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TECHNICAL SUPPORT FOR GEIS: RADIOACTIVE WASTE
ISOLATION IN GEOLOGIC FORMATIONS

Volume 1, Executive Summary

April 1978

Work Performed Under Contract No. W-7405-ENG-26

Science Applications, Inc.
Oak Ridge, Tennessee

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RADIOACTIVE WASTE ISOLATION
IN GEOLOGIC FORMATIONS

Volume 1

Executive Summary

April 1978

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TECHNICAL SUPPORT FOR GEIS:
RADIOACTIVE WASTE ISOLATION
IN GEOLOGIC FORMATIONS

Volume No.	Volume Title	Prepared by*
TM-36/1	Executive Summary	SAI
TM-36/2	Commercial Waste Forms, Packaging and Projections for Preconceptual Repository Design Studies	SAI
TM-36/3	Stratigraphies of Salt, Granite, Shale, and Basalt	D&M
TM-36/4	Baseline Rock Properties-Salt	D&M
TM-36/5	Baseline Rock Properties-Granite	D&M
TM-36/6	Baseline Rock Properties-Shale	D&M
TM-36/7	Baseline Rock Properties-Basalt	D&M
TM-36/8	Repository Preconceptual Design Studies: Salt	PBQD
TM-36/9	Drawings for Repository Preconceptual Design Studies: Salt	PBQD
TM-36/10	Repository Preconceptual Design Studies: Granite	PBQD
TM-36/11	Drawings for Repository Preconceptual Design Studies: Granite	PBQD
TM-36/12	Repository Preconceptual Design Studies: Shale	PBQD
TM-36/13	Drawings for Repository Preconceptual Design Studies: Shale	PBQD
TM-36/14	Repository Preconceptual Design Studies: Basalt	PBQD
TM-36/15	Drawings for Repository Preconceptual Design Studies: Basalt	PBQD
TM-36/16	Repository Preconceptual Design Studies: BPNL Waste Forms in Salt	PBQD
TM-36/17	Drawings for Repository Preconceptual Design Studies: BPNL Waste Forms in Salt	PBQD
TM-36/18	Facility Construction Feasibility and Costs by Rock Type	PBQD
TM-36/19	Thermal Analyses	SAI
TM-36/20	Thermomechanical Stress Analysis and Development of Thermal Loading Guidelines	D&M
TM-36/21	Ground Water Movement and Nuclide Transport	D&M
TM-36/22	Nuclear Considerations for Repository Design	SAI
TM-36/23	Environmental Effluent Analyses	SAI

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ABSTRACT

This volume, Volume 1, "Executive Summary," is the first of a 23-volume series, "Technical Support for GEIS: Radioactive Waste Isolation in Geologic Formations," Y/OWI/TM-36, which supplements the "Contribution to Draft Generic Environmental Impact Statement on Commercial Waste Management: Radioactive Waste Isolation in Geologic Formations," Y/OWI/TM-44. The series provides a more complete technical basis for the preconceptual designs, resource requirements, and environmental source terms associated with isolating commercial LWR wastes in underground repositories in salt, granite, shale, and basalt. Wastes are considered from three fuel cycles: uranium and plutonium recycling, no recycling of spent fuel, and uranium-only recycling.

This volume briefly describes the Alternate Repository Preconceptual Design Studies that led to the generic repository designs and the determination of the repository's resource requirements and environmental effluents.

PREFACE

Project Background

One of the major problems related to the production of electricity by light-water nuclear reactors is the management of radioactive wastes generated by the use of nuclear fuel. However, the subject is considered amenable to a rational solution, and the technology involved is considered to be well within the capabilities of our present-day technological base.

An important step toward the realization of an effective waste management program is the preparation of the generic environmental impact statement for commercial waste management. A pivotal issue in waste management is the means of providing long-term, permanent storage of these wastes in a manner that best assures their isolation from the biosphere. Analyses spanning two decades have generated the widely supported concept for providing final isolation of these nuclear wastes in deep geologic formations. Therefore, the Office of Waste Isolation* was assigned the responsibility for preparation of the sections of this generic statement that examine deep geologic waste isolation.

The original concept for deep geologic disposal was first advanced in 1957 when a National Research Council Advisory Committee of the National Academy of Sciences suggested the burial of solid radioactive wastes in salt deposits. To date, the majority of the research, development, and demonstration (RD&D) activities have been in salt. The current United States Department of Energy (DOE) National Waste Terminal Storage (NWTS) program calls for the selection of two sites overlying suitable salt formations by 1979, followed by the construction and start-up in 1985 of one NRC-licensed repository designed for the permanent disposal and isolation of commercial nuclear wastes in a salt formation at one of these two sites. In addition to this activity in salt, vigorous RD&D programs have been initiated to determine the appropriateness of various hard rocks as host media for a repository.

*Operated by Union Carbide Corporation-Nuclear Division for the Department of Energy.

The deep geologic isolation portion of the generic statement considers repositories located in salt, granite, shale, and basalt. The repositories are designed to handle wastes from the nuclear fuel cycle in which the spent fuel is considered a waste (no reprocessing), or from either of two fuel cycles that include reprocessing--the cycle with uranium and plutonium recycle and that with only uranium recycle. To prepare this contribution, the Office of Waste Isolation contracted with Dames & Moore, Parsons Brinckerhoff Quade & Douglas, Inc., and Science Applications, Inc. In order to prepare this description, generic sites were defined, preconceptual repository designs completed, and resource requirements and effluents from the repositories identified. The preconceptual repository designs for the salt host formation were based on more than two decades of analysis and in-situ experimentation. The data base upon which the repository designs for the non-salt host formations were based is much more sparse, since repository-oriented analyses of these formations have been proceeding only for the last couple of years. Additional analyses were performed for each of the host rocks during the course of these studies. Details of this additional technical work are described within the twenty-three volumes of this report.

Volume Summary

This volume, the Executive Summary, briefly describes the study conclusions (Section 1.0). Section 2.0 of this volume briefly outlines the developments that led to these alternative repository preconceptual design studies, the study objectives and scope, and the major study assumptions. Important aspects of the analyses that led to the generic repository designs, the determination of the repository's resource requirements, and the estimates of the environmental effluents are briefly discussed in Section 3.0.

1.0 CONCLUSIONS

These preconceptual design studies have refined the design of a reprocessing waste repository for a bedded-salt site and have shown that a very similar design may be appropriate for the storage and isolation of spent fuel. Further, similar preconceptual repository designs were developed for alternative host rocks (granite, shale, and basalt). In each case, the designs were for generic sites intended to represent potential repository locations, but not necessarily optimal sites. For each host rock, any specific site could be more or less favorable than the generic conditions for the isolation of nuclear waste, and site-specific analyses will certainly be required.

Major conclusions of these generic engineering studies that supported the development of the repository descriptions and environmental effluent estimates are:

- o Surface facility designs would be very similar for each rock type and nuclear fuel cycle analyzed.
- o No reasons were found to contradict the previously expressed position that repository facilities can be designed in bedded-salt formations which meet occupational safety, stability, as well as short-term and long-term nuclear safety requirements.
- o Repository facilities can probably be designed in the non-salt alternative rock types (granite, shale, and basalt) which meet occupational safety, stability, and short-term nuclear safety requirements.
- o For the non-salt host rocks, a potential long-term safety problem exists associated with groundwater leaching and migration of radionuclides.
- o For the non-salt host rocks, canister heat loads may have to be reduced from those established for salt to avoid

canister and waste-form overheating as a result of the lower thermal conductivities of the non-salt rocks.

- o If ready-retrievability of the waste is to be required for a repository, a major engineering requirement is the establishment of the length of such a period early in the design phase and holding that time constant during the final design phase.
- o Longer-term (20 to 50 years) provisional storage with ready-retrievability from a geologic repository seems technically feasible with reasonable economics.
- o Two major uncertainties in the knowledge of rock behavior as it applies to a waste repository are the thermomechanical interactions in the immediate vicinity of the waste and the thermal-hydrologic interactions in the host rock and its surrounding strata.
- o Effluents from the normal operation of a repository are predicted to meet NRC and EPA standards for similar facilities.
- o Effluents from "worst-case" operating accidents are predicted to be modest; conservative analyses predict concentrations requiring "24-hour NRC notification," but not requiring "immediate NRC notification."
- o Major uncertainties exist in the data base for nuclide leaching from spent fuel or the various waste forms, the groundwater chemistry to expect at repository depths, nuclide solubility in the groundwater, and nuclide transport through the various rocks. These uncertainties are of particular concern for the non-salt host rock designs.

Perhaps the major generic issue needing further analysis is the possibility that, for the non-salt repository sites, thermally induced groundwater flow patterns might allow radionuclide transport into the environment. Even if the most favorable hydrologic conditions exist at

an undisturbed non-salt site, care must be taken to assure that thermally induced hydrologic pathways are not produced that would cause such transport. Preliminary analyses performed during these studies, utilizing conservative assumptions regarding nuclide leaching and solubility and which assume no dilution or dispersion during transport, indicate that a possible problem exists for the conditions specified for the generic non-salt sites. This problem does not exist for the salt repository, because no circulatory groundwater occurs in the salt layer, as evidenced by the existence of the salt bed. Furthermore, it appears unlikely that a thermally induced hydrologic pathway to the salt repository would occur. For example, the probability of fracturing in the impermeable layers between the salt bed and any aquifer in the salt stratigraphy can be controlled by limiting the thermal load of the repository. In both instances, of salt and non-salt, the coupling of the induced heat and potential hydrologic pathways needs further investigation.

All aspects of mine excavation, containerized waste handling and emplacement, interim storage, and retrievability appear to be amenable to engineering solution. Analyses should be performed to determine optimum techniques to assure canister integrity during the retrievable phase of repository operation. The thermal restrictions used during these preconceptual designs, especially those associated with the maximum canister and waste temperatures, should be verified. The values of thermal properties of the various rocks need to be verified in in-situ environments, and the fracturing potential of the various non-salt rocks under the thermal stress of HLW or spent-fuel emplacement need analytical and experimental investigation. These types of analyses, coupled with a better understanding of the heat-removal potential of a repository ventilation system, can lead to acceptable repository designs.

The concept of ready-retrievability assumed in these studies requires that the waste be retrievable for a defined period of time with approximately the same effort and at approximately the same rate at which the waste was placed into the repository. During this period, the repository rooms must remain open to waste removal equipment, and it is assumed that the waste containers must remain integral. Such requirements have considerable impact on repository design and operation and therefore need

to be specified early in the design phase. Retrievability periods of varying lengths can dictate different design approaches; thus, the period must be defined and held constant throughout the final design stages. For example, these studies show that the alternative of 25-year retrievability appears to be achievable through modifications in the engineering design of the reference repository and/or by reducing the thermal load during the extended retrievable period. The optimum approach to retrievability from the various host rocks should be determined through further investigation.

These preconceptual design studies indicate that, for a given rock type, varying the waste form or the canister packaging has little effect on repository design at the preconceptual level. Obviously, a complete understanding of the waste to be received, its packaging, and anticipated receipt rates will be required for a detailed design. These factors will affect all aspects of the repository design, such as the equipment chosen, the waste-handling stream designs, and the operational procedures. The analysis of the alternative waste data set that includes combustible waste forms indicates that the normal preconceptual repository design could prevent unacceptable releases of radioactivity to the environment even in the event of a fire. However, such a fire involving these wastes would contaminate the repository room in which the waste was stored as well as portions of the repository ventilation system, thus causing potentially serious operational problems and delays. If a decision is made to accept combustible wastes at a federal repository, it seems most reasonable to take steps to minimize the receipt of combustible wastes in order to prevent such difficulties.

Effluents from the normal operations of these generic repository designs appear to meet existing NRC and EPA standards for continuous steady-state releases from similar facilities. The predicted maximum release concentrations from the "worst-case" operational accidents are modest; they are estimated to require notification of the regional NRC representative within 24 hours. These predicted releases would not, however, fall into the category requiring immediate notification.*

*Determined by comparing predicted maximum release concentrations with the federal regulations in 10CFR20.403 and the associated allowable concentrations in Appendix B, Table II.

