3. PROPOSED ACTION AND ENVIRONMENTAL IMPACTS

3.1 PROPOSED ACTION

The proposed action is to select borosilicate glass as the waste form for immobilizing SRP high-level radioactive waste in the DWPF. Borosilicate glass was utilized as the reference waste form in the DWPF EIS.[1] The environmental consequences of selecting borosilicate glass are within the envelope of effects discussed in the DWPF and disposal system EISs.[2] The assessment also shows that the environmental effects of disposing of SRP high-level waste as a crystalline ceramic form would not differ significantly from the projected effects for disposal of the borosilicate glass form.

3.2 PROPOSED WASTE FORM

The proposed waste form for immobilization of SRP high-level radioactive waste is borosilicate glass. In the glass-making process, the high activity fraction of this waste is mixed with glass-forming chemicals and melted at 1150°C. Tests on glass made with actual and simulated waste on a small scale, and glass made with simulated waste on a large scale, indicate that borosilicate glass can accommodate different SRP waste compositions and provide acceptable levels of the following attributes:

- Waste loading
- Leach rate
- Thermal stability
- Resistance to radiation effects
- Impact resistance.

3.2.1 Description of Borosilicate Glass Waste Form

Borosilicate glass is an amorphous material formed by melting SiO_2 together with the oxides of elements such as sodium and boron. Borosilicate glass was chosen as the proposed waste form for SRP waste from among other glasses because it combines a relatively low melting temperature, 1050 to 1150°C, and high waste solubility with acceptable leach resistance and thermal and radiation stability.[3] Because of its amorphous nature, borosilicate glass can accommodate a wide range of waste compositions while maintaining favorable product and processing characteristics.

Aluminosilicate glasses have been proposed as an alternative to borosilicate glasses. However, the melting temperature of typical aluminosilicate glass is approximately 1400°C compared to the melting temperature of 1150°C for borosilicate glass. A higher melting temperature would require more development of electrode materials and ceramic refractories and would probably result in decreased melter life. Also, off-gas problems from the melter would be appreciably increased. Since the aluminosilicate glasses offer little if any improvement in chemical durability over the borosilicate glasses, it was judged that they did not justify the increased processing problems and expense.

The borosilicate glass waste form to be produced in the DWPF will consist of about 46 wt % SiO_2, 11 wt % B_2O_3, 20 wt % alkali oxides, and 23 wt % other components. This includes a waste loading of about 28 wt % (primarily oxides of iron, silicon, aluminum, manganese, and uranium). A typical composition of the glass waste form is given in Table 3-1.[4]

TABLE 3-1

Typical Composition of SRP Waste Glass

	Concentration, wt %	
Component	Waste Glass	Contribution From Waste
SiO_2	46.3	4.8
Fe_2O_3	5.9	5.9
Fe_3O_4	2.8	2.8
Na_2O	16.3	3.8
B_2O_3	10.9	-
Li_2O	4.2	-
MnO_2	1.6	1.6
Al_2O_3	3.2	3.2
NiO	0.6	0.6
MgO	1.6	0.2
U_3O_8	1.2	1.2
CaO	1.0	1.0
TiO_2	0.7	-
ZrO_2	0.4	-
La_2O_3	0.4	-
Other solids*	2.9	2.9
	100	28

* "Other solids" include zeolite, undissolved salts, and radionuclides. Chemically, radionuclides are less than 0.1% of the waste.

The borosilicate glass waste form is made by melting a mixture of glass frit (i.e., glass former) with a wet slurry of waste in a joule-heated melter.* The molten glass is poured into canisters, 0.61 m in diameter by 3.0 m long, each containing approximately 1480 kg of glass waste. Characteristics of the reference glass canister are given in Table 3-2.[5]

TABLE 3-2

Characteristics of Reference Borosilicate Glass Waste Canister

Characteristic	Reference Borosilicate Glass[5]
Waste loading, wt %	28
Waste form weight per canister, kg	1480
Total weight of waste canister, kg	1930
Waste form density, g/cm^3	2.75
Canister material	304L stainless steel
Canister dimensions	0.61 m in diameter 3.0 m in length 0.95-cm wall
Heat generation, W/Canister (5-yr-old sludge plus 15-yr-old supernate)	423
Heat generation after 1000 years, W/Canister	<1
Radionuclide content, Ci/canister (5-yr-old sludge plus 15-yr-old supernate)	150,000
Radiation, R/hr at 1 m	2900

Borosilicate glass has been studied for the immobilization of SRP high-level waste since 1974 (Appendix B). Initial development was directed toward demonstrating the feasibility of vitrifying SRP waste through laboratory-scale tests with simulated and actual SRP wastes.[3,6] Several glass-former compositions (frits) were investigated to improve both processing and product performance

* Heating is supplied by passing alternating current through opposing pairs of electrodes positioned in the molten glass.

characteristics. In 1977, large-scale vitrification tests began with simulated SRP waste.[7] As a result of these large- and small-scale tests, glass frit compositions have been systematically improved, leading to the current frit composition, Frit 131.[8]

The properties of the borosilicate glass waste form are primarily determined by five of the glass components: silica, alkali (Na_2O and Li_2O), boron, alumina, and iron oxide. The alumina and iron oxide come from the waste itself and are particularly important determinants of the durability (mechanical stability and resistance to leaching by groundwater) of SRP waste glass.

3.2.2 Waste Form Properties

In the following sections, leach resistance, important physical properties relating to mechanical and thermal stability, and radiation stability of borosilicate glass are discussed.

3.2.2.1 Leaching Properties

Leachability is a very important property for evaluating waste forms.[9] In a multi-barrier geologic waste repository, interaction of the waste form with groundwater is the most plausible means to transfer radioactive materials to man's environment, although repository sites are being selected in those formations in which water intrusion in significant quantities is unlikely.

The most important determinants of the leachability are the borosilicate glass composition, the composition of the leachant, the leachant temperature, and the duration of exposure of the borosilicate glass to aqueous attack. Leachability is less affected by the presence of other waste package components, lithostatic pressure, or hydrostatic pressure.[4] The above factors and their effects on borosilicate glass leachability are summarized in Table 3-3. Leachability of the borosilicate glass waste form is discussed in detail in Reference 4.

At temperatures in the range of those expected for leaching of SRP waste glass in a repository (25 to 55°C), steady-state leachabilities are of the order of 10^{-3} to 10^{-4} g/m^2•day. At these temperatures, leachabilities decrease from initial values of 10^{-1} to 10^{-3} g/m^2•day, depending on the radionuclide, and then gradually approach the steady-state values.[6,10,11] Steady-state leachabilities for cesium, strontium, and plutonium in glasses containing actual SRP waste are shown in Table 3-4.

TABLE 3-3

Factors Affecting Leach Resistance of Borosilicate Glass Waste Form

Factor	Effect
Waste Loading and Composition	Increasing waste loading from 28 to 35 wt % decreases leachability by about 1/2.
Leachant Composition	Leach rates for two simulated groundwaters, brine and silicate, are typically within a factor of 5.
Leachant pH	Very little effect is expected over pH range for repository groundwaters (pH 5 to pH 9).
Duration of Exposure to Groundwater	Initial leachabilities (<28 days) are 10^{-1} to 10^{-3} $g/m^2 \cdot d$; steady-state values are 10^{-3} to 10^{-4} $g/m^2 \cdot d$.
Leachant Temperature	Decrease in temperature from 90 to 40°C results in about a factor of 10 decrease in initial leachabilities, depending on species leached and glass composition.
Leachant Flow Rate	For groundwater flows expected in repositories (<1 m/yr), variation in leachability would be small.
Pressure	Increase in pressure tends to decrease leachability, but the effect is small.

TABLE 3-4

Leachability of Actual Waste Glass in Distilled Water Based on Strontium, Cesium, and Plutonium

Waste	Element	Steady-State Leachability,* $g/m^2 \cdot d$	Release Fraction Per Year**
Tank 13	Strontium	2.6×10^{-4}	1.6×10^{-6}
	Cesium	2.5×10^{-4}	1.5×10^{-6}
	Plutonium	4.6×10^{-4}	2.8×10^{-6}
Tank 16	Strontium	1.8×10^{-4}	1.1×10^{-6}
	Cesium	2.1×10^{-4}	1.3×10^{-6}
	Plutonium	2.2×10^{-4}	1.3×10^{-6}

* Room temperature; area-to-volume ratio approximately $0.1 \ cm^{-1}$.

