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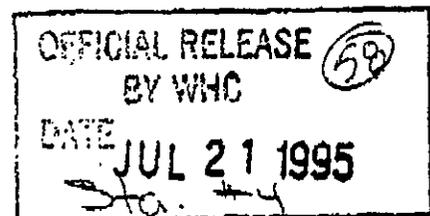
7. Abstract

Describes the options for long term storage and disposal of the cesium and strontium capsules. These capsules were made using cesium and strontium that was extracted from the waste stored at Hanford.



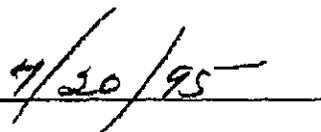
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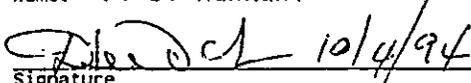
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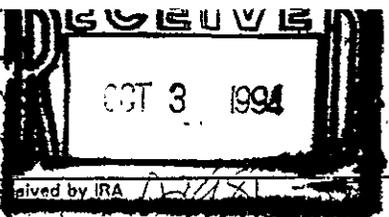
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PRELIMINARY TANK WASTE REMEDIATION SYSTEM
ENVIRONMENTAL IMPACT STATEMENT
ENGINEERING DATA PACKAGE FOR
DISPOSITION OF CESIUM AND
STRONTIUM CAPSULES

September 1994

F. D. Nankani

Westinghouse Hanford Company
Richland, Washington

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LIST OF TERMS

CFR	U.S. Code of Federal Regulations
CPF	Capsule Packaging Facility
D&D	Decontamination and decommissioning
DOE	U.S. Department of Energy
DWPF	Defense Waste Processing Facility
DWSF	Dry Well Storage Facility
HLW	High-level waste
HWVP	Hanford Waste Vitrification Plant
OCRWM	Office of Civilian Radioactive Waste Management
Tri-Party Agreement	<i>Hanford Federal Facility Agreement and Consent Order</i>
WAPS	Waste Acceptance Preliminary Specifications
WCD	Waste form and canister description
WESF	Waste Encapsulation and Storage Facility

PRELIMINARY TANK WASTE REMEDIATION SYSTEM
ENVIRONMENTAL IMPACT STATEMENT
ENGINEERING DATA PACKAGE FOR
DISPOSITION OF CESIUM AND
STRONTIUM CAPSULES

1.0 INTRODUCTION

The purpose of this document is to present basic data needed to programmatically assess four alternatives for the ultimate disposal of the existing cesium chloride ($^{137}\text{CsCl}$) and strontium fluoride ($^{90}\text{SrF}_2$) capsules. The four alternatives are as follows:

1. No-action Alternative for Remediating Cesium and Strontium Capsules
2. *Hanford Federal Facility Agreement and Consent Order* (Tri-Party Agreement) (Ecology et al. 1994) Alternative: Storage and Disposal of Cesium and Strontium Capsules
3. Tri-Party Agreement Alternative: Vitrification of Cesium and Strontium Capsules
4. Onsite stabilization and disposal,

The second alternative, Tri-Party Agreement Alternative: Storage and Disposal of Cesium and Strontium Capsules, is recommended as the reference alternative. The reference alternative is a more moderate case predicated on an as low as reasonably achievable approach, balancing concern for operator safety and environmental impacts with economic costs. The reference alternative for these capsules is ultimate disposal of the capsules in an offsite geologic repository.

Engineering studies have been identified for the various processing options within each of these programmatic alternatives. Results of these engineering studies will not be available before the issuance of this data report and hence are not dealt with in this report.

1.1 BACKGROUND

Facilities currently exist at the Waste Encapsulation and Storage Facility (WESF) on the Hanford Site for temporary storage of the capsules. The capsules could be stored for 30 years or more at WESF. Corrosion data from Bryan (1987) indicate that attack from cesium chloride at the storage temperatures will be very low, so capsule containment should last for at least 30 or 40 years, perhaps up to several hundred years. However, recent episodes of capsule failure during irradiator service have occurred. Long-term storage may require placement of additional containment around all capsules. Operation of WESF requires that the connecting B Plant also operate, but tentative missions envisioned for B Plant, call for deactivation by the year

2001. Therefore, either WESF must be modified to operate independently or a new standalone facility must be constructed for indefinite storage of the capsules.

Long-term storage of the capsules at the Hanford Site is not likely to be considered institutionally viable. Therefore, continued storage at WESF or at a standalone facility is considered only as an interim step until the capsules can be processed for permanent disposal. Current information indicates that an offsite geologic repository may be available and accepting waste by September 30, 2010.