Before detailed repository designs can be accomplished, an indication of federal, state and local requirements that may be imposed on the repositories must be forthcoming. These include potential requirements regarding the safeguarding of nuclear materials, guidelines on acceptable environmental releases, licensing requirements, and such policy-oriented decisions as the length of any retrievability period.

2.0 GENERIC DISCUSSION

Preconceptual repository design studies have been performed in support of the Generic Environmental Impact Statement on Commercial Waste Management being prepared under the auspices of the Department of Energy. This chapter of the Executive Summary describes the events which led to these preconceptual design studies (Section 2.1), the objectives (Section 2.2) and scope (Section 2.3) of the studies, and the baseline assumptions (Section 2.4).

2.1 BACKGROUND

In 1976, Battelle Pacific Northwest Laboratory (BPNL) was directed by the Energy Research and Development Administration (ERDA, which is now the Department of Energy, DOE) to prepare a Generic Environmental Impact Statement (GEIS) for Commercial Waste Management. The Office of Waste Isolation (OWI) was assigned the responsibility of preparing the sections of this statement which examine isolation of commercial wastes in deep geologic formations. These sections specifically describe the repository facilities required to receive, emplace, and store, or isolate, wastes at generic sites for deep geologic repositories, the resources utilized in constructing these facilities, and the effluents generated that may have an environmental impact.

The initial OWI contribution¹ was submitted to BPNL in December 1976. That report described facilities required to handle waste from a nuclear fuel cycle using uranium and plutonium recycle (Total Recycle) and to place these wastes in a generic bedded-salt formation. At the time this initial document was prepared, the fuel cycle incorporating uranium and plutonium recycle was the reference fuel cycle of the commercial nuclear industry, and over two decades of research, development, and demonstration activities pointed to the acceptability of bedded salt as a repository host formation.

Since December 1976, several events have occurred which are having a significant impact on commercial waste management programs. In particular, response to the presidential policy outlined by President Carter in

his National Energy Plan dated April 29, 1977,² has redirected these programs.

This presidential policy defers any U.S. commitment to advanced nuclear technologies that are based on the use of plutonium while the U.S. seeks a better approach to the next generation of nuclear power than is provided by plutonium recycle and the plutonium breeder. President Carter further stated that the U.S. will defer indefinitely commercial reprocessing and recycling of plutonium, as well as the commercial introduction of the plutonium breeder, and concentrate on evaluating alternative breeders, advanced converter reactors, and other fuel cycles, with emphasis on non-proliferation and safety. The commercial waste management programs have been focused toward the spent unprocessed fuel (SURF) cycle in which there is no reprocessing and the spent fuel itself is considered waste.

In response to this policy reorientation, the scope of the generic statement on commercial waste management was expanded. In a meeting on February 7-8, 1977, between representatives of OWI, ERDA-OR, ERDA-RL, ERDA-HQ, and BPNL, OWI was requested to perform an accelerated series of preconceptual repository design studies considering alternative fuel cycles and alternative rock formations in support of the expanded scope.

The initial step in the accelerated series of preconceptual design studies was to modify the December 1976 version of OWI's contribution to the generic statement to describe the isolation of spent unprocessed fuel in bedded salt in a manner comparable with the total recycle waste isolation. To accomplish this, in March 1977 OWI contracted with two firms: the architect-engineer firm of Parsons Brinckerhoff Quade & Douglas, Inc. (PBQD) and the multi-disciplinary firm of Science Applications, Inc. (SAI). This effort resulted in a May 1977 version of OWI's contribution.³

The next stage of the accelerated series of preconceptual design studies was organized to incorporate comparable consideration of the alternative host rocks of granite and shale and the handling of wastes from the fuel cycle in which only uranium is recycled. To accomplish this work, OWI contracted with three firms: (1) PBQD, to generate the preconceptual repository designs and to analyze the costs associated with the various

alternatives; (2) the geotechnical firm of Dames & Moore (D&M), to analyze the geophysical properties of the various host formations of interest for potential geologic isolation of commercial waste; and (3) SAI, to analyze nuclear considerations of the repository designs and determine thermal releases and environmental effluents that may be associated with these facilities under normal operations or accident conditions. In addition, SAI assisted OWI in project planning and organization and in the publication of the OWI generic statement contribution. This effort resulted in the September 1977 version of the OWI contribution.⁴

Subsequently, the final phase of these studies were initiated with all three firms which emphasized the generation of preconceptual repository designs in basalt and analysis of a set of waste forms defined by BPNL. These expanded studies have resulted in the April 1978 version of OWI's contribution to the Generic Environmental Impact Statement on Commercial Waste Management.⁵

The technical work performed by the three contractors during these preconceptual repository design studies is presented in this twenty-three volume report entitled "Technical Support for GEIS: Radioactive Waste Isolation in Geologic Formations." The complete report consists of this Executive Summary and twenty-two topical volumes describing in detail the problems analyzed, technical approach as chosen, assumptions made and results of the analyses, including their integration into the repository design.

2.2 STUDY OBJECTIVES

The primary objective of the preconceptual repository design studies was to develop the technical and economic data required to generate generic designs for a deep geologic repository appropriate for the storage of radioactive wastes from specific alternative fuel cycles in specific alternative geologic media, and to identify the resource requirements and effluents from these repositories. This information is to form the basis for the determination of the environmental impacts of deep geologic waste isolation in the Generic Environmental Impact Statement on Commercial Waste Management.

This primary objective can be divided into the following subordinate objectives:

- o Specify generic sites for each host rock for which the repositories are to be designed.
- o Determine reasonable host-rock properties from available data.
- o Obtain waste form and packaging descriptions and waste projections.
- o Determine acceptable thermal loadings for each of the repository designs.
- o Develop preconceptual repository designs.
- o Assure proper consideration of nuclear aspects in the preconceptual designs.
- o Estimate construction and operation costs.
- o Determine resource requirements and environmental effluents.

In the preconceptual designs, major emphasis was placed on meeting the basic goal of waste management--that is, a dose as low as reasonably achievable (ALARA) to man and his environment. In addition, emphasis was placed on the safety and viability of mining operations, retrievability of the waste forms for a specified period of time, maintainability of the facilities, and minimization of the impact upon the surrounding geologic formations. Economic factors were considered secondary to addressing satisfactorily these other basic design factors.

2.3 SCOPE

The preconceptual repository design studies investigated the storage and isolation of commercial radioactive wastes from each of three light-water reactor (LWR) fuel cycles in each of four host rock formations.

The host rock formations considered include bedded rock salt, crystalline rocks (granite), argillaceous rocks (shale), and volcanic rocks (basalt). The depth of potential host formations was limited to between

one and three thousand feet. Since this is a generic study and specific sites have not been chosen, descriptions of generic sites were developed for the studies rather than using specific site data.

The three LWR fuel cycles considered in the studies are: the reprocessing fuel cycle in which uranium and plutonium are recycled (total recycle), the spent unprocessed fuel (SURF) cycle in which spent fuel is considered a waste, and the reprocessing cycle in which only uranium is recycled (U-only recycle). For each of the reprocessing fuel cycles, the waste forms analyzed include vitrified glass or calcined high-level waste, cladding, intermediate-level transuranic waste and low-level transuranic waste. For the SURF cycle, unprocessed spent PWR and BWR fuel assemblies are received at the repository in canisters, and it is assumed that some low-level waste will be received, which may or may not be transuranic.

A baseline set of waste forms, packaging arrangements, and waste projections was utilized to develop repository designs for wastes from each of the above fuel cycles in each host rock formation. Also, an alternative set of waste data defined by Battelle Pacific Northwest Laboratory (BPNL) for each of the fuel cycles was employed for additional repository designs in the bedded-salt formation.

2.4 MAJOR ASSUMPTIONS

2.4.1 Waste

The waste received at the repository is assumed to be that generated during various light-water reactor fuel cycles. All wastes are assumed to arrive at the repository in suitable stainless steel canisters or carbon steel drums or, in the case of some of the alternative BPNL wastes, steel boxes. All repository designs were based on receipt of solid, non-combustible wastes only; however, special analyses were performed to determine the impact of storing certain forms of combustible wastes as well. If a damaged canister is received, such a canister would be overpacked at the repository. If a canister is excessively contaminated with surface activity upon receipt, it would be decontaminated before handling. No other packaging operations would be required.

The wastes are assumed to arrive at the repository at various ages, that is, various lengths of time after the fuel from which the wastes are generated was discharged from a reactor. For the baseline data set, high-level waste from both recycle cases is assumed to be 10 years old at receipt; and all other waste forms from those cycles are assumed to be 5 years old. Spent fuel from the SURF cycle is assumed to be 10 years old, and low-level waste 1 year old. For the alternative data set, the age of high-level waste upon arrival at the repository is dependent upon availability of reprocessing capacity and can be as young as 6.5 years old; however, the average age at receipt would be about 10 years old during the time required to fill the first repository. The spent fuel in the alternative data set is assumed to be 5.5 years old when it arrives, and all other wastes are assumed to be 5 years old.

The waste generation rates and, thus, the repository waste receipt rates for the baseline case are based on the OWI modification of ERDA's "mid-case" forecast of 1976.⁶ In the modified forecast, 480 GWe nuclear power generating capacity would be on line by the year 2000 A.D. The alternative waste data set is based on 400 GWe capacity by the year 2000 A.D.

2.4.2 Repository Size and Operation

It is assumed that the repository will begin accepting waste late in fiscal year 1985. Construction schedules, in particular, reflect this assumption. The year that construction begins depends on rock type, but in each case receipt operations begin in 1985.

Uncertainty in waste projections, along with the fact that a number of different projections are being proposed by various groups and agencies, led to the approach of choosing a constant subterranean acreage independent of fuel cycle and rock type. Then, given a waste projection data set for a given fuel cycle, the year may be computed in which the reference repository in a given rock type would reach design waste emplacement densities.

The subterranean and surface acreages were chosen to be 2000 and 200 acres respectively. These specific values were assumed because they represent reasonable design values. The subterranean acreage is large enough to accept the waste through the year 2000 or later, depending upon

fuel cycle and rock type. The surface acreage is sufficiently large to contain all of the surface buildings, the railroad and road systems, and the excavated rock from those rooms that would be filled during the retrieval period, yet it is conveniently sized for security and surveillance.

As indicated above, all wastes are assumed to arrive at the repository in appropriate containers. All waste handling is assumed to be remote or from appropriately shielded vehicles. Although, within a given repository design, waste emplacement areas may be at slightly different depths, no waste emplacement area was allowed to overlie another emplacement area.

2.4.3 Waste Retrievability

The assumption of a waste-retrievability concept probably has the greatest impact on repository design and operation. The concept of waste retrievability assumed in these studies requires that the waste be retrievable for a defined period of time with approximately the same effort and at approximately the same rate at which the waste was placed into the repository. During the period of time that the waste is retrievable, the repository rooms must remain open to waste placement and removal equipment, and the waste containers must remain structurally intact. It is not obvious that waste retrievability will be required for an NWTs repository, but such a concept has been assumed in these repository designs. The assumption of a retrievability requirement complicates repository design and operation and increases costs.

Two reasons for requiring waste retrievability can be postulated. First, retrievability of all wastes for a period of time might be required for further verification of deep geologic repository concepts, reactions of the host rock, and facility operational procedures. Engineering considerations dictate that the period of retrievability be fixed prior to detailed engineering design. A period of five years has been selected to define this operational mode in these designs.