** Calculated for a full-size DWPF canister assuming a five-fold increase in release rate due to increased area from fabrication-induced fracture.

Because the SRP high-level waste varies in composition (Table C-1, Appendix C), the effects of waste composition on leachability have been determined. In general, addition of SRP waste improves the leach resistance of the glass over that of the frit alone, primarily because of its iron and aluminum content (the major components in SRP waste). Increasing waste loading from 28 wt % (the reference loading) to 35 wt % decreases leachability by about a factor of two. Radionuclide leach rates may vary by up to a factor of five from the average over the expected range of waste glass compositions.[4,11,12]

The effects of leachant composition on glass leaching have also been studied because of expected differences in the composition of groundwater from potential repositories. The tests have shown that leachants (such as deionized and distilled water) which have low pH buffering capacity are generally more aggressive (by up to a factor of 10) than simulated repository groundwaters. However, over the range of expected repository groundwater compositions (pH 5 to pH 9), variations in pH will not significantly affect leachability.[13,14] Leach rates measured in simulated brine and silicate groundwaters are typically within a factor of 5.[4,15]

As the waste form surface temperature decreases to the ambient repository temperature due to the decay of Sr-90 and Cs-137 (Figure C-1), the leachability of the glass waste form will also decrease.* Depending on the radionuclide leached, initial (short-term) leach rates decrease by about a factor of ten as temperature is decreased from 90 to 40°C.[4,6,12] Steady-state leach rates decrease by about a factor of four over the same temperature range.[4] Thus, if the waste package should fail prematurely so that leaching occurred at 80°C (the projected maximum temperature of the design basis SRP waste glass in a wet salt repository), steady-state leach rates would be about a factor of four higher than those given in Table 3-4.

In the repository, SRP waste glass would be leached in the presence of repository minerals and multibarrier components. Tests of the interactions between SRP waste glass and other possible components of a repository system demonstrate that SRP waste glass is compatible with current repository concepts.[13] In general, the leachability decreases slightly in the presence of potential repository minerals.[4] Potential canister (304L stainless steel) or overpack (Ticode 12) materials have little effect on the leachability. Potential backfill materials can have large beneficial interactions, and materials have been identified which have beneficial effects on glass leaching.[4]

Early results from a study of leaching mechanism of borosilicate glass suggest that the observed reduction in leach rate with time results from an adherent surface layer of oxides which forms on the glass surface and which subsequently retards leaching from the waste form matrix.[4] The controlling leaching process then becomes diffusion to and through the surface layer. Solubility limits of the waste elements in the leaching environment, however, may ultimately determine the release rate from the waste form.

3.2.2.2 Physical Properties

The importance of the mechanical and thermal properties of the waste forms is discussed briefly in Appendix B. In general, the thermal and mechanical properties of borosilicate glass are expected to be more than adequate for both normal and accident conditions that might be experienced in production, interim

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* Because of the barriers provided by the waste package and the repository, groundwater is not be expected to contact the waste form for at least 1,000 years after emplacement. At this time, the temperature of the waste form would essentially be that of the ambient repository temperature.

storage, transport, or emplacement. Also, for all normal operations, the waste canister will provide the necessary structural support. Typical mechanical and thermal properties of borosilicate glass are given in Tables 3-5 and 3-6.

A particularly important characteristic is the waste form's ability to withstand impact forces without generating and dispersing a large quantity of fines. Canisters containing Savannah River glass have demonstrated the ability to survive a 9-m drop without rupture. When subjected to impacts of 10 J/cm^3 in drop tests, samples of borosilicate glass generated very small fractions of respirable particles (Table 3-5).

Except in severe accidents, the greatest stresses to the borosilicate glass waste form will probably arise from temperature changes during cooling from the melt. Both bulk and surface cracks have been observed in initial tests with full-size canisters of simulated waste glass. However, both kinds of cracking can be limited either by controlled cooling or by use of fins in the canister. Thus, the increased surface area from cracking is not expected to increase the fractional release rate from a DWPF canister by more than a factor of five (compared to the uncracked monolith).[4,17]

In the unlikely event of a high temperature excursion (such as a fire), no volatilization would occur, and the glass would devitrify only if the temperature were maintained over 500°C for extended periods of time.[18] Because leach tests have shown that the release rate of long-lived alpha-emitting radionuclides (actinides) is not affected by devitrification, a high temperature excursion would not have a significant effect on the performance of borosilicate waste glass in the repository environment.[4]

3.2.2.3 Radiation Stability

Stability against the effects of self-irradiation is an important determinant of the waste form's long-term durability in a repository. The major cause of radiation effects in waste forms is the displacement of atoms caused by alpha particles and alpha recoil resulting from the decay of the actinide elements.[15]

Extensive radiation damage studies on borosilicate glass, including doping tests with Pu-239 and Cm-244, indicate that the performance of glass in a repository should not be affected significantly by self-irradiation for periods of 10^6 years or more.[19]

TABLE 3-5

Mechanical Properties of Borosilicate Glass[4]

Property	Borosilicate Glass
Tensile Strength, MPa	57
Compressive Strength, MPa	550
Young's Modulus, GPa	67
Poisson's Ratio	0.18
Density, g/cm^3	2.75 (100°C)
Fraction of Fines Generated in Impact of 10 J/cm^3, %	0.14 to 0.18*

* Reference 16. *Fraction of particles less than 10 micrometers in size.*

TABLE 3-6

Thermal Properties of Borosilicate Glass[4]

Property	Borosilicate Glass
Thermal Conductivity, W/m•K	0.95 (100°C)
Heat Capacity, J/g•K	0.83 (25°C)
Thermal Diffusivity,* m^2/s	3.8×10^{-7}
Linear Thermal Expansion Coefficient, K^{-1}	10.9×10^{-6}
Softening Point, °C	502
Annealing Range, °C	450-500

* Calculated from other properties.

3.2.3 Waste Form Processing

In the DWPF reference process, the sludge fraction of the SRP high-level waste is reacted with hot caustic in the waste tanks, if desired to reduce the aluminum content in the sludge, then washed with water to remove soluble salts. The sludge slurry is then pumped to the DWPF for vitrification. A schematic diagram of the borosilicate glass vitrification process is shown in Figure 3-1.[20]

In the DWPF, the slurry is mixed with glass-forming additives (and with any radionuclides recovered from supernate processing), heated to drive off excess water, and then fed to an electric-conduction heated, ceramic-lined melter operated at 1150°C. Here, the slurry will dry and then form a molten glass, which will be poured into a canister. After cooling to ambient temperatures, the canister will be decontaminated, sealed by welding, and then stored onsite until shipped to a federal repository for disposal.

3.2.4 Development Requirements and Goals

The vitrification process has been demonstrated on a small scale with actual waste and on a large scale with simulated waste. Each of the other key steps in the overall reference immobilization process has also been demonstrated. Laboratory tests with both simulated and actual waste have demonstrated that a durable glass waste form can be produced for SRP waste.

Optimization studies are continuing in the following areas:

- Increased solids content of melter feed slurries. Increasing the solids content from 40 to 50 wt % nearly doubled melter throughput and increased process reliability in laboratory tests.

- Increased waste content in glass. The feasibility of increasing the waste content in glass from 28 to about 35 wt % waste oxides has been demonstrated. This increase would reduce the required number of canisters at the DWPF, transportation costs, and overpack and emplacement costs at the repository, as well as improving the form's leach resistance.

- Improved glass compositions. New glass compositions have been developed which should improve melter operation and waste form performance. In laboratory tests with these glasses, corrosion of melter materials and glass volatility were reduced, compared to the reference composition. Improved frit compositions also resulted in a decrease in leachability by up to a factor of 15 (compared to the reference composition).