1.2 DESCRIPTION OF CAPSULES

The cesium and strontium capsules are double-walled containers for the highly radioactive materials recovered from Hanford Site high-level waste (HLW). The capsules were produced in the 1970's and 1980's. The oldest capsules are about 20 years old.

1.2.1 Cesium Capsules

The cesium capsules contain cesium chloride. The cesium chloride in each capsule includes thousands of curies of ^{137}Cs . The amount of ^{137}Cs in each capsule decreases rapidly with time because ^{137}Cs has a half-life of about 30 years.

The cesium capsules consist of stainless steel inner and outer containers. As many as 1,328 cesium capsules (Bender 1994a) are expected to be in storage in the WESF water pools by mid-1996. The number of capsules available for storage and disposal is subject to change depending on the uses in the next several years. Some of the uses may include destructive testing of some of the capsules. Other capsules are now at several offsite locations, but will be returned to WESF.

The outer capsule dimensions are 52.77 cm long and 6.67 cm in diameter (see Figure 2-4 for more details). Decayed to January 1, 1995, the average radioactivity in each cesium capsule is about 40.1 kCi. The peak radioactivity in a cesium capsule is about 54.38 kCi. The average power per capsule is about 192.59 watts. The peak power is about 261.17 watts.

Hundreds of the cesium capsules were transported offsite to commercial irradiation facilities in the 1980's. Due to capsule failure at a commercial irradiation facility, all of the these capsules are being returned to WESF as rapidly as possible. Many other capsules were transferred to laboratories for testing and research purposes. Many capsules were subjected to destructive testing.

1.2.2 Strontium Capsules

The strontium capsules contain strontium fluoride (SrF_2). The SrF_2 in each capsule includes thousands of curies of ^{90}Sr . The amount of ^{90}Sr in each capsule also decreases rapidly with time because ^{90}Sr has a half-life of about 28 years.

The strontium capsules consist of hastelloy inner and stainless steel outer containers. As many as 601 strontium capsules (Bender 1994b) are expected to be stored in WESF by mid-1996. The outer capsule dimensions are 51.05 cm long and 6.67 cm in diameter (see Figure 2-4 for more details).

Decayed to January 1, 1995, the average radioactivity in each strontium capsule is 38.47 kCi. The peak radioactivity in one capsule is about 93.27 kCi. The average power per capsule is about 260.07 watts. The peak power is about 630.53 watts.

Some of the strontium capsules were transferred to laboratories for testing and research. All of the whole strontium capsules remain in storage in WESF water pools.

2.0 WASTE INVENTORY AND SITE DESCRIPTION

2.1 SITES COMPRISING CATEGORY

The capsules are currently stored in an onsite location, namely, the Waste Encapsulation and Storage Facility (WESF) (Figure 2-1) which is annexed to B Plant, located in the 200 East Area.

2.2 ENGINEERED FEATURES OF WESF

The facility layout is provided in Figure 2-2. The capsules are stored in a series of water-filled basins. Storage of capsules is in an active mode that requires cooling water, makeup water, ventilation, and facility maintenance. The capsules are designed as shown in Figure 2-3.

2.3 WASTE VOLUME AND COMPOSITION

The characteristics of the capsules are given in Table 2-1. The total volume of the cesium chloride and SrF₂ capsules is approximately 4 m³, based on outer capsule dimensions given in Figure 2-3.

In addition to the onsite capsules now in WESF, the inventory of capsules located offsite, and capsules in the process of being returned to the Hanford Site, will be considered as the total capsule inventory. The final number of capsules may vary because of offsite inventory and potential future uses of certain capsules for both research and analysis purposes. Inventory updates are maintained quarterly. Table 2-2 presents details on the estimated number of capsules and capsule overpack canisters.

The WESF site is unique because the wastes it contains are in relatively pure form and are encapsulated and concentrated. The salts produced for encapsulation are generally of high purity. The impurities contained within the salts are largely nonradioactive and nontransuranic contaminants.

2.4 MAXIMUM SITE INVENTORIES

Since there is only one site in this category, this information is not applicable.

2.5 POSTIMPLEMENTATION CONFIGURATION

In all cases, WESF is abandoned as a capsule storage facility and is decontaminated and decommissioned. Table 2-2 provides information on the number of overpack canisters required for various alternatives, which are described in more detail in later chapters.