A second reason for retrievability stems from the concept of interim storage for waste, which is most applicable for the spent unprocessed fuel (SURF) cycle. In this cycle, it may be desirable to recover the spent fuel assemblies in order to reprocess them at a later date. If this option

is to be maintained for a specified time, engineering design must ensure that each spent-fuel assembly placed in the repository is retrievable for that entire period. A retrieval period of 25 to 100 years might be considered appropriate for the SURF cycle. At least a twenty-five-year period would appear to be required under this concept because of the lead-time necessary to establish a reprocessing capability after a reprocessing decision has been made. In this second concept, the reprocessing waste could possibly be returned to the same repository from which the spent fuel was retrieved. In these studies, twenty-five-year retrievability is analyzed as an alternative design for the SURF cycle.

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6. Office of Waste Isolation/Union Carbide Corporation, Nuclear Division, Nuclear Waste Projections and Source Term Data for FY 1977, Y/OWI/TM-34, Oak Ridge, TN, April 1978.

3.0 TECHNICAL DISCUSSIONS

The scientific and engineering studies that supported OWI's deep geologic contribution to the Generic Environmental Impact Statement on Commercial Waste Management are briefly described in this chapter. For these brief descriptions, these studies have been delineated as follows:

- o Definition of Waste
- o Geotechnical Investigations
- o Baseline Repository Designs
- o Thermal Loading
- o Hydrology and Waste Containment
- o Radioactive Waste Handling
- o Decommissioning, Sealing, and Long-term Monitoring
- o Resource Requirements and Environmental Effluents
- o Alternative Concepts
- o Cost Estimates

Within each brief description, the locations of more detailed discussions are specified. In several cases, these locations are in more than one of the volumes of this support document to the generic statement.

3.1 DEFINITION OF WASTE

The waste-data requirements to support these preconceptual repository designs included: definition of types of wastes, their contents (including radionuclide distribution as a function of time), waste forms, waste and shipping container descriptions, overpack descriptions, number of waste containers per shipping container, surface dose rate and contamination levels, heat generation rate per container and its decay, and annual waste receipt rates. Assumptions in the definition of the baseline waste data set to be received at the repositories were that only solid, non-combustible wastes in appropriate containers would be received.

The baseline set of waste data for these studies was defined by OWI.¹ This data set identifies high-level waste (HLW) for each of the reprocessing cycles, in either a vitrified glass or a calcined form, cladding waste (CW), an intermediate-level transuranic waste (ILW-TRU), and a low-level transuranic waste (LLW-TRU). For the spent unprocessed fuel (SURF) cycle, a single PWR or BWR (spent-fuel assembly) was assumed packaged in a canister and LLW that may or may not be transuranic was defined. The quantities of these wastes were based on an ERDA "mid-case" forecast which was modified to reflect 480 GWe nuclear power generating capacity is on-line by the year 2000 A.D.¹

The original baseline HLW densities of 3 cubic feet per metric ton of heavy metal charged to the reactor (ft^3/MTHM) for the vitrified waste and $1.5 \text{ ft}^3/\text{MTHM}$ for the calcined waste, defined in Reference 6, were satisfactory in bedded salt. However, with these densities and the baseline canister designs, the canister temperature limits set to assure retrievability (Section 3.4) were exceeded in the other rock types. Although any of several techniques might be employed to reduce the canister temperatures attained by storage in these rock types, such as other canister designs, the technique chosen in these studies was to modify the HLW densities. In granite, shale, and basalt, vitrified waste densities of 5, 7 and $7 \text{ ft}^3/\text{MTHM}$ respectively were chosen, (associated calcined waste densities are 3.5, 5.5 and $5.5 \text{ ft}^3/\text{MTHM}$).

The HLW from the recycle of uranium and plutonium was assumed to be a 3:1 mixture of waste from uranium fuel and mixed-oxide fuel (MOX) in the definition of HLW heat generation rates. During repository operation, the ratio of MOX-to-U-fuel never exceeds this ratio; therefore, these assumptions ensure that the canister and repository heat-loading limits will not be exceeded within the repository designs.

In addition to this baseline waste data set, an alternative data set was defined by Battelle Pacific Northwest Laboratories (BPNL) and is presented in Part II of Volume 2 of this report. This alternative data set is based on a scenario in which 400 GWe nuclear power-generating capacity is on-line by the year 2000 A.D. The major differences between this data

set and the baseline data are the definition of a vitrified glass form of HLW only, the specification of several categories of ILW-TRU, a different specification of LLW-TRU, the packaging of ILW-TRU in either larger canisters or in drums, and the packaging of LLW-TRU in either drums or metal boxes. In addition, this data set allowed waste deliveries by truck or rail, whereas the baseline designs considered only rail deliveries.

In the alternative data set, the HLW heat load per canister varied with the year of receipt, reflecting the changing composition of the waste caused by recycling. BPNL specified the HLW density as about $2\text{ft}^3/\text{MTHM}$. As in the baseline data case, in order to accomplish these repository designs, the waste was diluted when the HLW heat load per canister began to exceed that which would generate the limiting canister or waste center-line temperatures.

The larger variety of waste containers and categories of waste, as well as the flexibility of truck or rail deliveries in the alternative data set made the repository designs slightly more complicated and a little more costly, but did not have any impact on the basic concepts of the repository designs and their operation.

3.2 GEOTECHNICAL INVESTIGATIONS

Since a generic environmental impact statement is not for a specific site, and since no sites have been selected for commercial waste repositories, it was decided to define a generic site for each type of host rock being considered. In this case, the term "generic" simply means "not - site-specific." The composition and physical properties of each host formation and its associated stratigraphic column were chosen so as to be representative of average, but not optimum, conditions which might be encountered at possible repository locations within the continental United States. Geotechnical criteria used in selecting the possible locations to be considered in developing the generic data included: a host formation depth between 1000 and 3000 feet below the surface, a host formation thickness of at least 150 to 250 feet, and a host formation lateral extent that is large compared with 2000 acres. The technique used in generating the

generic data was subjective scientific judgment, rather than a rigorous mathematical technique. The approach began with a comprehensive literature search for rock property and stratigraphic data, followed by the selection of representative formations and stratigraphic sections, and then a generation of the generic site description. This approach was employed for each host rock: salt, granite, shale, and basalt.

To specify each generic host rock, both intact and rock-mass data defining mechanical, thermal, hydraulic and mineralogical properties were used. The intact rock properties included: mineralogic composition, density, natural moisture content, porosity, the stress-strain properties represented by Young's modulus and Poisson's ratio, tensile strength and uniaxial compressive strength, and the thermal properties of heat capacity, thermal conductivity, and coefficient of linear thermal expansion. In addition, for the non-salt rocks, horizontal and vertical permeabilities were defined. Similar rock-mass properties were defined, with the exception that for salt, the property of creep was defined through a mathematical relationship.

Details of the stratigraphic data reviewed and the generic stratigraphies developed for each of the four host rocks are presented in Volume 3 of this report. Each of the generic stratigraphies, with the exception of basalt, was developed considering several geologic formations. In the case of basalt, the siting criteria of depth, thickness, and lateral extent limited the data to the Columbia River Flood Basalts in the Northwestern part of the United States.

Details on the intact and rock-mass property data are presented in Volumes 4 through 7 of this report for salt, granite, shale, and basalt respectively.

The studies indicate that an adequate data base exists for most of the rock properties required for preconceptual or conceptual design studies. Tables listing these data are presented in the respective volumes. However, since many of these properties vary widely from site to site, detailed design activities will require site-specific data. In general, state-of-the-art investigative techniques appear to be adequate for obtaining these site-specific data, with the possible exception of data on fractures,

faults, and other discontinuities. These latter data are particularly important for the non-salt rocks: granite, shale, and basalt.

The main inadequacy of the present data base lies in both laboratory testing and longer-term in-situ behavior of the various host rocks when heated and irradiated. Particularly difficult would be the gathering of data on the influence of heat on groundwater movement. In-situ testing programs are presently being performed by the Department of Energy in Conasauga shale in Tennessee and in Gulf Coast region domed salt, or are being planned by the Department of Energy in Nevada shales and in Columbia River flood basalts.

3.3. BASELINE REPOSITORY CONCEPTS

An artist's concept of a deep geologic repository is presented in Figure 3-1. This conceptual layout provides some insight into the integration of surface facilities, mine shafts, and underground mine areas required for a waste repository. The surface facilities for a typical repository design, illustrated in the surface site plan in Figure 3-2, can be constructed on about 150 to 200 acres, and the associated subterranean waste emplacement horizon might span about 2000 acres. This perspective is typical of a repository designed for either spent-fuel-assembly storage/disposal or for disposal of the high-level wastes (HLW), cladding wastes (CW), intermediate-level wastes (ILW), and low-level wastes (LLW) generated by the reprocessing of spent fuel assemblies.

The surface facilities for a typical repository design are characterized by three major structures plus the normal complement of administrative and supporting structures. The major buildings are a canistered waste handling building which features shielded and remote handling facilities for the highly radioactive wastes, a low-level waste handling building designed for contact handling, and a mine operations building. Each of these buildings would have associated with it a mine shaft connecting it to the underground operations. The mine operation building is connected to the men-and-materials shaft, through which all mining and waste emplacement workers would move to and from their underground work stations, and

through which all excavated rock would be moved to or from the surface. The canistered waste and low-level waste buildings are connected to shafts dedicated to the lowering of each respective waste.

A basic consideration in repository subterranean design and operation was that excavation activities and waste-handling activities are physically separated at all times. Movement of the highly radioactive canistered high-level wastes or spent-fuel assemblies would be by a low-speed transporter vehicle. These wastes would be loaded onto the transporter in the immediate vicinity of the canistered waste shaft. The transporter would proceed to the room where the waste is to be placed. The waste canister would then be lowered into a hole or a trench in the floor of the excavated room. In the event that a retrievability mode of operations is selected for the early years of operation, a concrete shielding plug would be lowered into the hole or over the trench above the waste to facilitate subsequent removal of the waste if that should be desired. Otherwise, crushed rock would be poured into the hole or trench above the waste canister. More than adequate radiation shielding would be provided by about seven feet of crushed rock. Palletized drums of low-level wastes would be transported by forklifts to storage rooms in a separate area of the mine and stacked in the open rooms. Upon a decision to terminate the retrievable mode of operations, all filled rooms would be backfilled with crushed rock and as subsequent rooms became filled with waste, they too would be backfilled. During decommissioning of the repository, the remainder of the void spaces at the repository horizon would be backfilled and all shafts sealed.

A representative layout of the subterranean area of a repository is presented in Figure 3-3. This example is for reprocessing wastes from the total recycle case; however, layouts for wastes from the other fuel cycles are similar. The excavation of the subterranean facilities requires mining between 30 and 95 million tons of rock, depending on the fuel cycle and the host rock formation. A summary of the quantities of excavated rock being handled for each of the combinations of fuel cycle and host rock is presented in Table 3-1. In the various designs, up to 60% of the excavated rock is eventually used to backfill the excavated rooms and

corridors. This backfilling process minimizes any possible long-term surface subsidence as the repository region slowly reconsolidates. The higher volumes of granite and basalt being excavated result from the higher extraction ratios allowed for these stronger rocks.

The salt excavations are anticipated to utilize continuous mining machines; consequently, no simultaneous blasting and nuclear waste handling is anticipated onsite. On the other hand, because of the hardness of the non-salt rocks and in order to accomplish timely construction of the non-salt repository designs, the use of conventional drilling and blasting is required. For any of the repository designs evaluated in this study, all excavations could not be completed prior to the first waste arrivals. Therefore, simultaneous blasting and nuclear-waste handling would occur at the non-salt repositories, which reinforces the desire to keep mining activities and nuclear waste handling activities physically separated in these facilities.

The construction aspects of these various facilities do not appear to present any serious difficulties. The 2000-acre emplacement horizon assumed in these studies compares in excavation size to existing large mines in the United States.

The waste capacities of the various repository designs are presented in Tables 3-2 through 3-5 for salt, granite, shale, and basalt respectively. The basic approach in these repository designs was to assume the repository would encompass about a 2000-acre subterranean area, then to determine how much waste could be placed into the repository. The quantity of waste is thus mainly dependent upon the allowed extraction ratio and areal thermal loading for the higher heat-generating wastes. These design parameters are chosen to assure room and pillar stability during the operational phase and no disruption of the formation stability even after decommissioning. Finally, the capacity is somewhat affected by the design decision to allocate the repository space to the various waste types in such a fashion that all waste areas are filled simultaneously. That is, receipt of HLW, ILW, and LLW would terminate at approximately the same time.