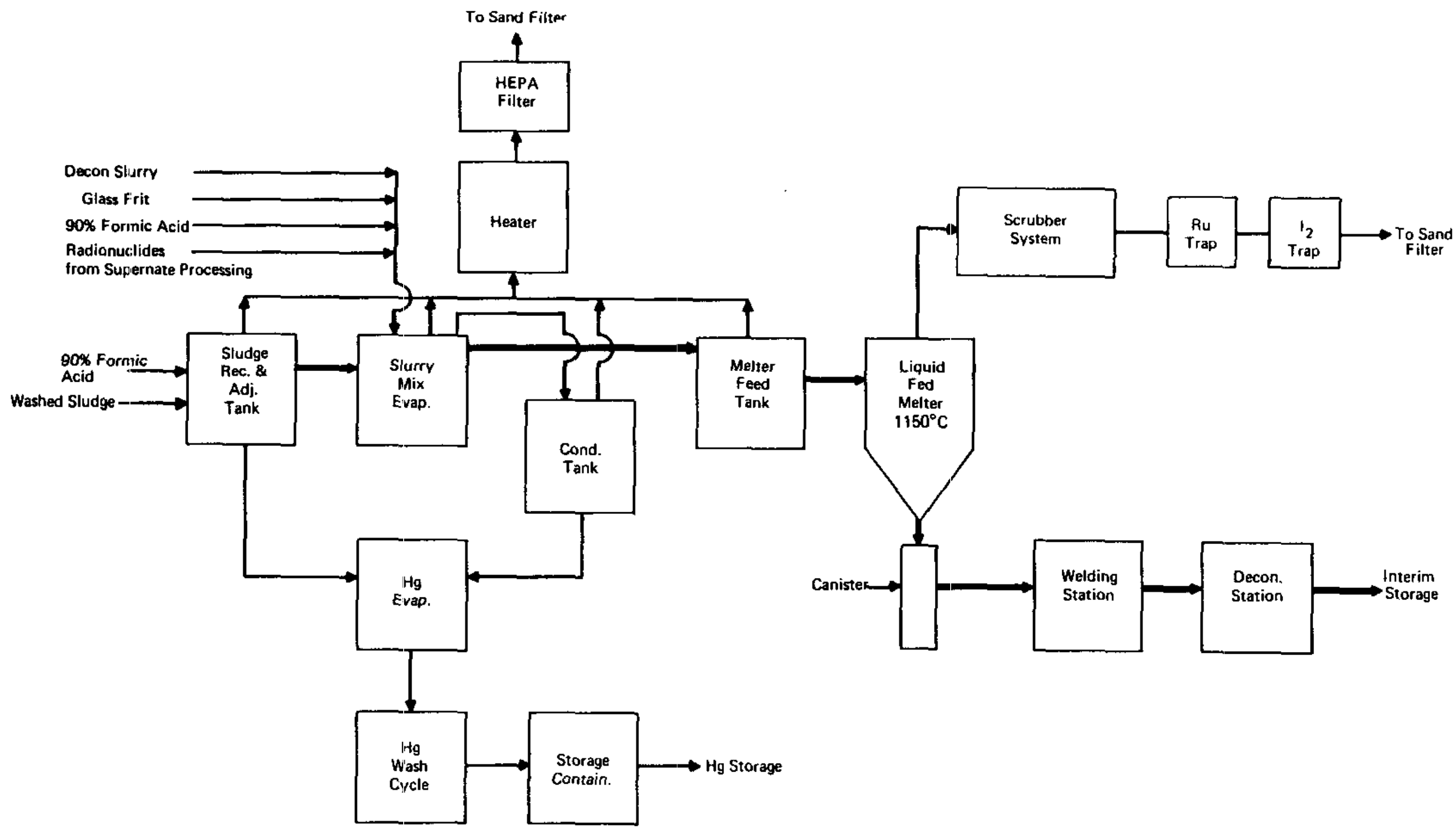

FIGURE 3-1. Borosilicate Glass Process Flowsheet[20]

- Minimizing thermal fracture in glass waste forms. Small-scale tests indicate that glass fracture during cooling from the melt can be reduced by controlled cooling and by preventing the molten glass from wetting the canister wall.[4]

- Improved repository system materials. Small-scale tests have identified promising repository backfill and other materials which reduce leach rates by up to a factor of 80.

3.2.5 Regulations and Criteria

The DWPF will be operated in conformance with all applicable Environmental Protection Agency (EPA) and DOE radiation guides for both onsite workers and the offsite public. Permits and approvals needed for the production of borosilicate glass in the DWPF were summarized in Table 6.1 of the DWPF EIS.[1]

The DWPF waste form will be shipped to a federal repository in a package that complies with applicable transportation regulations. These regulations and the responsible federal agencies are addressed in Appendix D of the DWPF EIS.

Proposed criteria and regulations that apply to federal repositories are being developed by the EPA and the Nuclear Regulatory Commission (NRC). The NWTS Program of DOE is responsible for repository operations and has proposed draft product specifications on the waste form to aid in ensuring satisfactory performance in the repository. Compliance with these repository requirements is summarized in the following sections.[21]

3.2.5.1 EPA Criteria

Although the EPA has not yet published environmental standards for high-level waste disposal, EPA has developed many internal working drafts of these criteria. The current version of the draft rule, 40 CFR 191, consists of two parts: Subpart A specifies standards for management of high-level waste and would be applicable to DWPF operations, and Subpart B contains standards for disposal and would be applicable to repository operations and closure.

Based on the latest internal draft EPA regulations, the selection of borosilicate glass as the DWPF waste form would contribute to the overall disposal system's conformance with the draft standards for management in Subpart A.

The draft criteria relating to disposal of high-level waste (Subpart B) contain projected performance requirements for repository operations in terms of total curies released to the accessible environment over a 10,000-year period. The risk assessments for typical repositories given in Section 3-4 show that virtually no activity is released in the 10,000-year period covered by the EPA criteria.

Although the number of health effects (or premature deaths) was not used as a numerical standard in the draft criteria, EPA did state that a "projected release could reasonably be limited to a level that would correspond to 1000 premature deaths over 10,000 years for a 100,000 MTHM* repository." Because the full SRP waste inventory represents an equivalent 3200 MTHM, any comparison to the EPA value for premature deaths should show that the risk is equal to or less than 32 premature deaths (10 premature deaths per 1000 MTHM). Risk analyses performed for SRP waste in a salt repository (Section 3-4) show that the dose to the affected population integrated over 10,000 years following disposal would not cause any deaths in the "best estimate" case. For an extreme case of adverse repository conditions, approximately 0.000026 premature death is estimated to occur. This is about 1 million times less than the EPA value. Under these same adverse conditions, population dose integrated over one million years is equivalent to, at most, one additional cancer.

3.2.5.2 NRC Regulations

While the NRC has no jurisdiction over defense nuclear facilities such as the DWPF, the Energy Reorganization Act of 1974 provides the NRC with specific licensing and regulatory authority over DOE facilities used primarily for the receipt and long-term storage (disposal) of high-level waste. Proposed NRC technical criteria for regulating the disposal of high-level radioactive waste in geologic repositories (10 CFR Part 60) were published for comment on July 8, 1981 (46 Fed. Reg. 35280). Most of the criteria in the proposed draft regulations pertain to repository siting, design, construction, operation, and decommissioning; however, two sections entitled Performance Objectives (10 CFR 60.111) and Requirements for the Waste Package and Components (10 CFR 60.135) relate to the waste form itself.

One of the proposed performance objectives requires that the waste package contain the waste for at least 1,000 years. This requirement on the waste package is outside the scope of this environmental assessment, but this assessment assumes that the use of borosilicate glass would contribute to the overall waste package meeting the proposed waste form performance objectives.

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* MTHM - Metric tons of heavy metal.

Another performance objective requires that the engineered system (i.e., the waste packages and the underground facility) be designed such, that after the first 1,000 years, the release rate of any radionuclide into the geological setting be less than 10^{-5} parts per year. Borosilicate glass, as part of the multibarrier approach for the waste packages, can contribute to meeting these requirements if it has leach rates $<10^{-4}$ parts per year.[22] The projected long-term release rate for the DWPF borosilicate glass waste form is below 10^{-4} parts per year, as discussed in Section 3.4.3.3.