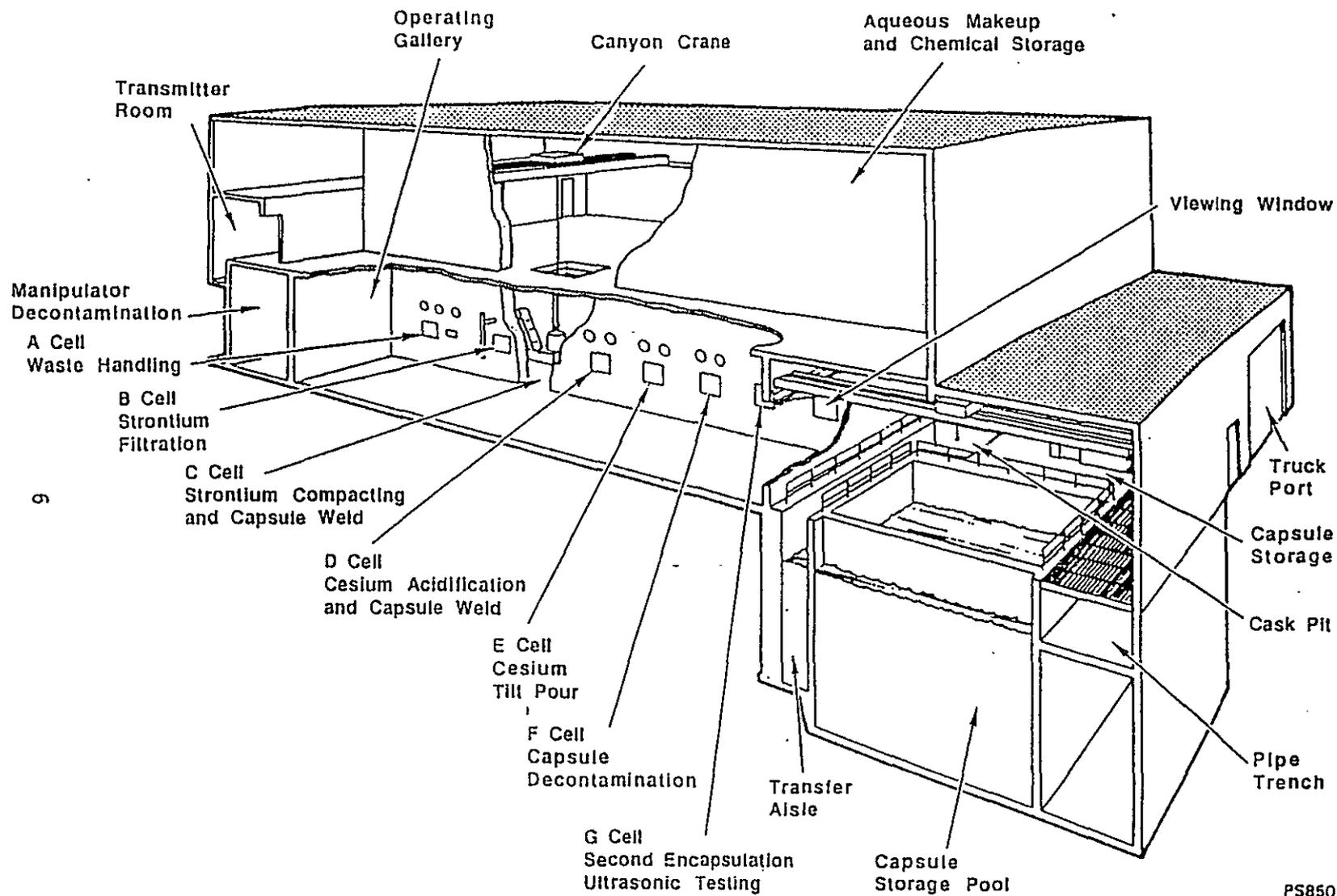
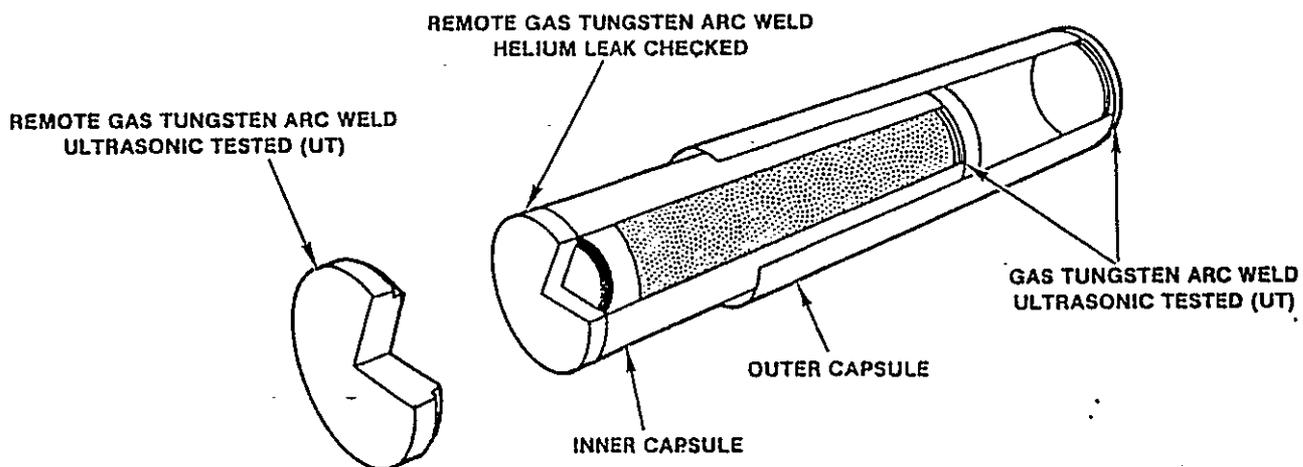


