million Ci) of this activity is represented by tritium.

Table 8-9 provides a summary of residual radionuclide source terms at NTS for the aforementioned activities.

Potential Impacts on Surrounding Populations

Population impacts from NTS activities are most effectively discussed in terms of activities that resulted in the introduction of radioactivity into the 1) atmosphere, 2) surficial soils, and 3) at subsurface depths, where radioactivity could in the future become available to the groundwater regime.

While maximum human exposure from atmospheric releases essentially coincided with peak periods of past nuclear detonation, human exposures to radioactivity introduced below ground is primarily a concern of the future.

Past Impacts Associated with Atmospheric Releases

In 1979, the Department of Energy (DOE) launched a major effort, called the Off-site Radiation Exposure Review Project (ORERP), with the principal objective of collecting, organizing, and analyzing all relevant documents and data pertaining to fallout and resultant exposure to off-site population groups in the vicinity of NTS. Since that time, more than

Table 8-9. Summary of Remaining Radioactivity on the NTS

Source	Area Affected	Media	Major Known Isotopes	Depth Range	Remaining Inventory (Ci)
Atmospheric & Tower Tests	Aboveground Nuclear Weapon Proving Area	Surficial Soils & Test Structures	Americium Cesium Cobalt Europium Strontium	Above land surface	~ 20
Safety Tests	Aboveground Experimental Areas	Surficial Soils	Americium Cesium Cobalt Plutonium Strontium	At land surface	~ 35
Nuclear Rocket Development Station	Nuclear Rocket Motor, Reactor, and Furnace Testing Area	Surficial Soils	Cesium Strontium	Less than 10 feet	~1
Shallow Land Disposal	Waste Disposal Landfills	Soils & Alluvium	Dry-packaged low-level & mixed wastes	Less than 200 feet	~ 500,000[a]
Greater Confinement Disposal	Monitored Underground Waste Disposal Borehole	Soils & Alluvium	Tritium Americium	60 meters	~ 9.3 million[a] (~ 10,000 feet^2)
Shallow Borehole Tests	Underground Nuclear Testing Areas	Soils & Alluvium	Americium Cesium Cobalt Europium Plutonium Strontium	Less than 200 feet	~ 2,000 at land surface, unknown at depth
Deep Underground Tests	Underground Nuclear Testing Areas	Soils, Alluvium, & Consolidated Rock	Tritium, fission, and activation products	~ 600 meters	112 million[b]

[a] Inventory at time of disposal (not corrected for decay).

[b] The 112 million Ci represents that fraction of the total underground source term (estimated to be 300 million Ci) which is within 100 m of the water table. It is this fraction that is available to the ground water regime and is, therefore, referred to as the hydrological source term.

200,000 documents have been amassed, and exposure estimates for discrete fallout events have been derived from empirical measurements and computer-projection models (ANS90).

External exposure estimates were originally published in 1986 (ANS86) and updated in 1990 (ANS90). The total collective external exposure from 1951 through 1975 for all communities was estimated to be 86,000 person-R, with the greatest exposures occurring in Saint George, Utah; Ely, Nevada; and Las Vegas, Nevada. Summaries of the distribution of individual cumulative external exposures are provided in Tables 8-10 and 8-11, which identify three discrete time periods. By far, the largest collective and individual exposures occurred between 1951 and 1958. During the period from 1961 to the time of the Limited Test Ban Treaty, no individuals are known to have received cumulative external exposures greater than 0.5 R. The 480 individuals who received exposures between 0.1 and 0.5 R lived in small ranch communities just north and northeast of the NTS. From 1963 to 1975, cumulative external exposures were small, with only six individuals (at the Diablo Maintenance Station) receiving more than 0.1 R.

The contribution of dose resulting from inhalation and ingestion of radionuclides was not considered in earlier exposure estimates. Investigators from the Desert Research Institute (DRI), Colorado State University (CSU), and the Lawrence Livermore National Laboratory (LLNL) are now systematically reconstructing the internal dose to individuals for all locations and test events at NTS. The computer code PATHWAY was developed to predict radionuclide ingestion by residents in the arid regions around NTS following radioactive fallout deposition (WHC90). PATHWAY simulates the transport of approximately 21 fallout radionuclides through agricultural ecosystems to humans and accounts for agricultural conditions of the southwestern United States during the 1950s. Outputs can be generated that are specific to age, sex, and radionuclides. For the inhalation pathway, estimates will be based on empirical air sampling measurements, fallout data, and meteorologic records.

NTS Health Studies

Numerous population groups exposed to fallout from NTS weapon tests have been studied for health effects. Those studied include civilian populations in the Utah - Nevada area and military participants in weapons testing. Most of these studies assessed the incidence of leukemia and thyroid disorders among the exposed populations.

Table 8-10. Exposure Summary by Major Time Period of the Locations with Recorded External Gamma Exposures, the Mean Location Exposure, and the Population Weighted Exposure[1]

	Time period		
	1951 to 1958	1961 to LTBT[2]	LTBT[2] to 1975
Collective exposure (Person-R)	84,400	610	320
Number of locations with recorded exposure	260	74	72
Mean location exposure (R)	1.3	0.048	0.017
Population weighted exposure (R)	0.47	0.031	0.002

[1] Source: ANS90

[2] Limited Test Ban Treaty signed August 5, 1963

Table 8-11. Distribution of Individual Cumulative External Gamma Exposure by Exposure Range During the Three Major Time Periods

Exposure Range (R)	Persons within Exposure Range		
	1951 to 1958	1961 to LTBT	LTBT to 1975
<0.01 to 0.1	61,000	180,000	180,000
0.01 to 0.5	80,000	480	6
0.5 to 1.0	19,000	0	0
1.0 to 5.0	20,000	0	0
5.0 to 10.0	520	0	0
10.0 to 15.0	45	0	0
Total	180,000	180,000	180,000

Source: ANS90

The results of key leukemia and thyroid studies involving NTS population groups are summarized below.

- A 1979 study reported an apparent twofold increase in the rate of leukemia mortality among Utah residents born between 1951 and 1958 in "high exposure counties"

(LYO79). Some scientists view this finding with skepticism because of a possible misinterpretation of the dose distribution and the paradox that the rates of cancer at other anatomical sites were lower in "high exposure" areas than those in "low exposure" areas.

- Machado et al. (MAC87) reported similar findings of an excess of childhood leukemia deaths in three "high exposure" southwestern Utah counties among individuals younger than 15 years of age who were born before the tests ended. These authors suggested the possibility that the transient increase of radiation-induced childhood leukemias followed the peak fallout deposition between 1953 and 1957.

- Johnson (JOH87) identified radiation-induced cancers among Mormon families in southwestern Utah exposed to fallout between 1951 and 1962 and venting of underground nuclear detonations between 1962 and 1979. This study was found to suffer from methodological deficiencies related to the selection of study subjects, the methods of obtaining medical information and cancer diagnosis, and the interpretation of data (ICR91).

- Caldwell et al. (CAL83) reported an excess incidence of leukemia but no overall excess of other cancers among the 3,224 military personnel who participated in the 1952 Smokey nuclear test. Through 1977, 9 cases of leukemia had occurred, compared with 3.5 cases expected. The recorded average external dose was 520 mrem. A similar study of 5,000 other individuals who had participated in 24 detonations found no leukemia excess (ROB83).

- Another population group studied since 1965 for thyroid disorders includes a cohort of about 2,600 public school students who as infants lived in proximity to the Nevada Test Site in Utah and Nevada. The prevalence of thyroid abnormalities in these children has been compared to that in a control group of 2,219 children selected from a county in Arizona that was presumed to have received little or no fallout from the Nevada Test Site. Thyroid doses were primarily the result of ingestion of milk contaminated with radioiodine. Cumulative thyroid doses among study subjects were estimated to range from 30 to 700 rad (MAY66). Incidence of thyroid neoplasms was first reported in 1974 and 1975 (RAL74, RAL75). Although the rate of thyroid neoplasms among the Utah/Nevada subjects of 5.6 per 1,000 was higher than that of Arizona control subjects (3.3 per 1,000), the difference was statistically insignificant. In a follow-up study conducted in 1985-1986, in which 3,122 of the original 4,819 subjects were reevaluated, the rate of thyroid neoplasms in the Utah/Nevada subjects of 24.6 per 1,000 was again slightly but insignificantly higher than the Arizona subjects (20.2 per 1,000) (RAL90). The authors previously concluded that living near the Nevada Test Site in the 1950s had not resulted in a statistically significant increase in thyroid neoplasms among exposed subjects when compared to control subjects of the same age and gender.

It is now generally accepted that a fundamental limitation in all previous NTS studies was that individual radiation exposures were uncertain or lacking because individual residence histories for study subjects were unknown, and/or reliable exposure rates for many locations were not available at the time of the study.

In response to DOE's previously cited Off-site Radiation Exposure Review Project that amassed exposure data on a county-by-county basis for all or part of seven western states, the National Cancer Institute (NCI) sponsored two major studies to determine whether there were any effects of fallout on the public near the NTS (WAC90).

The first NCI-sponsored study was intended to examine whether leukemia in the state of Utah was related to fallout radiation doses. Dose estimates for the Utah leukemia case-control study were recently reported by Simon et al. (SIM95). The primary objective of the dosimetry task was to estimate the total observed dose from all pathways to the active marrow by summing exposure from each event at each location where the individual resided. External exposure from radionuclides deposited on the ground presented by far the most significant dose contribution to the active marrow.

The second NCI-sponsored study was a reevaluation of the earlier thyroid study. This study reassessed exposures to the same cohort of subjects identified in the 1965-1970 study and reexamined subjects for thyroid neoplasia. Results of this study were reported by Kerber et al. (KER93) and more recently by Till et al. (TIL95). Their reassessment of the study cohort demonstrated a statistically significant dose-response relationship between exposure to radioiodines from open-air weapon tests at the NTS and the occurrence of thyroid neoplasms (carcinomas and benign neoplasms). It should be noted, however, that the association was not statistically significant for thyroid carcinomas alone.

In summary, most of the airborne radioactivity released during the detonation to which nearfield residents were exposed has been widely dispersed in the atmosphere, greatly diluted in the terrestrial biosphere, or decayed in the more than 30 years since the last atmospheric test. Therefore, future human exposures from past atmospheric tests can be assumed negligible.

Potential Future Exposures Associated with Current Soil Contaminants

The potential for significant future exposures to area residents of NTS is limited to those soil contaminants that in time may migrate down through the unsaturated zone and encounter ground water that may subsequently be withdrawn for human use and consumption. Currently radionuclide inventories residing in surficial or shallow strata are unlikely to reach an aquifer. DOE considers only radionuclides from deep underground tests that were deposited beneath the water table or within 100 m of the top of the water table as the potential hydrological source term (DOE96).

As previously noted, the hydrological source term available to the ground water regime is estimated to be 112 million Ci, of which about 100 million Ci is represented by tritium. There is considerable uncertainty about the actual quantity of tritium that can enter the ground water regime. Uncertainties involve the extent to which radioactivity is securely trapped in the melt glass matrix formed in the detonation cavity and the nearfield impact of the detonation on ground permeability.

The shock wave and compressive forces from the tests can, on one hand, enhance permeability by creating fractures nearby; on the other hand, these forces may decrease permeability by closing pre-existing fractures.

Tritium, as water, is considered by far the most mobile radionuclide present in the subsurface environment surrounding the underground test cavity. With its half-life of about 12 years, the estimated 100 million Ci hydrologic source term of tritium represents the major radionuclide of concern for the next 200 years.

Risks Associated with Tritium Migration

Proposed changes in NTS operations as well as DOE's policy of reviewing sitewide impacts under the National Environmental Policy Act (NEPA) have prompted the need for a new Environmental Impact Statement (EIS) for the NTS (DOE93d). The draft EIS (DOE96), issued in January 1996, assessed doses and risks from past activities and future operations expected to be carried out under each of the following four alternatives:

- Alternative 1: No Action. The DOE would continue to support ongoing program operations, but no new initiatives would be pursued.

- Alternative 2: Discontinue Operations. Under this option, only services required to continue the protection of human health and safety would be performed, inclusive of environmental monitoring.

- Alternative 3: Expanded Use. Implementation of this alternative would involve expansion of many current activities and programs, but also enhance current remediation and waste management activities.

- Alternative 4: Alternate Use of Withdrawn Land. While defense programs would be discontinued, there would be increased activities for waste management, remediation, and nondefense research activities (e.g., solar energy).

The proposed NTS EIS alternatives, however, are not expected to change the current inventory or configuration of subsurface contamination. Thus, an assessment of future radiological impacts to off-site residents is considered identical among the proposed alternatives. The migration of tritium from discrete underground NTS test areas to locations outside the current site boundary and accessible to members of the public are of primary concern and have been evaluated in the draft EIS.

Table 8-12 provides summary data regarding doses and risks to hypothetical individuals. Individuals are assumed to ingest contaminated well water for a period of 70 years from the nearest accessible location. The 70-year lifetime exposure scenario coincides with the time of peak concentrations of tritium in ground water for each of the three underground test sites:

- Yucca Flat. Tritium concentrations migrating from Yucca Flat to Mercury, Nevada are not expected to reach the minimum detectable level of 1 pCi/L. Lifetime doses and risks are, therefore, negligible.

- Project Shoal Area. At the closest accessible location (the eastern boundary of the Project Shoal Area), tritium is expected to reach a maximum concentration of about 280 pCi/L in about 206 years, yielding a lifetime dose of 1.6 mrem. At the nearest existing public well, maximum concentrations are not expected to occur for 278 years, resulting in doses and risks that are nearly four orders of magnitude lower.

- Central Nevada Test Area. At the nearest existing public well, the time of maximum tritium concentration is not expected for more 400 years at concentrations that are small fractions of 1 pCi/L. Associated doses and risks at this location are essentially non-existent. Near the southern boundary, tritium concentrations as high as 1.2×10^8 pCi/L

Table 8-12. Doses and Health Risks to Exposed Individuals[a] from Subsurface Radioactivity

Test Location	Receptor Location	Arrival Time[b] of Peak Conc. (yr)	Peak Tritium Concentration (pCi/ℓ)	Lifetime Dose (mrem)	Risk of Fatal Cancer
Yucca Flat	Mercury, NV	100	< 1	3.0×10^{-5}	1.5×10^{-11}
Project Shoal Area[c]	Eastern boundary[d]	206	280	$1.6 \times 10^{+0}$	8.0×10^{-7}
Project Shoal Area[c]	Nearest public well	278	<1	2.0×10^{-4}	1.0×10^{-10}
Central Nevada Test Area[e]	Central Nevada Test Area boundary[d]	15	1.8×10^{8}	$8.0 \times 10^{+3}$	4.0×10^{-3}
Central Nevada Test Area[e]	Nearest public well	410	$<< 1$	1.8×10^{-17}	9.0×10^{-24}

[a] The maximally exposed individual is a hypothetical person who is assumed to obtain drinking water from a well at the receptor location for a lifetime of 70 years, centered around the time of peak tritium concentration in the well water.

[b] Time period from the underground test date to the arrival of the peak tritium concentration in well water at the receptor's location.

[c] Results based on analysis performed by Chapman et al. 1995 (CHA95).

[d] No public well currently exists at these locations.

[e] Results based on analysis performed by Pohlmann et al. 1995 (POH95).

had been predicted for 1983 (or 15 years after testing), yielding a lifetime dose of about 8,000 mrem, or an average annual dose of 114 mrem. In 1996, these concentrations would be reduced by more than a factor of two due to natural decay. However, there has been no confirmation of these concentrations by ground water sampling and assessment at this location.

8.1.6.4 Radiological Surveillance Around the Nevada Test Site

Since 1970, the EPA's Characterization Research Division (formerly named the Environmental Monitoring Systems Laboratory - Las Vegas or EMSL-LV) has assumed responsibility for the Off-site Radiological Safety Program (ORSP) at NTS and other U.S. nuclear test sites. Among ORSP's primary objectives are to systematically measure and document levels and trends of environmental radiation and radioactive contaminants in the vicinity of the test sites.

Off-site levels of radiation and radioactivity are assessed by gamma-ray measurements using highly sensitive pressurized ion chambers (PICs) and thermoluminescent dosimeters (TLDs); by sampling air, water, soil, milk, meats, food crops, and indigenous flora and fauna; and by in-vivo/-vitro bioassays of off-site population groups. Results of these measurements are collated and made available to the public in an annual report (DOE93a). Provided below is a brief description of the major elements of the ORSP and summary data for 1993, the most recent year of published data.

External Ambient Gamma Monitoring at NTS

External ambient radiation levels are measured independently by a network of 27 pressurized ion chambers and 127 thermoluminescent dosimeters located in various communities surrounding the NTS (Figure 8-4). Ambient dose and dose rates measured by these devices represent the combined sources of cosmic and terrestrial radiation. Ambient air dose levels ranged from 66 mR/yr at Pahrump, Nevada, to 166 mR/yr at Austin, Nevada, with an average absorbed tissue dose value of 97 mrem/yr. Observed variations in ambient dose rates reflect differences in altitude, soil composition, and meteorological factors. This average of 97 mrem/yr is considerably higher than the combined national average value of cosmic and terrestrial radiation level of 56 mrem.

Atmospheric Monitoring

A network of 30 continuously operating stations monitored airborne particulate radionuclides and radioiodines. An additional 14 sampling stations sampled for atmospheric tritium and noble gases. Data indicate that airborne radioactivity from diffusion, evaporation of effluents, or resuspension of radionuclides from past releases are currently below detection limits at off-site locations.

Using the CAP88-PC model and NTS radionuclide emission data, an effective dose equivalent of 0.004 mrem/yr was calculated for the off-site maximally exposed individual.

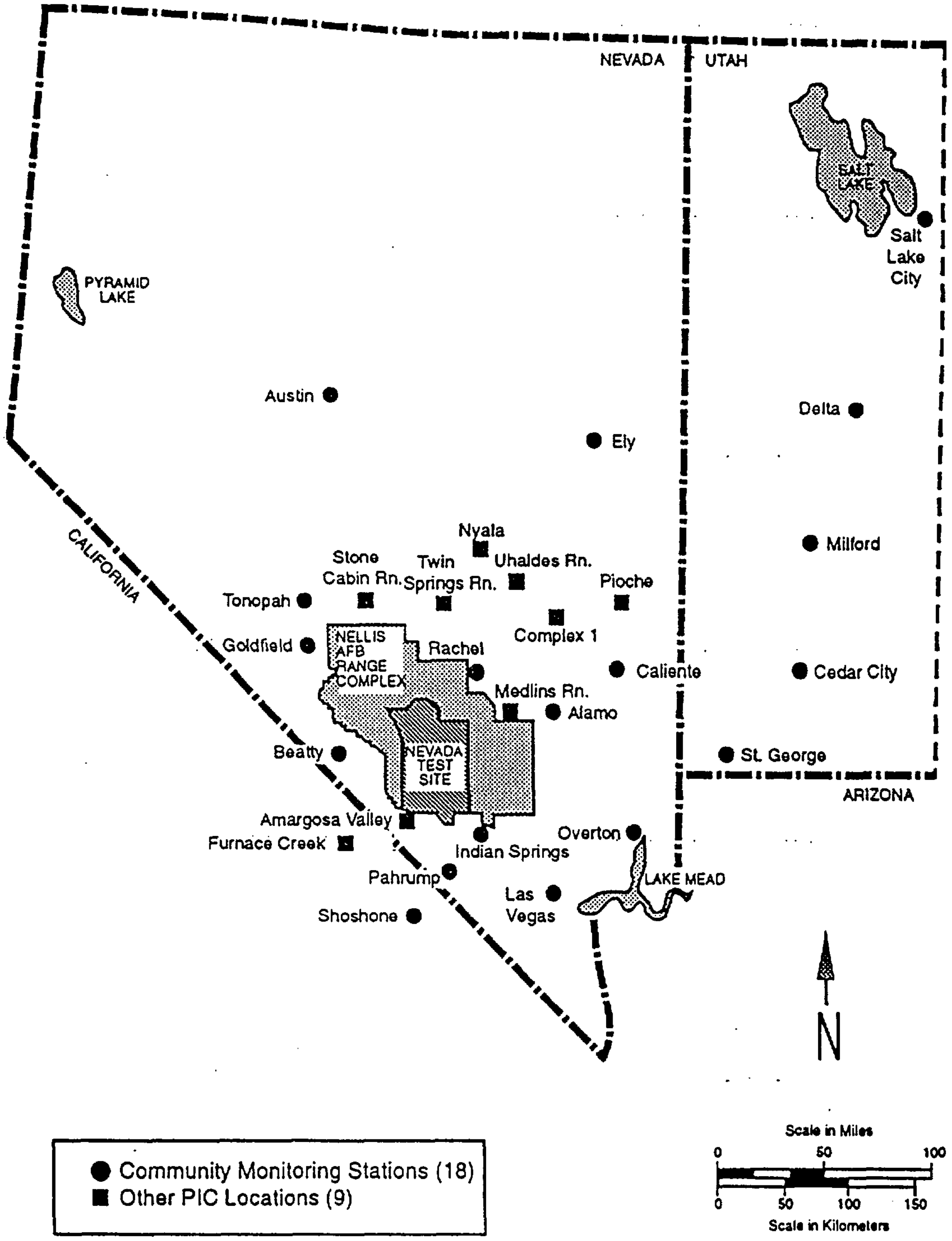


Figure 8-4. Pressurized Ion Chamber Network Station Locations - 1993

Monitoring of Local Food Products

A large variety of local foods that included milk, meat, vegetables, fruits, and wild game were obtained from specified locations to distances of up to 200 miles. Food products were analyzed for various radionuclides including H-3, Sr-89/90, and Pu-238/-239/-240. The Sr-90 levels in samples of animal bone remained very low, as did Pu-239/-240 in both bone and liver samples of domestic and game animals. Although a few milk samples contained measurable levels of Sr-90 and several fruit and produce samples contained measurable levels of Pu-239/-240 and Sr-90, their potential contribution to human internal exposures was considered insignificant.

Population Monitoring by Bioassay

Since 1970, the ORSP has been assessing representative members of the off-site populations for potential internal exposure from fallout. The off-site internal dosimetry program is designed to measure radionuclide body burdens among persons who were subjected to fallout during the early years of weapons testing as well as provide a monitoring system for present-day NTS activities and environmental conditions.

In 1993, this program included 158 individuals representing 54 families. Evaluation of participants includes a biannual whole-body count, lung count, and urinalysis. At 18-month intervals, participants also receive a comprehensive medical examination.

No transuranics were detected in any lung counts, and in general body burdens of participants were representative of any normal population when matched for age and sex distribution.

Ground Water and Long-Term Hydrological Monitoring

Since 1972, a Long-Term Hydrological Monitoring Program (LTHMP) has been implemented at NTS. Routine monitoring is conducted at specified on-site wells and at wells, springs, and surface waters in the off-site area around the NTS.

Because tritium is a product of nuclear testing that was found in significant quantities in underground test cavities and because of its high mobility, it is expected to be the first radionuclide to migrate. Therefore, tritium serves as a warning indicator of other potential radionuclide migration and correspondingly was the primary radionuclide analyzed in the LTHMP. Off-site sampling locations include 23 wells, 7 springs, and 2 surface water sites, which are sampled on a monthly basis.

In 1993 and over the past decade, detectable levels of tritium have been found in a limited number of samples obtained from surface water. In all cases, the tritium activities fall within the range of environmental levels and are thought to be the result of rainfall containing scavenged atmospheric tritium.

Summary of ORSP Results and Off-site Dose Estimates Pertaining to NTS Activities

For 1993, EPA's comprehensive off-site environmental surveillance program around NTS measured no levels of radiation that would contribute significant exposure to any member of the public.

Potential exposure from all pathways to members of the public due to NTS activities are estimated annually by two separate methods. The first calculates annual dose by means of computer effluent modeling (CAP88-PC), meteorologic, and demographic data. The second approach uses measurement data from the ORSP with conservative assumptions and standard dose conversion factors.

Based on computer modeling, the committed effective dose equivalent to the maximally exposed off-site resident for 1993 was estimated to be 0.004 mrem. Environmental sampling data estimate a comparable dose of 0.05 mrem/yr from NTS and non-NTS fallout. For the 80-km (50-mile) radius population of 21,750 individuals, a collective population dose of 1.2×10^{-2} person-rem was estimated for 1993. These doses are considered negligible when compared to the average ambient external gamma dose rate of 97 mrem/yr contributed by natural cosmic and terrestrial radiation alone.

8.2 CONSIDERATIONS OF CLIMATE CHANGE OVER THE REGULATORY TIME FRAME

The important aspects of climate (defined as the ensemble of weather conditions over time) relevant to the Yucca Mountain Rulemaking are precipitation and temperature variability. These parameters influence, directly and indirectly, infiltration rates in the area of the proposed repository.

"Variability" means the timing, rates of change, magnitude, and persistence of conditions at one or more levels. Inferences about variability are based on studies of past conditions in the region, as recorded by paleoenvironmental indicators, both geological and biological, and on the use of computer models of the atmospheric circulation to simulate both past and future climatic regimes. Models are tested by comparing their reconstructions of past climates with paleo-data; if they produce good simulations of past conditions, confidence in their forecasts of future climate is enhanced. Thus, paleo-data are essential in assessments of future climates.

However, another factor must be considered: the impact of human interference with naturally occurring climate variations. Large-scale changes in atmospheric composition have occurred and are almost certain to continue into the future (HOU92). The only tool to anticipate the consequences of such changes are general circulation models that may help to chart the future course of climate change. As the concentration of greenhouse gases in the 21st century will likely exceed anything the world has experienced for millions of years, it must be recognized that the paleo-record may not define the climate of the future. Unknown feedbacks, or abrupt changes in the climate system that have been rare in the past geological record may occur in the future.

Nevertheless, the paleo-record combined with realistic computer models of existing and future climate, incorporating all known interactions and feedbacks, provide the best set of tools currently available to define the potential limits of climate variability in the area of interest.

8.2.1 Past Climate Conditions and Variations

Global climate has changed over glacial to interglacial time scales in response to changes in orbital forcing (the relative position of the earth to the sun, with consequent changes in the geographical and seasonal distribution of incoming solar radiation). In simple terms, these changes altered the Pole-Equator temperature gradients, which led to changes in atmospheric circulation and the overall hydrological balance of the earth -- causing ice sheets to accumulate on the continents at high latitudes, the sea level to fall, and rainfall patterns in the tropics to shift.

Changes in incoming solar radiation alone were insufficient to bring these environmental changes about; they were amplified by internal feedbacks of the climate system itself, most probably through changes in atmospheric composition and the albedo (reflectivity) of the earth's surface. Such feedbacks led to reduced levels of carbon dioxide and methane (both greenhouse gases) and a higher overall albedo for the earth, due to more extensive snow and ice cover, and more extensive deserts. However, at other times in the cycle of orbital changes, feedback mechanisms brought about increases in greenhouse gases and other changes in the climate system, eventually leading to rapid destruction of the ice sheets and abrupt deglaciations. The growth and decay of ice sheets affected the atmospheric circulation, displacing jet streams equatorward, causing massive increases in rainfall in previously dry areas.