The draft regulation on waste package requirements (10 CFR 60.135) directly includes some requirements on the waste form: the waste form must be solid, consolidated (nondispersible), and noncombustible. In addition, 10 CFR 60.135 requires that the waste package: contain no materials that are explosive, pyrophoric, or chemically reactive; contain no free liquids; be designed to contain the wastes during transportation, emplacement and retrieval; and be uniquely identified. These requirements are compatible with borosilicate glass.

3.2.5.3 DOE Specifications

The NWTS Program is developing waste form performance criteria which will include performance specifications and data requirements for high-level waste forms for geologic isolation. These performance criteria reflect all currently proposed EPA and NRC criteria that are pertinent to geologic isolation. The NWTS program has recently proposed a corresponding set of interim product specifications that include five categories of requirements (operational safety, release rate by leaching, criticality, identification, and performance testing) in three time periods:

- Operational Period (100 years after fabrication)
- Containment Period (next 1000 years)
- Isolation Period (succeeding 10,000 years).

Borosilicate glass meets the NWTS Program specifications, as described in the following paragraphs.

Operational Period. Potential safety hazards during the operational period involve damage to the canister and waste form by dropping or other impacts, or damage by fire that would allow radioactivity to escape. Resistance of borosilicate glass waste canisters to damage by impacts and thermal excursions was noted in Section 3.2.2.2.

Similarly, borosilicate glass meets all proposed criteria with respect to combustibility, pyrophoricity, explosive properties, toxicity, and criticality.

Finally, specifications related to identification of canisters, conservatism of models used to predict long-term performance, characterization test data, and quality assurance programs can be satisfied by borosilicate glass.

Containment Period. During the containment period when heat is being generated in significant amounts by radioactive decay, it is assumed that a corrosion-resistant overpack will prevent groundwater from contacting the immobilized waste. Thus, radioactive release from the waste package by high-temperature leaching will not occur. It was earlier noted that the DWPF waste package will not, in fact, exceed 80°C at a waste surface exposed to leaching in a salt repository.

For the SRP defense high-level waste, which is characterized by low heat generation and radioactivity, the borosilicate glass waste form has demonstrated excellent thermal and radiation stability and is not expected to deteriorate during the 1000-year containment period. However, it is doubtful that such a containment period is necessary for SRP waste canisters.

Isolation Period. The waste form characteristic that is most important during the isolation period is the radionuclide release rate due to leaching, which has been tentatively specified by the NWTS Program to be less than 10^{-4} parts per year.[22] The position taken by the NWTS Program is that this release rate should be met under a variety of repository conditions to satisfy the proposed NRC criteria.

Information presently available from leach tests under simulated repository conditions indicates that the borosilicate glass waste form will meet long-term release rates of less than 10^{-4} parts per year.

3.3 AFFECTED ENVIRONMENT

3.3.1 Defense Waste Processing Facility (DWPF)

The Savannah River Plant occupies an approximately circular area of 78,000 hectares (192,000 acres) in South Carolina, 37 km southeast of Augusta, GA. The site borders the Savannah River, which forms the South Carolina-Georgia border, for about 27 km. The plant site (Figure 3-2), the DWPF site (Figure 3-3), and their environmental characteristics are described in Reference 1.

3.3.2 Transportation

The environment affected by shipping SRP high-level waste canisters is also described in the DWPF EIS.[1]

3.3.3 Generic Geologic Repository

The DOE program for isolating high-level waste emphasizes disposal in mined repositories located in stable geologic formations 600 to 1200 meters below the earth's surface.[23] The goal is to find sites in suitable rock formations that meet environmental, regulatory, and institutional requirements. Screening will identify potential sites, which will then be characterized to assess the sites' suitability for a repository. Characterization includes surface studies, boreholes to repository depth, and finally exploratory shafts.

The geologic waste repositories will be the subject of separate NEPA documentation. Appendix D gives a generic description of the repositories as a basis for determining the conditions to which the waste form will be exposed during geologic disposal, and for estimating the potential environmental consequences of repository operations and closure.

The repository site performance criteria include topics such as site geometry, geohydrology, geochemistry, geologic characteristics, tectonic environment, surface characteristics, environmental characteristics, and socioeconomic conditions.[24] Site performance and repository design features will be emphasized to ensure containment, and to provide natural and man-made barriers to waste movement. Waste migration will be further impeded by placing the repository where there are low rates of groundwater flow.[25]