Figure 2-2. Waste Encapsulation and Storage Facility.

	FORM	LOADING	CENT OF THEORETICAL DENSITY BASED ON TOTAL VOID SPACE OF CAPSULE	TEMPERATURE			
				AIR		WATER	
				CENTER LINE	SURFACE	CENTER LINE	SURFACE
STRONTIUM FLUORIDE	COMPACTED POWDER	150 kCi (MAX)	68	860°C	430°C	860°C	71°C
CESIUM CHLORIDE	MELT-CAST	70 kCi	65	450°C	200°C	327°C	58°C



	CAPSULE									
	INNER					OUTER				
	MATERIAL	WALL THICKNESS	OUTSIDE DIAMETER	TOTAL LENGTH	TOTAL CAP THICKNESS	MATERIAL	WALL THICKNESS	OUTSIDE DIAMETER	TOTAL LENGTH	TOTAL CAP THICKNESS
STRONTIUM FLUORIDE	HASTELLOY C-276 (UT)	0.305 (UT)	5.72	48.39	1.02	STAINLESS STEEL 316-L (UT)	0.277 (UT)	6.67	51.05	1.02
CESIUM CHLORIDE	STAINLESS STEEL 316-L (UT)	0.241 (UT)	5.72	50.10	1.02	STAINLESS STEEL 316-L (UT)	0.277 (UT)	6.67	52.77	1.02

NOTE: ALL DIMENSIONS ARE IN cm

Figure 2-3. Typical Cesium and Strontium Capsule.

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Table 2-1. Characteristics of Existing Capsules.

Characteristics	As Filled	Sr (601 capsules)				Cesium (1328 capsules)				
		Jan. 1, 1995	Jan. 1, 2000	Jan. 1, 2015	Jan. 1, 2020	As filled	Jan. 1, 1995	Jan. 1, 2000	Jan. 1, 2015	Jan. 1, 2020
Cumulative MCI	32.66	23.11	20.5	14.3	12.7	73.9	53.2	47.4	33.6	29.9
Cumulative kW	220.8	156.3	138.6	96.9	85.9	355.3	255.7	228.0	161.5	144.0
Average kCi	54.36	38.47	34.14	23.85	21.16	55.7	40.1	35.75	25.33	22.58
Average W	367.43	260.07	230.78	161.24	143.08	267.6	192.59	171.69	121.64	108.44
Highest curies loading, kCi	146.6	93.27	82.76	57.82	51.31	74.5	54.38	48.48	34.34	30.62

NOTE: The values for megacuries and highest curies loading reflect only parent radionuclide activity for ^{90}Sr - ^{90}Y and ^{137}Cs - $^{137\text{m}}\text{Ba}$. A half-life of 30 years is assumed for cesium and strontium.

Table 2-2. Estimated Capsule and Capsule Canister Details.

Waste Type	Geologic Disposal (2010)				
	Number of Capsules	Heat Load (kW)	Average kW per capsule	Number of Canisters*	Average number of capsules per canister
Strontium capsules	601	109.2	0.18	93	6.5 ^a
Cesium capsules	1,328	181.19	0.13	215	6.2 ^a
Total	1,929	290.39	-	308	-
	Onsite Stabilization and Disposal and No-Action				
Strontium capsules	601	109.2	0.18	217	2.8 ^b
Cesium capsules	1,328	181.19	0.13	367	3.8 ^b
Total	1,929	290.39	-	584	-

^aBased on thermal limit of 1.17 kW/canister (strontium) and 0.8 kW/canister (cesium) geologic repository heat load limits. A half-life of 30 years is assumed for cesium and strontium. Note: These heat limits are based on a 3-m long canister. With the reference high-level waste (HLW) glass canister now being 4.42-m long, the number of capsules per canister could be increased while keeping the radial heat flux the same. Since the reference HLW canister has not yet been officially approved for use with the cesium and strontium capsules, the more conservative size canister was used to estimate the total number of canisters generated.