The number of containers within a filled repository is only a measure of the number of units handled, but is not a direct measure of the amount of waste in a repository. This lack of direct correlation occurs because of the variations in waste forms and because thermal restrictions dictated different densities for heat-generating waste per HLW container for the various host rocks. A better comparison of the capacities of the various host rock repositories is accomplished by comparing the repository thermal capacities of the heat-generating wastes, or the fuel equivalence of the storage capacities in metric tons of heavy metal charged to the reactor represented by the waste stored. Using this latter comparison technique, Table 3-2 shows that a 2000-acre repository in salt supports about equal amounts of electrical power generation, whether one of the reprocessing cycles is employed or spent fuel is disposed of. On the other hand, Tables 3-3 through 3-5 indicate that for each of the non-salt host rocks, a 2000-acre repository will support almost twice the fuel equivalence in the form of spent fuel assemblies than it will support in the form of HLW. It is further seen that the 2000-acre granite or basalt repository designs support about 20% more generating capacity than the salt repository designs for the reprocessing cycles and about 90% more for the spent fuel cycle. This considerable difference in the spent fuel cycle capacities is due to two factors. First, BWR canisters in the granite and basalt repositories were assumed to be placed in trenches as close as physically possible and yet remain capable of being independently supported, whereas the emplacements in salt and shale repositories were limited to individual holes at least six feet apart. If further analyses determine that trenches could be employed in these latter host rocks, the capacities of these repositories could be increased. Second, the properties of granite and basalt allowed these repositories to be designed for higher heat loads per acre in the spent fuel cycle. In each case, the shale repositories hold the least amount of waste.

The basalt data on repository capacities for the SURF cycle (Cycle II) presented in Table 3-5 is appropriate for a repository preconceptual design in basalt using an allowed thermal loading constrained only by formation stability and room and pillar stability. However, further analyses

after this design was completed (See Volume 19 of this Report.) indicated that the areal thermal loading should be reduced to as low as 115 kW/acre to assure that the spent fuel cladding does not exceed 200°C during storage. This reduction in areal thermal loading would result in a 20% to 30% reduction in the spent fuel capacity of the SURF repository designs in basalt shown in the text and tables of these study reports.

Detailed discussions of these baseline repository designs and the decisions that led to the designs are presented in the following volumes of this report:

- Volume 8 ... Salt Preconceptual Designs
- Volume 9 ... Drawings for Salt Designs
- Volume 10 ... Granite Preconceptual Designs
- Volume 11 ... Drawings for Granite Designs
- Volume 12 ... Shale Preconceptual Designs
- Volume 13 ... Drawings for Shale Designs
- Volume 14 ... Basalt Preconceptual Designs
- Volume 15 ... Drawings for Basalt Designs

In each case, the volume contains design data for each of three fuel cycles.

3.4 THERMAL LOADING

The emplacement of heat-generating radioactive wastes (HLW and spent fuel) in a deep geologic repository gives rise to thermal and thermomechanical phenomena which need special consideration in the design stage. Over the past two decades, thermal analyses and experiments have been conducted on the burial of heat-generating waste in salt.²⁻¹¹ A summary of much of this work and an integration of work presently being conducted is presented in a recent document by Dr. J. E. Russell of OWI.¹² In addition, further detailed analyses are presented in Volume 19 of this report. Analyses and experiments are presently being conducted on waste burial in the various non-salt rocks. Perhaps the most detailed analyses to date are

presented in this report (Volumes 19 and 20). Results are not yet available from in-situ experiments being performed under the waste management program in Louisiana, Tennessee, Nevada, and Washington. In addition, in-situ experiments are being conducted as a joint US-DOE and Swedish program in the strip mine in Sweden.

Two factors summarize the concerns of the impact of the imposed heat: first, the requirement for ready-retrievability of the waste for a pre-determined period of time and, second, measures to foster both waste confinement within the geologic formation and isolation from the biosphere. In these preconceptual design studies, an iterative procedure was utilized to determine the design values for such parameters as the thermal loading of each canister and the areal loading within the repository. As a convenient nomenclature, analyses and results associated with the waste canister itself and within a few feet of the waste canister are called "very-near-field," those associated with the repository horizon (e.g., room and pillar stability) are labeled "near-field," and those associated with the region beyond the repository horizon are called "far-field."

For the isolation of high-level waste, the following steps were followed in the iterative analysis:

- Step 1: Choose thermal and thermomechanical criteria.
- Step 2: Propose a conservative room and pillar design without consideration of an imposed thermal loading.
- Step 3: Make near-field heat-transfer calculations to determine range of areal thermal loading of interest.
- Step 4: Make very-near-field heat-transfer calculations to generate very-near-field temperature profiles as a function of areal thermal loading and canister loading.
- Step 5: Make near-field rock mechanics calculations to determine areal thermal loading that assures room and pillar stability.
- Step 6: Determine maximum canister load from Step 3 data for the areal thermal load from Step 4.

Step 7: Make far-field thermal and rock mechanics calculations to assure that far-field design limits are not exceeded.

If any of the tentative limits are exceeded in any of the above steps, the previous steps are revised and repeated until the calculational results indicate that the limits are not exceeded.

In the case of spent fuel repository analyses, the above procedure was modified slightly. Since it was decided to place a single PWR or BWR spent fuel assembly into each canister, the thermal load for a given canister was determined, and step 6 above was not required. Steps 1 through 3 were followed by steps 5 and 7. Very-near-field heat transfer calculations were then performed to determine if canister or spent fuel temperature limits were exceeded.

The thermal and thermomechanical limits used in these preconceptual design studies are summarized in Table 3-6. It must be emphasized that these limits are based upon the best available data at this time. As such, they should be re-evaluated as more data become available. In addition, these limits definitely require re-evaluation on a site-by-site basis prior to detailed site design. Detailed descriptions of the rationale for these limits are presented in Sections 2 and 3 of Reference 12, Volume 19 of this report, and Section 3 of Volume 20 of this report.

The thermal load limits generated by these analyses and used in the preconceptual repository designs are presented in Table 3-7, along with the controlling factors associated with each limit. In these analyses, the baseline canister-emplacement design was an overpacked canister placed in a hole lined by a carbon steel sleeve. In general, the void space between the sleeve and the hole was assumed to be backfilled with crushed rock. In each of the HLW calculations, a six-inch annulus of crushed or fractured rock was assumed. This particular assumption appears to be most appropriate for the hard rocks (granite, shale, and basalt). Since salt, on the other hand, tends to reconsolidate and to experience creep with increased temperature rather than fracture, the maximum canister loads presented in Table 3-7 may be conservative for this host rock. For example if no backfill is placed around a sleeve in a salt hole, calculations

indicate that an initial thermal load of about 5 kW would be acceptable for vitrified glass HLW (See Section 3 of Volume 19 of this report.).

In these analyses, the initial heat generation for an HLW canister was varied by changing the assumed density of the waste in the canister, but keeping the volume of the waste constant. In the case of spent fuel, each canister was assumed to contain a single spent PWR or BWR assembly; consequently, canister load limits were not computed for spent fuel.

In order to design the spent fuel repository, thermal loading limits for the various host rocks were established based on preliminary analyses, which assumed there would be no difficulty in placing a canister containing a single spent-fuel assembly into any of the host formations. These limits are the ones presented in Table 3-7 and are the ones upon which the preconceptual repository designs are based. Subsequent thermal analyses during these studies indicated that to achieve the 150, 190, and 130 kW/acre local areal thermal loadings for salt, granite, and shale respectively, and assure ready-retrievability of the spent fuel for 5 years, the annulus between the spent fuel sleeve and solid formation should not be filled with crushed or fractured rock. The analyses indicate that conduction of heat through an annulus of crushed rock several inches thick is a less effective heat transfer mechanism than radiation across the gap. If a six-inch annulus of crushed rock did exist around the canisters, these areal thermal loadings may have to be reduced to about 140 kW/acre for salt and granite, and about 90 kW/acre for shale in order not to exceed a spent fuel cladding temperature limit of 200°C. Similarly, the subsequent analyses indicate that 190 kW/acre is not achievable in a basalt repository for spent fuel if the 200°C limit for spent fuel cladding temperature is to be maintained. If the basalt around the canister emplacement does not fracture, a thermal loading of about 136 kW/acre appears to maintain the cladding temperature limit. On the other hand, if a six-inch annulus of crushed basalt exists, a maximum thermal loading as low as 115 kW/acre might be required. These lower thermal loadings could reduce basalt repository capacities 20% to 30% below the preconceptual designs generated during these studies for spent fuel.

In addition to the calculations directly supporting the preconceptual repository designs, the impact of varying the value of several design parameters that are important to heat transfer have been analyzed. The results of these parametric calculations are presented in Volume 19 of this report.

Of particular interest would be analyses of the thermal and thermomechanical limits that affect repository design and the sensitivity of the design to variations in their assumed values. Since the amount of fracturing around a canister is crucial to repository design, thermomechanical stress analyses and in-situ tests would appear to be warranted in the near future, especially in the cases of the non-salt rocks. Since reductions in HLW waste density below about 3 ft³/MTHM may have serious economic impacts on the rest of the fuel cycle, studies to determine other means of maintaining thermal and thermomechanical limits should be performed. The analyses in this report have assumed no coupling between groundwater inflow and the repository temperatures; in the case of the non-salt host rocks, this coupling should be studied analytically and, if possible, in in-situ tests.

3.5 HYDROGEOLOGY AND WASTE CONTAINMENT

The hydrogeology of a potential repository site is a crucial aspect of the feasibility of deep-geologic waste isolation at that site. The most important, and perhaps the only realistic, potential mechanism for nuclide transport from a deep geologic repository in the periods during which isolation is required involves the leaching, dissolution, and transportation of these radionuclides by groundwater flowing through the repository region. In addition, the existence of groundwater near the repository horizon creates operational problems if the repository region is not isolated from such groundwater by impermeable strata. These operational problems can probably be overcome by groundwater control systems at reasonable increases in repository complexity and cost; however, the long-term impact of site hydrogeology on waste containment must be analyzed on a site-by-site basis to determine the feasibility of waste containment. Details of the analysis performed in these areas during the course of these

studies are presented in Volume 21 of this report. The following paragraphs summarize the highlights of the results.

Since salt itself is an impermeable rock and many examples exist of both bedded and domed salt formations of sufficient continuity and extent for repository location, it appears that no circulating groundwater problems exist for a repository located in salt. Care must be taken in the design of a repository so that groundwater won't be introduced into the host salt formation through the fracturing of any impermeable strata that may be isolating the salt formation from existing groundwater. This can be controlled by limiting the areal thermal loading in the repository as was done in these designs.

Although there appears to be no circulating groundwater; data presented in Volume 4 of this report indicates that most salt formations have a natural moisture content generally less than 0.5 percent by volume. This moisture content is in the form of small brine inclusions within the salt. Theoretical calculations and experimental observations indicate that these brine inclusions migrate toward a heat source at a rate directly proportional to the existing temperature gradient.¹³ The rate of this brine inflow to a cavity such as an HLW canister hole has been estimated to be less than one liter per year, with a cumulative flow into a canister hole not exceeding thirty liters in fifty years.¹³⁻¹⁵ During the operational phase of a repository, water from this amount of brine inflow would easily vaporize and be transported out of the mine through the ventilation system. After the operational phase, there does not appear to be sufficient water to create a nuclide migration path.