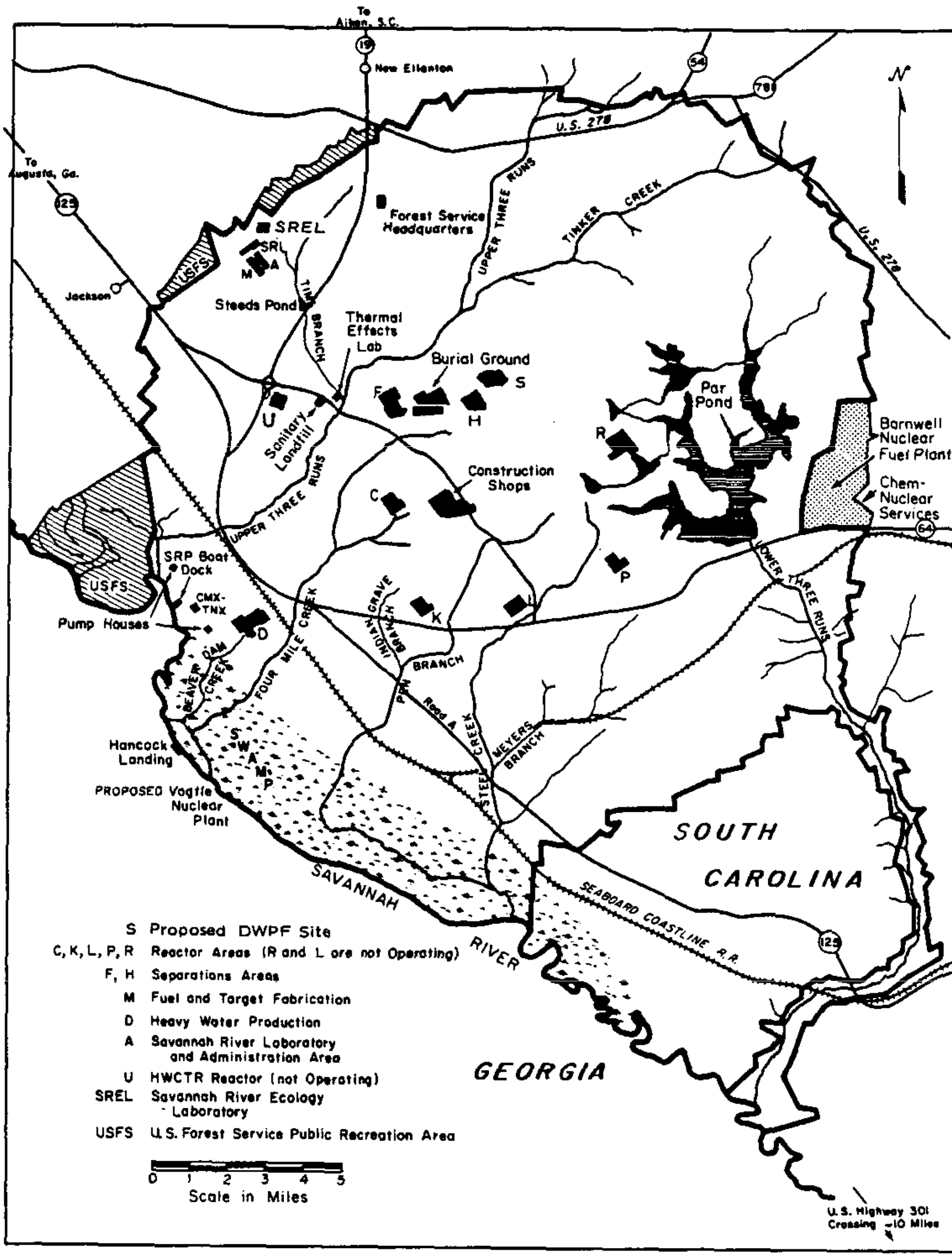

FIGURE 3-2. The Savannah River Plant Site

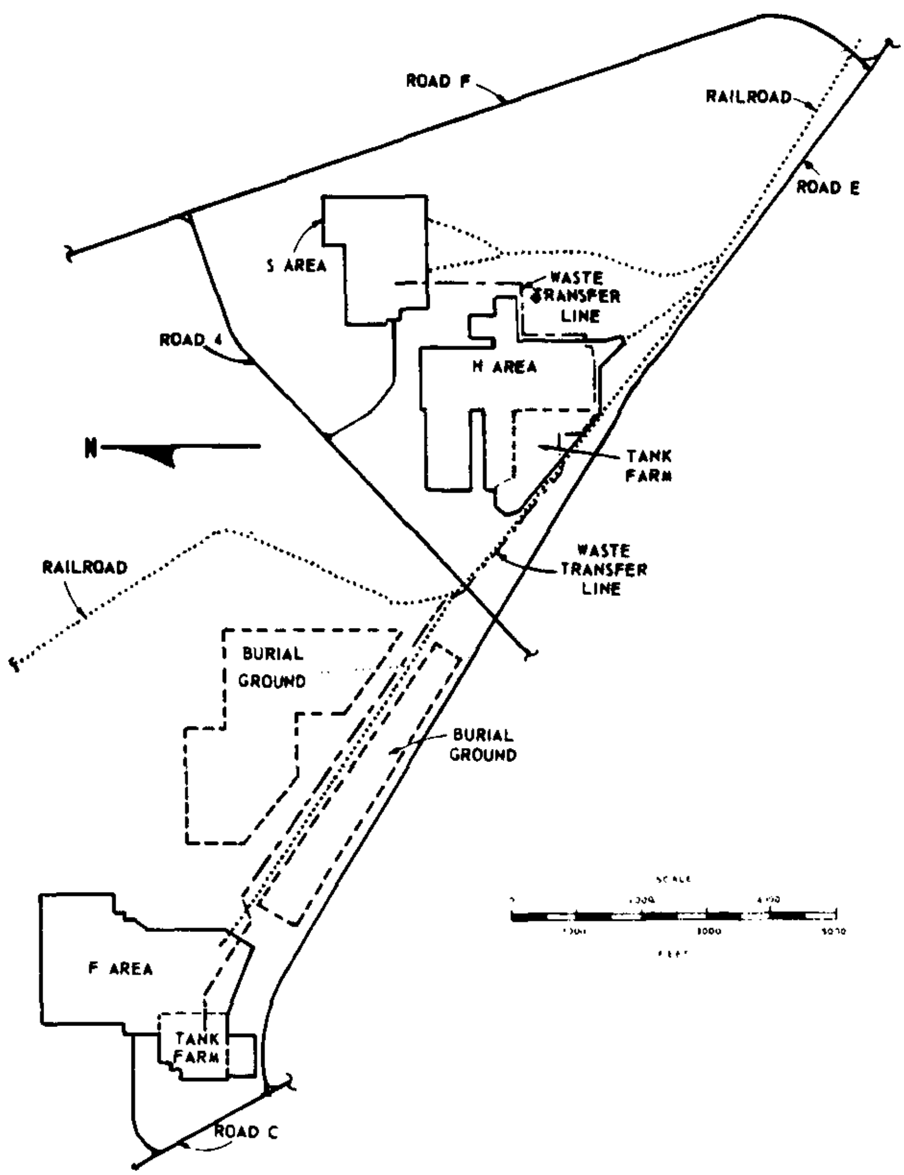

FIGURE 3-3. Proposed Location of the DWPF in S Area at the Savannah River Plant

3.4 ENVIRONMENTAL CONSEQUENCES

3.4.1 Preparation, Interim Storage, and Transportation of Borosilicate Glass Waste Canisters to Repository

The environmental impacts of immobilizing the SRP high-level radioactive waste in a borosilicate glass waste form, storing the immobilized waste at SRP until a geologic repository becomes available, and transporting the waste to a geologic repository are assessed in Reference 1. Socioeconomic effects and resource consumption from immobilization operations are minimal, and radiological effects to the public are projected to be much below normal background levels. Nonradiological effects from transportation are anticipated to be similar to those experienced with conventional common carriers. All operations will be within regulatory limits.

3.4.2 Repository Operations

3.4.2.1 Overpacking*

At the repository site, plans are for each canister of immobilized high-level waste to be sealed in an overpack designed to prevent leakage for 1000 years after the repository is closed. The overpacking will involve transferring the canister from the transport cask, handling during lag storage, placing the waste canister into the overpack, and sealing the overpack by welding.[26]

The greatest risk during the overpacking operation would be the accidental dropping of a canister onto an unyielding surface, causing breaching of the canister. Proposed DOE product specifications require the waste canister to survive a 9-m drop test (over twice the height to which a canister normally would be raised during handling) without breaching. With the proposed overpacking, the canister would be additionally protected, for example, by a carbon steel reinforcement can and by an outer titanium can. (A canister containing borosilicate glass has already passed the proposed drop test.)

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* Such overpacking is a proposed requirement by the Nuclear Regulatory Commission draft of 10 CFR 60. It is designed to protect waste from contact with groundwater during an initial heat pulse period. Since the heat output of the SRP high-level waste is too low to produce a significant heat pulse, overpacking the DWPF canister may not be necessary.

The overpacking operation is performed in a conventional hot-cell in which the ventilation pattern is controlled, and all exhausts are passed through prefilters and then HEPA filters before being released to the atmosphere.

3.4.2.2 Emplacement

Emplacement includes loading the waste package into a shielded transfer cask, moving the cask to the waste hoist, lowering the hoist and cask about 640 m to the underground excavation, transferring the cask to an underground transporter, moving through underground corridors to the storage room, and emplacing the waste package into a hole in the floor of the storage room. The hole is backfilled with crushed host rock, and a concrete plug is placed on top to close the hole.

The descent of the shielded transfer cask in the waste hoist has potential for severe damage to the canister if the hoist should malfunction and allow the canister to fall freely. However, because of multiple safety features designed into the hoist, a 2000-ft fall of the waste hoist is estimated to have a probability of about 10^{-5} per year. If the fall were sufficient to breach the canister, impact tests on the borosilicate glass waste form show that less than 0.2% respirable fines would be produced in such an impact.[16]

To result in any harm to the public, hoist failure must coincide with failure of the underground ventilation system. This system is one of the major engineering features in the repository, and includes roughing filters, HEPA filters, water sprays, demisters, and multiple fans. Underground ventilation would be diverted through the multiple exhaust filter arrangement only in the event of a release of radioactivity. The probability of failure of exhaust filters is estimated to be 10^{-4} per year. The combined probability of a hoist failure and a simultaneous filter failure is 10^{-9} per year.[27]

All other operations would limit the free fall to 1.2 times the canister length (about 4 m), and are covered by the existing specification that the canister must survive a 9-m drop test without breaching. In current plans, the canister would, in fact, be doubly encapsulated in the overpack during the entire emplacement sequence.

3.4.2.3 Retrieval

Should retrieval of the waste be required after emplacement, it is assumed that only the waste canister could be retrieved because the overpack assembly would most likely be bound in the burial hole (e.g., due to creep of salt). The retrieval scenario further assumes that the emplacement room and access corridors have been backfilled, but that the repository is still accessible.

The processes associated with retrieval of the waste package include the following:

- Location of emplacement tunnel (if sealed)
- Re-excavation of emplacement tunnel (if backfilled)
- Location of waste package (determine verticality)
- Overcoring to expose top surface of containerized waste package
- Cutting overpack and removing the overpack head pieces
- Extracting waste canister into shielded transfer cask.

After the canister is raised into the transfer cask, the cask would be moved to the main hoist and brought to the surface. At the surface, the canistered waste form would be placed in shielded storage for further disposition. The canistered borosilicate glass has the required mechanical strength to survive such an operation.