^bBased on a Dry Well Storage Facility heat load of 0.5 kW/canister.

*Estimates based on heat loads only. Physical constraints may increase the actual number of canisters.

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3.0 DESCRIPTION OF CAPSULE MANAGEMENT OPTIONS

The process steps for each of the four alternatives for the disposal of cesium and strontium capsules are addressed in this chapter. The disposal alternatives are as follows:

1. No-Action Alternative for Remediating Cesium and Strontium Capsules (No-Action)
2. Tri-Party Agreement (Ecology et al. 1994) Alternative: Storage and Disposal of Cesium and Strontium Capsules (Geologic Disposal)
3. Tri-Party Agreement Alternative: Vitrification of Cesium and Strontium Capsules
4. Onsite stabilization and disposal.

Currently all capsules onsite are maintained within the water basins at WESF. Capsules received from offsite will be added to those at WESF. As warranted the capsules may be repackaged. The capsules and basin would be inspected and maintained as necessary.

3.1 NO-ACTION ALTERNATIVE FOR REMEDIATING CESIUM AND STRONTIUM CAPSULES

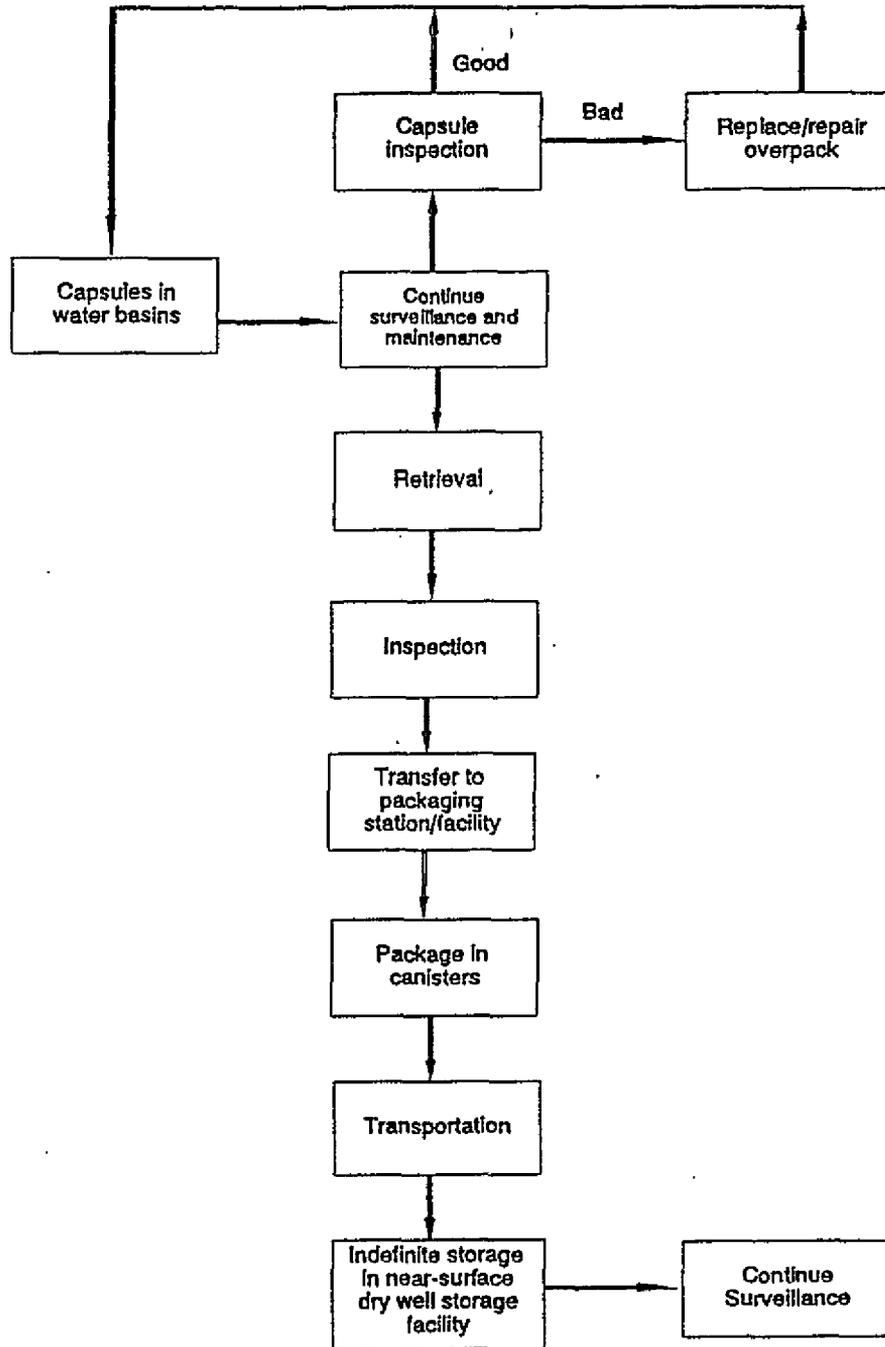
The no-action alternative continues storage of the capsules in the water basins until the capability for overpacking the capsules is provided. The capsules will be placed in the overpack canisters. The overpack canisters will be transferred and placed in a storage facility for an indefinite period. Security and institutional controls would remain in place to protect workers and the public.