The situation is quite different for the non-salt rocks that have been investigated: granite, shale, and basalt. In the development of the generic stratigraphies for these non-salt rocks, sufficient groundwater was predicted to exist at the generic sites to cause substantial amounts of groundwater flow into an operating repository and groundwater flow through a decommissioned repository. The groundwater scenario predicted for these non-salt rock repository horizons follows three basic steps. First, groundwater will flow into the operational repositories at rates on the order of 400,000 gallons per day for a 2000-acre granite or

basalt repository and on the order of 5,000,000 gallons per day for a 2000-acre shale repository. After decommissioning, groundwater will flow into the repository horizon at ever-decreasing rates until all the void spaces left by the backfilling process are filled and hydrologic equilibrium is established. Equilibrium might be reached in a shale repository in only a few years, whereas in a granite or basalt repository this may take a few tens of years. Finally, after hydrologic equilibrium is established, a much slower rate of groundwater flow will exist through the repository horizon, with an upward component of flow induced by buoyant forces from the heat that has been deposited in the strata.

During the operational phase of the non-salt rock repositories, groundwater management systems can be engineered to handle the inflow predicted. Even if most of the groundwater did not evaporate, and if it happened to become contaminated through contact with waste containers or with the waste from breached containers, decontamination of this water would be a manageable problem. Such system concepts are discussed in Section 5 of Volume 22 of this report.

Although the data base to support long-term nuclide leaching and migration analyses is very sparse, an attempt was made to estimate the possibility of such migration (Sections 7.0 and 8.0 of Volume 21 of this report). Analysis of the existing data on borosilicate glass or spent fuel pellet dissolution and nuclide leaching indicates that, if the waste containers* are breached, a large fraction, if not all, of the radionuclides could be available for dissolution in the groundwater within a few tens of years. Taking into account nuclide solubility, nuclide decay, and the available data on nuclide retention by various rock types, a hydrologic transport analysis for the generic non-salt repository site conditions was performed. This analysis did not take into account dispersion or dilution. The hydrologic transport mechanism at the generic non-salt sites seems to be sufficiently slow that radioactive decay precludes significant concentrations of the shorter-lived nuclides from reaching the biosphere.

* Note that for spent fuel the "waste container" includes both the steel canister and the Zircaloy cladding on the fuel.

Most of the longer-lived nuclides are either in essentially insoluble forms or are predicted to be retained by the rocks through which they migrate. The preliminary analysis of reprocessing waste isolation indicates a possibility that ^{99}Tc , which will probably be in a soluble pertechnetate form in the reprocessing waste, might reach the biosphere in concentrations greater than allowed in potable water by present EPA standards and 10CFR20 regulations. A possibility exists that the groundwater chemistry could reduce the pertechnetate form to the insoluble TcO_2 form, in which case technetium would no longer pose a migration threat. In the case of spent fuel, technetium will be either a pure metal or TcO_2 ; thus, it will not be soluble in most groundwaters. However, the iodine in the spent fuel is in the form of soluble iodides, and the preliminary analysis indicates the possibility that ^{129}I might reach the biosphere in unacceptable concentrations through such a hydrologic transport from the non-salt repositories. Since these preliminary analyses indicate a potential groundwater contamination problem, more detailed analyses need to be performed taking into account dispersion and dilution. Further, the data base to support such analyses needs to be improved.

Since, in salt, the formation itself appears to present reasonable radionuclide containment and isolation, canister integrity beyond a retrievable phase of repository operation is not as important an issue. However, since under some hydrogeologic conditions the possibility of nuclide migration from the non-salt repositories exists, the question of long-term canister integrity in these formations could become more important. Some discussion on this issue of canister integrity is in Section 6 of Volume 22 of this report.

Of all the issues associated with deep geologic repositories, hydrogeology and waste containment is understood least, particularly for non-salt rocks. Information is lacking on as basic an issue as whether the rock mass permeability and porosity of such rocks as granite, shale, and basalt will increase or decrease in the elevated temperatures of a repository environment. Uncertainty exists in the leach rate and nuclide migration data, especially at elevated temperatures. Data on the durability of canister materials in geologic repository environments is almost

non-existent. Research and investigation are needed in in-situ groundwater velocity fields, especially on the effects of induced temperature gradients on such fields, the impacts of joints and fractures, diffusion and convection of radionuclides, and natural or induced geochemical retardation.

3.6 WASTE HANDLING CONSIDERATIONS

The handling of nuclear wastes dictates the consideration of a number of issues particular to the radioactivity of the waste. The issues that have been analyzed in these preconceptual repository designs include:

- o Baseline Design Assumptions
- o Safeguards Considerations
- o Occupational Radiation Exposure
- o Radiation Monitoring
- o Contaminated Groundwater Treatment
- o Canister Integrity
- o Critical Mass Calculations

These issues are discussed in detail in Volume 22 of this report. Summaries of these discussions are presented herein.

Baseline Design Assumptions

In order to design the receiving, handling, and storage facilities for a waste repository, it is necessary to specify types of waste that will be received, sizes of the packages, maximum decay heat loads, radiation and contamination levels, and methods of package handling, as well as similar information on the shipping containers for the packages. The values of properties that were the basis for the preconceptual repository designs and their impact on the designs are summarized in Volume 22.

In the designs presented in Volumes 8 through 17, it was found that state-of-the-art waste handling procedures and equipment (such as hoists, elevators, and conveyors) were appropriate for repository operations.

Remote-handling procedures and special shielded transporters were assumed to be required for high-level and most intermediate-level wastes. In the baseline waste designs, the low-level wastes were defined such that contact handling was possible; thus, these wastes could be transported on pallets carried by unshielded forklifts. However, in the designs for the BPNL alternative waste forms, because of higher surface radiation, it was found that a shielded forklift was required to handle the low-level waste. This same forklift design could handle some of the intermediate-level waste categories in the alternative data set. (See Section 3.5 of Volume 22 of this report.)

Safeguard Considerations

The need for safeguards and the extent of potential safeguards are discussed in Section 2.0 of Volume 22 of this report. There is currently great uncertainty as to the level of safeguards that will be required for the NWTS facility, since the safeguards requirements for the entire nuclear industry are being strengthened, and since the requirements for a waste repository have yet to be developed by NRC. At the present level of safeguard analysis for a waste repository, no specific fuel cycle-dependent criteria have been identified. A proposed safeguards program is described, which relies on control of personnel access at the perimeter and item accountability of canisters and drums received. This level of safeguards is felt to be reasonable because of the physical properties of the waste.

Occupational Radiation Exposure

The objective of the radiation-control program is to ensure that the radiation doses to individual operating personnel are below federal limits and that the cumulative staff dose is as low as reasonably achievable (ALARA). Such considerations for a waste repository are presented in Section 3.0 of Volume 22 of this report. The establishment of an ALARA program is discussed, as well as the factoring of the ALARA program into repository design, operation, and maintenance. The cost/benefit aspects of ALARA are discussed as it applies to the repository. It is recognized that measurable dose commitments can arise from maintenance activities;

therefore, a program for the reduction of radiation doses from these activities is discussed. Dose commitments from accident conditions are examined, even though accidents are not considered in ALARA calculations.

Implementation of a radiation-control program during the design phase requires radiation zoning and a use of proper radiation-attenuation curves. Appropriate radiation-zone limits and attenuation curves are set forth.

Separate OWI contracts are generating preliminary designs for the HLW and ILW shielded transporter or a transporter with a shielded transfer cask. This transporter is too large and slow to be a viable transport vehicle for the ILW and LLW in drums and boxes defined by BPNL. Therefore, preliminary calculations were performed to determine the amount of shielding that would be required for safe, forklift handling of these wastes.

Radiation Monitoring Program

During the operational phase of the repository, monitoring requirements will probably be similar to those for other nuclear facilities. The radiation-monitoring program described in Section 4.0 of Volume 22 of this report will ensure radiological protection of both the onsite personnel and the offsite population during the operation and decommissioning phases of the repository. This program will measure repository contributions to offsite exposure from direct radiation, radioactivity in breathing air, deposited radioactivity, and radioactivity in consumed materials (water, milk, plants, meat, fish, etc.). The program described also includes a description of the radiation monitoring network, the instrumentation, the data acquisition and control system and applicable regulations. Specific requirements for long-term radiation monitoring have not been defined yet for a waste repository, but are being analyzed as part of the NWTS program.

Canister Integrity Considerations

Canister integrity is essential to ensure retrievability of the canister. In addition, canister integrity may be important in delaying nuclide migration from non-salt repositories. The major factors affecting canister integrity during normal operation of the repository are described

in Section 6.0 of Volume 22 of this report. The major sources of canister stress are waste solidification and internal pressurization; these stresses can be kept to acceptable levels. Canister designs and operational procedures can be defined to accommodate predicted corrosion during the retrieval phase of repository operations; however, further understanding of corrosion processes is required to predict the effectiveness of canister integrity in delaying nuclide migration.

Nuclear Criticality Calculations

A waste repository will contain appreciable amounts of fissionable isotopes, but, to form a critical mass, one or more of these isotopes would have to proceed through a complex and highly unlikely process of selective leaching, selective transport, and/or selective deposition. In the event that such a process could occur, preliminary calculations of PuO_2 critical mass requirements under various conditions have been performed. These are described in Section 7.0 of Volume 22 of this report.

In addition, criticality considerations during the handling and transport of HLW and spent fuel are discussed. No circumstances are envisioned in which criticality can occur during the operational phase of the repository.

3.7 DECOMMISSIONING, SEALING, AND LONG-TERM MONITORING

Other studies presently being conducted under the direction of the Office of Waste Isolation are analyzing these aspects of repository operation and waste isolation. Preliminary results from some of these studies have been incorporated into the generic environmental impact statement; however, no additional work in these areas has been performed under the preconceptual alternative repository design studies.

3.8 RESOURCE REQUIREMENTS AND ENVIRONMENTAL EFFLUENTS

Detailed discussions of the resource requirements and environmental effluents associated with the construction and operation of a deep-geologic waste repository are presented in Volume 23 of this report. Both

normal operations and releases from accidents are analyzed. Specific consideration is given to the repository design details for each host rock and the parameters associated with each waste form from each fuel cycle.

The major resource requirements for the construction and operation of a deep-geologic waste repository are listed in Table 3-8. All surface acreage is committed only until the repository is decommissioned, after which time the government will maintain surveillance, but the land can be returned to normal surface use. Even during the operational phase of the repository, surface rights might be leased selectively on some of the surface area overlying the 2000-acre mine but lying outside the fenced facilities area. The temporarily committed surface area not directly overlying the mine area (that is, some 16,000 acres) will experience no restrictions on surface usage, but drilling and mining will be controlled and only selectively allowed. In some states, fee simple acquisition of this 16,000 acres may be required. A schematic of the surface area requirements is presented in Figure 3-4. All other resource requirements seem to be typical of large mining operations or nuclear material handling facilities. Water and power availability would have to be assessed on a site-specific basis.

Estimates of the maximum airborne emissions from repository construction and operations (excluding waste-related emissions) are listed in Table 3-9, including a comparison of estimated concentrations with occupational exposure guidelines. In every case the emissions are below occupational exposure guidelines, even though conservative assumptions were employed. Site-specific considerations will have to consider the integrated effect of this facility with others in the area (if there are any); and perhaps, for example, better treatment of the coal-fired boiler exhaust would be required. All emission-control equipment requirements are well within the present state-of-the-art.

The estimated maximum waste-related releases to the atmosphere during normal operations are presented in Table 3-10. These releases stem from contamination of canister surface and undetected leaking canisters. In every case, the concentrations as they are released to the atmosphere are a small fraction of the federal regulations for continuous release to

unrestricted areas (10CFR20, Appendix B, Table II), even without considering atmospheric dispersion. Again, all emission-control equipment (roughing filters and double HEPA filtration) is well within the state-of-the-art.