3.4.3 Long-Term Effects of Isolation

A geologic repository will be designed to control long-term radionuclide releases to levels that conform with applicable requirements. Consequence analyses of the of high-level waste disposal in geologic repositories generally conclude that the isolating qualities of the geologic media will dominate the performance of the disposal system.[28,29,30]

Once the waste is placed in a repository, natural processes over the geologic time frame could allow groundwaters to enter the repository, corrode the canister, contact the waste form, and cause the leaching of radionuclides. Contaminated groundwater would then migrate to the accessible environment (surface or underground water supplies that are used by humans). Studies of repository performance conclude that this process would be the only major contributor to the risk of human exposure.[30] Any doses to humans would occur at least thousands, and as much as millions, of years after repository closure because long periods of time would be required for the waste to leach and for the contaminated groundwater to traverse the distance between the repository and the accessible environment. Also, radionuclide travel in the groundwater generally would be retarded by sorption in the geologic media.

As a result of these time delays, which allow most of the radionuclides to decay, and the large volumetric dilution that would occur during transport, calculated doses are insignificant when compared with the effects of other natural toxic substances in the earth's crust.[31] They are also small when compared with the exposure to man from natural radioactive sources.[32,33]

3.4.3.1 Repository System Performance Models

Over geologic time periods (~10^6 years), the release of radionuclides from the repository will be governed primarily by barriers formed by the surrounding geologic media, and then by the waste form and by the engineered barriers. Geochemistry of the potential repository media is reasonably well known, and this information can be used to predict the long-term behavior of the disposed waste. As discussed in Section 3.4.3.3, migration of the radioactive components is expected to be retarded by the solubility limits of the dissolved waste and by chemical interactions (such as sorption) with the engineered barriers and the repository rock.

Several studies have analyzed the long-term performance of geologic waste isolation systems.[28-30,34-37] Typically, these studies use mathematical models to simulate and assess the behavior of the waste form, the repository site, and the overlying rock in pathways along which radionuclides could be transported to the human environment.* Values and ranges for geologic and waste form properties determined from geologic exploration and laboratory tests are used to represent interactions between the waste elements and components of the isolation system. Although the details of the analyses may differ, these studies have generally concluded that the exposure to future generations from isolated high-level wastes will be very small and that the doses will be controlled primarily by the geologic media and less so by the engineered barriers of the repository.

A typical model of the waste form/repository/site system is illustrated in Figure 3-4. Such models can be divided into three major subsystems:

- Release rates of radionuclides from the waste form and repository.

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* Several of these studies for commercial high-level waste and spent fuel are reviewed and compared in Reference 30.

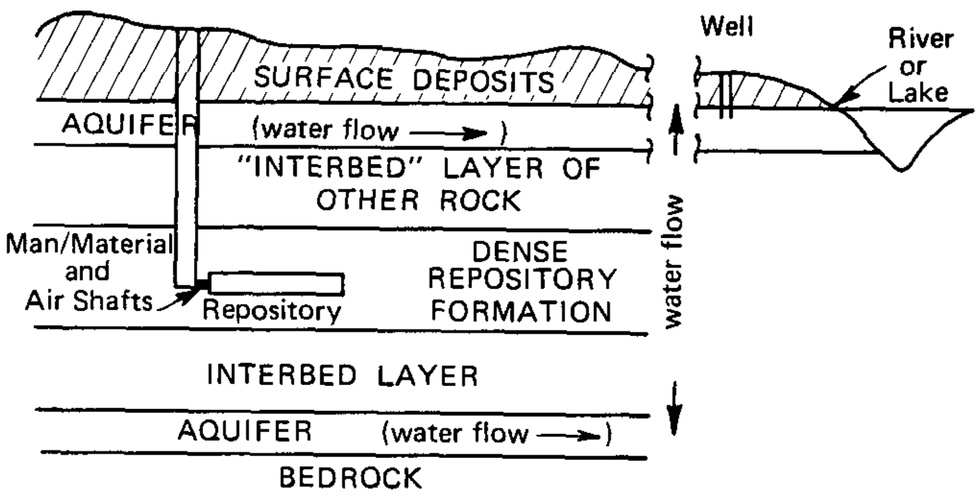

FIGURE 3-4. Typical Repository Site Model

- Hydrologic transport of radionuclides through the rock formations to a freshwater aquifer.

- Transport to and uptake by humans. Dose models are based on human-use patterns for surface water bodies (lakes and rivers) or wells drilled into an aquifer.

Several approaches have been used in evaluating the above processes which might lead to human exposure. "Deterministic" analyses choose specific values for the parameters and calculate the performance of a defined system. "Sensitivity" analyses identify which components have the most influence on the performance of the isolation system. "Uncertainty" analyses recognize that no repository can be modeled exactly; properties can be estimated only within an approximate range of values. Rather than select the "worst" possible value for each property, the analyses can treat all of the uncertainties simultaneously by a "Monte Carlo" technique. The result is a probability distribution of doses for the modeled system.

Although repository design, operations, and closure will be conducted to minimize detrimental effects on the surrounding rock, the geologic media will not be returned to their exact original state.[38] Assessments of long-term isolation, therefore, must also consider the possibility that engineered and natural barriers could deteriorate.

3.4.3.2 Performance Assessment for SRP Waste

An assessment of dose-to-man was performed for SRP waste in potential geologic repositories by Lawrence Livermore National Laboratory (LLNL).[34,35,39] This assessment included uncertainty and sensitivity analyses for undamaged ("uneventful") repositories, as well as analyses of the consequences of events which might disrupt the repository and surrounding geologic media.

Results of these analyses indicate that, under most circumstances, peak doses from SRP waste disposal will be much less than 1% of the dose from natural background radiation. Also, predicted health effects are many orders of magnitude lower than those caused by other sources. For a typical repository, credible events which might damage the repository would not significantly affect human exposure. Waste form release rates generally affect expected peak doses only if the doses are already negligibly small. For a "poor" repository site, which could yield higher, but still low doses, the waste form had little effect. These general results have recently been corroborated by an analysis which used the repository performance assessment model developed by Pacific Northwest Laboratory (PNL) for spent fuel disposal.[28,29]

Bedded Salt. Using uncertainty analyses, LLNL performed extensive studies of dose-to-man from SRP waste in a bedded salt repository.[34,35,39] Water from a lower aquifer (Figure 3-4) was conservatively assumed to permeate the salt layer to initiate the release of radionuclides from the waste. The radionuclide-containing water was then assumed to rise to an upper aquifer, from which it might be extracted by a well or might eventually contaminate surface water. Results of these processes are summarized in Table 3-7, in terms of the "best estimate" and "90% confidence level" doses for three cases:* (1) peak dose to an individual drawing all his drinking water from a well located 1.6 km downgradient from the repository; (2) peak dose to the average individual in a population residing in a river system that is fed by the upper aquifer 20 km from the repository; (3) total dose to the river system population over periods of 10^4, 10^5, and 10^6 years after repository closure.

The waste form's effect on repository system performance was assessed by assuming a mean fractional release rate of 5×10^{-6} parts per year from a waste package in salt, and associated standard deviations of one and two orders of magnitude. For the more extreme cases in the uncertainty analyses, the package release rates were generally higher than the mean. As discussed in the next section, the quoted release rate was estimated for a cracked borosilicate glass monolith, based on laboratory leach tests, making the highly conservative assumption that dissolution is not limited by solubility or by interaction with other package materials and/or rock.

The sensitivity of population dose and potential health effects to the release rate of the waste package is shown in Figure 3-5.[34] Dose is relatively insensitive to release rates greater than about 10^{-6}/yr for the least optimistic choices of geologic parameters (the 90% confidence level). For the "best estimate" case, doses vary appreciably with release rate less than $\sim 10^{-5}$ parts per year; however, these doses are already extremely small. Therefore, the properties of the repository site will dominate over waste form leach resistance in determining dose-to-man.