A flowsheet for the no-action alternative for the cesium and strontium capsules is shown in Figure 3-1. This option consists of the following basic elements:

- Continue storage of capsules in the water basins at WESF.
- Build capability to allow loading of capsules into overpack canisters.
- Place capsules in overpack canisters.
- Place canisters in dry well storage following production.
- Store capsules canisters in a dry well for an indefinite period with continued surveillance.

The cesium and strontium capsules will be stored in the water-filled basins until the capability for overpacking is provided. Overpacking may also be accomplished by performing the necessary modifications to WESF to allow overpacking. During the storage period, the capsules and basins will be

Figure 3-1. Flowsheet for No-Action Alternative for
Remediating Cesium and Strontium Capsules.



DP087\FIG-3-1
JES (9/14/94)

periodically inspected and maintained as necessary. For purposes of generating the data tables for this alternative, it is assumed that no-action will occur at WESF until about 2010 (when the offsite geologic repository will be ready to accept waste) to allow capsule cooling. It is also assumed that, with maintenance, the WESF basins will not have to be replaced.

Detailed engineering studies now planned may show a more feasible date for removing the capsules from the WESF.

Capsules will be removed from the WESF basins and transferred to an inspection cell for survey, decontamination or overpacking, and a calorimetry check for heat content. The capsules will then be transferred to a storage vault until they are needed, after which the capsules will be placed in racks and inserted into the overpack canisters. (For purposes of generating impact data for this alternative, a separate capsule packaging facility is assumed in Appendix A, Section A1.1. However, detailed planned engineering studies may indicate it is better to perform the capsule packaging operation in WESF.) The canisters are sealed, inspected, and surveyed for radioactive surface contamination. Lag storage space (an air-cooled vault) is provided for the sealed canisters. Further analysis is required to optimize the overpack design for dry well storage. The canisters are then placed in near-surface dry wells to provide long-term, passively cooled storage (see Appendix A, Section A2.2). A total of 672 dry wells (584 canisters plus 15 percent contingency) would be required under current assumptions, occupying about 18,000 m² in a grid pattern. The capsules will be left in dry well storage for an indefinite period. For purposes of this analysis, continued surveillance of the facility is assumed until 2090 as shown in Figure 3-2.