Southern Nevada and the Great Basin experienced such dramatic changes, which, together with lower temperatures, led to aquifer recharge and the filling of many closed basins with extensive lakes. Such changes are clearly documented in geological studies of raised shorelines, exposed lake sediments and their sedimentological and geological characteristics. Lake level variations extending back into the last glaciation are best known; they are generally well-dated and have been studied in many areas of the western United States. Observed changes are well supported by a variety of biological evidence, particularly that obtained from the analysis of packrat middens, which contain discrete samples of local vegetation in the vacinity of the packrat nests, from particular intervals in the past. When lake levels were high, vegetation was generally more extensive; some areas that are arid today were forested. Hydrological changes in the arid western United States do not coincide in detail with the record of continental ice volume changes, but it is clear that in broad terms, high lake levels were present when the Laurentide Ice Sheet was extensive, and that water levels fell in association with deglaciation. As noted by Smith and Street-Perrot (SMI83), "More than a hundred closed basins in the western United States contained lakes during the Late Wisconsin [the last episode of the ice ages], 25,000 to 10,000 yr B.P., but only about 10% of the lakes are perennial and of substantial size today...". Even in today's hyperarid Death Valley, there is evidence that an extensive lake occupied the basin between 21,500 and 11,900 yr B.P. (HOO72; SMI83).

The longer term record of hydrological variability is much harder to document, given the problems of dating and the real possibility that some paleo-lakes may have been caused by slight tectonic changes or other geomorphological factors. Furthermore, rapid changes in ice sheet size, as postulated from sedimentary records in the North Atlantic and elsewhere, may have resulted in very abrupt changes in the hydrological regime in the western United States, which are as yet poorly documented in the paleo-record.

If jet stream displacement, due to ice sheet growth and decay, is the principal factor in hydrological change in the western United States, there is good reason to suspect that a quite variable hydrological regime has influenced the region over glacial-interglacial timescales. Nevertheless, the more prolonged glacial episodes were dominated by cooler, wetter conditions, associated with higher infiltration rates, more vegetation, and the presence of many freshwater lakes in the Great Basin. Quantifying such changes is difficult, but Spaulding et al. (SPA83) estimates the limit at the last glacial maximum as approximately 6°C colder, with precipitation levels double those of today.

8.2.2 Potential Future Climate Conditions

Orbital variations clearly have driven the broad-scale variations of global climate over the last several million years at least, and these orbital variations are likely to be a dominant influence in the future. Since the orbital variations are periodic and predictable, their occurrence in the past and in the future can be calculated. Variations over the past million years have occurred within a fairly limited envelope, and predicted variations for the future show that (for at least the next 250,000 years) the expected orbital changes will stay well within this envelope. How such changes will affect climate can be assessed by using the solar radiation changes to force a global climate model to simulate both past and potential climate variations in the future.

Most studies attempt to reconstruct past changes where the simulations can be verified by observation, but a few attempts to forecast future changes have been made over the past 25 years, at varying levels of sophistication. Figure 8-5 shows the results of these efforts, with the overall parameter describing the output expressed (on the righthand side) in terms of global temperature. Obviously, the sophistication of such calculations has increased over the years, but most studies consistently predict that global climates over the next 60,000 years or so will gradually shift towards a full glacial mode, similar to that experienced 20,000 years before present (BP) during the most recent glacial period. Indeed, the trend towards such a state began a few thousand years ago, in the mid-Holocene.

The trend towards a glacial extreme is not monotonic but involves minor oscillations on a generally downward trend in temperature. Following the temperature minimum, there is some indication that conditions like those of today will not return again until about 120,000 years after the present (AP). It also appears that the "saw-tooth" nature of past climate variations, with slow declines to cold glacial conditions, followed by abrupt "terminations" of glacial conditions will also continue into the future.

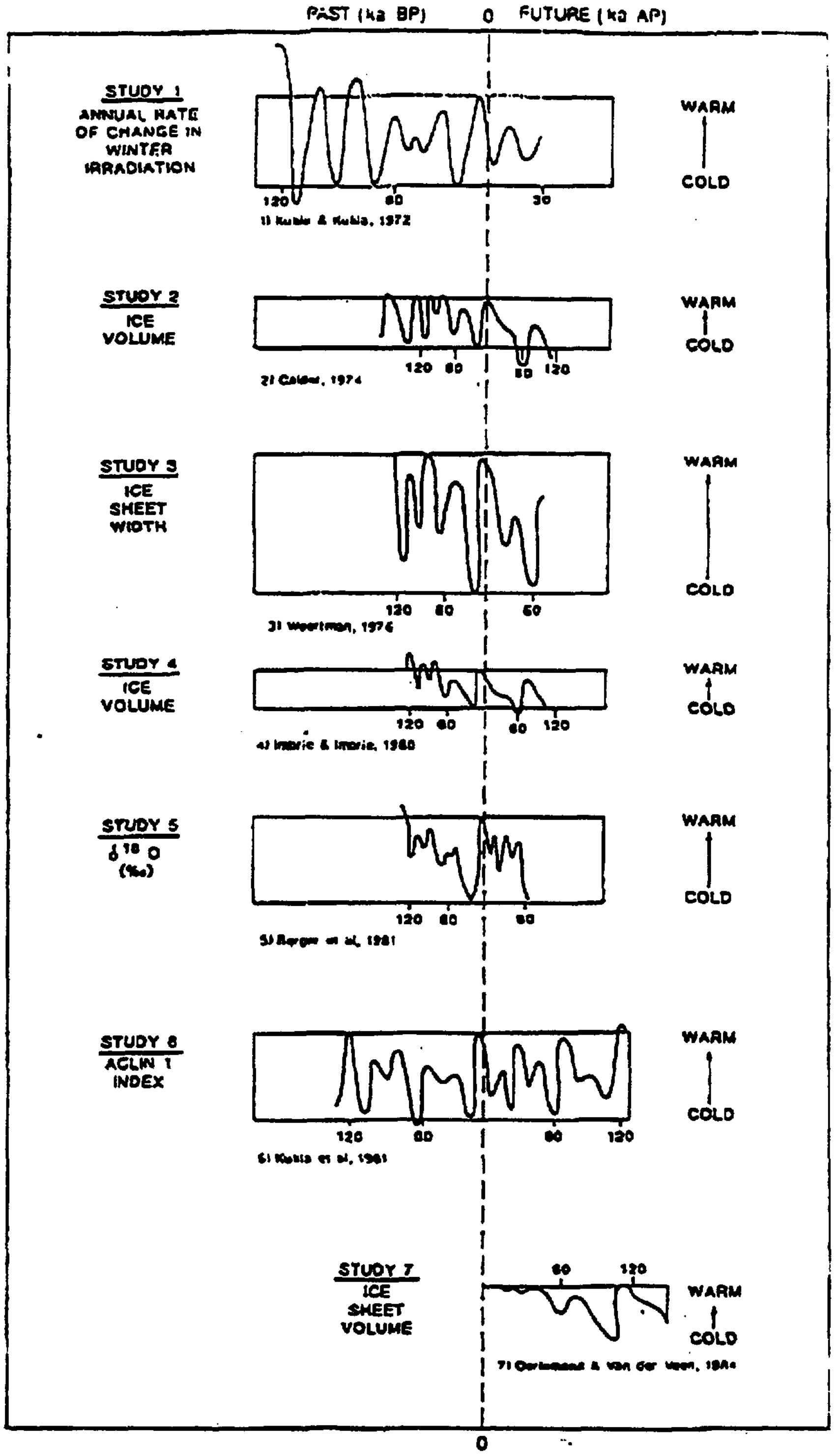


Figure 8-5. Future climates, expressed in terms of overall global temperature change, as predicted by seven different models driven by changes in orbital forcing. The boxes on each diagram delimit the last glacial and interglacial extremes. Dates are in years x 10^3 (from GOO92).

In general, the present arid climate conditions are expected to be maintained in the future. The Sierra Nevada Mountains, which lie to the west of Yucca Mountain, have a strong rain-shadow effect on the Yucca Mountain region, and this effect is expected to be maintained or enhanced in the future because the Sierra Nevada range is still increasing in elevation (DeW93).

These are very broad conclusions that do not allow for the high-frequency oscillations, superimposed on longer term trends, which have been seen, for example, in the Greenland ice cores, or in some marine sedimentary records from the North Atlantic, and most recently in the Santa Barbara Basin (BEH96). Such changes would be expected to occur in any future glaciation, since they appear to be integrally linked to the dynamics of ice growth and decay and their impact on ocean circulation (BRO94).

What these models do not consider is the potential *additional* effects of greenhouse gas increases on the radiative balance of the earth and, consequently, on the general circulation. It is generally believed that the small insolation changes brought about by orbital changes are insufficient by themselves to bring about glaciation, or indeed to terminate glaciations. The critical issue is the feedbacks, which may amplify the small radiative signal, with the ice sheets themselves playing a major role (via albedo effects, sea-level change, topographic influences on atmospheric circulation, effects on ocean thermohaline circulation, etc.). What is not clear whether any near-term increase in greenhouse gases (in the next few decades to centuries) would eventually be overwhelmed by the orbitally induced shift toward future glaciation, or if the warmer climate would preclude such a development by minimizing the necessary feedback mechanisms. Broecker (BRO75) termed this near-term warm episode a "super-interglacial" because it may involve temperatures higher than in any recent interglacial period. As such, it is difficult to predict what the overall consequences of such a unique state might be for the future evolution of climate.

One study of such a scenario used a 2.5D general computer model to assess both anthropogenic effects *and* orbital forcing (BER91). It begins by assuming that the Greenland Ice Sheet will be entirely consumed in the near term, but even so, *the general direction of long-term climate change towards glaciation is not changed*, although the peak timing of the next glaciation is delayed by about 5,000 years (Figure 8-6). However, this model is still fairly crude and does not incorporate many of the feedbacks that may be critical in the evolution of future climate. More experiments with transient climate simulations, using the next generation of coupled ocean-atmosphere general circulation models, will be needed to obtain a more sophisticated answer to this question.

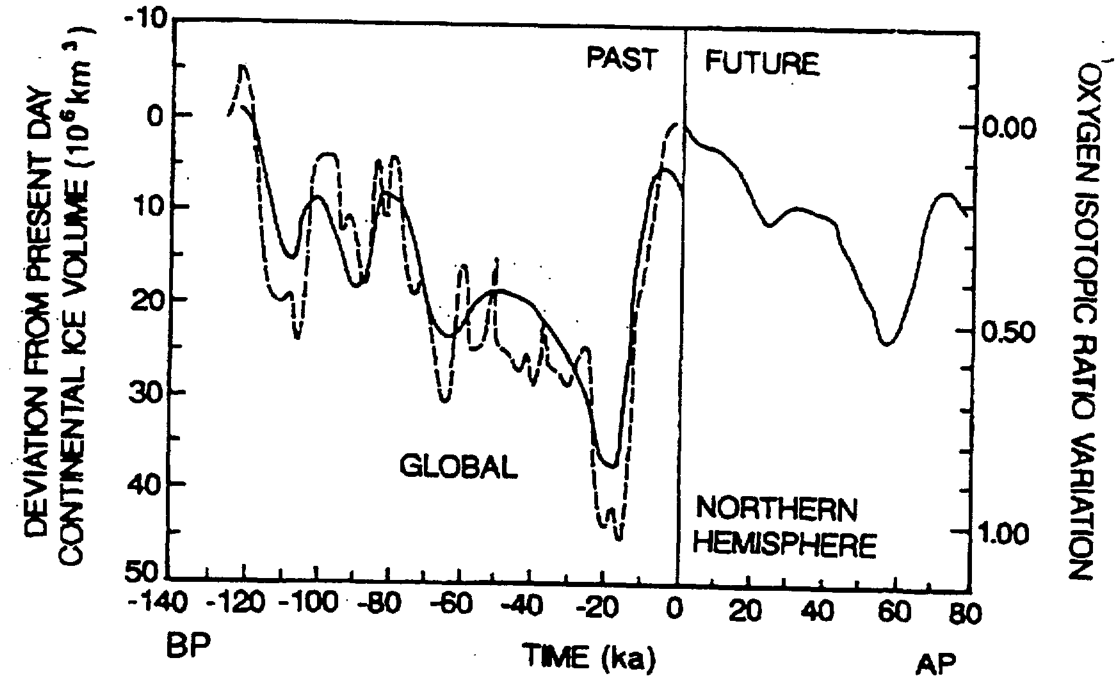


Figure 8-6. Model simulations (solid line) of past and future climate conditions, expressed in terms of changing ice volume on the continents, and including anthropogenic greenhouse effects in the immediate future. Dashed line gives past global ice volume changes as registered by oxygen isotope ratios in benthic foraminifera from the oceans (BER91).

At this stage, there is no compelling evidence that the world of the next million years will not be subjected to the same range of climate variations experienced over the last million years; however, in the near term (from the next few decades to several thousand years), an enhanced greenhouse effect will very probably bring about warmer conditions than have been experienced for thousands, perhaps even hundreds of thousands of years. This was the general conclusion of experts who were asked to assess the magnitude and direction of future climate change (cf. Figure 3-11 in DeW93). They estimate that the likely upper limit of a temperature increase in the mean annual temperature of the Yucca Mountain region would be about +2 to 3 °C. Whether this effect will persist for hundreds or thousands of years depends greatly on assumptions made about future energy consumption patterns and the overall availability of fossil fuels in the future. If society eventually limits fossil fuel consumption, this warmer episode may come to a close, with the naturally occurring trends then becoming dominant. Nevertheless, the possibility that a greenhouse gas-induced "super-interglacial" may lead to unanticipated pathways in the climate system and new climate states can not be entirely ruled out (BRO87).

However, the changes of greatest concern at Yucca Mountain are those associated with the "glacial climate mode" rather than with an "interglacial mode." Past history indicates that wetter conditions in the region have generally been associated with globally cooler climates, or with transitions to such climates, whereas interglacials have been arid. Currently, no evidence suggests that this basic pattern is likely to be different in the future. Hence, the immediate future climate of Yucca Mountain, dominated by anthropogenic effects, is likely to be as dry or drier than the present, but eventually cooler and wetter conditions will dominate the area during persistent glacial climate modes.

8.3 PROTECTION OF INDIVIDUALS

On the premise that potential releases of radionuclides from an undisturbed repository into the accessible environment will not occur for thousands of years following repository closure, any standard that intends to protect members of the public will require an assessment of performance at times far into the future. To determine whether the proposed repository will be able to comply with a standard that specifies a dose (or risk) limit, it is necessary to model exposure conditions to some individuals or representative group of individuals and then compare the calculated dose to that of the standard. In effect, this implies that a given standard must not only specify a dose (or risk) limit but also the individual or group of individuals for whom the calculation is to be made.

As a rule, performance assessments consider all of the main modes of exposure pathways: ingestion, inhalation, and external irradiation. For each pathway, however, parameters must be selected that are affected by the behaviors, habits, lifestyles, and other characteristics of individuals within the general population. The choice of who is to be protected can profoundly affect the calculated risk and whether or not the standard is achievable. A key issue in developing a standard, therefore, concerns how to select the individuals or group of individuals and what assumptions to make regarding conditions of exposure. For example, it is possible to select person(s) who, on the basis of certain factors such as age, pregnancy, or metabolic factors, may either receive larger doses or are more sensitive to radiation than others and whose vulnerability may be further amplified by assumptions about lifestyle and individual behavior.

In general, conservative assumptions result in calculations of dose/risk that may be considered bounding. In some instances, such bounding estimates can be very useful. If compliance can be shown with a bounding estimate, then by definition all other conditions must be assumed to comply, and there is no need for a more complex analysis. A common criticism of bounding estimates is that conservative assumptions are compounded and produce highly improbable conditions that overestimate future exposures. This would imply a standard that is unjustifiable, too stringent, or potentially unachievable by present-day technologies.

8.3.1 The Critical Group Concept

The release of radionuclides from the repository to the accessible environment is likely to expose a sizeable number of individuals. Within the exposed population, radiation doses and associated health risks can be assumed to vary from low or negligible levels to some maximal level. To avoid a standard based on persons who represent an extreme condition, the NAS Committee on Technical Bases for Yucca Mountain Standards has endorsed the use of the critical group in developing a standard. The critical group is defined by the ICRP (ICR77, ICR85) as a relatively homogeneous group of people whose location and lifestyle are such that they represent those individuals expected to receive the highest doses as a result of radioactive releases. As part of the critical group definition, the ICRP specifies the following additional criteria:

- Size - The critical group should be small in number and typically include a few to a few tens of persons.

- Homogeneity among members of the critical group - There should be a relatively small difference between those receiving the highest and the lowest doses. It is recommended that the range between the low and high doses not differ by more than a factor of ten or a factor of about three on either side of the critical group average.

- Magnitude of dose/risk - It is suggested that the regulatory limit defined by a standard exceed the calculated average critical group dose by at least a factor of ten.

- Modeling assumptions - In modeling exposure for the critical group, the ICRP recommends that dose estimates be based on cautious but reasonable assumptions.

In principle, the critical group concept adequately meets the objectives of protecting the vast majority of the public since these individuals can be expected to receive the highest doses based on cautious, but reasonable, assumptions. The group should be small enough to be relatively homogeneous with regard to factors that affect the magnitude of exposure and risk.

The ICRP, however, does not prescribe the lifestyle, habits, or conditions of exposure that may define a critical group in the far future. Rather, its generic recommendations are limited to the need to use current knowledge and the use of cautious but reasonable assumptions for characterizing future exposure scenarios.

According to current understanding, the principal route by which radionuclides contained in the repository could expose humans in the accessible environment is by way of contaminated ground water.

To proceed from contaminated well water concentrations at various locations and times to calculations of doses and risks to a critical group requires the development of a comprehensive exposure scenario that specifies discrete pathways and quantifies the intakes of individual radionuclides. Depending upon the potential uses of contaminated well water, prominent pathways for human exposure may include internal exposure from the ingestion of contaminated drinking water, vegetables, fruits, dairy products, and meats. For persons engaged in agricultural activities, internal exposure may also result from the inhalation of airborne contaminants resuspended from soil that has been irrigated with contaminated water. Over time, the buildup of soil contaminants could reach levels that also yield significant external doses.

The selection of an exposure scenario that is appropriate for a specified critical group, therefore, not only requires a complex array of parameter values that define potential radionuclide concentrations in various media to which individuals may be exposed, but must also provide quantitative descriptions that include where individuals live, what they eat and drink, and what their sources of food and water are. Not surprisingly, many key parameters needed to model human exposures at Yucca Mountain are highly site-specific. Parameters considered site-specific largely reflect the desert conditions of the sparsely populated Amargosa Valley. For example, the combined impacts of low rainfall, desert temperatures, and soil quality limit land use for farming. However, these very conditions mandate extensive irrigation of farm crops and use of local ground water for cattle. Under these conditions, contaminated well water has the potential for unusually high activity concentrations in all locally grown food products.

8.3.2 Probabilistic Scenario Modeling

The unique requirements for modeling repository performance and human exposure scenarios for times far into the future, as briefly described above, have emphasized limitations as well as uncertainties in performance assessment methodology. The need to provide numerical values for parameters that define human exposure pathways is a major source of uncertainty.

To account explicitly for uncertainties, the NAS Committee (NAS95) offered two probabilistic modeling approaches. The first, described in Appendix C of the NAS report, A Probabilistic Critical Group Approach, uses statistical methods and probability values to characterize members of the critical group. The second, The Subsistence-Farmer Critical Group, as defined in Appendix D of the report, also employs a probabilistic method but is more presumptive inasmuch as it identifies the subsistence farmer as its principal representative of the critical group. Provided below is a brief description of these two modeling approaches.

Approach #1: A Probabilistic Critical Group. In support of ICRP recommendations to use current knowledge and cautious but reasonable assumptions to identify the potential future critical group and conditions of exposure, the NAS Committee (NAS95) suggested the following steps for the Monte Carlo method that implements a probabilistic assessment:

Step 1: Identify general lifestyle characteristics of the larger population that includes the critical group.

Step 2: Quantify important characteristics, distributions of characteristics, and geographic locations of the potentially exposed population.

Step 3: Based on findings in Steps 1 and 2, model radionuclide transport for estimates of exposures to members of the critical group.

The first and second steps serve to identify the larger exposed population of which the critical group is a subset receiving the highest dose and, therefore, is at greatest risk. As noted previously, human exposure to ground water contaminants may involve several pathways.

Some pathways are likely to be more important than others and reflect the way in which contaminated water is used. Thus, specific information on location, living patterns, lifestyles, and economic activities of potential members of the exposed population can lead to the identity and characterization of the critical group. Based on current understanding, principal factors affecting the magnitude of individual exposure include 1) distance of residence from the repository, 2) level of dependence on local well water, 3) use of local well water for drinking, crop irrigation, livestock, etc., and 4) personal habits that affect food and water consumption. For example, if current population data were to show that individuals at maximum risk involved a cluster of residents whose potable water was supplied by a common well, the critical group might consist of a mix of economic lifestyles that could include casino workers, NTS defense workers, homemakers, office workers, professionals, skilled laborers, etc.

An important component of Step 3 is the superimposition of the critical group to area(s) that in the far future will overlay the contaminated aquifer at locations that may range from near maximum at the footprint of the repository to lower concentrations defined by the directional migration pattern of the contamination water plume. For each location, specific data would be sought that define the slope/topography of the land, quality of soil, depth to ground water, well productivity, and other factors. Taken collectively, data for each location would be used to determine its suitability and probabilistic future use for farming, residential, commercial, industrial, and other purposes that may affect the exposure of members of the critical group. To account for probabilistic land use, local ground water dependence, and numerous model parameter uncertainties, the NAS Committee recommended that probabilistic distributions of doses/risks be based on Monte Carlo simulations. In this method, data on the frequency distribution are sampled to provide input to generic model equations. In effect, the Monte Carlo method produces a single predicted value for each set of randomly selected parameter values. The results of numerous (hundreds to thousands) iterations of model solutions are then statistically analyzed to determine the distribution of model solutions. From such a distribution of predicted values, information is extracted that defines the best estimate of an average value (i.e., the most probable value), the range of potential values, and a measure of uncertainty of model predictions that reflects the collective uncertainties of input parameters (HEN92).

Approach #2: The Subsistence Farmer Critical Group. The model described in Appendix D to the NAS report specifies a priori one or more subsistence farmers and makes assumptions designed to define the farmer at maximum risk to be representative of the critical group.

(Note: Subsistence farming does not exclude commercial farmers who, in addition to cash farm products, raise food for personal consumption.)

The subsistence farmer of the future would have nutritional needs consistent with those of a present-day person and, like the subsistence farmer of today, would obtain most or all drinking water from an on-site well that would also be used in the production of all consumed food. The subsistence farmer would also be assumed to live his/her entire life at the same location.

Thus, the magnitude of the dose to a subsistence farmer will largely be defined by the space- and time-dependent probabilistic distribution of radionuclide concentrations in ground water at the point of water withdrawal.

Although a standard that incorporates the probabilistic critical group would allow the applicant to use more flexible assumptions in demonstrating compliance, this approach is relatively complex and more difficult to implement than the subsistence farmer critical group. Moreover, the assignment of probability values relating to land use, demands on natural resources, and human activities to the probabilistic critical group may be viewed as subjective and biased by the limitations that define present-day society.

Although the subsistence farmer approach is relatively simple and easy to understand, it may be unrealistically conservative. For example, the assumption might well be made that the subsistence farmer uses a well at the repository boundary, where contamination levels are highest, even though this location is unsuitable for farming. However, the model requires no speculation about the future or judgmental assumptions.

Moreover, the subsistence farmer approach is consistent with previous EPA programs and past EPA guidance (EPA92). These programs define a dose standard in behalf of the reasonably maximum exposed individual (RMEI). For example, the National Emission Standards for Hazardous Air Pollutants (NESHAPS) (40 CFR Part 61) requires estimation of the dose to a person assumed to reside at a location in the accessible environment where that individual would be expected to receive the highest dose.

The basic approach to modeling RMEI doses is to identify the most important exposure pathway(s) and input parameters. By using maximum or near maximum (i.e., 95th percentile) values for one or a few of the more sensitive parameters while assuming average values for others, it can be reasonably assumed that derived RMEI dose estimates correspond to near maximum exposures that could be received by any member of the exposed population. However, the inappropriate combination of highly conservative model parameters is avoided, since the ultimate objective of the RMEI approach is to define an exposure that is well above average exposures but within the upper range of possible exposures.

8.3.3 Support for Subsistence Farming at Yucca Mountain Using Present-day Data

8.3.3.1 Data Reported by the Nevada Agricultural Statistics Services

The Nevada Agricultural Statistics Service compiles annual comprehensive census data for all commercial agricultural activities reported to the Nevada Department of Business and Industry and to the U.S. Department of Agriculture. Data summarized below were extracted from the report issued in September 1995 and represent the most recent information (OWE95).

Of the 70.3 million acres of land in Nevada, about 8.8 million acres (about 13 percent) are used for agricultural purposes. By far, the largest percentage of agricultural land is rangeland (88 percent), with cropland and "other" representing the remaining 9 percent and 3 percent, respectively. The number of farms reporting census data for 1994 totaled 2,400, with an average of about 3,700 acres per farm. Adjusted for seasonal variations, the total labor force engaged in commercial agriculture numbered 54,000 persons, or about 4 percent of the total state population.

Table 8-13 summarizes livestock data for meat, milk, and, to a limited extent, wool and poultry/egg production. Of the 2,400 reporting farms, about two-thirds engaged in cattle ranching. Fewer farms engaged in commercial dairy farming (8.3 percent) and raising sheep (13.3 percent) and hogs (5 percent). In total, livestock generated $189.4 million in farm commodities in 1994.

Table 8-13. Livestock Data for 1994

	Cattle	Milk Cows	Sheep	Hogs	Chickens
No. of farms that raise livestock	1,600	200	320	120	9
Total inventory of livestock	490,000	21,000	91,000	9,000	---

Among the dominant crops raised commercially are alfalfa, wheat, barley, garlic, onions, and potatoes. Statistical data, including acreage planted, production yields, and total harvests are shown in Table 8-14. The total value of crop commodities for 1994 amounted to about $110 million.

Table 8-14. Crop Data for 1994

Crop	Acres Harvested	Yield/Acre	Total
Wheat	9,000	74.4 Bu/acre	670,000 Bu
Barley	4,000	85.0 Bu/acre	340,000 Bu
Alfalfa hay	240,000	4.3 t/acre	1,032,000 tons
Other hay	230,000	1.6 t/acre	368,000 tons
Alfalfa seed	13,000	680 lbs/acre	8,840,000 lbs
Garlic	1,650,000	7.5 t/acre	12,375,000 tons
Onions	1,500,000	21 t/acre	31,500,000 tons
Potatoes	8,000	15.6 t/acre	123,098 tons

8.3.3.2 Data Collected by EPA Study Team

The census information provided by the Nevada Agricultural Statistics Service contains only limited statistical data by counties. Moreover, census data are limited to commercial farming and thus do not include subsistence farmers and local gardeners who grow vegetables and fruits exclusively for personal consumption. For these reasons, EPA organized a study team to obtain additional data for Nye County and, specifically, the Amargosa Valley, the region considered most likely to be at risk from ground water releases. Under contract to EPA, the team examined various documents prepared by Nye County, obtained regional data on ground water usage, interviewed state and county officials by phone, and conducted a site visit in November 1995. Principal persons providing information and supporting the site visit included: Steve Frishman, State of Nevada, Agency for Nuclear Projects; Marcia Housel, Planning Information Corporation of Nye County; Michael Delee, Amargosa Valley Chamber of Commerce; George Blanship, Nye County Board of Commissioners, Nye County Documents; and Brad Mettam, Yucca Mountain Assessment Office. Findings of the EPA study team are summarized below.

Nye County: An Overview

Nye County encompasses the proposed Yucca Mountain Site and represents a land area of 18,000 square miles, most of which is owned by the Federal government. Only about 7 percent of the land (about 822,000 acres) is owned privately. Currently, Nye County has a resident population of about 24,000 persons, which is expected to nearly double by the year 2010. Economic and social factors responsible for the 4.6 percent annual growth rate in recent years remain but could be tempered in future years by availability of water and other resources.