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* Results of uncertainty analyses show the relative likelihood of possible doses or health effects for the parameter ranges used in the model. For example, the 90% confidence level dose is the dose that equals or exceeds 90% of the doses that are calculated by varying parameters over their possible ranges. The best estimate value represents the dose for which there are equal probabilities that doses would be greater or smaller.

TABLE 3-7

Dose-to-Man from SRP Waste in a Bedded Salt Repository

	Dose from Repository: Best Estimate	Dose from Repository: 90% Confidence Level	Dose From Natural Background Radiation
Peak dose to a maximum individual, 1.6-km well, rem/yr	6×10^{-5}	1×10^{-2}	1×10^{-1}
Peak dose to an average individual, river system,* rem/yr	3×10^{-9}	2×10^{-7}	1×10^{-1}
Total population dose, river system,* person-rem			
10^4 yr	$<2 \times 10^{-8}$	2×10^{-1}	1×10^{8}**
10^5 yr	9×10^{0}	9×10^{2}	1×10^{9}**
10^6 yr	2×10^{2}	2×10^{3}	1×10^{10}**

* River system fed by aquifer 20 km from repository.

** Assumes a constant population of 100,000 people.

The best estimate of peak dose to the well user is about three orders of magnitude below background radiation. Even this small dose is believed to be pessimistic because of the conservatively high estimate used for the release rate. The population dose integrated over one million years is equivalent to less than one excess cancer, even at the 90% confidence level. In contrast, for a population of 100,000, more than 180 people per year would die from cancer from all causes, based on 1978 data for cancer incidence in the U.S. This would amount to about 1.8×10^{8} cancer deaths over one million years compared to less than one potential death caused by the geologic isolation of SRP waste.

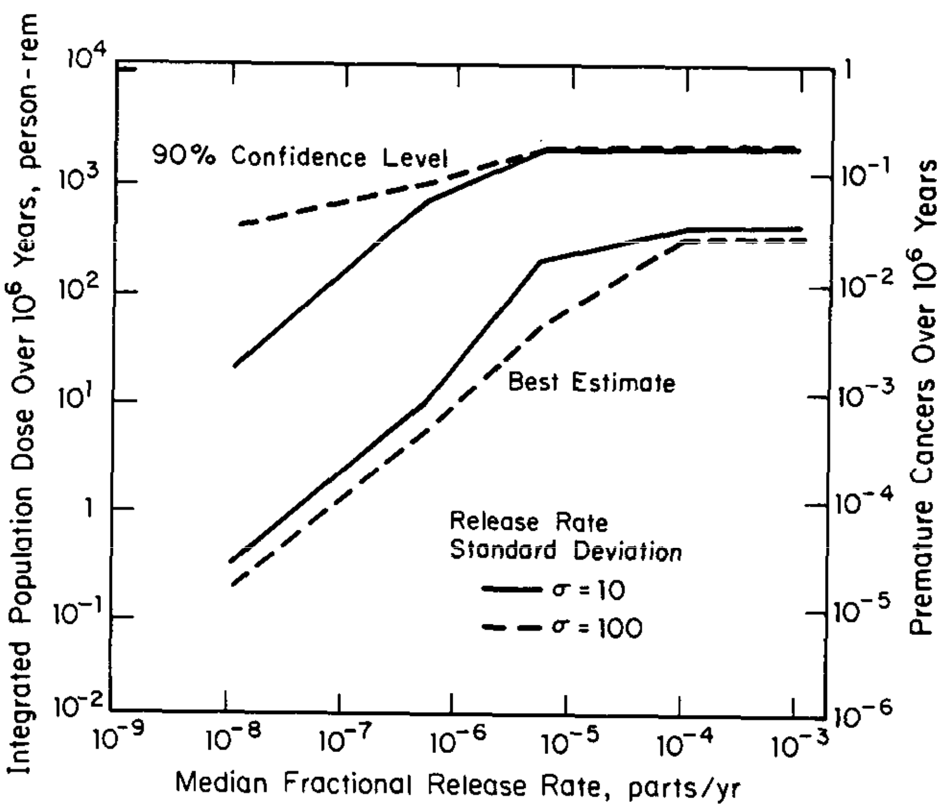

FIGURE 3-5. Sensitivity of Population Dose and Health Effects to Waste Package Release Rates

LLNL also modeled flaws and "disruptive" events, which could damage the integrity of the repository.[34] "Best estimate" doses for these cases, which include an undiscovered borehole into the repository and fault movement, are summarized in Table 3-8.

TABLE 3-8

Dose-to-Man from SRP Waste in a Disturbed Salt Repository

	Peak Individual Dose, 1.6-km Well, rem/yr	Total Population Dose Over 10^6 yr,* person-rem
Uneventful	6×10^{-5}	2×10^{2}
Fault through repository	6×10^{-3}	2×10^{3}
Failed or undetected borehole	5×10^{-3}	1×10^{3}
Deteriorated backfill	6×10^{-4}	1×10^{3}
Breccia pipe	3×10^{-4}	3×10^{2}
Dose from background radiation	1×10^{-1}	1×10^{10}**

* Based on river system fed by aquifer 20 km downgradient from repository.

** Assumes constant population of 100,000 people.

These flaws rarely increase the expected dose by more than an order of magnitude. For the 90% confidence level and higher, dose commitments actually decrease for some disruptive events.[34] Groundwater, which could pass through the entire area of an "uneventful" repository, is instead channeled along the more-permeable flows. Thus flow of water could bypass all or part of the waste in the repository.

For the disturbed salt site, reducing the waste form release rate by an order of magnitude always gave less than a ten-fold reduction in dose.

For the most severe cases modeled, LLNL showed that simple repository design features, such as providing a permeable "bypass" for groundwater underneath the repository, could reduce the doses significantly.[40]

In another study, dose-to-man calculations for SRP waste were performed with a PNL risk analysis model used previously to analyze the storage of spent fuel in a salt repository.[28,29] Results summarized in Figure 3-6 as a function of fractional release rate and groundwater travel time, generally agree with those of the more detailed LLNL analysis. The doses are generally less than 1% of background (i.e., less than 1 mrem/yr) even for very poor repository sites (i.e., short groundwater transport times).*