3.2 TRI-PARTY AGREEMENT ALTERNATIVE: STORAGE AND DISPOSAL OF CESIUM AND STRONTIUM CAPSULES (REFERENCE ALTERNATIVE)

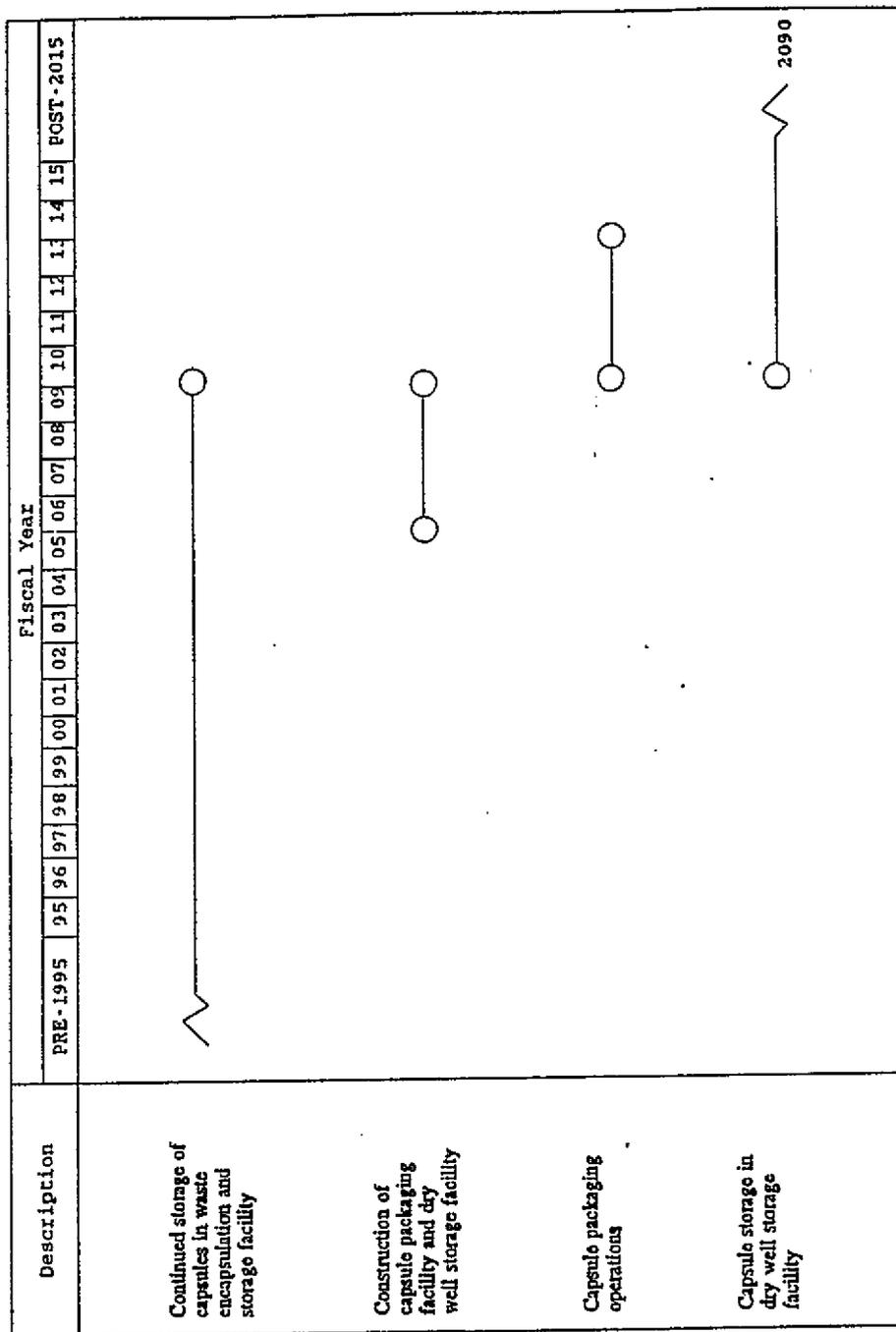
This second alternative (geologic disposal) continues storage of the capsules within the water basins at WESF until an offsite geological repository is completed. As warranted the capsules may be repackaged.

A flowsheet for the storage and disposal alternative for cesium and strontium capsules is shown in Figure 3-3. The basic components of this option include continued storage of capsules in the WESF water basins until a geologic repository is available.

For geologic disposal, an average of 6.5 strontium capsules or 6.2 cesium capsules will be placed in each canister, as based on assumed repository heat limits, yielding a total of about 509 (308 plus 201 contingency) capsule canisters (see Table 2-2). The waste acceptance criteria for repository disposal may require changes to the overpack canister that is proposed in this document. Options for change include an improved waste form and/or multi-barrier containers, if the current design proves to be inadequate.

Following placement of the capsules in canisters, the canisters are assumed to be transported to an offsite geologic repository for disposal. It is assumed the canisters will be transported by rail for storage at an offsite repository. Transportation details are given in Appendix B, Section B2-2.

Figure 3-2. Schedule of Activities for the No-Action Alternative for Remediating Cesium and Strontium Capsules.



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Figure 3-3. Flowsheet for Tri-Party Agreement Alternative: Storage and Disposal of Cesium Strontium Capsules (Reference Alternative).

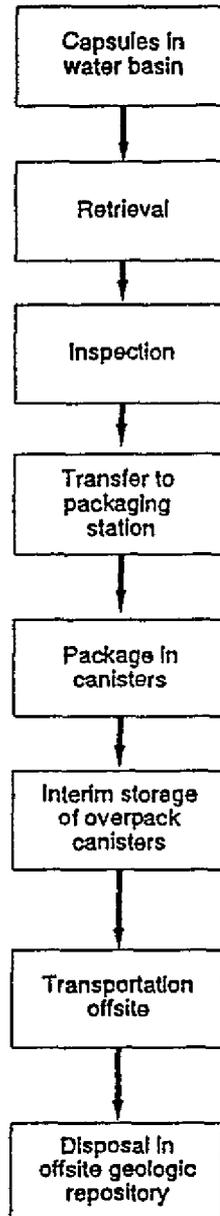


Figure 3-4 provides a schedule of the assumed sequence of activities.