Water Usage and Water Restrictions. For Nye County, ground water comes primarily from one of these four water basins:

- Ralston Valley ground water, the source for Tonopah's municipal water supply.
- Big Smoky Valley ground water, the source for Round Mountain's municipal water supply and the source for agricultural, domestic, commercial, and industrial uses throughout the valley.
- Amargosa Valley ground water, the source for agricultural, domestic, commercial, and industrial uses throughout the valley.
- Pahrump Valley ground water, the source for the Central Nevada Utility Company, other water purveyors in the valley, and agricultural, domestic, commercial, and industrial uses throughout the valley.

Under Nevada statutes, the Nevada Division of Water Resources retains the right to issue water permits and regulates water uses in areas where the ground water is being depleted or is at risk of depletion (designated basins). Permits are required for all large-quantity ground water users, which include public as well as agricultural projects. Permits are issued on a priority basis, which generally favors projects that have a high public interest over private water usage. Permits are categorized by end use but do not specify crops when ground water is used for irrigation purposes. Water usage for commercial crop irrigation is limited to 127 L/m^2 per month, which corresponds to 60 inches per year.

A noted exception to the regulated use of water pertains to private wells. A well situated on private land is excluded from permit requirements if consumption is restricted to a single dwelling and is limited to less than 800 ft^3 per day. This amount of water is considered adequate to support subsistence farming and is, therefore, a significant factor in the scenario model described in Section 8.5.

Social and Economic Characteristics. National census data for 1990 (NYE93f) reveal the following statistical profile for Nye County residents:

- Average age: 36.5 years
- Sex: Male: 53%
  Female: 47%
- Median family income: $34,196
- Per capita income: $15,454
- Size of labor force: 50% of total population
- Unemployment rate: 5.4%

Whites constituted 92 percent of the population, with native American Indians representing the largest minority group at 3.6 percent. Minorities with smaller representation included Hispanics, African-Americans, and Asians.

Nearly two-thirds (about 62 percent) of Nye County residents lived in urban areas. Of the rural residents, only a small fraction (about 200 individuals) were classified as belonging to the "farm population." For persons aged 25 years or older, 75 percent completed high school and 9.5 percent held a college degree.

The most common occupation among residents was classified as "Precision Production, Craft, and Repair," which incorporates construction trades and extraction (mining) industries. However, Nye County is characterized by widely separated communities with limited interaction between them. County-wide statistics are, therefore, potentially misleading since the economic bases of communities within the county vary greatly.

Amargosa Valley Demographics

Amargosa Valley, an unincorporated town in southern Nye County, is the closest community to the proposed repository, approximately 16 miles from the site. Amargosa Valley includes some 550 square miles of desert, with only 1,143 individuals dispersed over the 352,000 acres. These individuals live in a number of small clusters scattered throughout the town boundary. Unlike most communities, Amargosa Valley has no well-defined town center consisting of commercial establishments, although this appears to be slowly changing.

Although agriculture was established by the early 1900s, serious settlement began only in the 1950s and 1960s and was stimulated by the availability of low cost land. The difficulty and cost of harvesting crops in this desert community (mainly cotton and alfalfa) caused many of the early farms to fail and become abandoned. However, in recent years, there has been a steady change as pumping has become more efficient and new agricultural ventures have been initiated.

Most of the income in the area has come from mining operations. Until the late 1980s, the main employers in the area were two mining operations, the American Borate Company (ABC) and International Mineral Ventures (IMV). The closure of ABC initially reduced the town's prosperity and population, but recently a partial reopening of ABC plus increased employment at other industrial minerals facilities have resulted in slow but steady economic growth since 1990.

According to Nye County's Baseline Economic and Demographic Projections: 1990-2010 (NYE93c), Amargosa Valley employment in 1990 was 219 persons within Amargosa and 26 out-commuters. The distribution within industry is shown in Table 8-15, along with baseline projections through 2010. According to the Nye County Overall Economic Development Plan (OEDP) (NYE93a), the County's highest unemployment rate was in Amargosa Valley, at 11.9 percent.

The average per capita income of $10,809 was substantially below the Nye County average of $15,454, and 25.6 percent of the population was below the poverty level. Of the 561 persons over the age of 16, 412 were in the labor force, with 49 persons unemployed and looking for work. The industry employing the most persons was mining.

Table 8-15. Reference Baseline Projections (February 19, 1993): 1990-2010
Total Employment: Full- and Part-time Jobs - Amargosa Valley
(NYE93c)

	1990	1995	2000	2005	2010
Farm & Agr Svc	30	29	29	28	27
Mining	44	39	44	47	50
Construction	11	8	9	9	10
Manufacturing	5	6	6	7	7
Trans. & Pub. Util.	8	10	13	14	16
Wholesale	1	1	1	1	1
Retail	22	36	56	77	98
Fin. Ins. Real Est.	8	11	15	18	20
Services	80	106	129	156	183
Government	11	11	12	10	10
TOTAL	219	255	315	368	423

Field Observations Pertaining to Agricultural Activities at Amargosa Valley

A tour of the Amargosa Valley focused on current agricultural activities, including those with a limited operating history. The first farm visited had grown barley and alfalfa this past year with a yield of about 1.5 tons per acre per cutting. With 5 to 6 cuttings per year, the yearly yield was estimated at 10 tons per acre. Recently several pistachio trees have been planted, which are expected to bear nuts within a few years (Photo #1). For the future, the farm-owner anticipates raising cattle and estimates that his land could produce 60 head of cattle per year.

A second and much larger farm (Funeral Mountain Ranch) visited also raised alfalfa. Alfalfa grown commercially here is utilized as 1) "green chop" for consumption by local dairy farmers, 2) bailed and shipped to California (Photo #2), and 3) dried/pelletized for shipment to Japan.

Two different irrigation methods are used in Amargosa Valley. The first uses a "pivot" in which water is sprayed from a rotating arm that irrigates a circular area. The second method divides the land into small berms, which are typically flooded twice a month.

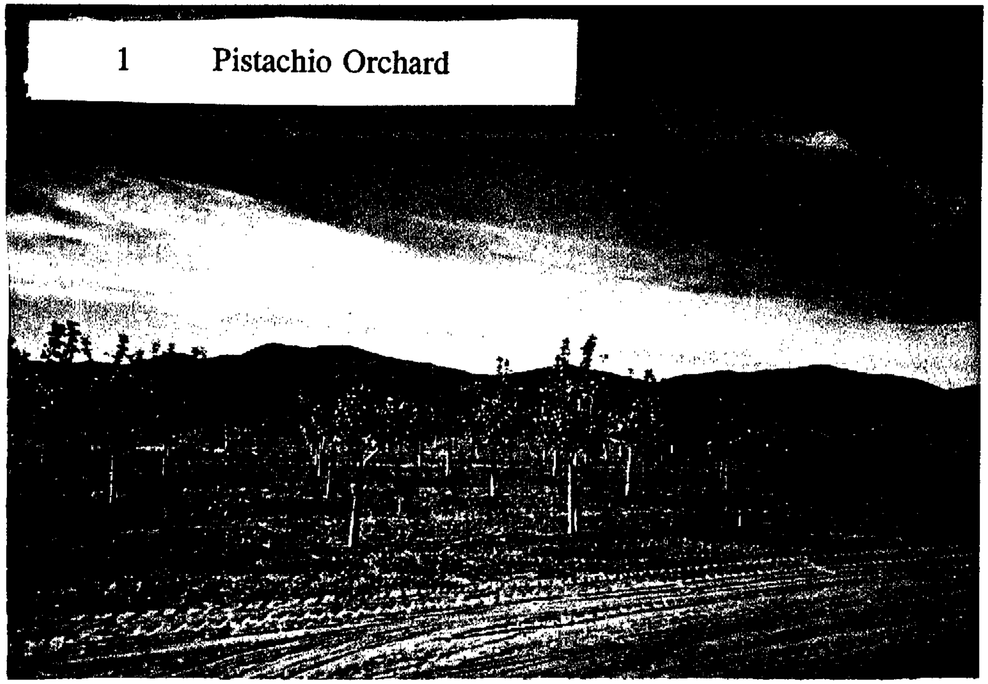
A third farm visited was a large dairy farm with 2,800 cows, of which 2,300 are milk producers. This farm has been in operation for only one year. Its milk is shipped unprocessed into California. Due to the size of the operation, cows do not graze but are fed locally grown "green chop". Due to the success of this farm, a second and third dairy farm of comparable size are currently under construction.

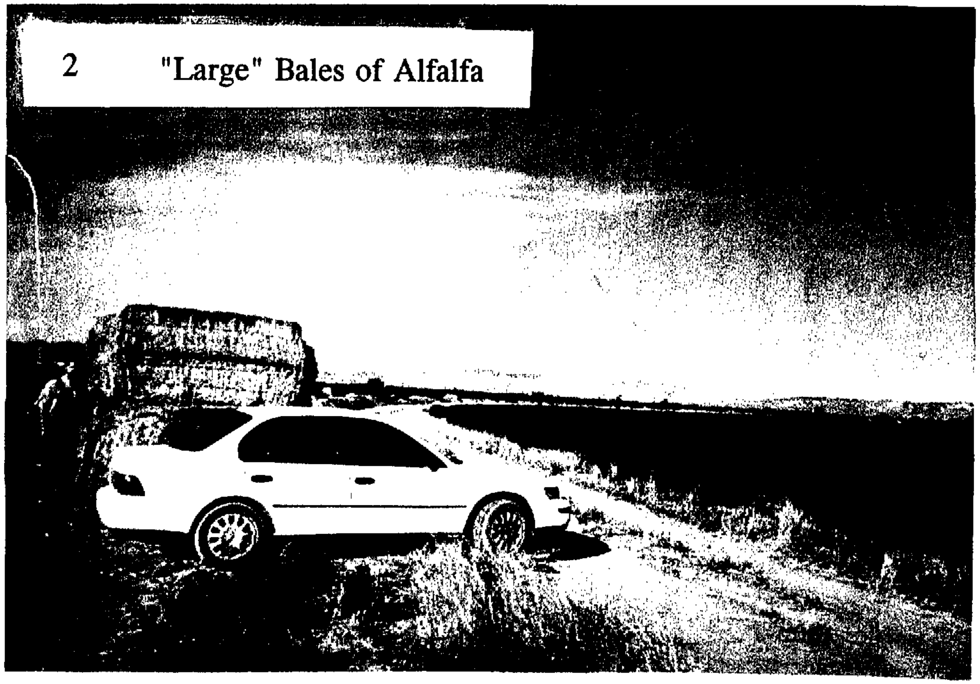
The study team visited four additional farms, including the following:

- a small farm raising pigs, sheep, and ducks (Photo #3),
- a farm growing primarily vegetables that are sold locally,
- a small fruit-tree orchard that was originally planted as an experiment to determine the feasibility of growing apricots, peaches, and figs (Photo #4),
- a sod farm, which ships and sells its products outside the valley.

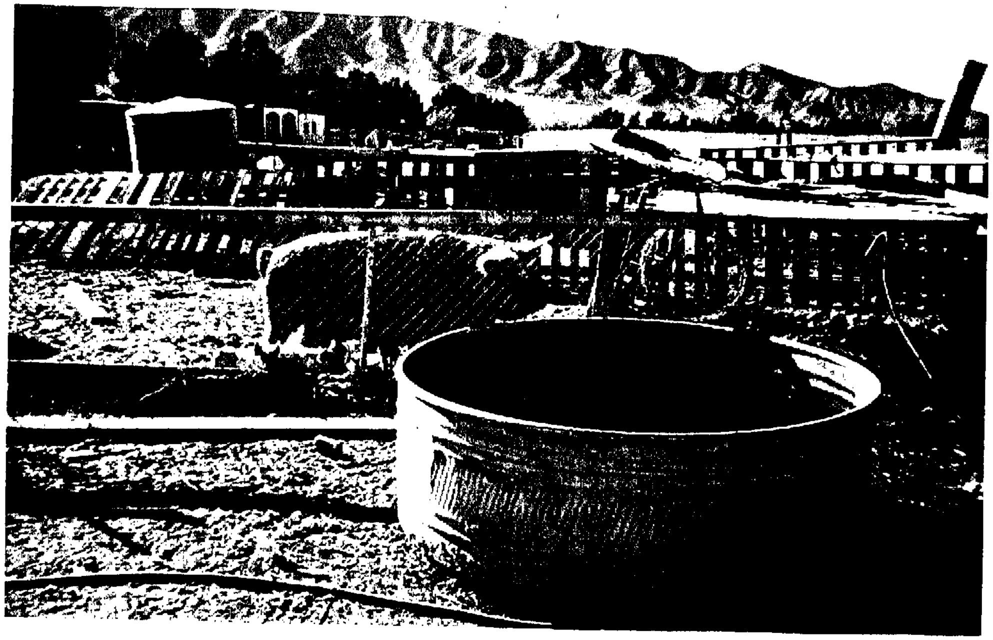
At least two unusual farming activities currently exist in Amargosa Valley. The first is a catfish farm, where catfish are raised in tanks filled with locally pumped ground water. This facility is currently a small operation that sells its products principally to local residents. Another small prototype operation raises ostriches. Currently, this facility is in its startup mode where ostriches are raised for the purpose of building a breeding colony. Once established, the ostriches will be marketed for meat.

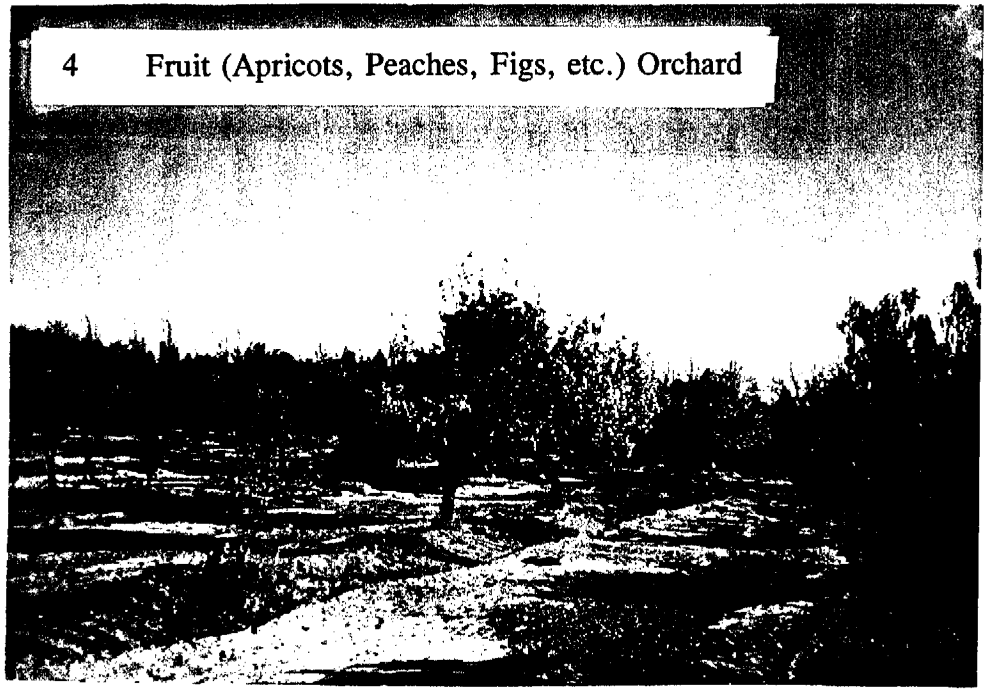
Water resources of Amargosa Valley are currently over-allocated. Although only 26,000 acre-feet of water per year have been apportioned, standing water permits would allow the removal of 43,000 acre-feet. State officials, however, acknowledge that only a fraction of the allocated water rights is being used, and a review is currently underway to rescind permits in cases where water has not been used in the past five years. A review of county documents (NYE93a, NYE93c, NYE93d, NYE93e) suggests that water usage declined from 9,672 acre-feet in 1985 to 4,109 acre-feet in 1990, with nearly all water used for irrigation.


1 Pistachio Orchard


2 "Large" Bales of Alfalfa

3 Pig, Sheep, and Duck Farm




4 Fruit (Apricots, Peaches, Figs, etc.) Orchard

8.4 DESCRIPTION OF THE HUMAN INTRUSION SCENARIO

On the human intrusion issue, the National Research Council (NAS95) in its Technical Basis for Yucca Mountain Standards, reached the following findings:

- There is no technical basis for predicting either the nature or the frequency of occurrence of intrusions.
- There is no scientific basis for making projections over the long term of either the social, institutional, or technological status of future societies.
- There is no scientific basis from which to project the durability of government institutions over the period of interest, which exceeds that of all recorded human history.
- Some degree of continuity of institutions, and hence of the potential for active institutional controls, into the future might be expected; but there is no basis in experience for such an assumption beyond a time scale of centuries. There is no scientific basis for assuming the long-term effectiveness of active institutional controls to protect human intrusion.
- There is no technical basis for making forecasts about the reliability of passive institutional controls.
- There is no scientific basis for estimating the probability of inadvertent, willful, or malicious human action.
- It is not feasible to make meaningful predictions about the probability of advertent or inadvertent intrusion.

Based on these findings, the NAS made the following observations:

- Although it can not be proven, it is believed that a collection of prescriptive requirements, including active institutional controls, record keeping, and passive barriers and markers, will help to reduce the risk of human intrusion, at least in the near term. The degree of benefit is likely to decrease over time.
- Because it is not technically feasible to assess the probability of human intrusion into a repository over the long term, it is not scientifically justified to incorporate alternative scenarios of human intrusion into a fully risk-based compliance assessment that requires knowledge of the character and frequency of various intrusion scenarios. However, it

is possible to carry out calculations of the consequences for particular types of intrusion events. Such calculations might be informative in the sense that they can provide useful insight into the degree to which the ability of a repository to protect public health would be degraded by intrusion.

To address the human intrusion issue on an adequate basis, the NAS made the following recommendations:

- The repository developer should be required to provide a reasonable system of active and passive controls to reduce the risk of intrusion in the near term.
- EPA should specify in its standard a typical intrusion scenario to be analyzed for its consequence to the repository performance.

This section of the BID presents background information relevant to human intrusion scenarios that can be developed for the Yucca Mountain Repository Site. The assumptions made about the intrusion scenarios follow the guidelines provided by the NAS. This discussion is organized as follows:

- the potential causes of intrusion were first discussed by summarizing the current resource potential in the vicinity of Yucca Mountain and how that may influence future intrusion;
- possible intrusion scenarios are then presented for each of the resources that are likely to occur in the vicinity of Yucca Mountain; and
- the assumptions that apply to each scenario and the parameters that would be used to calculate the consequences of each intrusion scenario are then provided.

8.4.1 Site Resources as Potential Cause for Intrusion

Extensive exploration, development, and mining have occurred in the Great Basin of Nevada, and extensive histories and lists of mineral deposits are available. Predicting future economic conditions and what materials in the vicinity of the site may be considered resources, or how they may be explored or produced is not feasible. However, the consequences of an intrusion scenario based on exploration and/or production of a present day resource can be evaluated using current methods and technologies, assuming similar types of intrusion in the future. The information on current and historic natural resources in the vicinity of Yucca Mountain is presented to establish plausible background data for scenarios based on current resource exploration and/or production.

The discussion of mineral resources, based on information from Miklas and Fiero's reports (MIK92; FIE86), is focused on oil and natural gas, geothermal resources, and metallic ores. Other mineral resources that are or may be present at or in the vicinity of the Yucca Mountain site, such as gravel, building stone, and pumice, are excluded from this discussion. Such minerals are abundantly present in other parts of the region. Moreover, since they have a low bulk value, they can only be profitably extracted from large scale, shallow surface workings. It is unlikely that surface operations could impact the proposed repository.

8.4.1.1 Petroleum and Natural Gas Resources

The Great Basin of Nevada, in which Yucca Mountain is located, has the potential for petroleum deposits. This is because excellent reservoir rocks and structures (faults and folds) exist and source beds (rocks in which petroleum might have formed) are also present. However the complexity of the geology, due to deformation resulting from tectonic forces, makes exploration difficult and costly. Also, the potential size of the reservoirs is limited due to the high degree of faulting in the region. This is evidenced by the nature of the reservoirs currently being produced in the immediate vicinity of Yucca Mountain.

Oil and natural gas have been produced in Nye County (Railroad Valley) and Eureka County (Pine Valley). Both sites are about 100 to 300 km northeast of the Yucca Mountain site. The fields are relatively small, and production is on the order of several hundred to a few thousand barrels per day. Given that all production to date has come from Tertiary basins in the Sevier orogenic belt between the Devonian/Mississippian Antler highland and the Paleozoic continental shelf, the potential for petroleum resources in the vicinity of Yucca Mountain is rated as low.

8.4.1.2 Geothermal Resources

In general, the Basin and Range Province, which contains the Great Basin subprovince, is an area of elevated heat flow (about 2 heat flow units [HFU]) relative to other continental settings (about 1 HFU). This is believed to be due to the thin crust and near melting conditions at the crust/mantle boundary. In Nevada alone, there are nearly 300 thermal springs and warm water wells. However, the hot spring activity is concentrated in the west-central and north-central parts of the state. Yucca Mountain is located in an area of moderately elevated heat flow (1.5 to 2.5 HFU). The Eureka heat flow, on the order of 0.75 to 1.5 HFU, is immediately to the north of Yucca Mountain and is thought to be below the average heat-flow values for the region due to underflow of intrabasinal ground water. A geothermal test well drilled on Pahute Mesa, approximately 40 km north of Yucca Mountain, found maximum water temperatures on the order of 60 to 90°C, well below current geothermal resource values. Warmer temperatures (125°C) were found at a depth of 12,140 feet (3,700 m); however, this is below the depth considered to be economical for low temperature geothermal production, which is 1 km .

8.4.1.3 Mineral Resources

Disseminated Gold/Silver and Uranium Deposits

Disseminated gold/silver deposits have fueled the Nevada precious metals boom over the past 15 years. These are low-grade deposits (0.01 to 0.1 oz/ton cutoff grade) worked by open pit operations involving minimal milling and cyanide leaching technologies. Base metals are generally low in these deposits, while mercury, arsenic, thallium, and antimony are elevated. In the Great Basin subprovince, these deposits occur in both sedimentary and volcanic host rocks.

Sedimentary rock that hosts gold/silver deposits is located predominately in the northern and western portions of the Great Basin, and is primarily between the Sierra Nevada mountains to the west and the Paleozoic eastern assemblage of the continental shelf. In addition to clustered deposits in sedimentary rock, the Carlin, Getchall, and Cortez metallogenic trends are recognized. Host rock is variable, ranging from calcareous through clastic sedimentary rocks, with some preference for argillaceous or carbonaceous carbonates. The Roberts Mountain thrust of the Antler orogeny, north of the Yucca Mountain site, marks a fairly sharp boundary between gold-bearing deposits northwest of the thrust and barren mineral deposits southeast of the thrust.

While relatively few of the disseminated gold/silver deposits in Nevada are hosted in volcanic rock, those that are have been associated with a magnetic anomaly along the Walker Lane Belt that includes Yucca Mountain. However, the host rock is generally Tertiary andesites, silicic tuffs, and volcaniclastic sedimentary rocks, none of which have been found in the vicinity of Yucca Mountain.

Most of the uranium production in Nevada has been from disseminated tertiary deposits and associated veins at the Apex mine in Lander county north of Yucca Mountain. Volcanic hosted uranium in sub-economic concentrations is found in silicic volcanics at the McDermitt Caldera in the northwest corner of Nevada.

Hot Springs Deposits of Gold/Silver and Mercury

Hot spring deposits of gold/silver and mercury in the Great Basin are generally young (1 to 3 million years) and associated with silicified rhyolite plugs and/or geothermal systems. Most of the hot spring deposits in the subprovince are in the Walker Lane Belt.

Porphyry Deposits of Copper, Molybdenum, and Gold/Silver

Calc-alkaline porphyry deposits associated with fossil hydrothermal systems contain many of the richest copper and molybdenum deposits in the Great Basin. While ore grades are generally low (0.5 to 1.0 percent Cu, 0.01 to 0.1 percent Mo), there is typically a high grade enriched cap. Although concentrations of gold and silver are very low, byproduct recovery from the large volumes of ore processed for copper and/or molybdenum is economic.

Porphyry intrusions are typically 0.5 to 3 km in diameter and lie at depths of 1 km. The deposits in the province generally date from 50 to 70 million years, although the deposits at Battle Mountain, Nevada and Bingham, Utah are dated from 35 to 40 million years. The relatively few copper and molybdenum deposits in the Great Basin lie far north of Yucca Mountain. The richest porphyry deposits in the Basin and Range Province are south of the Great Basin in southern Arizona and New Mexico. Given the relatively young age of Yucca Mountain, less than 17 million years, it is unlikely that porphyry deposits are to be found in the area.

Skarn and Carbonate-Hosted Deposits

Skarn and carbonate-hosted deposits in the Great Basin have been exploited for a variety of base and precious metals, including iron, tin, tungsten, copper, zinc, lead, molybdenum, gold, and silver. However, these deposits are largely limited to the northern Great Basin and the Porphyry Copper Block of Arizona/New Mexico to the south.

Epithermal Vein Deposits

In the Great Basin, through-going normal faults and associated fracture sets are clearly correlated with mineralization trends for a variety of metals. Veining is largely controlled by normal and slip-strike faulting resulting from caldera formation, although thrust faulting has also played a role. Thus, vein deposits are generally classified on the basis of the normal and slip-strike fault orientation as high-angle, listric, and detachment.

Many of the historic mining districts in the Great Basin, including Comstock, Bodie, and Tonopah, exploit polymetallic vein deposits. These deposits are usually mined for high-grade gold and silver, but economic concentrations of antimony, lead, zinc, copper, manganese, and uranium have also been developed.

Near Yucca Mountain, gold and silver have been produced from vein deposits in the Wahmonie District (25 km east), Bare Mountain (15 km west), and in the Bullfrog Hills (30 km west).

Breccias (Gold/Silver)

Breccia deposits (in pipes, stockwork fractures, and brecciated fault zones) of gold, silver, and base metals are widely dispersed in the Great Basin. Such deposits have been identified at Paradise Peak, Borealis, Victoria, and Ortiz in Nevada, northwest of the Yucca Mountain site. While such deposits are widespread, it should be noted that they contain only a small fraction of the total reserves of the region.

Massive Sulfide (Copper, Lead, Zinc)

Small deposits are found throughout the southern Basin and Range in Arizona and in the north-central Great Basin at Big Mike and Mountain City in Elko County, Nevada. Of volcanic origin, such deposits are believed to form at tectonic plate margins where seawater circulates near the vents of submarine hydrothermal systems.

Roll-Front (Silver, Uranium)

While roll-front uranium deposits are associated with much of the uranium that has been discovered and mined on the Colorado Plateau, no such deposits have been discovered in the Great Basin subprovince. Indeed, only a single sandstone roll-front deposit of silver (at Silver Reef, Utah) has been discovered in the subprovince.

Placer (Gold, Platinum)

Placer deposits in the Great Basin are fairly common, and, due to limited lateral transport, are almost always found in close proximity to the parent deposit. Exceptions include the Snake River, Idaho and Spring Valley, and Nevada placers, which are not associated with a lode deposit. Gold and platinum placers in Nevada are found north of the Yucca Mountain site at Round Mountain, Battle Mountain, and Manhattan.