A number of accident scenarios have been considered and conservative assumptions applied to the amount of radioactive material released during the accidents. Estimated maximum releases in each category of interest are presented in Table 3-11. These maximum releases consider all the accident scenarios, as well as the various fuel cycles and waste forms. A major breach of a calcined HLW canister that results in a complete spillage of its contents generates the worst predicted accidental release. However, the magnitude of even this release is such that notification of the director of the appropriate NRC Regional Office is required within only 24 hours of the accident; immediate notification is not required. A similar accident involving vitrified glass HLW or spent fuel was predicted to release only about 25 times the 10CFR20 limit for continuous release of β - γ particulates to the unrestricted area. These estimates of maximum 24-hour average concentrations in an unrestricted area occur at the repository 200-acre fence-line and assume worst-case, fumigation atmospheric conditions. Site-specific calculations will have to be performed when detailed designs are available.

3.9 ALTERNATIVE CONCEPTS

The three major alternative concepts analyzed during these preconceptual repository design studies involved the impact on repository designs of: (1) a longer term (25-year) retrievability period for spent fuel isolation, (2) an alternative set of waste descriptions, packaging, and projections supplied by BPNL; and (3) the acceptance of a combustible waste alternative, also specified by BPNL. Each of these alternatives and their impacts are summarized in this section. Other alternative concepts considered in the Generic Environmental Impact Statement, but not analyzed in sufficient depth to warrant further discussions in this technical support document, include:

- o Alternative concepts for handling peak arrival rates, including temporary onsite or offsite storage, acceptance of younger waste and the simultaneous operation of more than one repository.
- o Alternative designs to reduce repository acreage requirements including multilevel repositories and the integration of waste types in a repository room or area.
- o The concept of storing spent fuel that is younger than 10 years old and its impact on repository capacity for each of the host rocks.

Similarly, alternative deep geologic concepts are briefly discussed in Section 9.2 of the generic statement, but the information was gleaned from earlier analyses, and no original work was performed in this area under these studies. The alternative deep-geologic concepts include solution-mined cavities, disposal in a matrix of drilled holes, rock-melting concepts, and subseabed geologic disposal.

3.9.1 Twenty-Five-Year Retrievability of Spent Fuel

The possibility was examined for using a subsurface facility for interim retrievable storage of spent unprocessed fuel (SURF) assemblies for a period of up to 25 years after emplacement. This concept has a different objective than that of the reference design, in which a limited retrievable storage period may be necessary for operational verification. In this concept, "25-year retrievability" means that each canister is retrievable for 25 years from the date of its own emplacement. Should a decision be made during the retrievability period to reprocess SURF assemblies stored in a repository, it would be possible to recover them and transport them to a reprocessing facility without undue hazard.

Subsequent to reprocessing, the HLW from these assemblies could then be returned to the same facility for terminal storage. At that time, there would probably be no need to provide for 5-year retrievability, since the concept would have been validated by the earlier storage period for spent-fuel assemblies. On the other hand, if during the retrievability period a firm decision were made never to reprocess, the repository filled with spent fuel could be decommissioned using procedures similar to those for the baseline repository concept.

Two approaches have been investigated that appear to accomplish 25-year retrievability: (1) reduction of the local thermal-loading limit to 25 percent (salt) or 42 percent (non-salt) of the 5-year retrieval cases, as shown in Table 2-7, or (2) removal of sufficient heat from the mine area to assure retrievability. This second approach has been analyzed only for a repository in salt; however, the results are probably applicable to the other potential host rocks. With respect to this second approach, it has been determined that by increasing the ventilation requirements for the repository, sufficient heat could be removed from the repository to allow an average repository thermal loading equivalent to that for the 5-year retrieval case in Table 3-7, yet achieve 25-year retrievability.

One major impact on repository design of either of these two approaches is the possibility that thicker sleeves might be required to assure canister integrity for this longer period of time, especially in salt. Further, if sleeves are required for canister integrity, they will be needed for each canister placed into the rock formation and not just for those emplaced during the first 5 years of operation as in the baseline repository concept. These sleeves are a substantial fraction of repository costs; therefore, further analyses regarding their requirements and specifications for emplacement in each type of host rock are certainly warranted.

A relatively minor impact of either approach is an increase in the design height of the storage room in the salt repository from 22 feet to 25 feet to allow for additional creep. This increase in room height will require a third pass by presently available mining machines and, thus, will increase repository construction costs. All other repository dimensions, including all dimensions in granite and shale, are assumed to remain identical to the reference repositories. It is conceivable that other design changes such as stress relief slots in the ceilings of the storage rooms or variations in buttress and yield pillar dimensions could raise the thermal loading limits for 25-year retrievability, but these have not been evaluated.

The approach that reduces the local thermal-loading limits would not impact the repository capacity as severely as it might appear because, in many cases, the baseline design loadings could not be achieved because of physical restrictions on space (See Tables 3-2 through 3-5.). The actual reduction in capacity to achieve 25-year retrievability was just over 50 percent for salt, granite, and basalt and about 30 percent for shale. However, if the baseline repository design for basalt had taken into account the reduced thermal loading dictated by spent fuel cladding temperature limits, the imposition of 25-year-retrievability limitations would have caused only about a 30 percent reduction in capacity over this modified baseline design.

In the 25-year retrieval case, in which average repository thermal loadings equivalent to the 5-year retrieval case are maintained, excess heat is removed by continuously ventilating the filled repository rooms. It has been shown that up to 80 percent of the heat can be removed using ambient air at reasonable velocities (see Appendix A of Volume 19). However, the salt repository designs reflecting this approach (Volume 9) were based on assuring removal of 40 percent of the heat, thus allowing the local areal thermal loading to approach 60 kW/acre. The major impact of this approach is in the ventilation system design. Specifically,¹ the peak ventilation requirements are increased more than 60 percent over the baseline case;² a larger supply air shaft is required,³ an additional exhaust air shaft is needed,⁴ and an exhaust ventilation building is required.

Details of these repository designs and the thermal and thermomechanical analyses that support these designs may be found in the following volumes of this report:

Salt Designs	Volumes 8 and 9
Granite Design	Volumes 10 and 11
Shale Design	Volumes 12 and 13
Basalt Design	Volumes 14 and 15
Thermal Analysis Report	Volume 19
Thermomechanical Analysis Support	Volume 20

In addition, analyses of the environmental effluents from these designs may be found in Volume 23.

3.9.2 Repository Designs for BPNL Alternative Waste Data

As mentioned in section 3.1 of this volume and described in detail in part II of Volume 2 of this report, an alternative waste data set was supplied by Battelle Pacific Northwest Laboratories. This data set describes wastes for the same three fuel cycles as the baseline data set: uranium and plutonium recycle, spent-fuel cycle, and uranium-only recycle. The major differences between this data set and the baseline data are the definition of only a vitrified glass form of HLW with a higher average heat load per canister, the specification of several categories of ILW-TRU, a different specification of LLW-TRU, different packaging arrangements for ILW-TRU and LLW-TRU, and different arrival rates for the wastes. In addition, the alternative data set has both truck and rail deliveries, whereas the baseline set considered only rail deliveries.

Since the decay properties of the wastes in the two data sets are very similar, there is no impact on repository local thermal loading limits as long as canister thermal loading limits are not exceeded. The impact of the canister thermal loading limits on the wastes as defined by BPNL is discussed in Sections 3.1 and 3.4 of this volume.

Preconceptual repository designs for wastes from each of the fuel cycles were completed for a salt host formation in order to compare these designs with those for the baseline data. Detailed descriptions of these designs and comparisons with equivalent baseline waste repository designs are contained in Volumes 16 and 17 of this report.

Only minor modifications of facility designs were required to accommodate this alternative waste scenario. Waste receipt areas were modified to accommodate truck and rail deliveries; waste-handling paths were modified to accommodate the different waste types and their arrival rates; and the operations in the "Low-Level-Waste-Waste Building" were modified to accomplish remote and shielded waste handling. There is very little difference between the capacities of these alternative waste designs and those of the baseline waste designs. Further, the maximum radioactive releases from the two sets of repository designs are very similar during

normal operations and from the various accident scenarios that have been analyzed, although the maximum releases may be from different circumstances. Compare Table 3-13 with Table 3-10 and Table 3-14 with Table 3-11.

Therefore, these studies indicate that the alternative waste set defined by BPNL can be handled in repository designs that are very similar to the potential reference designs with essentially the same resource utilization and environmental effluents.

3.9.3 Combustible Waste Alternative

To date, nearly all repository design analyses assume that all waste delivered to the repository is non-combustible. One alternative that was investigated in these preconceptual design studies was the storage of some combustible wastes in a repository to assess the potential consequences of a fire. The combustible waste forms considered in this analysis were from a BPNL waste data alternative (Case 3d, described in Part II, Appendix B of Volume 2 of this report). The combustible waste consists of ILW and LLW drums of contaminated trash composed of paper, rags, PVC, neoprene, latex, and wood. Two accident scenarios were defined in which a canister is breached and a fire started: a "worst case" scenario in which a complete content spillage occurred and a scenario in which two six-inch holes are punctured through the canisters. For each case, parameters determined included the rate of burn, the heat generated, and the quantities and rates of effluents released to the room in which the fire occurred and into the ventilation system. The effluents were categorized as ash, fission products, actinides, and activation products. These analyses are discussed in detail in Section 4 of Volume 23 of this report.

The results of these analyses indicate that from the complete combustion of a single ILW drum, about 6×10^{-10} Ci of particulates would be released in 22 minutes (2.7×10^{-11} Ci/min). For a typical repository exhaust stack flow rate (250,000 cfm), the release at the stack would be 3.9×10^{-15} μ Ci/ml which is on the order of five times less than NRC limit for continuous releases in an unrestricted area. For these releases to get to an unrestricted area, a further reduction in concentration of several orders of magnitude would occur because of atmospheric dispersion.

The amount of reduction depends on atmospheric conditions; however, assuming worst case, fumigation atmospheric conditions yields a 24-hour average concentration of 2.3×10^{-16} $\mu\text{Ci/ml}$ in an unrestricted area for a period of less than half an hour (a couple of orders of magnitude below NRC limits).

Further analyses show that the transfer of a fire to other drums was unlikely. This result is consistent with data on handling and storing drums of combustible waste on the surface. Operations at the Savannah River Plant, where 252,000 m^3 of miscellaneous solids were stored during 1953-1975, experienced less than a dozen small fires, most of which were attributable to spontaneous combustion. Further, none of these drum fires spread to other drums.¹⁶ However, the environmental releases from such a fire are sufficiently small that a transfer to several drums would not exceed tolerable limits. The major impact of such a fire would be the resulting interruption of repository operations and the complexity of the clean-up after the fire.

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ARTIST'S CONCEPT OF A DEEP GEOLOGICAL REPOSITORY

TYPICAL REPOSITORY SITE PLAN

REPRESENTATIVE SUBTERRANEAN LAYOUT OF A DEEP GEOLOGIC REPOSITORY

A

CENTRAL AREA...

200 acres within inner controlled area, fenced, contains all surface facilities.

B

INNER CONTROLLED AREA...

Fee simple acquisition of about 1800 acres (1-mile radius), no drilling or mining allowed at all, some surface rights may be leased selectively.

This area encloses the maximum extent of underground development.

C

OUTER CONTROLLED AREA...

Acquisition of Perpetual Subterranean Rights* for about 16,000 acres (2-mile annulus). All existing holes plugged. Drilling and mining to be controlled and only selectively allowed, so that the safety of the repository is not compromised. No restrictions on surface usage.

*In some states, fee simple acquisition may be required.

Figure 3-4. SCHEMATIC OF SURFACE AREA REQUIREMENTS FOR A FEDERAL WASTE REPOSITORY.

TABLE 3-1
EXCAVATED MATERIAL SUMMARY

Medium	Cycle ^a	Volume Mined (10 ⁶ Cubic ft)	Weight Mined (10 ⁶ Tons)	Total Backfill (10 ⁶ Tons)	Permanent Disposal (10 ⁶ Tons)
Salt	I	677	45	26	19
	II	526	35	20	15
	III	632	42	24	18
Granite	I	691	57	26	31
	II	1020	84	41	43
	III	594	49	23	26
Shale	I	400	32	18	14
	II	475	38	23	15
	III	400	32	18	14
Basalt	I	678	61	28	33
	II	1030	93	47	46
	III	600	54	25	29

^aCycle I = Uranium and Plutonium Recycle.
Cycle II = Spent Unreprocessed Fuel (SURF) Cycle
Cycle III = U-Only Recycle.