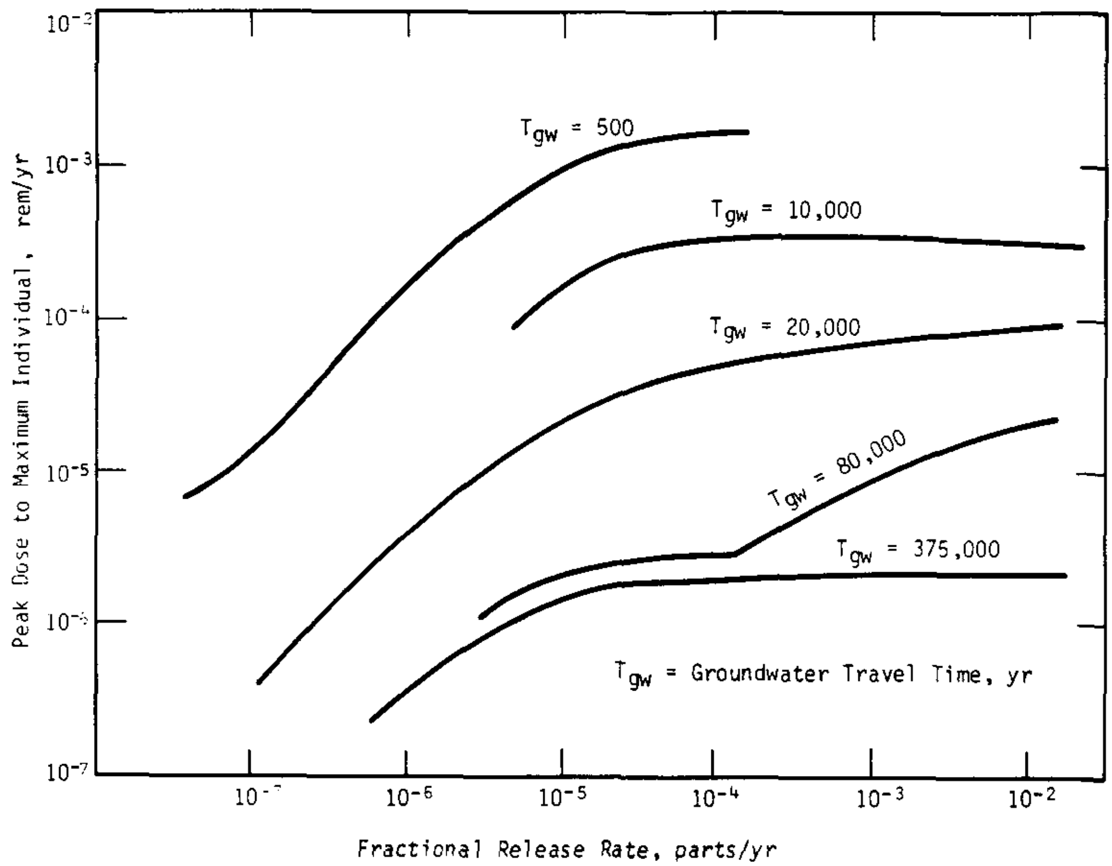

FIGURE 3-6. Dose-to-Man from SRP Waste in a Salt Repository

* The PNL study assessed the importance of groundwater travel time--the time necessary for water in an aquifer to reach a discharge point on the earth's surface. The "fractional release rate" is the rate of release into the aquifer; delays and dilution before the waste reaches the aquifer were not considered.

Basalt. LLNL also used the uncertainty analysis approach to calculate individual and population doses for SRP waste stored in a basalt repository.[34] The basalt results are summarized in Table 3-9. As in the analyses of bedded salt, maximum doses are much less than natural background.

TABLE 3-9

Dose-to-Man from SRP Waste in a Basalt Repository

		Dose from Repository		Dose From Natural Background Radiation
		Best Estimate	90% Confidence Level	
Peak dose to a maximum individual, 1.6-km well, rem/yr	Basalt	1×10^{-3}	4×10^{-2}	1×10^{-1}
	Ratio (Basalt/Salt)	15	4	—
Total population dose, over 10^6 yr,* person-rem	Basalt	1×10^{3}	2×10^{3}	1×10^{10}
	Ratio (Basalt/Salt)	5	1	—

* Based on river system fed by aquifer 20 km downgradient from repository.

The basalt doses are generally higher than the salt doses, but these differences are small at the 90% confidence level. The waste form has a somewhat smaller effect on dose for the basalt repository than for the salt repository. As for salt, the properties of the basalt repository and surrounding geologic media dominate over the waste form durability in determining dose-to-man.

Other Geologic Media. Doses have been calculated for disposal of commercial high-level waste in other geologic media considered for high-level waste disposal. Results are similar to those described above. Those studies that used pessimistic geologic and waste release parameters typically predicted doses around 1% of natural background radiation, while results of more realistic studies gave doses two to three orders of magnitude lower.[30]

3.4.3.3 Radionuclide Release Rate in Repository

The release of radionuclides from the vicinity of the waste form will be governed by the repository design and characteristics of the surrounding geologic media. Most radionculdies immobilized in the waste form have low solubilities, and their sorption on engineered barriers, such as backfill material, and on the surrounding rock should significantly reduce the release rates below those predicted from typical leach tests on the waste form.

The effects of the repository environment on waste chemistry have been considered in only a few risk studies (for example, References 36 and 37). The rate of waste release is usually treated parametrically by estimating a "release duration" over which the waste form (or repository) will release all of its contents at a constant rate.[28,29] For specific waste forms, release rates based on laboratory leach tests are generally used. However, experimental data indicate that the release of waste from the engineered system may be very much slower than the release rates based on laboratory leach tests.[41-43]

Factors affecting the release of radionuclides from the engineered barrier system of the repository include groundwater flow, oxidation-reduction conditions, temperature, pH, solubility of the leached radionuclides, and interaction of radionuclides with surrounding materials (such as sorption). The effects of these factors on the release of radionuclides from the SRP borosilicate glass waste form are discussed below.

A repository in bedded or domed salt would be expected to have no natural groundwater flows, at least for long time periods. If water penetrates a salt repository, the flows would be extremely slow and would result in essentially static leaching conditions. Crystalline rock media (such as basalt, tuff, shale, and granite) are characterized by very slow movement of underground waters, and would also provide virtually static leaching conditions. Only for unlikely geologic or man-caused events could a significant flow of water pass through the repository.[34]

Natural groundwaters contain little dissolved oxygen. Under these reducing conditions, the actinides and technetium have such low solubilities that they would not dissolve at significant concentrations.[33] Most leaching tests, however, have been performed with water in contact with air; the soluble species measured in these tests are believed to overstate the actual release of these elements in a repository which fills with groundwater after closure. Whereas salt repositories are not expected to fill with water, repositories in granite and basalt are expected to be below the water table and, after closure, will slowly fill with water. In repositories which do fill with water after closure, water could dissolve oxygen from trapped air and create oxidizing conditions.

This dissolved oxygen would soon disappear, however, because of interactions with the rock.[37,41] Thus, long-term leaching of waste forms should be under reducing conditions which would tend to limit the solubilities of the radionuclides.

After the short-lived radioactive elements have decayed, temperatures in the repository will approach the ambient temperatures of the surrounding rock. Typical ambient temperatures for salt are around 35°C;[44] hardrock conditions would range from 20°C in granite to about 50°C in basalt.[45] Leaching and other waste element interactions would be expected to occur at these temperatures.

A range of radionuclide release rates that might occur in a repository can be estimated by using laboratory leaching data to establish an upper bound, coupled with available solubility data to provide a lower, more realistic estimate for the insoluble elements. For the LLNL analyses, fractional release rates in salt (5×10^{-6} parts per year) and basalt (10^{-5} parts per year) were conservatively estimated using available leaching data on borosilicate glass,[46,47] correcting for temperature, and assuming a five-fold increase in release rate due to fabrication-induced cracking. For insoluble radionuclides, such as most of the actinides and technetium, release rates would most likely be controlled by their solubilities in the groundwater. Release rates of actinides predicted from solubilities are generally orders of magnitude lower than the rates estimated from leaching data.[36,48]

Other interactions between the waste form, groundwater, and natural and engineered barriers could also lower release rates from those estimated based on leaching tests. For example, insoluble products of leaching can create a protective layer on the waste form's surface. Such protective layers have been observed on leached surfaces of borosilicate glass.[46,49]

Surrounding rock can also contribute to the retardation of waste migration by reacting with waste species. Although not representative of expected repository conditions, high-temperature leach tests of borosilicate glass in the presence of crushed granite, basalt, or salt, showed three orders of magnitude less uranium in solution with rock present than without the rock.[42,50] Silicon, sodium, and cesium concentrations in solution were also greatly lowered.[42]

Other materials in the repository can also limit the intrusion of water and impede waste transport. Backfill clays, for example, could delay the movement of actinides from the vicinity of the waste form canister for up to 100,000 years.[51] Other materials can control groundwater chemistry or strongly sorb radionuclides.[52] In addition, the presence of certain canister materials may lower leach rates; e.g., borosilicate glass leach rates have been observed to decrease by up to two orders of magnitude in the presence of lead.[43,46] Aluminum can also decrease leach rates.[41]

In summary, the complex interactions of the waste elements with other materials in the repository, their solubility limits, the long duration of groundwater travel, and sorption of the waste elements in the surrounding geologic media will combine to limit release of radionuclides to the accessible environment to values much lower than those estimated from simple laboratory leaching tests. In particular, the following effects are expected for some specific radionuclides:[33]

- The transport time of the most hazardous fission products, Sr-90 and Cs-137, would be long enough to permit their full decay.

- Sorption of long-lived actinides, such as americium and plutonium, would retard their movement through the geologic medium, permitting substantial decay before potential release.

- Weakly sorbed long-lived radionuclides, such as Tc-99, Np-237 and Ra-226, would be only slightly soluble in groundwaters expected in deep geologic formations. Thus, their movement with groundwater would also be retarded, and the potential hazard to humans would be reduced.

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