8.4.1.4 Other Materials

Other materials found in the Great Basin include barite, manganese, borax, mercury, beryllium, gallium/germanium, zeolites, and fluorspar. Of these, only zeolites and fluorspar are believed to occur in significant deposits near the Yucca Mountain site.

The unique ion exchange and sorptive properties of zeolite minerals find numerous practical applications, including molecular sieves and water softeners. While thick zeolite beds are present in the vicinity of Yucca Mountain, they are found only at great depth. Because of the low value of the resource, economic recovery currently relies on surface-mining techniques.

Fluorspar also has a number of industrial applications, primarily in chemical, ceramic, and metallurgical applications. The largest fluorspar-producing region of Nevada is the Bare Mountains, about 15 km west of Yucca Mountain.

8.4.1.5 Ground Water

Ground water is currently the only source of water in the area, and is used for domestic, agricultural, and industrial purposes. Water for site investigation requirements is obtained from two wells (J-12 and J-13) located approximately 3.5 miles from the proposed repository footprint. Currently, the J-12 and J-13 wells are the closest production wells to Yucca Mountain; there are no production wells situated on Yucca Mountain itself. These wells are completed in the welded Tertiary volcanic rocks (Topopah Spring Member of the Paintbrush Tuff). Wells in the volcanic aquifer range in depth from 850 to 3,500 feet and are capable of producing from 370 to 770 gpm, based on testing performed in 1967 reported in U.S. Geological Survey Water Supply Paper No. 1938. The water table beneath the proposed repository site is located within the Calico Hills and Crater Flat formations. The Crater Flat hosts the lower volcanic flow system, with the Calico Hills acting as an aquitard between the Crater Flat formation and the upper volcanic flow system within the Topopah Spring. Ground water quality in the volcanic aquifers is variable, being a complex function of many factors. The primary factor governing water quality in the volcanic rocks is the residence time of the water within the aquifer. Wells completed near recharge areas are likely to produce better quality water than those completed in areas with a high residence time. The two existing water supply wells, J-12 and J-13, are completed within the recharge area of Fortymile Wash, and produce water of good quality. Water beneath Yucca Mountain is generally found to be older and of poorer quality.

The Tertiary volcanic section beneath Yucca Mountain is underlain by a sequence of Precambrian and Paleozoic clastic and carbonate rocks. The Paleozoic carbonate sections constitute an aquifer of regional extent. Beneath Yucca Mountain the Paleozoic carbonate sequence is not considered a resource at the present because of the presence of a thick aquifer in the overlying volcanic rocks, the relatively great depth to the carbonate rocks, and water quality concerns in the deeper rocks. Geochemical, heat flow and hydraulic head data suggest that the carbonate aquifer system is interconnected with, and recharges the welded tuff aquifer beneath Yucca Mountain.

The configuration of the water table beneath the proposed repository is almost completely insensitive to the topography of the land surface. Accordingly, depth to water is considerably greater under Yucca Mountain than beneath the adjacent valleys. The increased depth to water greatly diminishes the resource value and potential of ground water directly beneath the ridge. In arid climates this fact generally holds for small- and medium-sized mountains or ridges, and would be known to those conducting an exploratory ground water drilling program, even assuming the complete loss of all current knowledge about the site. Thus, the likelihood of a ground water exploratory program selecting a drilling site on top of a ridge is small, since the additional depth to water beneath the ridge would increase drilling costs without any increase in the probability of successful productive wells.

8.4.1.6 Resource Summary

Within a 30-km radius of the Yucca Mountain site, there are six active gold- and silver-producing properties in the Bullfrog and Bare Mountain mining districts to the west of the site. Fluorspar is also produced from the Daisy Mine in the Bare Mountain district. A small amount of mercury (200 flasks) has been produced, at both the Thompson Mine on the north end of Yucca Mountain and at Bare Mountain. Borax is produced in the Amargosa Valley south of the site. Uranium, geothermal, and hydrocarbon resources have not been exploited in the vicinity of Yucca Mountain; however, hydrocarbon exploration (wildcat) wells have been drilled within 20 km of the site. Because of the historic lack of mineral and petroleum resource development at Yucca Mountain, it is likely that future development would not be undertaken. However, the proximity of reported resource development suggests that exploration drilling for minerals and petroleum at Yucca Mountain might occur in the future.

Ground water in the vicinity of Yucca Mountain is currently produced from the Tertiary volcanic welded tuff units. Yields from wells constructed in fractured volcanic rocks are generally high. Ground water is also found within the underlying Paleozoic carbonate unit, but the relatively great depths and poor quality of this water preclude it from being utilized as a resource at present. The resource value of ground water in this area depends on the depth to the water (as reflected in drilling costs) and water quality. Beneath Yucca Mountain the resource value of ground water is considerably lower than in the adjacent valleys, due to the increased depth to water and somewhat poorer ground water quality.

8.4.2 Types of Human Intrusion

The NAS recommends that EPA specify in its standard a typical intrusion scenario to be analyzed for its consequence to repository performance. Selecting an intrusion scenario for analysis entails judgment. To provide for the broadest consideration of what scenario or scenarios might be most appropriate, the NAS recommends that EPA make this determination in its rulemaking to adopt a standard. As a starting point, the NAS suggests a stylized intrusion scenario consisting of one borehole of a specified diameter drilled from the surface through a canister of waste to the underlying aquifer.

To provide the background for selecting a scenario, discussed below are the types of human intrusion scenarios that may be considered based on the resource potential as it is known today. Table 8-16 lists likely scenarios for each type of resource currently found in the vicinity of Yucca Mountain. Following the table are discussions of these likely scenarios.

Table 8-16. Likely Human Intrusion Scenarios for Different Types of Resources

Nature of Human Intrusion	Petroleum or Geothermal	Minerals	Welded-Tuff Aquifer	Carbonate Aquifer
(1) Borehole completed in repository		X		
(2) Borehole completed in welded-tuff aquifer beneath repository			X	
(3) Borehole completed in carbonate aquifer beneath repository and welded-tuff aquifer	X	X		X
(4) Aquifer testing with uncased borehole or test well			X	X
(5) Production well completed and placed into service			X	

8.4.2.1 Petroleum/Geothermal Related Intrusion

Human intrusion resulting from the exploration for petroleum or geothermal resources would be comparable due to the depth at which the resources are expected to be found in the area around Yucca Mountain. Petroleum typically is found in the Paleozoic carbonates, and the geothermal temperatures that make recovery economic are in the Precambrian basement rocks. Both resources are at depths that would require drilling through the repository horizon elevation and the welded-tuff aquifer, and into or through the Paleozoic carbonate aquifer (Scenario No. 3, Table 8-16).

Typically, petroleum and geothermal exploration holes are cased into competent rock (unfractured and minimal porosity) to provide a seal at the surface in the event that high pressure gases or liquids are encountered during drilling. The seal is expected to withstand the pressures anticipated, and is usually a 14- to 30- inch diameter pipe, depending on the largest drill bit expected to be used, set and cemented in the initial borehole that has been advanced into unfractured rock. Once the cement has set, a drill bit slightly smaller than the surface casing, typically 12 to 24 inches in diameter, is lowered to the bottom of the casing and the drill string advanced. Drilling is usually continued in the open hole (no casing) with cuttings or core samples being collected to identify the rock type being penetrated and to evaluate resource content or potential. Excess cuttings are collected in a pit adjacent to the drill rig and the fluid recirculated to flush more cuttings to the surface. If air drilling methods, which are common in geothermal exploration, are used, the fluid (air) from the cutting return stream is discharged directly to the atmosphere above the cuttings pit.

There are two categories of risk that could occur for this scenario. The first category of risk involves potential releases that result from the drill passing through the repository and associated waste, entraining or dissolving radioactive waste products, and carrying waste products to: 1) the surface in the cuttings return; 2) the welded-tuff aquifer, once it is reached, through the drilling fluid circulation path; and 3) to the deeper carbonate aquifer, once it is reached, through the drilling-fluid circulation path. The primary mechanism of contamination of the aquifer would be the circulation of contaminated drilling fluid. Petroleum exploration drilling commonly uses the direct rotary drilling method, which pumps the drilling fluid down the center of the drill pipe, exits the bit, and flushes the cuttings up the annular space between the hole and the drill pipe to the surface. The cuttings are collected on the surface in pits. Direct rotary air drilling methods, used to drill geothermal wells, would also discharge the cuttings to a pit, but would not recirculate contamination as readily because they do not recirculate the returning fluid. In both types of drilling, contamination can also be spread when the drill string is removed from the hole to change bits, test the formation, or abandon the hole.

The second category of risk involves potential releases from the borehole being abandoned and not being sealed. In this instance, material from the breached waste area could fall through the open borehole to the aquifer zones, and be dissolved and transported to the environment. If the borehole fills with water, material from the repository horizon could still sink through the water column or contamination could be circulated by density and/or thermal effects.

8.4.2.2 Mineral Exploration-Related Intrusion

Mineral exploration drilling has the potential to result in more than one scenario, as illustrated in Table 8-16. The basic variation is in the depth drilled, due to the high degree of uncertainty with respect to what might be considered a mineral resource in the future. Completion of an exploratory borehole in the repository horizon (Scenario No. 1, Table 8-16) is considered because it is conceivable that the radioactivity produced by the repository might be detected using remote sensing instruments and prompt further exploration that would require drilling.

Typically, mineral-exploration drilling is performed using relatively small diameter (nominally 3 to 7 inch) drills or coring bits with air or rotary wash. Mineral exploration holes are not cased except when the near surface materials are very unstable, which could preclude keeping the top of the hole open. The potential pathways to the environment are very similar to those discussed for petroleum or geothermal exploration, with the primary difference being the size of the borehole and associated quantity of material potentially removed and circulated. Coring is a frequently used method to provide direct visual identification of the material being penetrated and to permit analyses for the evaluation of ore grade. If coring were done when the drill is penetrating the repository horizon, it would be possible for a sample (as a 1.5- to 5-inch diameter by 30- to 48-inch long cylinder) of the waste material or contaminated materials from the repository to be brought to the surface.

Considering the known occurrences of mineral resources in the vicinity of Yucca Mountain, it is likely that an exploration borehole would be completed in the Paleozoic and older rocks that are beneath the volcanics that contain the repository horizon and the welded-tuff aquifer zone (Scenario No. 3, Table 8-16). Improper abandonment (hole left open with no backfilling) of a borehole like this could create a similar contamination circulation route to that described for an abandoned petroleum or geothermal exploration hole. Again, the difference is in the diameter of the hole, which would, in this case, tend to limit the size of material that could fall to the deeper aquifer zones as well as the amount of density and thermal circulation.

8.4.2.3 Ground Water Resource-Related Intrusion

The intrusion scenarios developed for ground water resources relate to exploration, aquifer testing, and well development and production, and are shown in Table 8-16.

Ground Water Exploration Drilling

Table 8-16 shows the possible borehole scenarios for the welded-tuff aquifer (Scenario No. 2) and carbonate aquifer (Scenario No. 3). The exploration for ground water resources would probably involve a direct rotary-air or water-configured rig using a drill bit or pneumatic hammer on the order of 6 to 8 inches in diameter. In arid regions, like the Yucca mountain area, air drilling is commonly used to minimize the amount of water required. A secondary benefit of drilling with air is, when ground water is reached, an estimation of aquifer yield can be made based on the water flow from the discharge line (bloey pipe) as drilling progresses through the aquifer. The exploration borehole would be advanced as rapidly as possible until the first water is noted in the return air, or an increased flow rate is identified from the annular space if water or drilling mud were used. In this case, where the repository horizon is above the aquifer, the repository materials could be penetrated and circulated to the surface for several minutes and would probably not be noticed until ground water begins to emit from the return line or annulus. Even at this time, it is not likely that the waste material would be recognized unless some type of radiation detector were being used at the drilling site. As described in the section describing petroleum and geothermal scenario, the contamination would be circulated to the surface with the flow up the annular space and, if water were used, returned to the aquifer by the mud pump taking water from the suction pit. If air were used, the recirculated air would not be as contaminated unless the compressor intake were in close proximity to the bloey line, which would be discharging the contaminated return air and cuttings.

Additional releases of the repository waste material and mixing with the aquifer fluids could result from removal of the drill string from the hole and reinserting it (tripping), as is done when drill collars need to be added or the bit must be changed. This is a random action depending on the depth of drilling, bit wear, and rock type, and could be exacerbated by drilling through repository waste containers (creating the need for a bit change). A more common tripping of the drill string is to recover core when a fixed core barrel is used. In this case, the drill string is removed every 30 to 48 inches of drilling, depending on the length of the core barrel, to recover the cored rock. In some instances, a wire-line coring device is used to preclude the necessity of removing the entire drill string from the hole. The core barrel in this case is lowered inside the drill rod, attached to the bit and, once the core barrel is full, pulled to the surface using a wire line on a hoist. Wire-line coring is most frequently used in mineral exploration drilling due to the smaller core diameters (typically less than 2.5 inches) needed primarily for mineral identification.

The case of water resource exploration drilling with an improperly abandoned borehole is similar to petroleum, geothermal, and mineral exploration, and is discussed in Sections 8.4.1.1 and 8.4.1.2.

Aquifer Testing

Aquifer testing scenarios are shown in Table 8-16 (Scenario No. 4) for the welded-tuff and carbonate aquifers. Aquifer testing could be performed during drilling in the uncased borehole, which is typically done when drilling is performed using an air rotary rig. For more extensive testing, aquifer testing would be performed in a well constructed in the exploratory borehole.

Testing in the open (uncased) borehole is referred to as drill-stem testing. It is performed using the air flow from the compressor(s) to lift the water from the aquifer zone, by inserting the drill pipe near the bottom of the hole and injecting air, causing the fluid column to rise to the surface and/or entraining the water in the air stream to remove it from the borehole. This method creates a scouring action in the open borehole due to up-hole air/water mixtures reaching velocities on the order of 3,000 ft/min. A fluid stream moving this fast would produce erosion in the repository zone penetrated, increasing the material carried to the surface or falling into the borehole. Testing in this manner is usually of shorter duration (several minutes to a few hours) than aquifer tests performed in cased holes due to the potential erosion of the borehole walls as well as the estimating nature of the test. In some cases, an air-lift pumping system (a pipe or drill rod with an air line suspended inside to beneath the water table but not to the end of the pipe) can be lowered into the open hole and used to test the flow. This provides a more accurate flow measurement and eliminates the erosion on the borehole wall above the water table because the water flows inside the pipe to the surface.

The second testing-related scenario utilizes a well constructed in the exploratory borehole for testing the potential yield and evaluating the storativity of the aquifer. Aquifer depths in the vicinity of Yucca Mountain are currently in excess of 800 feet (244 m) and would require a well of at least 12 inches in diameter to permit setting a pump that would be capable of testing the aquifer adequately. Exploratory boreholes are not typically drilled large enough to facilitate constructing a well of that diameter; therefore conducting a test would require reaming the exploration borehole to a larger size. The reaming of the borehole would remove more material from the breached repository area and allow it to circulate, dissolve, or slough into the borehole. Once the reaming is completed, a casing with well screen in the aquifer zone would be set in the reamed borehole, gravel packed in the aquifer zone, a cement plug placed on top of the gravel pack, a bentonite slurry placed around the casing to the surface, and a cement plug placed around the upper few feet of casing to form a surface seal. This well would be constructed in the same manner as a production well, which is discussed in the next scenario. Testing would be performed by placing a pump in the screened zone of the casing and varying the pumping rate to evaluate the aquifer parameters and, after an optimum rate is selected, pumping the aquifer at that rate for several hours or days. During testing, the only release of radioactive contaminants from repository materials to the aquifer would be prior to or during well construction. After the well is constructed, the breached repository horizon would have been cased with solid pipe and isolated from the fluid stream.

Ground Water Production

As shown in Table 8-16, Scenario No. 5, the ground water production scenario is identified only with the welded-tuff aquifer because of the depth and reported poor water quality of the carbonate aquifer. In the event the production of the carbonate aquifer were to be considered, the scenario would only differ from that for the welded-tuff aquifer due to the depth and would mainly influence circulation time.

The construction of a production well is similar to the process described above for test-well construction, except that a production well is of larger diameter. A reverse rotary drill rig may be used for drilling production wells. In reverse rotary drilling, the fluid flows into the annular space at the surface and maintains a static head of water in the hole. The mud pump draws a suction on the drill rod and the fluid is pumped from the drill rod to the mud pit, where the cuttings settle out and the fluid flows back into the hole. In this scenario, the release to the environment occurs during the drilling of the well as contaminated fluid is circulated from and later past the repository horizon and into the mud pit and subsequently into the aquifer zone. Once the well is constructed, the repository horizon is sealed off and the primary source of contamination would be residual fluids in the well and aquifer.

After well construction is completed, the casing is pumped and surged to remove residual drilling fluids from the aquifer and to develop the gravel pack. This will remove the residual fluids from the casing and flush the aquifer. Well testing may be conducted to confirm the well performance prior to placing it into production. This will also flush the aquifer zone, with the fluids from all testing typically being discharged to a natural drainage feature, the mud pit, or the ground surface in the vicinity of the well. After all testing is completed, the well is connected to the distribution system and placed in service. Well water could be a sole source supply for a commercial application or combined with several other wells in a large facility or municipal supply system.

8.4.3 Parameters and Assumptions Associated with Ground Water Withdrawal

The potential release associated with ground water withdrawal results from the contamination of the aquifer being pumped. The aquifer considered for ground water withdrawal in the vicinity of Yucca Mountain is the welded-tuff aquifer. The primary parameters necessary to assess the consequence of intrusion are the aquifer pumping rate, the duration of pumping, aquifer properties, the degree to which the aquifer has been contaminated, and the nature of the contaminants.

Production wells are typically of large diameter (16 to 36 inches) to facilitate multistage turbine pumps that can lift water from the aquifer zone at the optimum flow rate, which can range from 500 to 1,500 gpm. For example, the intrusion scenario used by Sandia National Laboratories in TSPA-93 (DOE94a) assumed a production well intersected the repository and the well was drilled using a 24-in (0.6-m) bit.

Pumping rates ranging from 300 to 700 gpm were used on tests at the Nevada Test Site performed by the USGS (USG72), during which the welded-tuff aquifer was pumped at 370 gpm for four days at one well location and 697 gpm for four days at another well.

The assumed pumping duration for a production well would be based on how many gallons per day would be necessary to supply the user. This value would determine the duty cycle of the pump required. For example, pumping a well at 770 gpm continuously would produce 1 million gallons per day. In a production or test well, the screened zone would be the only source of contamination because the repository horizon above the aquifer would be cased with pipe to facilitate transporting water to the surface. This would require assumptions to be made for the well-drilling scenarios (air or water) to assess the amount and nature of residual contamination in the aquifer zone available to leach into extracted water if the well were eventually used as a supply. The properties of significance would be the nature of the permeability (i.e., primary porosity or fractures), physical and chemical properties of the welded-tuff aquifer (i.e., adsorption and redox potential), and the secondary mineralization and its influence on radionuclide transport and solutioning. Of equal importance are the assumed contaminants that have been introduced into the aquifer either as solids or in solution.

8.4.4 Parameters and Assumptions Associated with Human Intrusion

There are two broad categories of risk from radioactive material that could occur as a result of an intrusion into the repository of the type characterized by borehole scenarios. These are:

- Risks from materials brought directly to the surface by the intrusive activity
- Risks that arise from improper abandonment of an exploratory borehole that would compromise the integrity of the repository's engineered or geologic barriers

The radioactive materials brought directly to the surface by the intrusive activity would pose hazards to the intruders themselves and to the public. The NAS (NAS95) concluded that analyzing these risks is unlikely to provide useful information about a specific repository site or design and, therefore, should not provide a basis for judging the resilience of the proposed repository to intrusion. Accordingly, the NAS recommends that these risks not be considered in the compliance analysis. For these reasons, discussions of parameters and assumptions associated with these scenarios are not presented in this BID.

The consequences of abandoning a borehole that had intersected repository waste without plugging it with impermeable material would be long term. The importance of this scenario, as suggested by the NAS (NAS95), would be related to the creation of enhanced pathways to the environment (both to the atmosphere and to the aquifer), and not associated with the breaching of the canister, which will happen eventually even without human intrusion.

8.4.4.1 Factors of Consideration

To evaluate the human intrusion scenarios, the following factors or parameters must be evaluated and the assumptions associated with these factors must be made.

Institutional Controls

According to the NAS report, there is no scientific basis for making projections over the long term of either the social, institutional, or technological status of future societies. There is no scientific basis from which to project the durability of government institutions over the period of interest, which exceeds that of all recorded human history. On this time scale, human institutions have come and gone. Some degree of continuity of institutions, and hence of the potential for active institutional controls, into the future might be expected; but there is no basis in experience for such an assumption beyond a time scale of centuries. Similarly, there is no scientific basis for assuming the long-term effectiveness of active institutional controls to protect against human intrusion.

Furthermore, according to the NAS report, there is no scientific basis for making forecasts about the reliability of passive institutional controls. The likelihood that markers or barriers would persist, be understood, and deter intrusion cannot be assessed from a technical base.

Drill Depth and Hole Size

As noted by the NAS (NAS95) report, it is not feasible to predict which natural resources will be discovered or will become valuable enough to be the objective of an intruder's activity or to predict the characteristics of future technologies for resource exploration and extraction. These would affect the assumptions of drill depth and hole size.

Based on current practice, as discussed in Section 8.4.2, typical diameters of exploration boreholes and depths of penetration are:

Types of Exploration	Hole Size (inch)	Drill Depth
Petroleum/Geothermal	12 - 24	carbonate aquifer
Mineral	3 - 7	carbonate aquifer
Ground Water	6 - 8	welded-tuff or carbonate aquifer

Number of Boreholes and Borehole Location

Generally, resource exploration utilizes remote sensing, topographic, and geologic information to select drilling locations. However, when investigating a broad area like the region including Yucca Mountain, the spacing of exploration boreholes will vary for the various types of resources. Petroleum and geothermal resource exploration is performed to detect regional or structural trends that can extend for tens or hundreds of miles and thus exploration drilling typically involves a single hole in a region or within a geologic structural trend. Mineral exploration is carried out in an orderly manner, usually employing a grid. The initial grid size, when regional resources are being evaluated instead of localized vein-type deposits, may be a mile or more on center for boreholes. The grid spacing is decreased only if economic levels of target minerals are detected, which is not expected to be the case in the immediate vicinity of Yucca Mountain. Borehole locations could be on mountain top or in the low areas.

Exploration for ground water resources can be focused based on surface features or convenience to a user and, in such case, the exploration wells are typically clustered or linearly spaced a mile or more apart. For regional investigations of ground water resource potential, randomly and widely spaced boreholes are commonly used. In such case, a density of 1 well per 2,000 km^2 is reasonable, which provides adequate information on the nature and presence of a ground water resource. In the Yucca Mountain area, the most likely locations for ground water exploration would be the drainage basins that surround Yucca Mountain.

In terms of the number of boreholes to be assumed in the scenario, the NAS report suggests a stylized intrusion scenario consisting of only one borehole. The reason is that a single borehole scenario holds the promise of providing considerable insight into repository performance with the minimum complication. Under many conditions, the effect of multiple borehole presumably would be the sum of the effects of each taken separately, but circumstances when this assumption is invalid can also be conceived. Because construction of the scenario is arbitrary, the NAS report argues for the simplest case that tests the repository.

In determining the location of the borehole, the stylized single borehole scenario suggested by the NAS report postulates drilling from the surface through a canister of waste to the underlying aquifer. The emphasis would be on the creation of enhanced pathways to the environment as opposed to emphasis on breaching the canister, which will happen eventually even without human intrusion.

Borehole Sealing

According to the NAS report, the characteristics of future technologies for resource exploration and extraction, and whether future practice will include sealing of physical intrusions such as boreholes cannot be predicted.

A common practice in current exploration drilling is to leave the borehole open and allow it to backfill naturally or assume the mud-drilling fluid will act as a sealant. For air rotary drilling, there is no drilling fluid filler. For mud rotary drilling, the mud drilling fluid may lose its effectiveness as a sealant if the mud shrinks excessively as it dehydrates.

In an open abandoned borehole, the most likely materials to cause natural backfilling are the loose granular surface materials, or friable or loose tuffaceous formations. The only way that the tuffaceous material could be loosened to fall into the open borehole would be by erosion (running water), mechanical impact (scraping, etc.), or shock (seismic waves). If loose surficial materials were washed or ran into the open hole, backfilling of an abandoned borehole could take place relatively quickly. On the other hand, if the loose surface materials or materials from the borehole wall were too large to fall freely, they could plug or bridge (stick together) in the borehole. In such case, the top of the hole could be plugged, precluding backfilling.

Time of Drilling

According to the NAS report, the predictions for how long into the future institutional controls might survive and remain effective are arguable. The probability that an intrusion would occur in a given future time period, such as in any one year, cannot be assessed from a technical base.

Detection of Repository

The issue is whether drilling would penetrate a waste canister and whether a future intrusion would detect the existence of a repository and take remedial actions.

Two drilling companies were contacted concerning the likelihood of penetrating an intact waste canister with a drill being used in a conventional drilling operation. Mr. Leroy Jochum

(VIC96) stated that, irrespective of bit type (carbide, diamond, etc.) the drill would not penetrate the canister but would most likely be deflected. If the driller wanted to penetrate the canister, tools could be fabricated to cut the steel, but deliberate effort would be needed and it would take a long time.

Mr. John Horton (LAY96) also indicated that special effort would be required to penetrate the canister. It would require a concerted effort by the driller, possibly involving modification of the bit and a considerable amount of time. He mentioned laser/plasma drilling technology that is being developed by companies involved with DOE's Hanford site in Washington State, and stated that future technology might be able to penetrate a waste canister if adequate energy could be applied to the drilling face.

These professional opinions indicate that present-day drilling technology or future technologies could only penetrate a waste canister if the driller was dedicated to doing so. Conventional drilling methods, without special tools or spending an inordinate amount of time and effort, would not be able to breach the canister.

Also, it is conceivable that the radioactivity produced by the repository could be detected using remote sensing instruments and prompt further investigation.

In summarizing this issue, the NAS report concluded that the probability that a future intrusion would be detected and remediated, either when it occurs or later, cannot be assessed from a technical base.

Mechanism for Waste Reaching the Aquifer

In addition to the assumptions regarding the borehole, the nature of the waste intercepted and the circulation mechanism (fluid solutioning or material falling from the repository horizon to the aquifer) must be assumed in order to assess the source term.

8.4.4.2 Scenario Examples

The NAS report suggests a stylized intrusion scenario consisting of one borehole of a specified diameter drilled from the surface through a canister of waste to the underlying aquifer.

Examples of such a scenario are described below.

Example 1

An example of a scenario was provided by the NAS report that would postulate current drilling technology but assume sloppy practice, such as not plugging the hole carefully when abandoning it, after which natural process would gradually modify the hole. It is assumed that the intrusion occurs during a period when some of the canisters will have failed but the released materials would not otherwise have had time to reach the ground water. In this example, the original hole size, the modification of hole size by natural processes, and the mechanism and processes for waste to reach the aquifer must be assumed or analyzed.

Example 2

Another example is a hypothetical, non-mechanistic scenario that postulates that the entire content of a single waste canister is emptied through the abandoned borehole into the aquifer. In this example, evaluation of drilling technology, drill size, modification of hole size by natural processes, and the mechanism and processes for waste to reach the aquifer is unnecessary.

The location of the assumed borehole, in relation to the boundary of repository footprint and the location of a critical group, is a very important factor for both examples. This is discussed in Section 8.4.4.3.