^bTotal backfill is the tonnage mined less the tonnage associated with the trenches and drilled holes all multiplied by a backfill tonnage factor.

TABLE 3-2
BASELINE SALT REPOSITORY CAPACITIES

Cycle ^a (date filled)	Waste Type	Capacity (# of containers)	Repository Area (acres) ^b	Kilowatts per Container	Average kW/acre ^c	Megawatts Delivered to Repository ^d	Fuel Equivalence of Storage Capacity (MTHM)
I (2006)	HLW (Glass)	41,340	1,018	2.8	148.	116.	86,400
	IL-TRU	274,485	649	0.0003	0.164	0.08	151,000
	CW	57,055	135	0.022	12.9	0.14	139,800
	LL-TRU	571,200	08	0.0002	----	0.11	-----
II (2008)	PWR	134,648	700	0.55	125.	74.	62,000
	BWR	200,016	1,035	0.18	41.	36.	36,600
	LL-TRL	60,192	97	0.0002	----	0.01	-----
III (2007)	HLW (Glass)	44,460	1,018	2.6	148.	116.	93,000
	IL-TRU	246,780	649	0.0003	0.16	0.07	136,000
	CW	60,260	135	0.022	11.16	3.	148,000
	LL-TRU	81,600	14	0.0002	----	0.02	-----

^aCycle I = Total Recycle; Cycle II = Spent Unreprocessed Fuel (SURF) Cycle; Cycle III = U-Only Recycle

^bApproximate area is gross acreage associated with each storage section, including main corridors, exhaust corridors, and buttress pillars; it is not the net area used to calculate areal heat loads.

^cThis acreage is associated with local heat dissipation and, therefore, includes the rooms and adjacent pillars, but not acreage for corridors, buttress pillars, etc.

^dThermal power of waste at the time of waste delivery; does not account for thermal decay as a function of time.

TABLE 3-3

BASELINE GRANITE REPOSITORY CAPACITIES

Cycle ^a (date filled)	Waste Type	Capacity (# of containers)	Repository Area (acres) ^b	Kilowatts per Container	Average kW/acre ^c	Megawatts Delivered to Repository ^d	Fuel Equivalence of Storage Capacity (MTHM)
I (2008)	HLW (Glass)	85,332	1,042	1.7	110.	145.	107,000
	IL-TRU	304,520	495	0.0003	0.0006	0.02	167,500
	CW	62,720	102	0.022	18.24	3.	153,700
	LL-TRU	731,520	121	0.0002	----	0.15	-----
II (2017)	PWR	260,160	1,061	0.55	101.	143.	120,000
	BWR	394,600	643	0.18	50.	71.	72,000
	LL-TRU	89,040	19	0.0002	----	0.02	-----
III (2008)	HLW (Glass)	85,332	1,042	1.6	101.	136.	107,000
	IL-TRU	253,658	415	0.0003	0.231	0.08	153,000
	CW	62,510	102	0.022	18.24	3.	19,500
	LL-TRU	89,040	19	0.0002	----	0.02	-----

^aCycle I = Total Recycle; Cycle II = Spent Unreprocessed Fuel (SURF) Cycle; Cycle III = U-Only Recycle.

^bApproximate area is gross acreage associated with each storage section, including main corridors, exhaust corridors, and buttress pillars; it is not the net area used to calculate areal heat loads.

^cThis acreage is associated with local heat dissipation and, therefore, includes the rooms and adjacent pillars, but not acreage for corridors, buttress pillars, etc.

^dThermal power of waste at the time of waste delivery; does not account for thermal decay as a function of time.

TABLE 3-4

BASELINE SHALE REPOSITORY CAPACITIES

Cycle ^a (date filled)	Waste Type	Capacity (# of containers)	Repository Area (acres) ^b	Kilowatts per Container	Average ^c kW/acre ^c	Megawatts Delivered to Repository ^d	Fuel Equivalence of Storage Capacity (MTHM)
I (2000)	HLW (Glass)	47,472	620	1.2	177.	57.	42,500
	IL-TRU	159,340	762	0.0003	0.075	0.05	87,600
	CW	31,928	153	0.022	5.52	1.6	78,200
	LL-TRU	278,208	69	0.0002	----	0.06	-----
II (2006)	PWR	114,264	784	0.55	174.	63.	52,700
	BWR	166,704	304	0.18	146.	30.	30,500
	LL-TRU	52,904	19	0.0002	----	0.01	-----
III (2001)	HLW (Glass)	51,612	680	1.1	166.	57.	46,200
	IL-TRU	154,553	744	0.0003	0.075	0.05	85,000
	CW	34,231	165	0.022	5.52	1.7	83,900
	LL-TRU	52,992	19	0.0002	----	0.01	-----

^aCycle I = Total Recycle; Cycle II = Spent Unreprocessed Fuel (SURF) Cycle; Cycle III = U-Only Recycle

^bApproximate area is gross acreage associated with each storage section, including main corridors, exhaust corridors, and buttress pillars; it is not the net area used to calculate areal heat loads.

^cThis acreage is associated with local heat dissipation and, therefore, includes the rooms and adjacent pillars, but not acreage for corridors, buttress pillars, etc.

^dThermal power of waste at the time of waste delivery; does not account for thermal decay as a function of time.

TABLE 3-5

BASELINE BASALT REPOSITORY CAPACITIES

Cycle ^a (date filled)	Waste Type	Capacity (# of containers)	Repository Area (acres) ^b	Kilowatts per Container	Average kW/acre ^c	Megawatts Delivered to Repository ^d	Fuel Equivalence of Storage Capacity (MTHM)
I (2006)	HLW (Glass)	119,710	967	1.2	189.	144.	107,200
	IL-TRU	303,197	502	0.0003	0.22	0.1	166,800
	CW	62,451	104	0.022	35.2	3.	153,000
	LL-TRU	731,520	131	0.0002	----	0.15	-----
II ^e (2017)	PWR	268,524	1,155	0.55	167. ^e	148.	123,800
	BWR	393,480	637	0.18	142. ^e	71.	72,000
	LL-TRU	86,832	18	0.0002	----	0.02	-----
III (2008)	HLW (Glass)	119,710	967	1.2	189.	144.	107,200
	IL-TRU	263,943	440	0.0003	0.22	0.08	145,200
	CW	65,049	109	0.022	35.2	1.	159,400
	LL-TRU	89,040	19	0.0002	----	0.02	-----

^aCycle I = Total Recycle; Cycle II = Spent Unreprocessed Fuel (SURF) Cycle; Cycle III = U-Only Recycle.

^bApproximate area is gross acreage associated with each storage section, including main corridors, exhaust corridors, and buttress pillars; it is not the net area used to calculate areal heat loads.

^cThis acreage is associated with local heat dissipation and, therefore, includes the rooms and adjacent pillars, but not acreage for corridors, buttress pillars, etc.

^dThermal power of waste at the time of waste delivery; does not account for thermal decay as a function of time.

^eCapacities for this repository design were based on an allowed thermal loading constrained by formation stability and room and pillar stability; subsequent analyses indicate that the thermal loading might be constrained to as low as 115 kW/acre to assure that the spent fuel cladding does not exceed 200°C during storage.

TABLE 3-6

THERMAL AND THERMOMECHANICAL LIMITS FOR
PRECONCEPTUAL DESIGN STUDIES^a

Very-Near-Field Considerations:

--Maximum HLW Temperature	
• Glass	500°C
• Calcine	700°C
--Maximum Spent Fuel Cladding Temperature	200°C
--Maximum Canister Temperature	375°C
--Control of Decrepitation in Salt	260-320°C max. Salt Temperature
--Moisture Content of Salt	No limit established
--Maximum Fracture of Rock	6" annulus around canister

Near-Field Considerations:

--Room Closure During Retrievability Period - Salt	<10-15% of Original Room Opening
--Room Stability - Non-Salt, Strength-to-Stress Ratio	>2 within 5' of openings, >1 otherwise
--Pillar Stability - Non-Salt, Strength-to-Stress Ratio	>2 across mid-height of pillar

Far-Field Considerations:

--Maximum Uplift Over Repository	4 to 5'
--Temperature Rise at Surface	<1°F
--Temperature Rise in Aquifers	<10°F, ALARA

^aDetailed discussions of these limits are in Volumes 19 and 20 of this report: Y/OWI/TM-36/19, Thermal Analysis, and Y/OWI/TM-36/20, Thermo-mechanical Stress Analysis and Development of Thermal Loading Guidelines.

TABLE 3-7

THERMAL LOAD LIMITS FOR PRECONCEPTUAL REPOSITORY DESIGNS

	Thermal Load Limit (controlling factor) ^c			
	Salt	Granite	Shale	Basalt
Canister Limits During Retrieval Period ^a (kW)				
● Vitrified Glass HLW	3.2(A)	1.7(A)	1.2(A)	1.3(A)
● Calcined HLW	2.6(A)	1.6(A)	1.1(A)	1.1(A)
Canister Limits After Retrieval Period ^a (kW)				
● Vitrified Glass HLW	3.6(B)	2.3(B)	1.6(B)	1.9(B)
● Calcined HLW	2.9(B)	2.0(B)	1.5(B)	1.7(B)
Local Areal Thermal Loading Limits (kW/acre) ^b				
● 5-Year Retrieval--HLW	150(C,D)	190(C)	130(C)	190(C)
● 5-Year Retrieval--Spent Fuel	150(C) ^d	190(C) ^e	130(C) ^e	190(C) ^f
● 25-Year Retrieval--Spent Fuel	36(C)	80(C)	55(C)	80(C)

^aBased on 5-year retrievability and assumes a 6" annulus of crushed rock around sleeve.

^bAcreage includes rooms and adjacent pillars, but not corridors, buttress pillars, receiving areas, etc.

^cControlling factors: A = Canister temp. limit, B = Waste temp. limit, C = Room closure during operational phase, D = Earth surface movement, long term.

^dAlthough 150 kW/acre can be tolerated locally, on the average over the entire repository only 60 kW/acre can be tolerated because of D.

^eThis limit requires that the annulus around the spent fuel sleeve not be backfilled with crushed rock and that the resulting very-near-field rock temperatures not result in excessive fracturing of the rock.

^fThe basalt repository design is based on this limit; however, later analyses indicate that the controlling factor might be the spent fuel cladding temperature and that the local areal thermal loading limit might be as low as 115 kW/acre.