8.4.4.3 Consequence Analysis

Having defined the reference scenario, the principal questions are what consequence should be assessed and how the result should be interpreted.

According to the NAS report, the consideration of human intrusion cannot be integrated into a fully risk-based standard because the results of any analysis of increased risk as a consequence of intrusion events would be driven mainly by unknowable factors. The numerical value of the risk of adverse health effects due to intrusion is the product of the frequency of an intrusion scenario and the measure of consequences. However, the frequency of an intrusion scenario in the distant future is indeterminate.

The NAS report recommends that the Yucca Mountain standard require a consequence-only analysis without attempting to determine an associated probability for the analyzed scenario. The calculations of consequences would provide useful information about how well a repository might perform after an intrusion occurs. The key performance issue is whether the repository would continue to be able to isolate wastes from the biosphere, or if its performance would be substantially degraded as a consequence of an intrusion of the type postulated.

According to the NAS report, the performance of the disturbed repository should be assessed using the same analytical methods and assumptions, including those about the biosphere and critical groups, used in the assessment of the performance for the undisturbed case. This analysis should be carried out to determine how the hypothesized intrusion event affects the risk to the appropriate critical groups. The results of this calculation, however, constitute a conditional risk, that is, a risk assuming that the hypothetical intrusion occurs.

Because the probability is inherently unknowable, the most useful purpose of this type of analysis is to identify the incremental consequences resulting from the assumed scenario. Since human intrusion of some type might be likely at some time in the future, a repository should be resilient to at least modest inadvertent intrusions. In other words, a repository that is suitable for safe, long-term disposal should be able to continue to provide acceptable waste isolation after some type of intrusion.

The NAS report recommends that EPA should require that the conditional risk as a result of the assumed intrusion scenario should be no greater than the risk levels that would be acceptable for the undisturbed repository case. It is further recommended that the analysis not include risks to the intruder or those arising from the material brought directly to the surface as a consequence of the intrusion.

The single borehole scenario should not be interpreted as an estimate of the likely form or frequency of intrusion. It should not be interpreted as either an optimistic or pessimistic estimate of what might actually occur, because there might be no boreholes that intercept canisters, or there might be more than one. The simplest scenario that provides a measure of the ability of the repository to isolate waste and thereby protect the public health is the most appropriate scenario to use for this purpose.

There are three categories of future human intrusion events:

- Inadvertent intrusion where the intruder does not recognize that a hazardous situation has been created. This category has been the focus of discussion in the context of standard-setting and licensing.

- Inadvertent intrusion where the intruder recognizes that a radioactive waste repository has been disrupted and takes corrective actions. On the assumption that the corrective measures taken are effective and the repository is sealed, this category is not of concern. If, however, corrective actions are not taken or are ineffective, this type of intrusion is operationally the same as the above category.

- Intentional intrusion for either beneficial or malicious purposes. The NAS report considers it presumptuous to try to protect against the risks arising from the conscious activities of future human activities. However, given the potential energy value of the wastes intended for Yucca Mountain, this category of intrusion scenarios might be likely.

There are two broad categories of risk resulting from an intrusion into the repository of the type characterized by borehole scenarios. These categories are:

- Risks that arise from improper abandonment of an exploratory borehole, which would compromise the integrity of the repository's engineered or geologic barriers, either from the ground water pathway or from the air pathway.

- Risks from materials brought directly to the surface by the intrusive activity, either to the intruders themselves or to the general public.

Ground Water Pathway from Abandoned Borehole

The consequences of abandoning a borehole that has intersected repository waste without plugging it with impermeable material would be long term. The emphasis of this scenario, as suggested by the NAS report, would be on the creation of enhanced pathways to the environment, as opposed to the breaching of the canister, which will happen eventually even without human intrusion.

The NAS report recommends defining a region (exclusion zone) in which human activities are to be regarded as intrusion and to exclude that region from calculation of the undisturbed repository performance. The NAS report further recommends that the exclusion zone be within the repository footprint area.

It should be noted that the location of the assumed borehole is a very important factor. The closer the borehole location is in relation to the boundary of the repository footprint and the location of the critical group, the less time would be required for radioactive materials to travel to the critical group. A separate exposure scenario, with a different critical group, would be required for evaluation of the ground water pathway from an abandoned borehole, as compared to the evaluation of the undisturbed repository performance.

Air Pathway from Abandoned Borehole

In addition to the ground water pathway, an uncapped, abandoned borehole that penetrates into the repository could provide an air path for waste materials released to the atmosphere. The radionuclide of primary concern for this air release pathway is carbon-14 (^{14}C). The travel time for gaseous releases would depend on the failed waste canister's location in relation to the abandoned borehole, and the manner in which the repository's openings have been backfilled.

The maximum exposures are expected to occur from ^{14}C released from waste canisters that fail early and have short travel times to the surface, giving little time for radiological decay. Because the undisturbed travel time through the unsaturated zone to the surface is already short, an uncapped, abandoned borehole would have little effect on the maximum exposure. Those canisters that fail later, and on which the uncapped, abandoned borehole would have a greater effect, have more time for radiological decay and are expected to have lower exposures from the air pathway than the canisters which fail early.

In terms of ^{14}C release flux at the surface, while the release flux at the uncapped, abandoned borehole itself may be increased, the overall release flux would be unaffected by the borehole, and any changes in potential exposures would be highly dependent on the assumed location of the exposure individual relative to the borehole.

For these reasons, it is concluded that the air pathway need not be considered as a measure of repository performance in evaluation of human intrusion scenarios.

Waste Materials Brought to the Surface by Human Action

The radioactive materials brought directly to the surface by the intrusive activity would pose hazards to the intruders themselves and to the public. According to the NAS report, whenever highly dangerous materials are gathered into one location and an intruder inadvertently breaks in, that intruder runs an inevitable risk. This is not unique to a particular geologic repository, and all geologic repositories have this feature. In particular, for inadvertent human intrusion, it would not be feasible to take regulatory actions today to protect the intruders themselves against the risk of their actions, except that requirements associated with active or passive institutional controls might be helpful in this regard.

It is possible that an inadvertent intruder would not recognize or would irresponsibly ignore the hazard and would leave the cuttings on the surface so that further exposure would occur. However, the amount of such future cuttings might not be very different from one repository site or design to another, especially given the unknown nature of an intrusion. Analysis of this hazard, therefore, also does not provide information that is useful for judging the ability of the particular repository site and design to protect the public. It is also not feasible to take regulatory actions today to minimize these risks.

The DOE containment and isolation strategy defines three post-emplacement time periods: the containment period, in which the waste canisters remain intact; the transition period, during which canister failure and waste mobilization are gradually increasing; and the peak dose period, in which the canisters have failed and seepage of water into the repository is mobilizing the waste radioactivity and transporting it to the environment. The physical condition of the repository will change throughout these time periods and affect the circumstances of an intrusion scenario, as outlined below.

Intrusion During the Containment Period

During the containment period, the waste containers remain intact. An intrusion of the repository by drilling will either intercept an intact container or miss completely. If there is no interception of a container, there will be no evidence to the drillers that a waste repository has been penetrated. If an intact container is intercepted, it is unlikely that the drill bit will be able to penetrate the container easily. Advance of the drill bit will be stopped or severely slowed, leading to investigation of the cause for the resistance. If drilling persists, metal will be evident in the cuttings and it will be evident that something unusual has occurred. Drilling may continue as part of an investigation of the circumstances encountered, in which case portions of the container and the waste form intercepted by the drill bit will be brought to the surface, or the drilling effort at that location may be abandoned.

Intrusion During the Transition Period

During the transition period, containment degradation is occurring as a result of corrosion of the container and waste form caused by infiltration of water to the repository. A drilling intrusion of the repository might encounter an intact container, a partially degraded container, materials between the containers that contain no radioactivity and give no evidence of the existence of the repository, or materials between containers that contain radioactive waste material that has been mobilized and has migrated some distance from the emplacement location.

This latter type of encounter would give no indication of the existence of the repository unless the drilling cuttings were being monitored for radioactivity. An encounter with an intact or partially-degraded container would produce circumstances such as described above for the containment period, i.e., an effect on drilling progress, metal in the cuttings, and investigation of the situation or abandonment of the drill hole.

Intrusion During the Peak Dose Period

In the peak dose period, it can be assumed that all containers have failed and all metals have oxidized. The repository conditions will be similar to that of an ore body, with pockets of radioactive materials at locations where containers used to be, and waste radioactivity throughout the repository. Depending on the extent of lateral migration and dispersion of mobilized waste radioactivity, there may still be areas between the emplacement locations where no radioactivity is present.

Under these circumstances, there may never be any evidence to a drilling operation that a radioactive waste repository was penetrated. Evidence might be available if cuttings are being monitored for radioactivity or if it is noticed that some of the cutting materials composed of oxides of waste package materials are unusual. If neither of these signals of something unusual is received, drilling operations will proceed as planned.

In summary, the consequences of intrusion, as an incremental effect on expected repository performance, will depend on when the intrusion is assumed to occur. If, for example, intrusion is assumed to occur late in the containment period, the effect on expected waste isolation performance could be relatively large because no releases are expected otherwise to occur. However, the risks (probabilities and consequences) associated with such a scenario can be extremely small. If the intrusion is assumed to occur in the peak dose period as defined by DOE's waste isolation strategy, the same scenario would indicate negligible impact on undisturbed repository performance because nuclide release is already significant as a result of ongoing degradation, release, and transport processes.

8.5 EXPOSURE AND RISK ANALYSIS PER UNIT CONCENTRATION OF CONTAMINATED GROUND WATER

The primary source of potential radiation exposure to the public from the proposed geologic repository at Yucca Mountain is by the ground water pathway. Water present in surface soil that reaches the waste containers may in time corrode the waste package to the point of failure, dissolve certain radionuclides from the waste form, and transport them to the underlying aquifer. Radionuclides can enter the environment accessible to humans when water drawn from wells that access the contaminated aquifer is used for drinking and agricultural purposes.

In the near future, it will be necessary to evaluate the overall capability of the repository to meet the performance objectives specified in the applicable regulatory standard. This comprehensive evaluation, or total system performance assessment (TSPA), will require the explicit quantification of all processes that may lead to human exposure and acknowledgment of the uncertainty in the models employed.

The TSPA is a sequential multi-step process that brings together all relevant components of the waste containment and isolation system that potentially affect the release of radionuclides to the accessible environment, and the corresponding concentrations and doses associated with the release. The five major steps are summarized briefly below.

Waste Package Degradation. Given that the waste package must be breached in order for a release to occur, an important first step in TSPA is the modeling of waste package degradation by aqueous corrosion processes. In total, five individual corrosion models have been proposed that are based on engineering designs and near-field environmental conditions. These models provide, as output, the "failure" time for each waste package and the time for the initiation of waste-form alteration.

EBS Release. The repository near-field conditions also affect the dissolution of the radionuclides and their diffusion transport outside the engineered barrier system (EBS) into the geologic environs of the repository. Waste-form and radionuclide dissolution rates have been derived from experimental data obtained under a range of simulated environmental conditions. Three conceptual models have been proposed for use. These models incorporate advective and diffusive releases of select radionuclides from the engineered barrier systems of the repository to the adjacent host rock.

Unsaturated Zone Transport. Those radionuclides released from the EBS must now be modeled for transport through the unsaturated zone to the underlying aquifer. The rate of transport is complicated, however, because the radionuclide may exist either in solution or associated with solid particles (i.e., adsorbed) as defined by the distribution coefficient or K_d. Because K_d is strongly influenced by the characteristics of the aqueous composition and those of the absorbing surfaces, credible transport models for the Yucca Mountain Site must rely on K_d values that are empirically derived and therefore site-specific. Another modeling uncertainty for the Yucca Mountain Site is the possible presence of fracture flow and transport, which may lead to the formation of so-called "fast" paths.

Saturated Zone Transport. Radionuclides reaching the saturated zone (i.e., the aquifer) will invariably undergo dilution through diffusion mixing and saturated zone flow. Since the aquifer is the principal pathway for radionuclides to reach the accessible environment, the saturated zone's greatest impact lies in its potential dilution effect. Major factors and uncertainties affecting saturated zone transport (and dilution) involve aquifer recharge rate, anticipated water withdrawal rate, and future climate.

Biosphere Dose and Risk Modeling. Last, to proceed from probabilistic distribution of radionuclides in ground water to estimates of dose and risk to a specified critical group requires a detailed scenario model. The human exposure model must account for all significant pathways, provide numerical values for all relevant parameters, identify the uncertainty of parameter values, and evaluate the impact of this uncertainty on the overall model estimates.

Although much of the information needed for a TSPA has been collected, ongoing studies are seeking supplemental, as well as confirmatory, data on geophysical and hydrological parameters. Outstanding issues at this time pertain to waste packaging and engineering designs, waste emplacement configuration, and site-specific geological and hydrological parameter values.

For these reasons, stringent estimates of anticipated radionuclide concentrations in ground water are currently lacking and preclude predictions of actual doses and risks to a critical group.

Projected doses and risks to the critical group, however, can be modeled by arbitrarily defining a fixed contamination level for each individual radionuclide. A convenient approach is to set the concentration of a radionuclide contaminant in extracted well water to a unit concentration for which a corresponding dose and risk estimate can be predicted to a member of the critical group using a defined model scenario.

Values derived per unit concentration not only provide a preliminary understanding of the relative contribution of individual pathways to dose and risk but may actually be employed in the total system performance assessment at a later date. When data become available that will allow the prediction of ground water contamination levels, actual estimates of dose and risk to the critical group can be determined by a simple multiplication step.

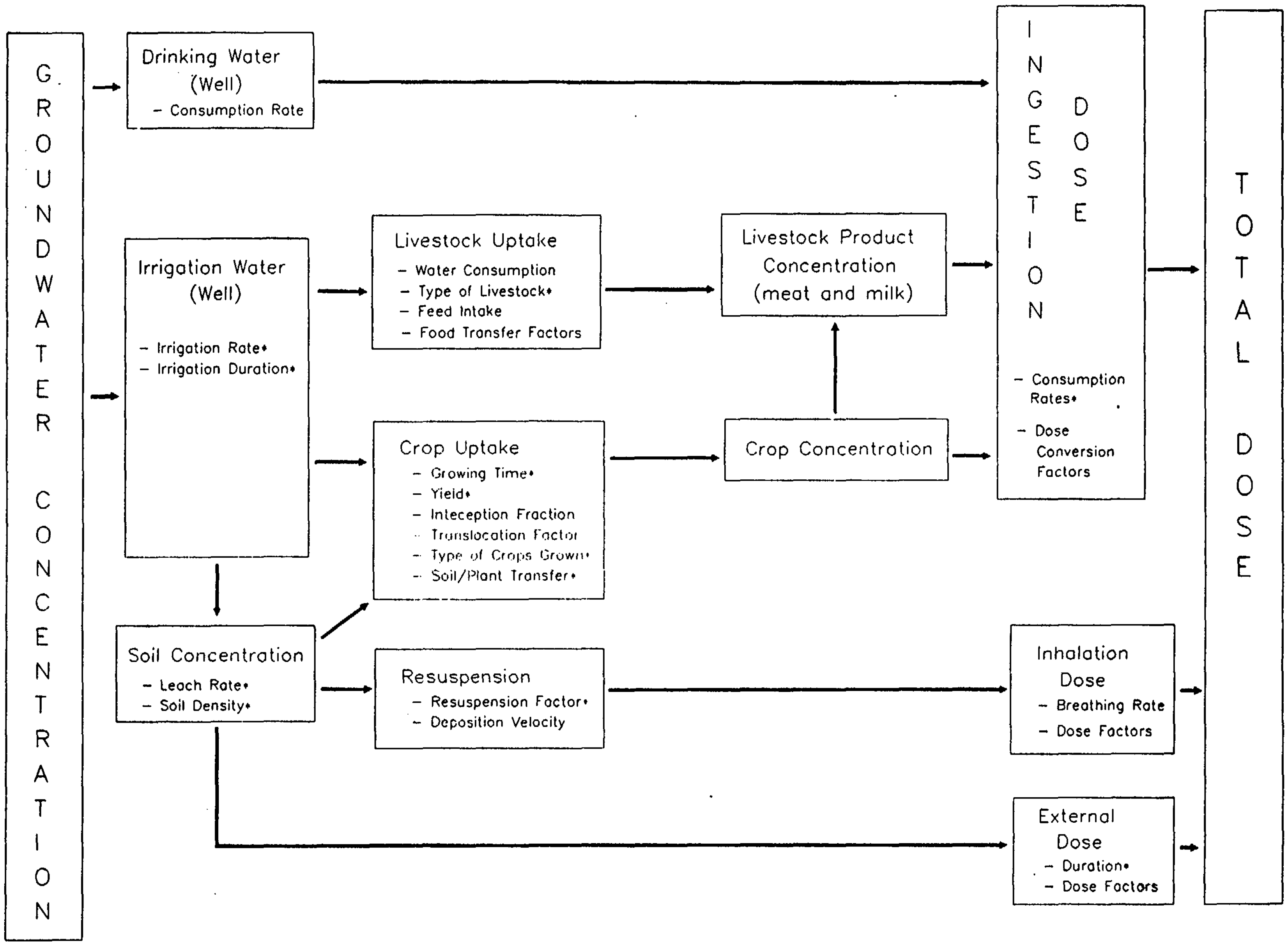
8.5.1 Description of the Model

Estimates of exposures and risks for well water containing 1 picocurie per liter (l pCi/L) of a specified radionuclide are based on the assumption that conditions believed to exist now will continue long into the future. Because demographic data indicate that persons similar to a subsistence farmer currently reside in the Amargosa Valley, the scenario selected involves a subsistence farmer residing in the southern vicinity of the proposed repository.

This scenario satisfies the basic criteria of the ICRP critical group by defining persons whose location and habits are such that they represent those individuals expected to receive the highest doses. The sample calculations provided here are considered an upper bound. The farmer is assumed to grow alfalfa, which is used for raising beef and milk cows. In addition, a garden lot is used to grow sufficient quantities of vegetables, fruits, and grain for home use. In effect, all food consumed by the farmer is assumed home-grown. All water required for drinking, crop irrigation, and livestock is assumed to be extracted from a single on-site well. The principal exposure pathways for the subsistence farmer are depicted in Figure 8-7 and lead to internal exposure from the ingestion of contaminated drinking water and food products and inhalation of resuspended (airborne) soil contaminants. External irradiation may result from outdoor exposure to irrigated soil when the contaminant emits penetrating gamma radiation.

Figure 8-7 also identifies numerous parameters that influence the dynamics of radionuclides in the environment, suggesting that calculation of dose is sufficiently complex to require a versatile and state-of-the-art system for modeling radiation doses associated with chronic environmental releases. Originally developed by Pacific Northwest Laboratory (DOE88a) to model exposures at the Hanford site, the GENII system implements NRC Regulatory Guide 1.109, along with dosimetry models recommended by the ICRP. To establish confidence in its results, the GENII code has also been subjected to rigorous peer review and meets the benchmarking requirements defined by the American Society of Mechanical Engineers (ASM89a, ASM89b). This code was also identified and recommended for performance assessment computer modeling. Among available computer codes, GENII-S is regarded as a proven, by the NAS Committee on the Technical Bases for Yucca Mountain Standards (NAS95).

The GENII-S code is a successor to the GENII Pacific Northwest Laboratories code. Developed by Sandia National Laboratories, the GENII-S was designed for use in the performance assessment of the Waste Isolation Pilot Plant (WIPP) (DOE93b). The GENII-S system includes interactive menu-driven programs that assist with scenario generation and data input requirements. Input requirements for each parameter value may then be selected from data libraries or other sources inclusive of site-specific information. An important feature of GENII-S is its versatility in conducting either a deterministic or stochastic analysis. Implementation of a stochastic analysis gives the user the option to assess the sensitivity and uncertainty of the model outputs.



• Identifies parameters that are best defined by site-specific values.

Figure 8-7. Ground Water Pathway Model for Subsistence Farmer

8.5.2 Overview of a Recently Completed Study

As part of the NRC's commitment to conduct an independent TSPA, the Commission's Division of Waste Management has undertaken various dose assessment analyses. In a recently completed study sponsored by the NRC and conducted by the Center for Nuclear Waste and Regulatory Analysis (CNW95—identified hereafter as the "CNWRA Study"), dose estimates corresponding to a unit ground water concentration were derived for 20 radionuclides for a subsistence farmer. The study's primary objectives were to identify pathways and parameters having the most influence on dose and dose variations. A Monte Carlo-based uncertainty analysis sampled 43 model parameters.

The basic model defined a subsistence farmer who grows alfalfa for livestock, including beef and milk cows, and vegetables, fruits, and grains for personal consumption.

The GENII-S code was used for modeling, whenever possible, site-specific (or site-compatible) parameter values for soil characteristics, meteorology, irrigation, and other agricultural practices.

8.5.3 A Critical Assessment of the CNWRA Study

In recognition of the existing CNWRA Study data and EPA's review of the study's basic approach and objectives, the selection of all model parameter values was critically reviewed and the resultant GENII-S output values were verified. The location of the farm is independent of consumption rates and dose per unit volume. Appendix 8.5 cites the parameters and parameter values reviewed. Presented below is an overview of the major parameter groupings that characterize the model.

Agricultural Data Used

Subsistence farming, as described in the CNWRA Study, is consistent with present-day agricultural practices for Nye County and incorporates all of the exposure pathways shown in Figure 8-7. Crops likely to be grown in a local vegetable garden were appropriately selected (MLS93) and grouped into categories of leafy vegetables, fruits, and root vegetables, as directed by the GENII-S code. Growing times for each group of crops were selected from

Chambers and Mays (CHA94) and reflect climatic conditions consistent with those at the Yucca Mountain Site.

Conversion of Areal Deposition into Mass Concentration. Contamination of feed and food crops is directly related to the areal deposition of contaminants by water irrigation. To convert the amount deposited by irrigation per unit area (areal concentration, C_A) into the mass concentration, C_M (per units mass) of soil or vegetation, it is necessary to divide the areal concentration by the soil density per unit area or by the mass of vegetation per unit area (termed the vegetation density or, in the case of farm products, the yield or weight per unit area). In soil, it is necessary to specify the depth of interest.

The areal soil density is given by $\rho_A = \rho z$, where ρ is the soil density (typically 1.6 to 2.6 g/cm^3 or 1600 to 2600 kg/m^3) and z is the depth of interest (cm or m). For determining uptake by plants from soil, root depths of 0.15 to 0.2 m are common, so that areal soil densities of 240 to 520 kg/m^2 are reasonable for this use. This calculational process assumes a uniform distribution of the radionuclide with depth, which would be more typical of tilled soil used for agriculture.

In the case of vegetation, the mass concentration C_M is obtained by dividing the areal concentration by the vegetation density Y_D (kg/m^2). As the amount of water in vegetation is highly variable and dependent upon collection and storage techniques, use of the dry-weight yield or dry vegetation density [$kg(dry)/m^2$] is preferable.

For Yucca Mountain, site-specific soil density values suggest a range of 1.2 to 1.8 g/cm^2, which for a 15-cm depth corresponds to a surface areal density of 180 to 270 kg/m^2. While this range of values appears appropriate for most vegetative pathways, alfalfa's tap root can grow to depths of several feet (STI91). The GENII-S code uses a two-compartment soil model that accounts for differences in densities between the tilled layer and the more dense lower layer. The higher density of the lower layer has less pore space available to hold water available for root uptake. For alfalfa, a conservative decision was made. A single-compartment soil model represented exclusively by the upper layer was selected.

Crop Yields. Estimates of crop yields were based on data provided by the Nevada Agricultural Statistics Services, which reflect state-wide yields for wheat, barley, potatoes, garlic, and onions. From these data, crop yields for leafy vegetables and root vegetables were assumed to have a range of 0.618 to 6.47 kg/m^2 and 0.769 to 20.8 kg/m^2, respectively. Because fruits are not commercially produced in this region, estimates of yields between 0.3 and 2.0 kg/m^2 were obtained from Snyder et al. (DOE94c).

Crop Interception Fraction. The crop interception fraction refers to the fraction of contaminants in irrigation water deposited on the plant surface. The interception fraction varies among crops and with geographic location. Based on laboratory and field study data that included the Yucca Mountain Site (LLL87), values were assumed to range from 0.06 to 1.0 with a best estimate of 0.40. While a value of 1.0 represents a theoretical upper limit, it is not considered excessively conservative in instances (1) of high-density vegetation which is characteristic of home gardens, (2) where irrigation is employed judiciously, and (3) where evaporation rate is high.

In summary, parameters selected for characterizing Yucca Mountain soil, crop growing times, crop interception fractions, and crop yields are representative of site-specific values or fall within the range of values cited in the scientific literature.

Radionuclides Considered

A total of 43 radionuclides has been identified to represent the bulk of radioactivity present in spent nuclear fuel and high-level DOE waste. However, not all radionuclides require aqueous analysis. Radionuclides that can be expected to exist in highest concentration in ground water are those with significant low K_d values. For modeling purposes, this characteristic of a radionuclide is expressed by its distribution coefficient, or K_d value. K_d expresses the ratio of the concentration of adsorbed isotope relative to dissolved isotope:

$$K_d\left(\frac{ml}{g}\right) = \frac{\text{concentration adsorbed on particles }\left(\frac{Ci}{g}\right)}{\text{concentration dissolved in water }\left(\frac{Ci}{ml}\right)}$$

The higher the K_d, the more likely an isotope can be expected to be bound to solid particles and prevented from reaching the ground water. With the exception of Pu-239, radionuclides with K_d values greater than 10 or 20 mℓ/g were not considered to contribute significantly to human exposure. Based on this criterion, TSPA 1995 (DOE95b) identified a total of eight radionuclides: C-14, Ni-59, Sc-79, Tc-99, Cs-135, Pb-210, Ra-226, and Np-237.

Radionuclides for potential consideration in the CNWRA Study were selected from among those cited in several other relevant performance assessment studies (DOE92, DOE93c, DOE94b, DOE94a). Final selection for inclusion in the analysis considered the water pathway and the need to account for the effects of retardation. Nevertheless, the 20 radionuclides selected (Table 8-17) are considered comprehensive since they include some with relatively high K_d values. The 20 radionuclides analyzed in the CNWRA Study incorporate the eight radionuclides considered in TSPA 1995 (TRW 1995).

Table 8-17. Radionuclides Selected for Analysis in CNWRA Study and Assumed K_d Values (CNW95)

Nuclide	TSPA 93*	GENII	RESRAD	Nuclide	TSPA 93	GENII	RESRAD
C-14	100	---	---	Th-230	100	630	60,000
Ni-59	0	72	1000	U-234	0	260	50
Se-79	0	19	---	U-238	0	260	50
Nb-94	100	120	0	Np-237	0	100	0
Tc-99	---	<0.1	0	Pu-239	50	260	2000
I-129	---	<0.1	0.1	Pu-240	50	260	2000
Cs-135	0	300	500	Am-241	100	260	20
Cs-137	0	300	500	Am-243	100	260	20
Pb-210	0	740	100	Cm-245	NG**	2600	0
Ra-226	0	560	70	Cm-246	NG	2600	0

* Represent minimum k_d values cited in TSPA 1993 (DOE94a).

** NG = not given in TSPA 1993.

Food Transfer Factors

Food transfer factors quantify the amounts of contaminants that may sequentially be transferred from soil to vegetation. When vegetation is used for animal feed, transfer factors must also be used to estimate contamination levels in meat, milk, and other food products. Radionuclide uptake by plants from soil has generally been described by an empirical concentration ratio CR, which is defined as:

$$CR = \frac{Radionuclide\ activity\ per\ unit\ mass\ of\ plant}{Radionuclide\ activity\ per\ unit\ mass\ of\ soil}$$

The radionuclide soil concentrations are generally expressed in terms of oven-dried soil weight. However, the radionuclide concentrations in crops are reported both in terms of fresh (or wet) weight and dry weight. The relationship between the fresh and dry concentration ratios is:

$$B_{iv} = CR(fresh) = \frac{CR(dry)}{FW/DW}$$

where FW and DW correspond to the fresh and dry weight, respectively. Due to variations in water content, the ratio of fresh to dry weights among leafy vegetables may vary from a low of about 7 for brussel sprouts to a high of 20 for lettuce. For root vegetables, ratios range from 4 (potatoes) to 18 (radishes) (NRC83).

The feed-to-beef and feed-to-milk transfer factors are also empirically derived constants that establish a relationship between the amount of an element (or radionuclide) that cattle and milk cows ingest chronically on a daily basis and the concentration of that element in edible meat or milk at equilibrium. The standard units are expressed as a ratio of pCi/kg of meat or milk to pCi/day of chronic intake contained in feed.

Concentration ratios and food transfer factors are affected by many processes and factors, some of which are site-specific. Empirical data for the area around the Yucca Mountain Site were reviewed in the CNWRA Study but dismissed as incomplete. Element-specific transfer factors used in the CNWRA Study (Table 8-18 (CNW95)) were therefore obtained from data published by the International Atomic Energy Agency (IAE94), the International Union of Radioecologists (IUR89), and by Oak Ridge National Laboratory (ORN82).

Table 8-18. Concentration Ratios and Transfer Factors Employed in CNWRA Study (CNW95)

Element	Concentration Ratio				Transfer Coefficient	
	Leafy Vegetables	Other Vegetables	Fruit	Grain	Beef	Milk
Am	1.2E-3	4.7E-4	4.7E-4	2.2E-5	4.0E-5	1.5E-6
Cm	1.1E-3	5.8E-4	5.8E-4	2.1E-5	3.5E-6 (ORN82)	2.0E-5 (ORN82)
Cs	1.1E-1	7.2E-2	7.2E-2	1.0E-2	5.0E-2	7.9E-3
I	3.4E-3	2.0E-2	2.0E-2	2.0E-2	4.0E-2	1.0E-2
Nb	5.0E-2	1.7E-2	1.7E-2	1.7E-2	3.0E-7	3.0E-7
Ni	1.8E-1 (ORN82)	3.0E-2	3.0E-2	3.0E-2	5.0E-3	1.6E-2
Np	6.9E-2	2.7E-2	2.7E-2	2.7E-3	1.0E-3	5.0E-6
Pb	1.1E-3	6.4E-3	6.4E-3	4.7E-3	4.0E-4	2.5E-4 (ORN82)
Pu	3.4E-4	2.3E-4	2.3E-4	8.6E-6	1.0E-5	1.1E-6
Ra	8.0E-2	1.3E-2	1.3E-2	1.2E-3	9.0E-4	1.3E-3
Se	2.5E-2 (ORN82)	2.5E-2 (ORN82)	2.5E-2 (ORN82)	2.5E-2 (ORN82)	1.5E-2	4.0E-3 (ORN82)
Tc	7.6E1	1.1E1	1.1E1	7.3E-1	1.0E-4	1.4E-4
Th	1.1E-2	3.1E-4	3.1E-4	3.4E-5	6.0E-6 (ORN82)	5.0E-6 (ORN82)
U	2.3E-2	1.1E-2	1.1E-2	1.3E-3	3.0E-4	4.0E-4

Transfer factors selected for analysis by CNWRA were compared to values 1) employed by EPA (NESHAPs), 2) cited in the literature, or 3) used as default values in other computer codes. Based on this review, it appears that except for iodine, the plant transfer values used by CNWRA are consistent with other values and appear appropriate for modeling the Yucca Mountain Site. For iodine, plant transfer factors appear low by one to two orders of magnitude.

With one exception, beef and milk transfer factors values cited for analysis also appear consistent with commonly used values. There appears to be a substantial discrepancy in the milk transfer factor for technetium. The cited value of 1.4E-4 is nearly 100-fold lower than those recommended by NRC's NUREG/CR-5512, EPA's NESHAPs, and the computer code PRESTO.

Water Consumption Rates for Drinking and Irrigation

A search for site-specific or representative values failed to yield useful information regarding water consumption rates for hot, dry regions, such as southwestern Nevada. To model potential exposure from drinking contaminated water, the CNWRA Study selected values from a nationwide Food and Drug survey (ROS92). Results indicated that water consumption rates in the United States are distributed log normally, yielding a geometric mean of 349 L/yr and a geometric standard deviation of 1.78. For modeling, CNWRA selected a 95 percent confidence interval, which corresponds to a range of 113 to 1081 L/yr.

These values are low when compared to generic values recommended by the EPA (EPA89), which assumes an average value of 511 L/yr and a 90th percentile value of 730 L/yr. In the context of ICRP recommendations to apply cautious but reasonable assumptions, national average values for estimating the amount of water consumed by a subsistence farmer in a desert environment appear inappropriate. For a subsistence farmer who is expected to spend 6.5 to 15 hr/day outdoors, higher values seem reasonable.

Contaminated water, when used for irrigation in addition to drinking, is also a potentially important pathway for human exposure. In Section 8-3, it was noted that the Amargosa Valley region south of the Yucca Mountain Site currently uses ground water for agricultural irrigation, which supports the likelihood of its future use by a subsistence farmer. However, suitable data do not currently exist for quantifying irrigation rates associated with small-scale or subsistence farming. For surrogate values, the CNWRA Study relied on water irrigation required for lawn maintenance in Nye County (MIL93). Data suggest a range of values between 26 and 84 inches per year, which corresponds to a growing season of 6 to 12 months per year.

A comparison of physiological parameters between lawn grasses and edible crops (moisture content, surface to volume ratio, root depth, etc.) suggests that these surrogate values provide a reasonable approximation. The appropriateness of these values is also supported by the fact that the range of irrigation values generally approximates the natural precipitation rates in parts of the country where irrigation is not required.

Buildup of Soil Contaminants

The chronic irrigation of farm land with contaminated water may result in a buildup of contaminants over time. A buildup of soil contaminants affects the ingestion pathways involving edible crops and animal feed, the inhalation exposure from resuspended soil particles, and the external exposure from contaminated ground surfaces.

Estimates of soil buildup are complex and reflect the rate of deposition of a contaminant and its rate of removal. Soil contaminants may be removed by several concurrent processes that include leaching (or washoff), crop uptake (and subsequent harvesting of crop), wind erosion, and radionuclide decay. These biophysical processes can be combined and represented by a simple removal rate constant. Removal rate constants, however, are not easily determined since they are radionuclide-specific and affected by a host of site-specific parameters.

Soil buildup was not modeled into the unit concentration dose estimates for the 20 radionuclides analyzed in the CNWRA Study. Dose estimates, in effect, reflect the combined annual external exposure and committed internal exposures associated with soil irrigation for a period of one year.

The potential impact of neglecting soil buildup is likely to vary depending on the radionuclide. When radionuclides are assumed highly soluble and exhibit high leach rates, buildup may be insignificant. For this condition, contamination of food crops may be dominated by external deposition on vegetative (leaf) surfaces.

As part of EPA's evaluation, the failure to consider soil buildup was assessed. Test runs were performed using leaching removal terms provided by the GENII code. For radionuclides with high leach rates, such as I-129 and Tc-99, dose estimates were unaffected by long-term irrigation, indicating that soil buildup was insignificant. For other radionuclides, doses were not significantly affected for irrigation times of 100 and even up to 1,000 years. With irrigation periods lasting 10,000 years, however, soil buildup for a limited number of radionuclides yielded a ten-fold increase in dose estimates.

An independent assessment by the Electric Power Research Institute (EPRI) cited provisional calculations for I-129 and Np-237 (EPR94). The EPRI results confirm that for the ground water release scenario, soil buildup from irrigation is insignificant. EPRI concluded that the need to assess the impact of long-term soil accumulation is limited when radionuclides can be assumed relatively soluble.

Inhalation and Soil Exposure Times

In the CNWRA Study, inhalation and soil exposure times are assumed to be the same. The minimum exposure time of 5,548 hr/yr is based on spending 73 percent of the time indoors with a 0.5 indoor exposure factor, and the remaining portion of the time outdoors in the contaminated area. The maximum exposure time of 7,117 hr/yr was based on an individual's spending 15 hr/day, 7 days/wk outdoors in the contaminated area (i.e., a 1.0 exposure factor), with the rest of the time spent indoors with an exposure factor of 0.5. A triangular distribution was assumed sloping to the minimum value from the maximum value, under the assumption that the farmer is likely to spend much of the day outdoors.

Selection of Dose Conversion Factors

Estimates of radiation doses that result from the ingestion, inhalation, or external exposure to radioactivity are based on dose conversion factors (DCFs) that make use of contemporary metabolic models and dosimetry methods. A critical choice in selecting DCF values for dose calculations relates to the solubility of the radionuclide contaminant in aqueous fluids. When ingested, a radionuclide that is highly soluble is readily absorbed from the intestinal tract to the blood stream where it may be metabolized and retained for long periods of time; similarly, a soluble contaminant that is inhaled may also be quickly removed from the lung and enter the blood stream where its fate is essentially that of an ingested radionuclide. For ingestion and inhalation pathways, DCFs are generally defined in terms of solubility by means of the f_1 (fractional uptake of nuclides from small intestine) and lung clearance class. Lung clearance, designated as D,W, or Y, refers to "days, weeks, or years" for the radionuclides to be removed from the pulmonary region of the lung.

For insoluble contaminants, the potential for internal exposure through inhalation versus ingestion is quite dissimilar. Inhaled insoluble radionuclides are not readily removed from the lung and may, therefore, result in long-term exposure of the lung and other tissues. Internal dose is minimized, however, when the insoluble contaminant passes through the digestive system without being absorbed.

The GENII code offers a choice of DCFs that correspond to very soluble, soluble, and insoluble states of individual radionuclides. A comparison of GENII DCF values with those in EPA Federal Guide Reports No. 11 and No. 12 (EPA88, EPA93b) shows that, when matched for solubility, there is generally good agreement.

A significant issue in the CNWRA Study is the selection of DCF values that assume all radionuclides to be insoluble. This choice is difficult to explain in light of the fact that radionuclides present in well water must be assumed to have been leached by infiltrating surface water from the waste package through engineered barriers, then traveled through hundreds of feet of unsaturated zone, and finally been introduced into ground water with nominal flow velocity and recharge rate. To account for the presence of radionuclides in extracted well water, it would be more logical to assume radionuclides are soluble.

Table 8-19 compares DCF values based on solubility for the 20 parent radionuclides modeled in the CNWRA Study and their respective radioactive daughters (radionuclide groupings are segregated by horizonal lines). For several radionuclides considered of primary importance to repository performance, differences between insoluble and soluble DCF values varied by as much as two orders of magnitude.

8.5.4 EPA Dose Estimates Per Unit Concentration for the Subsistence Farming Model

EPA calculated dose estimates per unit concentration for the subsistence farmer. The Agency used the GENII code and modified some parameter values employed in the CNWRA Study. The values modified include selected food-transfer factors, water-consumption rates, exposure durations, and DCF values:

- Transfer factors considered more appropriate for Tc-99 and I-129 are cited in Table 8-20.

- For water consumption, the range of values was revised to correspond to an average consumption of 730 L/yr, which is more than double the 349 L/yr value used by the CNWRA analysis.

- For modeling internal exposure, DCF values corresponding to a soluble chemical state of radionuclide contaminants were selected from EPA Federal Guidance Report No. 11 (EPA88).

Table 8-19. A Comparison of Insoluble DCF Values Selected in the CNWRA Study with Soluble DCF Values

Radionuclide	Insoluble DCF Values[1]			Soluble DCF Values[2]		
	Inhalation	Ingestion	Contaminated Ground	Inhalation	Ingestion	Contaminated Ground
U-234	4.02E-05	7.06E-09	7.48E-19	7.37E-07	7.66E-08	7.48E-19
U-238	3.57E-05	6.45E-09	5.51E-19	6.62E-07	6.88E-08	5.51E-19
Th-234	9.46E-09	3.64E-09	2.36E-17	8.04E-09	3.69E-09	2.36E-17
Pa-234	2.29E-10	5.98E-10	1.84E-15	1.98E-10	5.84E-10	1.84E-15
Th-230	8.04E-05	1.64E-07	7.50E-19	8.80E-05	1.48E-07	7.50E-19
Ra-226	2.21E-06	2.73E-07	5.44E-18	2.32E-06	3.58E-07	5.44E-18
Rn-222	0.00E+0	0.00E+0	1.65E-15	0.00E+0	0.00E+0	1.65E-15
Pb-210	0	0	2.48E-18	0	0	2.48E-18
Bi-210	3.79E-06	1.50E-06	1.05E-18	3.67E-06	1.45E-06	1.05E-18
Po-210	5.34E-08	1.71E-09	8.29E-21	4.18E-09	1.73E-09	8.29E-21
	2.33E-06	5.19E-07		2.54E-06	5.14E-07	
Tc-99	2.31E-09	5.98E-10	7.80E-20	2.77E-10	3.95E-10	7.80E-20
Se-79	2.57E-09	2.25E-09	2.07E-20	1.77E-09	2.35E-09	2.07E-20
Ra-226	2.21E-06	2.73E-07	5.44E-18	2.32E-06	3.58E-07	5.44E-18
Rn-222	0.00E+0	0.00E+0	1.65E-15	0.00E+0	0.00E+0	1.65E-15
Pb-210	0	0	2.48E-18	0	0	2.48E-18
Bi-210	3.79E-06	1.50E-06	1.05E-18	3.67E-06	1.45E-06	1.05E-18
Po-210	5.34E-08	1.71E-09	8.29E-21	4.18E-09	1.73E-09	8.29E-21
	2.33E-06	5.19E-07		2.54E-06	5.14E-07	
Pu-239	9.81E-05	1.54E-08	3.67E-19	1.16E-04	9.56E-07	3.67E-19
Pu-240	9.80E-05	1.54E-08	8.03E-19	1.16E-04	9.56E-07	8.03E-19
Pb-210	3.79E-06	1.50E-06	2.48E-18	3.67E-06	1.45E-06	2.48E-18
Bi-210	5.34E-08	1.71E-09	1.05E-18	4.18E-09	1.73E-09	1.05E-18
Po-210	2.33E-06	5.19E-07	8.29E-21	2.54E-06	5.14E-07	8.29E-21
Np-237	2.24E-04	1.85E-06	2.87E-17	1.46E-04	1.20E-06	2.87E-17
Pa=233	2.56E-09	9.62E-10	1.95E-16	2.24E-09	9.81E-10	1.95E-16
Ni-59	2.39E-10	5.44E-11	0.00E+00	3.58E-10	5.67E-11	0.00E+00
Nb-94	1.13E-07	1.96E-09	1.53E-15	9.76E-09	1.93E-09	1.53E-15
I-129	4.08E-08	6.75E-08	2.58E-17	4.69E-08	7.46E-08	2.58E-17
Cs-137	8.11E-09	1.28E-08	5.86E-16	8.63E-09	1.35E-08	5.86E-16

Table 8-19. (Continued)

Radionuclide	Insoluble DCF Values[1]			Soluble DCF Values[2]		
	Inhalation	Ingestion	Contaminated Ground	Inhalation	Ingestion	Contaminated Ground
U-234	4.02E-05	7.06E-09	7.48E-19	7.37E-07	7.66E-08	7.48E-19
Cs-135	1.21E-09	1.85E-09	3.33E-20	1.23E-09	1.91E-09	3.33E-20
Cm-246	1.50E-04	1.23E-06	7.85E-19	1.22E-04	1.00E-06	7.85E-19
Cm-245	1.50E-04	1.23E-06	8.70E-17	1.23E-04	1.01E-06	8.70E-17
Pu-241	1.83E-06	2.72E-10	1.93E-21	2.23E-06	1.85E-08	1.93E-21
Am-241	1.44E-04	1.19E-06	2.75E-17	1.20E-04	9.84E-07	2.75E-17
C-14	5.59E-10	5.59E-10	1.61E-20	5.64E-10	5.64E-10	1.61E-20
Am-243	1.46E-04	1.20E-06	5.35E-17	1.19E-04	9.79E-07	5.35E-17
Np-239	6.94E-10	8.89E-10	1.63E-16	6.78E-10	8.82E-10	1.63E-16
Pu-239	9.81E-05	1.54E-08	3.67E-19	1.16E-04	9.56E-07	3.67E-19
Am-241	1.44E-04	1.19E-06	2.75E-17	1.20E-04	9.84E-07	2.75E-17
Pa-231	3.38E-04	4.30E-06	4.07E-17	3.47E-04	2.86E-06	4.07E-17

[1] From GENII code.

[2] From EPA Federal Guidance Reports No. 11 and No. 12 (EPA88, EPA93b).

Table 8-20. Revised Concentration Ratios and Transfer Factors Assumed for Dose Estimates

Radionuclide	Leafy Vegetables	Other Vegetables	Fruit Grain	Beef	Milk
Tc-99	9.5E+00	1.1E+00	unchanged	unchanged	1.0E-02
I-129	unchanged	unchanged	unchanged	7.0E-03	unchanged

- For modeling external doses from contaminated ground surfaces, DCF values corresponding to EPA Federal Guidance Report No. 12 (EPA93b), Table III.3, were used.

- For inhalation and soil exposure, the maximum exposure time was set at 8,760 hr/yr. This is based on the assumption that a home provides little protection (e.g., windows would likely be open allowing the indoor airborne radionuclide concentration to approach the outdoor concentration). A minimum exposure time was set at 2,400 hr/yr, to represent a hired hand who would be working 8 hr/day, 6 days/wk. A uniform distribution between the two values was assumed.

- The dust loading value of 5.5E-05 g/m^3 was considered reasonable, but instead of assuming that this was a fixed value, it was assumed that the dust loading could range from 1.0E-05 to 2.5E-04 g/m^3 with a truncated log-normal distribution. These values represent the 1st and 99th percentile.

Table 8-21 provides average estimates of annual doses received by a subsistence farmer using well water containing 1 pCi/L of the radionuclide contaminant. Dose is expressed as the annual total effective dose equivalent (TEDE), which represents the sum of the annual external exposure and the 50-year committed dose from all internal exposure pathways. Average dose estimates and their associated standard deviations were calculated for a distribution assumed either normal or log-normal.

Probability plots from 125 realizations of sampled runs per radionuclide suggest that for most radionuclides, TEDE values are log-normally distributed. Equations for defining the geometric mean (x_g) and geometric standard deviation (s_g) and 95 percent confidence interval are identified below:

$$x_g = antilog \left(\frac{\Sigma \log x_i}{n}\right)$$

$$s_g = antilog \frac{\sqrt{\Sigma \; (\log x_i - \mu)^2}}{n}$$

where:

μ = the mean of log-transformed TEDE values for a given radionuclide,
n = the number of TEDE values for a given radionuclide (i.e., realizations).
x_i = a single TEDE value for a given radionuclide.

Table 8-21. Annual TEDE Values for Radionuclides Present in Well Water at 1 pCi/L

Radio-nuclide	Annual TEDE (rem/yr per pCi/ℓ)					
	Arithmetic			Geometric		
	Mean	Standard Deviation	95% Confidence Interval	Mean	Standard Deviation	95% Confidence Interval
U-234	8.45E-04	3.09E-04	2.39E-04 - 1.45E-03	7.91E-04	1.45E+0	3.82E-04 - 1.64E-03
U-238	7.39E-04	2.76E-04	1.98E-04 - 1.28E-03	6.89E-04	1.46E+00	3.28E-04 - 1.45E-03
Th-230	1.38E-03	5.18E-04	3.65E-04 - 2.40E-03	1.29E-03	1.45E+00	6.23E-04 - 2.67E-03
Tc-99	6.33E-06	2.21E-06	2.00E-06 - 1.07E-05	5.98E-06	1.41E+00	3.05E-06 - 1.17E-05
Se-79	5.98E-05	2.97E-05	1.59E-06 - 1.18E-04	5.32E-05	1.64E+00	2.02E-05 - 1.40E-04
Ra-226	4.38E-03	1.51E-03	1.42E-03 - 7.34E-03	4.12E-03	1.43E+00	2.04E-03 - 8.31E-03
Pu-239	8.76E-03	3.28E-03	2.33E-03 - 1.52E-02	8.18E-03	1.45E+00	3.95E-03 - 1.69E-02
Pu-240	8.79E-03	3.30E-03	2.32E-03 - 1.53E-02	8.22E-03	1.45E+00	3.97E-03 - 1.70E-02
Pb-210	1.67E-02	6.06E-03	4.82E-03 - 2.86E-02	1.57E-02	1.44E+00	7.68E-03 - 3.21E-02
Np-237	1.05E-02	3.82E-03	3.01E-03 - 1.80E-02	9.80E-03	1.45E+00	4.73E-03 - 2.03E-02
Ni-59	1.58E-06	9.31E-07	~0.00 - 3.40E-06	1.37E-06	1.72E+00	4.73E-07 - 3.97E-06
Nb-94	2.07E-04	5.84E-05	9.25E-05 - 3.21E-04	1.99E-04	1.35E+00	1.11E-04 - 3.58E-04
I-129	3.62E-03	2.15E-03	~0.00 - 7.83E-03	3.07E-03	1.80E+00	9.70E-04 - 9.72E-03
Cs-137	8.32E-04	4.62E-04	~0.00 - 1.74E-03	7.25E-04	1.70E+00	2.56E-04 - 2.05E-03
Cs-135	1.43E-04	8.68E-05	~0.00 - 3.13E-04	1.21E-04	1.81E+00	3.78E-05 - 3.87E-04
Cm-246	8.65E-03	3.24E-03	2.30E-03 - 1.50E-02	8.08E-03	1.45E+00	3.90E-03 - 1.67E-02
Cm-245	8.72E-03	3.26E-03	2.33E-03 - 1.51E-02	8.15E-03	1.45E+00	3.93E-03 - 1.69E-02
C-14	2.01E-05	5.36E-06	9.59E-06 - 3.06E-05	1.93E-05	1.32E+00	1.12E-05 - 3.33E-05
Am-243	8.45E-03	3.15E-03	2.28E-03 - 1.46E-02	7.90E-03	1.45E+00	3.81E-03 - 1.64E-02
Am-241	8.62E-03	3.23E-03	2.29E-03 - 1.50E-02	8.05E-03	1.45E+00	3.89E-03 - 1.67E-02

For TEDE values that approximate a normal distribution, the 95 percent confidence interval is given by:

$$x_i \pm (1.96)\, s_i$$

where x_i = the arithmetic mean and
s_i = the arithmetic standard deviation.

For log-normally distributed TEDEs, the 95 percent confidence interval is determined by the following equations:

lower 95% limit: $x_{gi}/s_{gi}^{1.96}$
upper 95% limit: $x_{gi}(s_{gi}^{1.96})$.

Table 8-21 indicates that for most radionuclides, the arithmetic mean exceeds the geometric mean by less than 10 percent. The table also shows which radionuclides contribute to relatively high doses. At a common concentration, the radionuclides yielding the highest doses are Pb-210, Np-237, Pu-239/-240, Cm-245/-246, and Am-241/-243. This information should not be interpreted to mean that these radionuclides will in fact dominate estimates of exposure, since actual dose estimates must account for the relative magnitude of the initial radionuclide concentrations in ground water at the point of extraction.

A delineation of total dose by individual pathways yielded some interesting results. Table 8-22 shows the average fractional contribution of internal dose from ingestion, inhalation, and external dose from ground contamination. Ingestion clearly is the predominant exposure pathway, contributing 90 to 99+percent to TEDE for all but one radionuclide. (For Nb-94, which emits two energetic gammas with a yield of 100 percent, the dominant pathway is external exposure from the contaminated ground surface.) For most radionuclides, drinking water and contaminated food products contributed nearly equally to the dose from ingestion. For C-14, I-129, Cs-137, Se-79, and Tc-99, food products contributed a significantly larger share. However, fractional contributions are based on average values, and for drinking water, they correspond to a consumption of 2 L/d (or 730 L/yr). A significant deviation from average values that define the ingestion pathway would have a corresponding impact on the fractional contribution(s) to TEDE values.

Estimates of Cancer Risks per Unit Concentration

Cancers, as a group, are classified as stochastic health effects. It is generally assumed that individual cancers are initiated by random ionization events involving individual cells. By definition, the severity of a stochastic health effect is independent of the dose. However, the risk of cancer is proportional to dose and generally assumed without threshold.

Table 8-22. Average Fractional Contributions to TEDE by Individual Pathways

Radionuclide	Average Fractional Contribution				
	Ingestion			Inhalation	External
	Drinking Water	All Other	Total		
U-234	4.30E-01	5.70E-01	1.00E+00	6.36E-06	1.08E-04
U-238	3.30E-01	7.60E-01	9.96E-01	6.62E-06	4.08E-03
Th-230	4.58E-01	5.41E-01	9.99E-01	4.53E-04	9.70E-05
Tc-99	2.30E-01	7.69E-01	9.99E-01	2.47E-07	1.04E-03
Se-79	1.60E-01	8.39E-01	1.00E+00	2.41E-07	4.21E-05
Ra-226	3.38E-01	6.17E-01	9.55E-01	4.43E-06	4.51E-02
Pu-239	4.50E-01	5.49E-01	1.00E+00	9.06E-05	5.13E-06
Pu-240	4.53E-01	5.46E-01	1.00E+00	9.02E-05	1.12E-05
Pb-210	4.47E-01	5.52E-01	1.00E+00	2.35E-06	2.53E-05
Np-237	4.29E-01	5.69E-01	9.98E-01	8.74E-05	2.37E-03
Ni-59	1.45E-01	8.55E-01	1.00E+00	1.84E-06	0.00E+00
Nb-94	2.51E-02	9.36E-02	1.19E-01	3.47E-07	8.81E-01
I-129	8.99E-02	9.09E-01	9.99E-01	7.32E-08	6.02E-04
Cs-137	6.11E-02	8.54E-01	9.15E-01	8.40E-08	8.50E-02
Cs-135	5.22E-01	4.78E-01	1.00E+00	7.00E-08	2.84E-05
Cm-246	4.56E-01	5.44E-01	1.00E+00	9.46E-05	1.11E-05
Cm-245	4.63E-01	5.36E-01	9.99E-01	9.39E-05	1.22E-03
C-14	8.58E-02	9.14E-01	1.00E+00	0.00E+00	0.00E+00
Am-243	4.49E-01	5.48E-01	9.97E-01	9.41E-05	3.11E-03
Am-241	4.64E-01	5.35E-01	1.00E+00	9.45E-05	3.90E-04

The methodology EPA has adopted for modeling cancer risk is primarily based on that developed by Land and Sinclair for the ICRP (LAN91). In turn, their models are largely based on empirical data derived from the Japanese Life-Span Study (LSS) involving Hiroshima and Nagasaki atomic bomb survivors. For most cancer sites, the EPA risk model is one in which age-specific, relative risk coefficients are obtained by taking a geometric mean of the coefficients derived from the LSS. The risk models are subsequently applied to estimate organ-specific fatal cancer risks per unit dose for a stationary population with mortality rates governed by 1980 U.S. cancer statistics and mortality rates.