TABLE 3-E COMMITMENT OF NATURAL RESOURCES FOR GENERIC REPOSITORY OPERATIONS

		Comments
<u>LAND (Acres)</u>		
Temporarily committed	18,000	Total surface area - disturbed and undisturbed.
--Disturbed area	200	Facility site; upon decommissioning this area will be returned to a natural state. Fee-simple acquisition of this land will be required to restrict drilling or mining activities.
--Undisturbed area	1,800	Area surrounding facility site; this area encloses the maximum extent of underground development. Fee-simple acquisition required. Surface rights may be selectively leased.
	16,000	Annulus surrounding repository area; acquisition of subterranean rights (or fee-simple acquisition if necessary) required. No restrictions on surface usage.
Permanently committed	2,000	Subterranean area required for repository; approximately a 200' formation thickness at a depth between 1000' to 3000'.
<u>EXCAVATED ROCK DISPOSAL (10⁶ Tons)</u>		
Salt repository design	19, 15, 18	For Total Recycle, SURF, and U-only Recycle respectively.
Shale repository design	14, 15, 14	For Total Recycle, SLRF, and U-only Recycle respectively.
Granite repository design	31, 43, 26	For Total Recycle, SURF, and U-only Recycle respectively.
Basalt repository design	33, 46, 29	For Total Recycle, SURF, and U-only Recycle respectively.
<u>WATER (10⁸ gallons)</u>		
Mining activities	32	Estimated mining requirement for non-salt rocks. No water requirement for mining activities in salt.
Other facility uses	40	Includes water for decontamination, radwaste treatment, and general plant operation.
<u>Mine water inflow disposal</u>		
--Salt mine	---	
--Shale mine	480	Over life of repository.
--Granite mine	35	Over life of repository.
--Basalt mine	35	Over life of repository.
<u>FOSSIL FUEL</u>		
Coal for electricity (10 ³ MT)	1,400	Equivalent to the consumption of a 30 MWe coal-fired power plant for 25 years.
--Equivalent electricity (10 ³ MW-hr)	3,900	
Coal for boiler operation (10 ³ MT)	1,200	

TABLE 3-9

ESTIMATED MAXIMUM AIRBORNE EMISSIONS FROM BASELINE
REPOSITORY CONSTRUCTION AND OPERATION^{d,f}

			Estimated Maximum Concentrations	Fraction of Occupational Exposure Guideline	Guideline
²²² Rn (from Rock)	Salt		4×10^{-14} $\mu\text{Ci/ml}$	10^{-6}	a
	Shale		8×10^{-11} $\mu\text{Ci/ml}$	0.003	a
	Granite		3×10^{-11} $\mu\text{Ci/ml}$	0.01	a
	Basalt		5×10^{-11} $\mu\text{Ci/ml}$	0.002	a
²²⁰ Rn (from Rock)	Salt		3×10^{-14} $\mu\text{Ci/ml}$	10^{-7}	a
	Shale		7×10^{-11} $\mu\text{Ci/ml}$	0.0002	a
	Granite		4×10^{-10} $\mu\text{Ci/ml}$	0.001	a
	Basalt		4×10^{-11} $\mu\text{Ci/ml}$	0.0001	a
H ₂ (corrosion, radiolysis)			37,000 ppm	40,000 ppm	e
<u>Diesel Exhaust</u>					
CO			2 ppm	0.04	b
Nitrogen Oxides (as NO)			17 ppm	0.7	b
HCHO (aldehydes)			0.04 ppm	0.02	b
Particulates			0.7 mg/m ³	0.14	b
<u>Coal-Fired Boiler Emissions</u>					
SO ₂			0.6 lb/10 ⁶ BTU	0.5	c
NO _x			0.2 lb/10 ⁶ BTU	0.3	c
Particulates			0.03 lb/10 ⁶ BTU	0.3	c

^aCode of Federal Regulations, Title 10: Energy, Part 20: "Standards for Protection against Radiation" (10CFR20) Appendix B, Table I, Revised January 1, 1977.

^bAmerican Conf. of Governmental Industrial Hygienists, Threshold Limit Values for Chemical Substances and Physical Agents in the Workroom Environment with Intended Changes for 1976.

^cCode of Federal Regulations, Title 40: Protection of the Environment, Part 60: "Standards for Performance for New Stationary Sources," (40CFR60), Revised July 1, 1976.

^dDoes not consider Waste-Related Emissions (See Table 2-10).

^eLower combustible limit from Handbook of Chemistry and Physics, 50th ed.

^fDetails of these release estimates can be found in Volume 23 of this report: Y/OWI/TM-36/23, Environmental Effluent Analyses, Section 1.0.

TABLE 3-10
MAXIMUM WASTE-RELATED RELEASE CONCENTRATIONS^a DURING
NORMAL OPERATIONS--BASELINE WASTE DATA^f

	Airborne Concentrations ^c (μCi/ml)			Highest Fraction of 10CFR20, App. B, Table II Guidelines ^e
	Cycle I ^d	Cycle II ^d	Cycle III ^d	
<u>Main Stack Release^b</u>				
Particulates (β-γ)	7 x 10 ⁻¹⁶	4 x 10 ⁻¹⁷	7 x 10 ⁻¹⁶	.04
(α)	7 x 10 ⁻¹⁷	6 x 10 ⁻¹⁹	2 x 10 ⁻¹⁷	.004
H-3		3 x 10 ⁻¹¹		.0002
C-14		3 x 10 ⁻¹⁴		3 x 10 ⁻⁷
Kr-85		2 x 10 ⁻⁹		.001
<u>Low-Level Waste Building</u>				
Particulates (β-γ)	7 x 10 ⁻¹⁷	3 x 10 ⁻¹⁸	7 x 10 ⁻¹⁸	.004
(α)	2 x 10 ⁻¹⁸	1 x 10 ⁻¹⁹	2 x 10 ⁻¹⁹	.0001

^aAssumes double HEPA filtration.

^bMain Stack processes air from the canistered waste building and the mine. A ventilation flow of 341,000 cfm is assumed.

^cConcentration upon release from stack, no atmospheric dispersion.

^dCycle designations: I = Uranium and Plutonium Recycle, II = Spent Fuel Cycle, III = Uranium-Only Recycle.

^eThese guidelines are for exposure in unrestricted area. These values are a direct comparison of stack releases; concentrations at the repository boundary would be even lower because of atmospheric dispersion.

^fDetails of these release estimates can be found in Volume 23 of this report: Y/OWI/TM-36/23, Environmental Effluent Analyses, Section 2.0.

TABLE 3-11

ESTIMATED MAXIMUM ACCIDENTAL RELEASE RATES--BASELINE WASTE DATA^e

	Cycle ^a	Waste Form	Maximum Stack Release Rate (Ci/Sec)		Est. Max. 24-hr Av Concentration in Unrestricted Area ^b μCi/ml	Releases Relative to 10CFR20 Limits ^c
			Initial	24-hr Av		
H-3	II	PWR	3.1×10^{-2}	2×10^{-4}	1.1×10^{-7}	0.55
C-14	II	PWR	9.3×10^{-5}	5×10^{-7}	2.8×10^{-10}	0.003
Kr-85	II	PWR	2.4	1.3×10^{-2}	7.1×10^{-6}	18.3
β-γ Particulates	III	Calcined HLW ^d	4.8×10^{-6}	2.5×10^{-8}	1.4×10^{-11}	700
α-TRU Particulates	III	Calcined HLW ^d	1.6×10^{-7}	8.3×10^{-10}	4.6×10^{-13}	23

^aCycle Designations: I = Uranium and Plutonium Recycle, II = Spent Fuel Cycle, III = Uranium-Only Recycle.

^bThe computation of maximum unrestricted area concentration assumes worst case, fumigation atmospheric conditions.

^c10CFR20 Limits are from Appendix B, Table II. A 24-hour notification accident is one which exceeds 500 times these limits.

^dMaximum accidental releases from vitrified glass HLW are on the order of 20 to 30 times less and from Spent Fuel 10 to 20 times less.

^eDetails of these release estimates can be found in Volume 23 of this report: Y/OWI/TM-36/23, Environmental Effluent Analyses, Section 3.0.

TABLE 3-12
SALT REPOSITORY CAPACITIES--BPNL WASTE

Cycle ^a (date filled)	Waste Type	Capacity (# of containers)	Repository Area (acres) ^b	Kilowatts per Container	Average kW/acre ^c	Megawatts Delivered to Repository	Fuel Equivalence of Storage Capacity (MTHM)
I (2008)	HLW (Glass)	33,240	920	2.31-3.47	148-149	105	101,100
	CW	35,182	696	0.14	49	5	146,700
	ILW Canisters	5,203					146,700
	ILW Drums (>1 R/hr)	457,101					146,600
	ILW Drums (<1 R/hr)	143,665	61				147,100
	LLW Drums	417,486					
	LLW Boxes	6,357					
II (2007)	PWR	129,736	808	0.853	167	110	59,800
	BWR	190,815	994	0.247	56	47	34,920
	LLW	56,100	33				
III (2008)	HLW (Glass)	32,299	925	2.61-3.44	147-150	108	98,200
	CW	35,180	700	0.14	49	5	146,700
	ILW Canisters	5,201					146,700
	ILW Drums (>1 R/hr)	457,065					146,600
	ILW Drums (<1 R/hr)	142,914	38				146,200
	LLW Drums	194,922					
	LLW Boxes	3,671					

^aCycle I = Total Recycle
Cycle II = Spent Unreprocessed Fuel (SURF) Cycle
Cycle III = U-Only Recycle

^bApproximate area is gross acreage associated with each storage section, including main corridors, exhaust corridors, and buttress pillars; it is not the net area used to calculate areal heat loads.

^cThis acreage is associated with local heat dissipation and, therefore, includes the rooms and adjacent pillars, but not acreage for corridors, buttress pillars, etc.

TABLE 3-13

ESTIMATED MAXIMUM WASTE-RELATED RELEASE CONCENTRATIONS^a DURING
NORMAL OPERATIONS--BPNL WASTE DATA^f

		Airborne Concentrations ^c (μCi/ml)			Highest Fraction of 10CFR20, App. B, Table II Guidelines ^e
		Cycle I ^d	Cycle II ^d	Cycle III ^d	
<u>Main Stack Release^b</u>					
Particulates	(β-γ)	1 x 10 ⁻¹⁷	2 x 10 ⁻¹⁷	1 x 10 ⁻¹⁷	0.001
	(α)	2 x 10 ⁻¹⁹	2 x 10 ⁻¹⁹	2 x 10 ⁻²⁰	1 x 10 ⁻⁵
H-3			1 x 10 ⁻¹²		5 x 10 ⁻⁶
C-14			2 x 10 ⁻¹⁴		2 x 10 ⁻⁷
Kr-85			1 x 10 ⁻¹⁰		0.0003
<u>Low-Level Waste Building</u>					
Particulates	(β-γ)	8 x 10 ⁻¹⁸	7 x 10 ⁻¹⁸	1 x 10 ⁻¹⁸	0.0004
	(α)	2 x 10 ⁻¹⁹	2 x 10 ⁻¹⁹	2 x 10 ⁻²⁰	1 x 10 ⁻⁵

^a Assumes double HEPA filtration.^b Main Stack processes air from the canistered waste building and the mine. A ventilation flow of 343,000 cfm and 386,000 cfm are assumed for cycles I and III and cycle II, respectively.^c Concentration upon release from stack, no atmospheric dispersion.^d Cycle designations: I = Uranium and Plutonium Recycle, II = Spent Fuel Cycle, III = Uranium-Only Recycle.^e These guidelines are for exposure in unrestricted area. These values are a direct comparison of stack releases; concentrations at the repository boundary would be even lower because of atmospheric dispersion.^f Details of these release estimates can be found in Volume 23 of this report: Y/OWI/TM-36/23, Environmental Effluent Analyses, Section 2.0.

TABLE 3-14
ESTIMATED MAXIMUM ACCIDENTAL RELEASE RATES--BPNL WASTE DATA^d

	Cycle ^a	Waste Form	Maximum Stack Release Rate (Ci/Sec)		Est. Max. 24-hr Av Concentration in Unrestricted Area ^b μCi/ml	Releases Relative to 10CFR20 Limits ^c
			Initial	24-hr Av		
H-3	II	PWR	3.7×10^{-2}	1.9×10^{-4}	1.0×10^{-7}	0.5
C-14	II	PWR	4.2×10^{-4}	2.2×10^{-6}	1.2×10^{-9}	0.013
Kr-85	II	PWR	3.9	2.0×10^{-2}	1.1×10^{-5}	28
β-γ Particulates	I	Glass HLW	3.0×10^{-7}	1.6×10^{-9}	8.8×10^{-13}	44
α-TRU Particulates	I	LLW-TRU	3.3×10^{-7}	3.5×10^{-9}	1.9×10^{-12}	95

^aCycle Designations: I = Uranium and Plutonium Recycle, II = Spent Fuel Cycle, III = Uranium-Only Recycle.

^bThe computation of maximum unrestricted area concentration assumes worst case, fumigation atmospheric conditions.

^c10CFR20 Limits are from Appendix B, Table II. A 24-hour notification accident is one which exceeds 500 times these limits.

^dDetails of these release estimates can be found in Volume 23 of this report: Y/OWI/TM-36/23, Environmental Effluent Analyses, Section 3.0.