To estimate radiation cancer induction (induction of all cancers: fatal and non-fatal), each organ-specific mortality risk estimate is divided by its respective lethality fraction, i.e., the fraction of radiogenic cancers at that site that are fatal. The EPA method for deriving cancer risk coefficients is described in detail in EPA Report 402-R-93-076 (EPA94) and more recently in the journal Health Physics (PUS95). EPA risk coefficients for relevant radionuclides are provided in Table 8-23.

Cancer risks associated with 1 pCi/L unit concentrations of ground water for each radionuclide were derived by multiplying the risk coefficients of Table 8-23 with pathway-specific quantities of radionuclides ingested, inhaled, and contaminating ground surfaces (Table 8-24). Because of the rigid link between dose and risk, standard deviations for cancer risks parallel the standard deviations previously defined in behalf of corresponding TEDE values. At unit concentration, radionuclides with the highest cancer risks include Pb-210, Ra-226, Pu-239/-240, and Am-241/-243.

Summary. To summarize information presented in this section, the following illustration is provided: when Pb-210 is present in well water at 1 pCi/L, and well water is used by the subsistence farmer for drinking and all phases of food production, the average total dose to the farmer is estimated at 0.016 rem (or 16 mrem) per year from all exposure pathways (see Table 8-21). Table 8-22 shows that this exposure is almost exclusively the result of ingestion, with nearly equal contributions from drinking water and contaminated food products. The lifetime fatal cancer risk associated with the annual TEDE of 16 mrem is estimated at about 1.7 chances in a million. For total cancer incidence (fatal and nonfatal), the lifetime risk is about 30 percent higher and estimated at 2.2 chances in a million (Table 8-24).

Table 8-23. EPA Cancer Risk Coefficients

Radionuclides	Cancer Mortality			Cancer Incidence		
	Ingestion (Bq^{-1})	Inhalation (Bq^{-1})	Ground Surface (per Bq y/m^2)	Ingestion (Bq^{-1})	Inhalation (Bq^{-1})	Ground Surface (per Bq y/m^2)
C-14	1.93E-11	1.31E-13	N/A	2.79E-11	1.89E-13	N/A
Ni-59	3.04E-12	8.00E-12	3.22E-13	5.00E-12	1.08E-11	6.21E-13
Se-79	N/A	N/A	N/A	N/A	N/A	N/A
Nb-94	1.08E-10	1.79E-09	1.84E-09	1.87E-10	2.22E-09	2.79E-09
Tc-99	2.49E-11	6.75E-11	6.22E-16	3.79E-11	7.81E-11	9.78E-16
I-129	5.06E-10	3.35E-10	1.56E-11	4.89E-09	3.30E-09	2.64E-11
Cs-135	8.27E-11	4.78E-11	N/A	1.22E-10	7.33E-11	N/A
Cs-137	5.71E-10	3.34E-10	N/A	8.54E-10	5.18E-10	N/A
Pb-210	1.46E-08	3.59E-08	2.41E-12	1.82E-08	4.51E-08	4.16E-12
Ra-226	5.26E-09	6.62E-08	8.21E-12	7.98E-09	7.35E-08	1.27E-11
Th-230	6.39E-10	4.22E-07	7.72E-13	1.01E-09	4.66E-07	1.33E-12
U-234	7.38E-10	3.58E-07	6.12E-13	1.20E-09	3.77E-07	1.13E-12
Np-237	6.65E-09	8.04E-07	3.05E-11	7.98E-09	9.33E-07	4.92E-11
U-238	7.31E-10	3.19E-07	4.81E-13	1.15E-09	3.36E-07	8.97E-13
Pu-239	7.12E-09	6.82E-07	2.89E-13	8.53E-09	7.51E-07	5.28E-13
Pu-240	7.11E-09	6.81E-07	5.86E-13	8.52E-09	7.51E-07	1.11E-12
Am-241	7.39E-09	8.96E-07	2.61E-11	8.87E-09	1.04E-06	4.26E-11
Am-243	7.36E-09	8.86E-07	6.21E-11	8.84E-09	1.03E-06	9.92E-11
Cm-245	7.54E-09	9.12E-07	8.68E-11	9.05E-09	1.06E-06	1.36E-10
Cm-246	7.48E-09	9.08E-07	5.10E-13	8.97E-09	1.05E-06	9.67E-13

Table 8-24. Cancer Risks Per Unit Concentration of Radionuclide

Radio-nuclide	Mortality Risk (Deaths/yr per pCi/L)				Cancer Incidence (Cancers/yr per pCi/L)			
	Arithmetic		Geometric		Arithmetic		Geometric	
	Mean	Standard Deviation	Mean	Standard Deviation	Mean	Standard Deviation	Mean	Standard Deviation
U-234	8.15E-08	2.98E-08	7.63E-08	1.45	1.32E-07	4.84E-08	1.24E-07	1.45
U-238	9.03E-08	3.35E-08	8.44E-08	1.46	1.45E-07	5.39E-08	1.35E-07	1.46
Th-230	5.97E-08	2.24E-08	5.58E-08	1.45	7.19E-08	2.69E-08	6.72E-08	1.45
Tc-99	3.99E-09	1.39E-09	3.76E-09	1.41	6.07E-09	2.12E-09	5.73E-09	1.41
SE-79	3.05E-08	1.51E-08	2.71E-08	1.64	4.25E-08	2.11E-08	3.78E-08	1.64
Ra-226	6.85E-07	2.21E-07	6.50E-07	1.39	1.04E-06	3.34E-07	9.84E-07	1.39
Pu-239	6.52E-07	2.45E-07	6.09E-07	1.45	7.81E-07	2.93E-07	7.30E-07	1.45
Pu-240	6.54E-07	2.45E-07	6.11E-07	1.45	7.81E-07	2.93E-07	7.30E-07	1.45
Pb-210	1.71E-06	6.20E-07	1.60E-06	1.44	2.19E-06	7.92E-07	2.05E-06	1.44
Np-237	5.93E-07	2.13E-07	5.57E-07	1.44	7.18E-07	2.57E-07	6.74E-07	1.43
Ni-59	8.49E-10	4.99E-10	7.33E-10	1.72	1.40E-09	8.21E-10	1.21E-09	1.72
Nb-94	8.26E-08	2.24E-08	7.95E-08	1.33	1.28E-07	3.41E-08	1.23E-07	1.32
I-129	2.46E-07	1.46E-07	2.08E-07	1.80	2.41E-06	1.44E-06	2.05E-06	1.80
Cs-137	3.49E-07	1.95E-07	3.03E-07	1.71	5.23E-07	2.92E-07	4.55E-07	1.70
Cs-135	6.20E-08	3.76E-08	5.24E-08	1.81	9.15E-08	5.55E-08	7.74E-08	1.81
Cm-246	6.47E-07	2.42E-07	6.05E-07	1.45	7.76E-07	2.91E-07	7.25E-07	1.45
Cm-245	6.53E-07	2.43E-07	6.11E-07	1.45	7.85E-07	2.92E-07	7.35E-07	1.45
C-14	6.86E-09	1.83E-09	6.62E-09	1.32	9.92E-09	2.65E-09	9.57E-09	1.32
Am-243	6.48E-07	2.39E-07	6.07E-07	1.44	7.84E-07	2.88E-07	7.35E-07	1.44
Am-241	6.48E-07	2.42E-07	6.06E-07	1.45	7.78E-07	2.91E-07	7.28E-07	1.45

In the future, TSPA model data will yield ground-water concentration values for Pb-210 and other radionuclides. Actual dose and risk estimates can then readily be derived from tables such as those presented in this section, which define dose and risk per unit concentration.

On a final note, it must be pointed out that the dose and risk data per unit concentration cited above ignored the observation that catfish are currently being raised in Amargosa Valley in tanks filled with locally pumped ground water. This potential pathway was not included because bioaccumulation factors provided in GENII (or cited in the scientific literature) are considered inappropriate. Under normal steady-state conditions, bioaccumulation factors assume that fish not only live in contaminated water but that the entire food chain exists in equilibrium with the contaminant(s). Fish raised commercially in tanks are provided commercial feed, which cannot be assumed to be contaminated. Additional research is needed to determine the limited bioaccumulation in fish raised under the unique conditions of commercial fish-farming.

APPENDIX TO SECTION 8.5

Input Parameters Employed by NRC Study (NRC95)

Parameter	Minimum	Best Estimate	Maximum	Distribution
Soil/Scenario Data:				
Soil/Plant Transfer Scale Factor	0.26		3.9	Log Normal
Animal Uptake Scale Factor	0.26		3.9	Log Normal
Surface Soil Plow Depth (cm)		15		Fixed
Surface Areal Soil Density (kg/m$^{2)}$	180		270	Uniform
Deep Areal Soil Density (kg/m^3)		1500		Fixed
Roots in Upper Soil (Fraction)		1.0		Fixed
Roots in Deep Soil (Fraction)		0.0		Fixed
External/Inhalation Exposure:				
Chronic Plume Exposure (hr)	5548	7116	7117	Triangular
Acute Plume Exposure (hr/yr)	5548	7116	7117	Triangular
Mass Load (g/m^3)		5.0E-5		Fixed
Soil Exposure Time (hr)	5548	7116	7117	Triangular
Home Irrigation Rate (in/yr)	26		84	Uniform
Home Irrigation Duration (mo/yr)	6		12	Uniform
Ingestion Exposure:				
Crop Resuspension Factor (/m)	1.66E-6		6.03E-5	Log Normal
Crop Deposition Velocity (m/s)		0.001		Fixed
Crop Interception Fraction (-)	0.06	0.4	1.0	Triangular
Soil Ingeston Rate (mg/day)		410		Fixed
Drinking Water Consumption (l/yr)	113		1081	Log Normal
Terrestrial Food Ingestion:				
Leaf Vegetable-Grow Time (days)	40		120	Uniform
Root Vegetable-Grow Time (days)	25		120	Uniform
Fruit-Grow Time (days)	65		95	Uniform
Grain-Grow Time (days)	60		90	Uniform
Leaf Vegetable-Irrigation Time (mo/yr)	3		8	Uniform
Root Vegetable-Irrigation Time (mo/yr)	2		8	Uniform

Input Parameters Employed by NRC Study (NRC95) (Continued)

Parameter	Minimum	Best Estimate	Maximum	Distribution
Fruit-Irrigation Time (mo/yr)	2		3	Uniform
Grain-Irrigation Time (mo/yr)	6		8	Uniform
Leaf Vegetable-Yield (kg/m^2)	0.618		6.47	Log Normal
Root Vegetable-Yield (kg/m^2)	0.769		20.8	Log Normal
Fruit-Yield (kg/m^2)	0.3	0.54	2.0	Triangular
Grain-Yield (kg/m^2)	0.471		0.605	Uniform
Leaf Vegetable-Holdup (days)		1		Fixed
Root Vegetable-Holdup (days)		14		Fixed
Fruit-Holdup (days)		14		Fixed
Grain-Holdup (days)		14		Fixed
Leaf Vegetable-Consumption Rate (kg/yr)	4.27		28.3	Log Normal
Root Vegetable-Consumption Rate (kg/yr)	11.3		231	Log Normal
Fruit-Consumption Rate (kg/yr)	10.2		208	Log Normal
Grain-Consumption Rate (kg/yr)	15.3		312	Log Normal
Animal Product Consumption:				
Beef-Consumption Rate (kg/yr)	22.1		157	Log Normal
Milk-Consumption Rate (kg/yr)	20.8		482	Log Normal
Beef-Holdup (days)		20		Fixed
Milk-Holdup (days)		1		Fixed
Beef-Contaminated Water (Fraction)		1		Fixed
Milk-Contaminated Water (Fraction)		1		Fixed
Fresh Forage Data:				
Beef-Dietary Fraction	0.3		0.82	Normal
Milk-Dietary Fraction	0.3		0.82	Normal
Beef-Grow Time (days)	30		62	Uniform
Milk-Grow Time (days)	30		62	Uniform
Beef-Irrigation Time (mo/yr)	3		8	Uniform
Milk-Irrigation Time (mo/yr)	3		8	Uniform
Beef-Yield (kg/m^3)	0.34		1.23	Uniform
Milk-Yield (kg/m^3)	0.34		1.23	U

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CHAPTER 9

SUMMARY SCENARIOS FOR COMPLIANCE ASSESSMENT

9.1 INTRODUCTION

In preceding chapters, information was presented that described the proposed Yucca Mountain repository in terms of its geophysical characteristics and engineering designs, and characterized a future standard that is likely to limit the dose to a specified individual(s). This individual(s) was defined as likely to be at highest risk from radionuclides released from the repository over a regulatory time period of one million years.

In Chapter 8, derived unit-concentration exposure and risk estimates were presented that were based on a conceptual model in which radionuclides were released into the biosphere from a repository through the following sequence: degradation and failure of the waste canister(s) through corrosion; release of radionuclides from the waste package into host rock; migration of radionuclides through the unsaturated zone into the aquifer (saturated zone); and dissemination of contaminated ground water to wells used for drinking and agricultural purposes. This scenario, involving a gradual release from an undisturbed repository, characterizes the most probable events and conditions of future human exposure and conforms with the primary objective of deep geological disposal: by virtue of siting and engineering design, a deep geologic repository is to provide long-term barriers that isolate wastes and limit the release of radionuclides into the biosphere.

An objective of deep geologic disposal is to isolate the wastes for a sufficiently long period of time such that most of the radionuclides will have decayed to natural background levels. While estimates of dose and risk for this gradual release process cannot be calculated with complete precision, there is substantial scientific basis for modeling the various processes that take into account parameter variabilities. By means of statistical processes, such as the Monte Carlo method (see Section 8.5), these uncertainties can be minimized, with the expectation of yielding dose/risk estimates that are reasonable.

In brief, the major release pathway involving ground water from an undisturbed repository at Yucca Mountain leading to human exposure is illustrated in Figure 9-1. The major reservoirs (source terms) containing radionuclides at various times following closure are depicted as

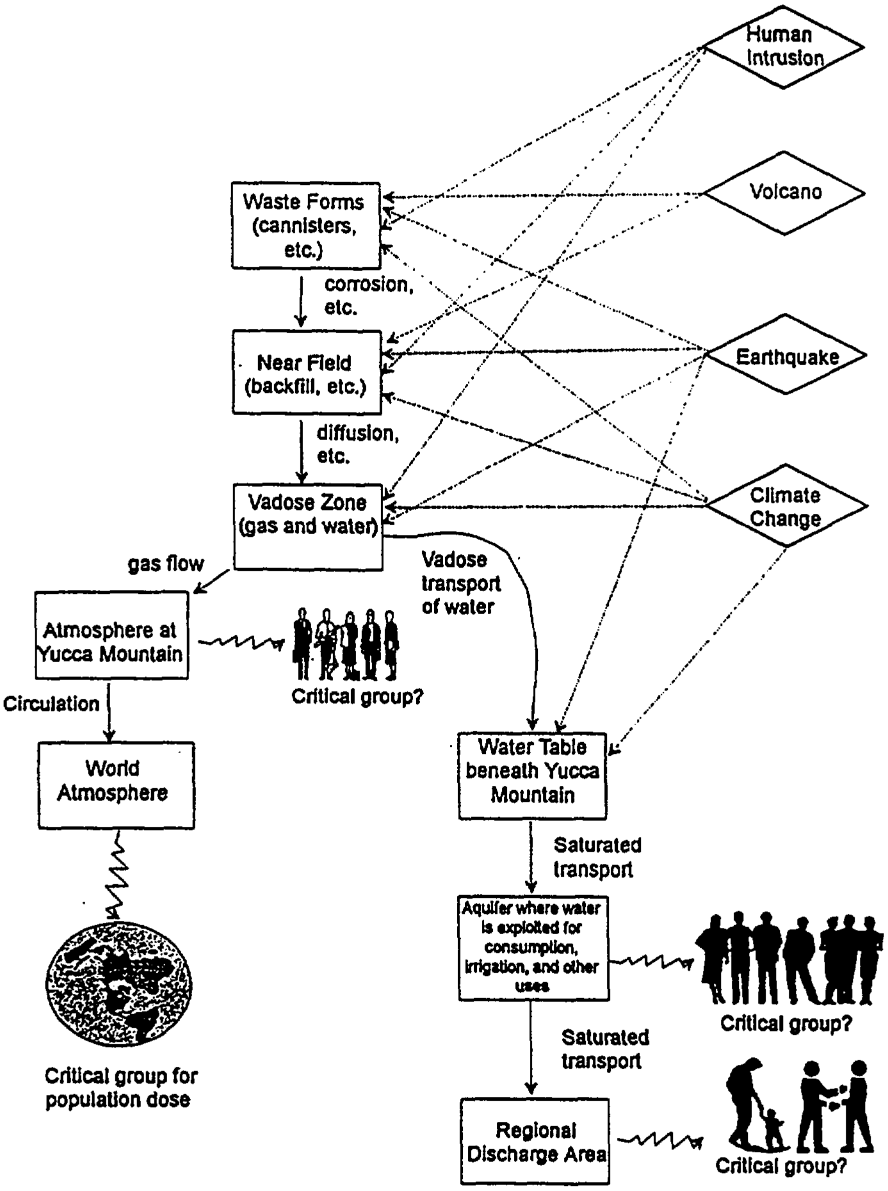

Figure 9-1. Schematic Illustration of the Major Pathways from a Repository at Yucca Mountain to Humans (NAS95)

rectangles. These reservoirs are not discrete with regard to their physical boundaries but rather form a continuum. Solid arrows between reservoirs represent the probable processes by which radionuclides are transported from one reservoir to another in an undisturbed repository.

Major processes and events with potential to modify normal behavior or drastically alter the physical integrity of reservoirs are shown in Figure 9-1 as diamonds. These modifiers are connected by dashed lines to those reservoirs upon which they are likely to have the most significant impact.

The difference between an event and a process is the time interval over which the phenomenon occurs relative to the time frame of interest; events occur over relatively short time intervals and processes occur over relatively long time periods. For example, a disruptive seismic event may occur over minutes, hours, or days. Even a volcanic eruptive cycle that may have time frames extending over several years must be considered an event when judged in context with the lifespan of humans and/or a million-year repository assessment period. Phenomena that exceed human life expectancy or occur over a significant portion of the period of regulatory concern are considered to be processes.

To be comprehensive and ensure maximum public protection, a standard must also consider 1) release pathways other than ground water and 2) improbable conditions that may lead to individual doses and risks well in excess of those specified for the undisturbed repository. A demonstration of compliance with such a standard, therefore, requires dose estimates in behalf of secondary release pathways and the prediction of improbable events and processes that may disturb the repository and their corresponding outcomes.

This chapter summarizes issues involved in developing a repository standard that addresses gaseous releases and improbable phenomena that have the potential for disturbing the disposal site.

9.2 GASEOUS RELEASES: A SECONDARY PATHWAY FOR HUMAN EXPOSURE

As previously noted, the primary pathway of radionuclide releases from an undisturbed repository involves the introduction of radionuclides into the underlying aquifer to wells where water might be used for purposes of drinking or agricultural irrigation. However, as shown in Figure 9-1, humans could also be exposed to radiation as a result of gaseous emissions from repository. Due to the ease with which gaseous contaminants are distributed in the atmosphere, human exposure is not limited to the nearfield population but extends to the world at large. The radionuclide with the highest potential for gaseous release and human exposure is carbon-14.

This section provides a brief overview of primary parameters that affect the timing and magnitude of gaseous releases, and additionally assesses bounding values for human doses.

9.2.1 Production and Early Containment of Carbon-14

Carbon-14 is produced in nuclear fuel as a result of neutron absorption by the following reactions: 1) N-14 (n, p) C-14 and 2) O-17 (n, He-4) C-14 (DAV79). Thus, the quantity of C-14 produced in fuel is governed by the amount of nitrogen and oxygen contained within the fuel core. The nitrogen content of LWR oxide fuel is variable, with an average value at about 25 μg/g (DAV79). The quantity of O-17 is relatively constant and is based on the stoichiometric relative abundance of O-17 to normal oxygen in uranium oxide (UO_2) fuel.

Carbon-14 in oxide fuels is assumed to exist as either CO_2 or low molecular weight hydrocarbons that in time are oxidized to CO_2. The total inventory of C-14 for the 63,000 tons of spent nuclear fuel is estimated to be 91,000 Ci (DOE94).

When estimating repository gaseous releases through the unsaturated zone to the accessible environment, parameters requiring consideration are 1) container performance and 2) the bulk permeability and retardation capability of the tuffs. The retardation of gaseous CO_2 flow is due to its exchange with the relatively immobile bicarbonate (HCO_3^-) in the pore water of the unsaturated zone.

Several waste-package designs have been considered by the DOE. All consist of a cylindrical metal container into which the waste is sealed with a gas-tight closure. The Site Characterization Plan (SCP) (SNL87) specified corrosion-resistant stainless-steel containers holding the equivalent of about 2 metric tonnes of nuclear fuel. Alternative waste-package designs under consideration would accommodate between 7 and 9 metric tonnes.

Another outstanding issue affecting container performance (and C-14 movement in the unsaturated zone) pertains to the emplacement of waste with regard to orientation and spacing between waste containers.

The option exists for placing containers in either vertical boreholes drilled in the floor of the repository emplacement drifts or horizontally in boreholes drilled into the walls of the drifts.

Because waste packages generate heat at the time of emplacement, their spacing within the repository determines the local areal power density (LAPD - given in kilowatts/acre). LAPD values are controlled by increasing or decreasing the spacing among nearest neighbor waste packages. Currently, the two thermal loadings, 57 kW/acre and 114 kW/acre, and the two waste-package designs produce four repository emplacement alternatives that have significant impacts on repository performance, inclusive of gaseous releases.

9.2.2 Impacts of Thermal Loading on Gaseous Releases and Transport

The emplacement configuration of heat-generating waste containers is likely to disturb the ambient environment of the repository in a number of ways. Waste-generated heat is expected to enhance vaporization of water within the tuff matrix and, at temperatures above 96°C, completely "dry out" the adjacent host rock. However, the fate of the displaced water is not clear. Since the buoyancy of heated water vapor would cause it to move upward, much of the water could exit the mountain through fractures, fissures, or fumaroles. In addition to the thermally displaced water, such a dry-out volume may also exclude any additional influx of surface water during the period of sustained heat generation.

Alternatively, vapor at the fringes of the dry-out zone could condense. Under conditions of condensation, convection cells could form, where condensed water is pulled back down toward the repository by gravity and capillary action only to be vaporized again. Pruess and Tsang (PRU93) and Buschek and Nitao (BUS93) predicted that a significant amount of water could be expected to be "refluxed" in fractures above the dry-out zone forming a hydrothermal umbrella. If large volumes of water are held in the condensation cap, the potential exists for pulsed influx through the dry-out zone and reentry into the repository in areas of lowest thermal output.

The impact of thermally displaced water in the vicinity of the repository has a dual affect on gaseous releases. Since most of the waste-container corrosion processes are known to be temperature and moisture dependent inclusive of 1) general aqueous corrosion, 2) steam corrosion, 3) pitting corrosion, 4) dry oxidation corrosion, and 5) stress corrosion, the potential impacts of waste emplacement and thermal loading on container failure are highly critical for modeling the time and release fraction of gaseous C-14.

Estimates of travel time for C-14 released from a container into the unsaturated zone are strongly affected by the moisture content. Under conditions of 100% humidity, C-14 is assumed to exist, the bulk of the time, as bicarbonate (HCO_3^-) in the slow-moving aqueous phase (ROS93). Conversely, within the dry-out zone, C-14 can be assumed to exist almost exclusively in the fast-moving gaseous form (CO_2).

In addition, repository heat can also affect chemical interactions with radionuclides that have been released from containers into the repository environs. Alterations of the zeolites could reduce the sorptive properties of minerals. It is also thought that the movement of condensed hot water could dissolve the faces of fractures, causing them to close under the pressure of the overburden. Equally, precipitation of dissolved minerals upon cooling could seal matrix fractures and limit movement (DOE94). In summary, the thermo-mechanical/chemical impacts associated with thermal loading, container integrity, and gas-flow transport are complex. At this time, there is a lack of consensus regarding their overall impact on repository performance. While it is generally assumed that a repository dry out has an overall positive affect in retarding container failure, potential scenarios exist that suggest otherwise. Equally uncertain is the impact of thermal loading on travel times of C-14 within the dry-out zone, zones exhibiting high thermal gradients, and zones containing large quantities of displaced/condensed water.

9.2.3 Estimates of Travel Time

Estimates of travel time for C-14 released from a failed container to the accessible environment are complicated by the fact that the radionuclide is likely to exist only a small portion of the time in gaseous form as $^{14}CO_2$, and the bulk of the time as bicarbonate (HCO_3^-) in the aqueous phase. In the bicarbonate form, C-14 moves more slowly than in the fast moving uncondensible gaseous form (ROS93). This "slowing," or retardation, must be incorporated into the travel-time calculations by dividing the short lived gas velocity at each point along the flow path by a retardation factor that accounts for the longer time and limited movement of C-14 in the aqueous bicarbonate phase. Travel-time probability distributions can be determined by coupled calculations of gas and heat flow (i.e., time-dependent temperature distributions in the repository environs).

Because estimates of early waste-container failure and release of C-14 are currently highly uncertain, travel times for release of C-14 have been estimated at 1,000-year intervals following waste emplacement. For example, Figures 9-2 and 9-3 show travel-time histograms for 57 kW/acre and welded-tuff bulk permeability of 10^{-11} m^2 at 1,000 and 10,000 years (DOE94):

- At the early time of 1,000 years, temperature gradients in the vicinity of the repository are high due to the large heat output. Correspondingly, gas velocities in the nearfield ("dry-out zone") are larger than in the far field. Calculated C-14 travel times range from 200 to 600 years.
- At 10,000 years, heat has been conducted outward and temperature gradients have been reduced, resulting in estimated travel times that range from 500 to 1,200 years.

From the foregoing, it is obvious that the magnitude of potential atmospheric releases is greatest if containment failure were to occur early. At early times, transport velocities can be expected to be maximal, and the reduction of C-14 by natural decay is minimal.

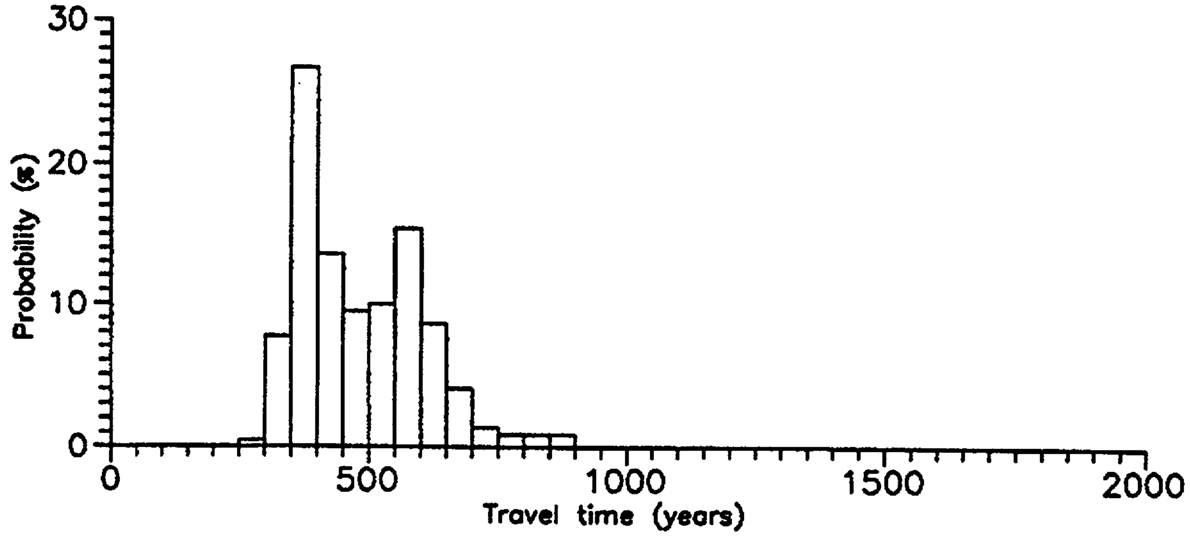

Figure 9-2. Retarded travel times of C-14 from the repository to the atmosphere for particles released at 1,000 years. Welded-tuff bulk permeability of 10^{-11} m^2.

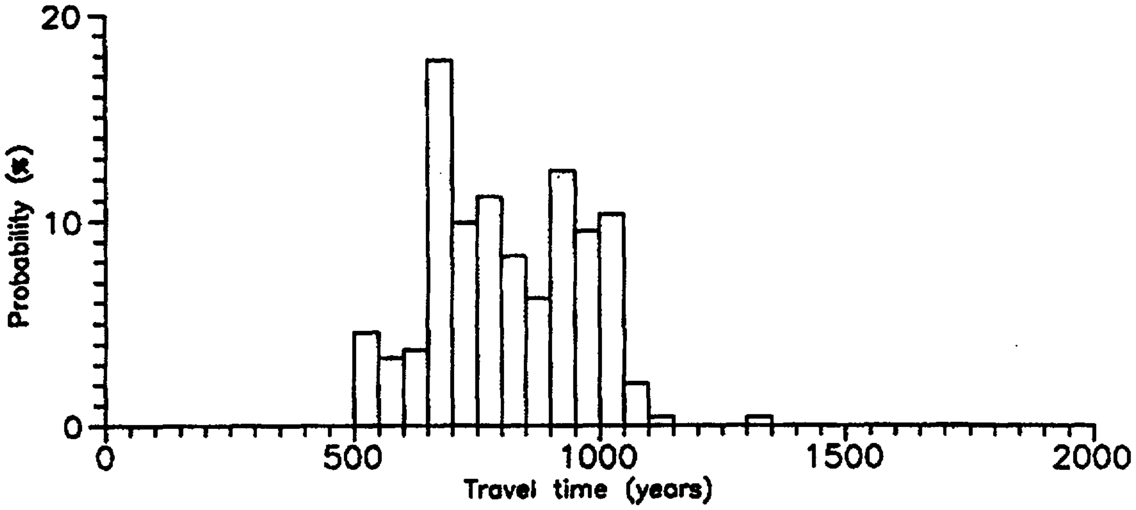

Figure 9-3. Retarded travel time of C-14 particles from the repository to the atmosphere for particles released at 10,000 years. Welded-tuff bulk permeability of 10^{-11} m^2.

It is reasonable, however, to expect that individual container failures will occur over a long period of time that would substantially broaden the range of C-14 travel times in the unsaturated zone from as little as 200 years to as much as 1,800 years. The period of C-14 release into the accessible environment is further delayed by the fact that, at time of canister failure, only a small percentage of the C-14 inventory has leaked from the fuel matrix into the void spaces of the container for instantaneous release. Barnard et al. (BAR92) estimate that this quick release fraction is likely to represent between 1.25 and 5.75 percent of the total inventory. The slow release of the larger remaining fraction of C-14 from the fuel matrix to the container and into the repository environs following container failure is likely to further extend the time period during which C-14 can be expected to enter the accessible environment. With a physical half-life of 5,730 years, it is reasonable to conclude that the quantity of C-14 that will reach the accessible environment will be substantially less than the 91,000 Ci inventory existing at the time of waste emplacement.

9.2.4 Distribution of C-14 in the Biosphere

The gaseous releases of C-14 from the repository will, in time, distribute itself globally and assume a near-equilibrium condition among the various reservoirs of exchangeable carbon within the biosphere. These reservoirs already contain large quantities of C-14.

Carbon-14 is produced naturally by cosmic ray interaction in the upper atmosphere. The reaction of cosmogenic neutrons with N-14 yields on average 38,000 Ci of C-14 annually to a global inventory of 3.8 million curies in the atmosphere (UNS77). This natural source has been augmented by man-made sources. The principal anthropogenic source comes from the detonation of thermonuclear devices, which is estimated to have introduced about 9.6 million curies of C-14 into the atmosphere.

Both cosmogenic and anthropogenic C-14 are readily oxidized to produce carbon monoxide and subsequently carbon dioxide (HAG65; TEL71). Thus, virtually all atmospheric carbon exists in the form of carbon dioxide. Atmospheric CO_2 enters the carbon cycle of exchangeable carbon reservoirs through plant photosynthesis, a process by which plants in the presence of sunlight convert CO_2 to chlorophyll (i.e., $nH_2O + nCO_2 \rightarrow \text{sunlight} \rightarrow (CH_2O)_n + nO_2$). Chlorophyll is a major component of the global food chain of terrestrial and marine species. In time, organified carbon in plant and animal matter is again catabolized to yield CO_2 and H_2O. Under steady-state conditions, the specific activity of C-14 per gram of total carbon strives toward a common value in all major reservoirs with exchangeable carbon. These reservoirs include (1) the atmosphere, (2) the terrestrial biosphere, and (3) the ocean-surface layer and marine life.

Currently, the distribution and specific activity of atmospheric C-14 are in states of flux that result from two independent mechanisms. Firstly, the episodic introduction of C-14 from thermonuclear detonations into the atmosphere is steadily being assimilated into the other major exchangeable carbon reservoirs. Excess atmospheric carbon is being absorbed into the marine environment and terrestrial biosphere with a half-time disappearance of about six years (NCR93). Secondly, the specific activity of the atmospheric carbon is being reduced by the combustion of fossil fuel and its associated release of CO_2 that has been depleted of C-14 through natural decay.

Starting with the cosmogenic background concentration of about 6 pCi/g of atmospheric carbon, the episodic introduction of 9.6 million curies of C-14 from nuclear weapons testing during the decades of the 1950s and 1960 caused specific activity levels to rise to a peak level of nearly 11 pCi/g carbon. Currently, the atmospheric specific activity level has declined to an estimated level of 7 pCi/g carbon.

9.2.5 Dose Modeling and Exposure Estimates

For practical reasons, estimates of human exposure to C-14 assume that the specific activity of C-14 in the atmosphere in gaseous CO_2 form is equal to that of organically bound carbon contained in all plant and animal products that may be ingested as food. Thus, pathways for internal exposure may involve inhalation and ingestion.

For all practical purposes, under a steady-state distribution of C-14 in the environment, the inhalation of $^{14}CO_2$ contributes insignificantly when compared to the ingestion pathway and may, therefore, be eliminated from dose consideration.

Upon the ingestion of organically bound C-14, the uptake, retention, and excretion by the body involve numerous pathways that correspond to biologic half-times ranging from less than one hour to several years. Even for a specific category of organic molecules such as proteins, turnover times are highly variable. While structural proteins show relatively long turnover times, other proteins such as enzymes, plasma albumin, and hemoglobin have relatively short turnover times. When all protein compartments are considered, the half-time of carbon (and therefore C-14) is estimated to be 119 days (NCR93). For fats, which are largely stored in the body as adipose tissue, the biological half-time of carbon is estimated to be 99 days; and for carbohydrates, the half-time is estimated to be one day. For a daily dietary intake of 300 g of carbon, a weighted biologic half-time of about 39 days is obtained. For dosimetric purposes, the ICRP has suggested a biological half-time of 40 days for C-14 (ICR82).

For steady-state environmental conditions, estimates of individual organ and whole body doses from ingestion have been derived by Killough and Rohwer (KIL78). Their model assumes that the specific activity of C-14 (i.e., pCi/g carbon) in the human body will, in time, be the same as that observed in environmental media, inclusive of all plant and animal food products. Correspondingly, the model takes into account the carbon content of individual tissues and organs that will be subject to the beta-ray exposure of C-14. At the present specific activity of C-14 in the atmosphere of 7 pCi/g carbon, C-14 is estimated to contribute an annual dose of about 1.5 mrem to humans throughout the world.

9.2.6 Dose Estimates from Repository Releases

Global Doses. Gaseous release of C-14 from the proposed Yucca Mountain repository will disperse itself globally and, therefore, lead to relatively constant exposures among individuals within the world community. The global distribution model yields a population dose estimate of 399 person-rem per curie of C-14 in behalf of a world population of 12.2 billion over a 10,000-year period (EPA96). Using this dose-conversion factor and conservatively assuming that the entire repository inventory of 91,000 Ci of C-14 is released, an average individual dose of about 0.0003 mrem/yr is estimated. In relationship to the current annual dose of 1.5 mrem from C-14, this potential incremental dose has been considered negligible (NAS95).

Local Doses. Radioactive carbon dioxide released from a localized source diffuses into the atmosphere and is diluted by convection, diffusion, and atmospheric turbulence. Before extensive mixing, however, the effluent plume at ground level may contribute to elevated specific activity levels of C-14 in local flora and fauna. For a ground-level release and applying conservative model parameters, specific activity levels of C-14 in local biota were estimated to reach levels up to three-fold of that expected from the global average (KIL78).

For the Yucca Mountain site, local doses from the gaseous release of C-14 may be estimated by standard transport and diffusion models. Model parameters must consider site-specific meteorological data, C-14 travel times and release rates, and source area. As discussed above, the travel time, release rate, and source area are closely linked to the waste emplacement configuration and thermal loading. Additionally, an assessment of C-14 releases should also address spike releases associated with the fraction of the C-14 inventory located between the inner wall of the waste container and the spent fuel. This quick release fraction may represent 1.25 to 5.75 percent of the total inventory that instantaneously escapes from the container upon failure.

9.2.7 Potential Impacts of C-14 that are Non-Radiological

A frequently voiced concern uniquely associated with some contaminant radionuclides involves the transmutation effect and its potential for inducing molecular disorientation. The potential impact of chemical transmutation is of particular concern for genetic macro-molecules of DNA and RNA. Chemical transmutation refers to the fact that when a radioactive isotope emits a beta particle, it also undergoes chemical transformation due to the change in atomic number. For example, when C-14 undergoes radioactive decay, it becomes nitrogen. When such atoms are incorporated in critical molecules such as DNA, the resulting change in atomic number, recoil, or excitation may give rise to biologic effects, including mutation, beyond those induced by the attendant ionizing radiation. At issue, therefore, is whether or not dose-response values, involving cytogenetic/genetic effects for absorbed radiation energy, might underestimate the hazards presented by these potential radionuclide contaminants. Potential impacts of transmutation have been reviewed by the National Academy of Sciences. In their first report, the NAS BEIR I Committee concluded (NAS72):

> *. . . that the genetic effects of decays of H-3, C-14, and P-32 can, in fact, be attributed almost entirely to their beta radiation and that the contribution from transmutation is so small in comparison that it is justified to consider the main effect to come from the radiation emitted when the isotope disintegrates.*

However, in the Committee's subsequent report (BEIR III), evidence was acknowledged which indicated a modest transmutation effect when C-14 (and H-3) occupied highly specific locations within DNA (NAS80). Although the Committee concluded that it still seems unlikely that neither H-3 nor C-14 decay are significantly underestimated by considering only the ionizing radiation dose accumulated by germ-line cells, a thorough review of the literature may, at some future date, be warranted to determine if the conclusions stated in 1980 can be validated by scientific data available at the time of the Yucca Mountain total system performance assessment.

9.3 DEVELOPMENT OF SUMMARY SCENARIOS

9.3.1 Identification of Improbable Phenomena

For a regulatory time frame that can extend to one million years, it is reasonable to conceive of circumstances defined by various natural and human-induced events and processes that may result in some persons at some time being exposed to levels well in excess of anticipated levels considered acceptable for an undisturbed repository. In recognition of the need to address repository performance under disturbed conditions, the NAS Committee on Technical Bases for Yucca Mountain Standards stated the following:

> *. . . the probabilities and consequences of modifications by climate change, seismic activity, and volcanic eruptions at Yucca Mountain are sufficiently boundable that these factors can be included in performance assessments that extend over this time frame. . . . The challenge [therefore] is to define a standard that specifies a high level of protection but that does not rule out an adequately sited and well-designed repository because of highly improbable events.* (NAS95) (Emphasis added.)

Substantial difficulties are likely to be encountered in making these predictions. Both the NRC and EPA have explicitly recognized that no analyses of compliance will ever constitute an absolute proof; the objective instead is a reasonable level of confidence in analyses that indicates whether limits established by the standard will be exceeded. Thus, in 40 CFR 191 (Appendix B), the EPA stated the following in behalf of a disturbed disposal system:

> *In making these various predictions, it will be appropriate for the implementing agencies to make use of rather complex computational models, analytical theories, and prevalent expert judgement relevant to the numerical predictions. Substantial uncertainties are likely to be encountered in making these predictions. In fact, sole reliance on these numerical predictions to determine compliance may not be appropriate; the implementing agencies may choose to supplement such predictions with qualitative judgement as well.* (EPA85) (Emphasis added)

Similarly, in 10 CFR 60, the NRC acknowledged that for performance assessment " . . . it is not expected that complete assurance that they [performance objectives and criteria] will be met can be presented." (NRC81)

Events and processes that may require consideration are not limited to those identified by the NAS and shown in Figure 9-1. Over the years, numerous reports have identified generic events and processes that do not consider geographical or site-specific features (DOE74, DOE79, BUR80, IAE83, AND89, and DOE90a). Table 9-1 represents a consolidated listing that was used as a starting point in the development of disruptive scenarios for the Waste Isolation Pilot Plant (WIPP).

Table 9-1. Potentially Disruptive Events and Processes (DOE91)

Natural Events and Processes	Human-Induced Events and Processes
Celestial Bodies: Meteorite Impact Surficial Events and Processes: Erosion/Sedimentation Glaciation Pluvial Periods Sea-Level Variations Hurricanes Seiches Tsunamis Regional Subsidence or Uplift Mass Wasting Flooding	Inadvertent Intrusions Explosions Drilling Mining Injection Wells Withdrawal Wells Hydrologic Stresses: Irrigation Damming of Streams and Rivers Repository- and Waste-Induced Events and Processes: Caving and Subsidence Shaft and Borehole Seal Degradation Thermally-Induced Stress Fracturing in Host Rock Excavation-Induced Stress Fracturing in Host Rock Gas Generation Explosions Nuclear Criticality

9.3.2 Screening of Events and Processes

Not all events and processes cited in Table 9-1 need necessarily be considered for Yucca Mountain. Phenomena such as erosion, sedimentation, etc. are certain to occur during the million-year time frame, which suggests that these phenomena are part of the base-case scenario. The effects of other events (e.g., sea-level variations, hurricanes, seiches, and tsunamis) are restricted to coastal areas. To analyze the potential relevance of events and processes to a specific repository site, three criteria must be considered:

1 probability of occurrence,
2 physical reasonableness, and
3 consequence.

To analyze the likelihood of a given event, it is most desirable to express its probability of occurrence in quantitative terms that draw on scientific data. Physical reasonableness as a screening criterion is a qualitative estimate of low probability that reflects subjective judgement. For subjective probability, the ICRP states:

> *. . . a number is assigned to the likelihood of an event occurring in a defined period of time, as a measure of the degree of belief that the event will actually occur during that time . . . The assignment can be made on the sole basis of subjective judgement, no statistical experience being needed. The result is conceptually identical to a traditional probability and can be used in the same way.* (ICR85a) (Emphasis added)

In instances where events are assigned subjective probabilities of occurrence, the ICRP offers an additional note of caution:

> *It is important to distinguish between the degree of belief and the idea of confidence limits applicable to an estimate of probability, which itself has some associated uncertainty.*

The third screening criterion is consequence. An assessment of consequence determines whether the event or process either alone or in combination with other phenomenon may adversely affect performance of the repository.

On the basis of these criteria, a proposed future standard may, for example, specify that events and processes with less than a specified chance of occurring within the regulatory period do not have to be considered in scenarios used to demonstrate compliance with the standard. Conversely, physically reasonable events and processes with significant impacts and probabilities greater than a threshold value would be considered for scenario development. The following illustrates a process for screening of events and processes.

The likelihood of a disruptive event and its consequence must also be defined temporally. For some events (e.g., meteorite impact), the probability of occurrence over time is a constant. For these cases, the probability of events occurring within a year's time interval can be assessed from Poisson statistics. For other types of events, the probability of occurrence will vary with time after repository closure, or it may be co-dependent on the occurrence of other time-dependent events. This second and more complex event scenario is described in ICRP Publication 46 (ICR85b) and is illustrated in Figure 9-4. For this type of event-induced scenario, the probabilistic annual individual dose rate is a function of both the time of occurrence of the initiating event, t, and the time elapsed since its occurrence, (T-t).

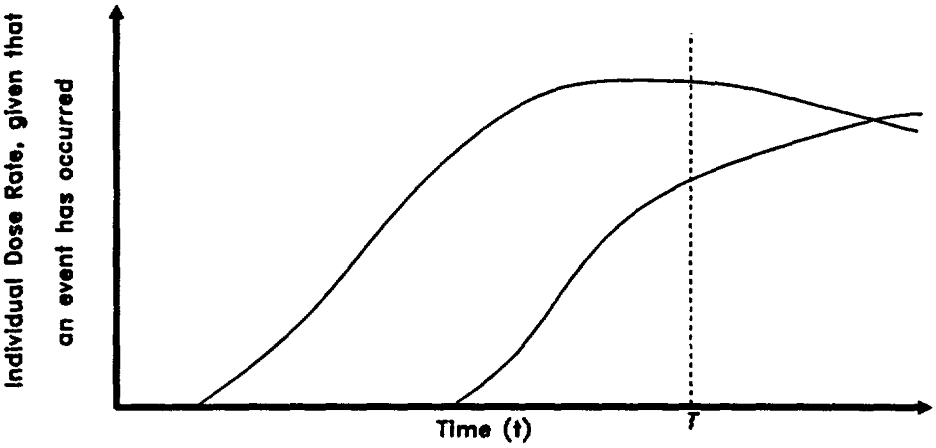

Figure 9-4. An Illustration of Hypothetical Individual Dose Rates Associated with a Disruptive Event Happening at Two Different Times after Disposal of Radioactive Waste

9.4 ASSESSMENT OF COMPLIANCE WITH A STANDARD

9.4.1 Construction of Criterion Curves

The occurrence of a given event or process at a repository site does not preclude the occurrence (or probability of occurrence) of other events and/or processes. For a complete performance assessment, all probable events and processes that may contribute to the release of radionuclides (and individual exposures) for each time interval must, therefore, be considered.

In Appendix B of 40 CFR 191, the EPA states:

> *. . . whenever practicable, the implementing agency will assemble all of the results of the performance assessments to determine compliance . . . into a 'complementary cumulative distribution function' that indicates the probability of exceeding various levels of cumulative release. When the uncertainties in parameters are considered in a performance assessment, the effects of the uncertainties considered can be incorporated into a single such distribution function . . . .*(EPA85)

For compliance purposes, a performance assessment may be constructed in which the complementary cumulative distribution function (CCDF) is represented by the mean CCDF.

To construct a mean CCDF (as well as other CCDFs, such as 5th, 50th, 95th percentile), four criteria have been identified (DOE91):

1. the set of scenarios analyzed must describe all reasonably possible future states of the disposal system,
2. the scenarios in the analyses should be mutually exclusive so that radionuclide releases and probabilities of occurrence can be conveniently associated with specific scenarios,
3. the cumulative releases of radionuclides (consequences) for each scenario must be estimated, and
4. the probability of occurrence of each scenario must be estimated.

A common methodology for developing a CCDF may involve the use of "event trees" (DOE83). An event tree is an inductive logic method that identifies all possible outcomes of a given initiating event. Analyses of this type are commonly used to assess potential accidents at nuclear power plants (NRC90). While suitable for power plant accident analyses in which events and processes are compressed in time, event trees were found to be difficult for summary scenario development that produced reasonable numbers of mutually exclusive, well defined scenario outcomes that could be analyzed probabilistically (DOE90b).

A preferred approach for developing summary scenarios and a CCDF involves the use of "logic diagrams" as described in SAND80-1429 (DOE90a). Logic diagrams are similar to event trees but are not defined by a strict temporal relationship between events and processes. Following the initiating event, at each junction within the diagram, a yes/no decision determines whether the next event or process is added to the scenario. As such, logic diagrams produce many more scenarios since all possible combinations of events and processes are developed and all scenarios are viewed as mutually exclusive.

Based on the aforementioned criteria of probability of occurrence, reasonableness, and consequences, scenarios defined by logic diagrams are collated to yield the mean CCDF.

9.4.2 ICRP Use of Criterion

The ICRP recommends the use of a criterion curve in setting nuclear safety standards pertaining to probabilistic events (ICR85a). Such a standard uses, as its criterion, a biological endpoint that the standard intends to control, such as the risk of inducing a fatal cancer or the closely related parameter of an annual dose limit. Figure 9-5 illustrates a criterion curve in which the maximum probability (i.e., $P \approx 1$) represents the base-case condition of gradual release by an undisturbed repository and corresponds to an annual dose rate of a few to tens of millirem. Salient features of the criterion curve include the following:

- a probability of one for annual doses of up to tens of mrem,
- a region of inverse proportionality between probability of occurrence and individual doses resulting in stochastic health effects,
- a non-proportional region of the dose range, which marks the transition between stochastic and non-stochastic health effects, and
- a constant probability for doses that are lethal.

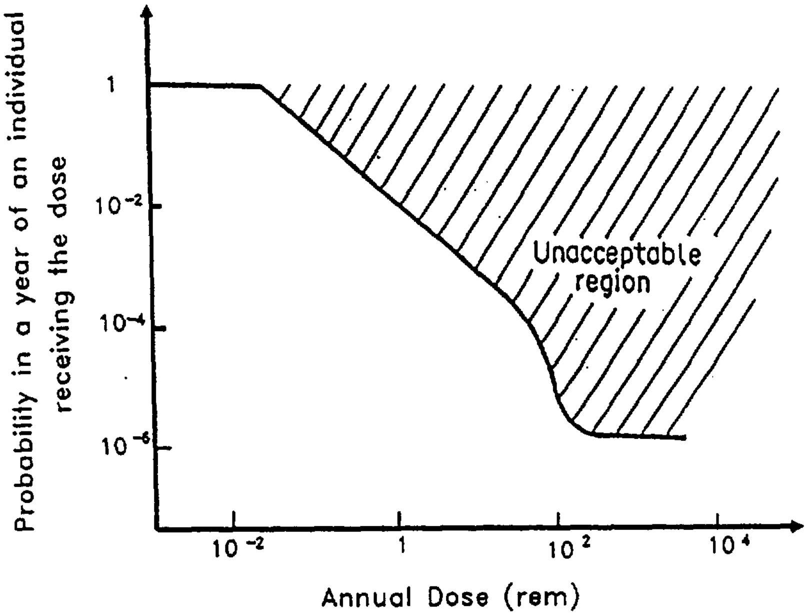

Figure 9-5. Sample Criterion Curve that Defines Acceptable Dose Based on Probability to a Member of the Critical Group (ICR85a)

For the range of doses in which only stochastic effects occur, the relationship between probability and dose is inversely linear, with values representing the product of the probability of the doses from all contributing scenarios and the probability of a health effect per unit dose. In the non-proportional region, where individual doses are in the hundreds of rems, the shape of the criterion curve approximates a sigmoid relationship that takes into account the increasing probability of death. In the lethal dose range, the probability is constant because the consequence to the individual is the same regardless of the dose received.

A criterion curve similar to the one proposed by the ICRP can be incorporated into a standard in which the probability of occurrence of all possible events and their consequent doses (or risks) to individual members of the critical group are numerically specified. Compliance with a criterion curve would require the implementing agency to demonstrate with reasonable assurance that all points of a CCDF fall within the acceptable region of the criterion curve.

9.4.3 Using an Expected Value to Determine Compliance

A total system performance assessment employs a quantitative approach that projects the disposal system. Key questions that must be addressed in the performance assessment are: "How reliable are the models employed in the performance assessment?" and "What is the uncertainty in the results of the performance assessment?" Preceding portions of this document have acknowledged uncertainties associated with all major elements affecting repository performance. Important sources of uncertainties pertain to the appropriateness of scenario selection far into the future; the variability and/or lack of knowledge regarding many parameter values employed by the models; the reliability of historical data in predicting the time-dependent probability of future events that may disrupt the repository; and the complex but uncertain interaction of independent variables on repository performance. While the uncertainty for some of the sources can be reasonably quantified (e.g., quantities of food and water ingested by humans), others are considerably more difficult (e.g., the probability of human intrusion). While there are no rigorous techniques for quantifying or eliminating uncertainties, several techniques for mitigating their impacts have been proposed by Bertram-Howery and Hunter (BER89), as summarized in Table 9-2.

Table 9-2. Techniques for Quantifying or Reducing Uncertainty in the Performance Assessment

Type of Uncertainty	Technique for Assessing or Reducing Uncertainty
Scenarios (Completeness, Logic, and Probabilities)	Expert Judgment and Peer Review Quality Assurance
Conceptual Models	Expert Judgement and Peer Review Sensitivity Analysis Uncertainty Analysis Quality Assurance
Computer Models	Expert Judgment and Peer Review Verification and Validation* Sensitivity Analysis Quality Assurance
Parameter Values and Variability	Expert Judgement and Peer Review Data-Collection Programs Sampling Techniques Sensitivity Analysis Uncertainty Analysis Quality Assurance

* to the extent possible.
Source: BER89

The EPA has acknowledged that performance assessments will contain uncertainties and that many of these uncertainties cannot be eliminated. Accordingly, the EPA has previously stated that:

> *. . . standards must accommodate large uncertainties, including uncertainties in our current knowledge about disposal system behavior and the inherent uncertainties regarding the distant future.* (EPA85)

Uncertainty and sensitivity analyses are, therefore, important aspects of performance assessment. Uncertainty analysis involves determining the uncertainty in model projections that results from imprecisely known (or variable) model input parameters; and sensitivity analysis involves determining the contribution of individual input parameters to the uncertainty in model predictions.

Although several alternatives are available, the preferred approach to uncertainty and sensitivity analysis is the Monte Carlo method. A Monte Carlo analysis is based on the use of a probabilistic method for selecting parameter values from a known or assumed distribution of values. For each iteration, the Monte Carlo algorithm samples a complete set of randomly selected inputs to calculate the aggregate measurement of interest (i.e., dose or risk). When a sufficient number of iterations have been performed, measurement estimates may then be ordered from highest to lowest to produce a distribution of results that can be characterized like any other distribution by its mean, median, and standard deviation. Because Monte Carlo analyses commonly employ the full range of values for each input parameter, it is recognized that the analysis produces unreliable values at either extreme of the distribution. For this reason, values below the 5th and above the 95th percentile are commonly excluded from consideration.

Because of the many uncertainties associated with the events and processes affecting repository performance, probability distributions of human exposure (and risk) are likely to vary over several orders of magnitude within the 5th and 95th percentile range. An important limitation of such a probability distribution is that no single value is correct in predicting future exposures. The probability distribution, however, does identify mean and median values, which represent expected values of dose (or risk) most likely to be received by individuals considered at maximum risk. To that extent, the EPA (EPA85) has previously acknowledged that the most probable (or expected) value of a probabilistic distribution of estimated radiation exposure may be used to demonstrate compliance:

> *. . . the implementing agencies need not require that a very large percentage of the range of estimated radiation exposures . . . fall below limits [of the standard]. The Agency assumes that compliance can be determined based on best estimate predictions (e.g. the mean or the median of the appropriate distribution, whichever is higher).*

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