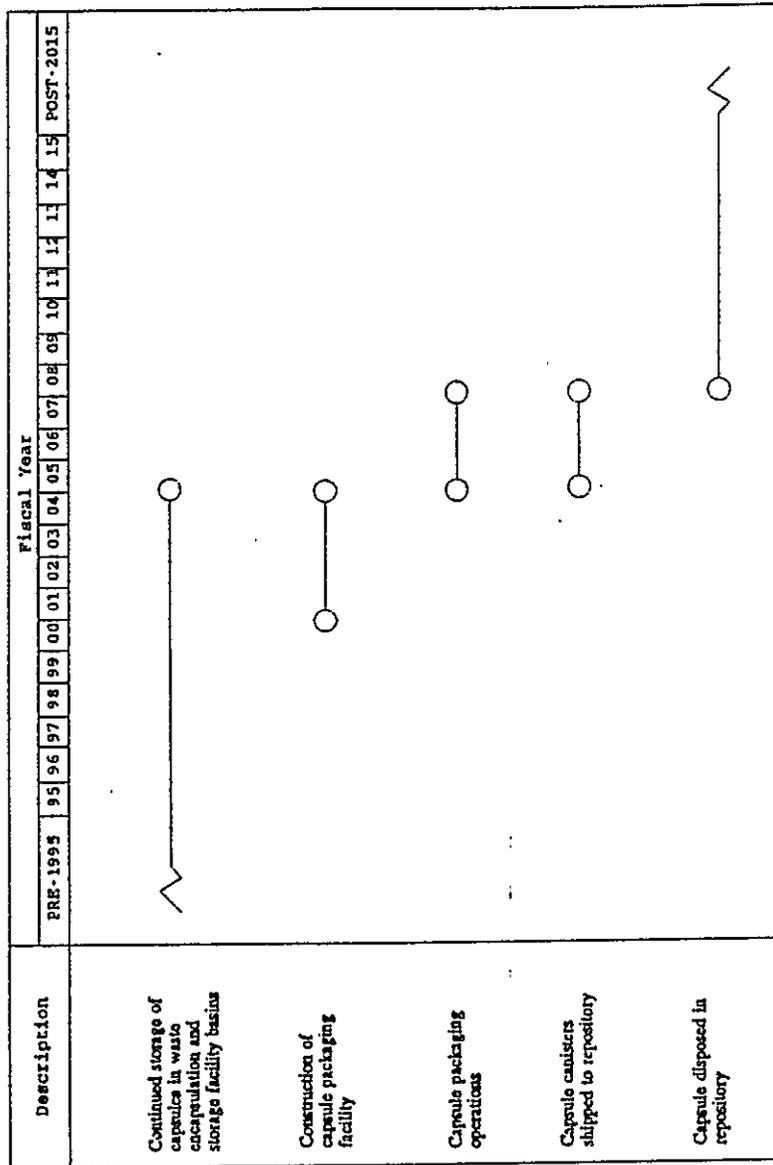
For purposes of this analysis, a repository is assumed to be available after 2010; if the repository is not available at that time, the capsules would continue to be stored at WESF, where additional nuclide decay would occur. This option also includes placement of capsules in overpack canisters (see Appendix A, Section A1.1), and placement of the overpack canisters in an offsite repository.

The *Final Environmental Impact Statement, Disposal of Hanford Defense High-Level, Transuranic and Tank Wastes* (DOE 1987) presents the above described scenario as the preferred alternative for disposal of cesium and strontium capsules. The same alternative is also described as the preferred plan in the *Hanford Waste Management Plan* (DOE-RL 1987). This preferred option for disposal calls for the capsules to be overpacked into canisters suitable for placement in the HLW repository. In this plan, up to four intact capsules would be placed in a canister of the same external dimensions as used by the Hanford Waste Vitrification Plant (HWVP). The characteristics of the HWVP canister are documented in the *Hanford Waste Vitrification Plant Preliminary Waste Form and Canister Description--Fiscal Year 1990 Update* (Colburn 1990). The internal structure of the canister would be modified to restrain the capsules and to provide heat transfer to the canister walls. Modifications to existing facilities (such as WESF) on the Hanford Site may be required to overpack the capsules.

In July 1985, the U.S. Department of Energy's (DOE) Office of Civilian Radioactive Waste Management (OCRWM) established a waste acceptance process as the means by which HLW producers would be allowed to dispose of wastes in the commercial repository. The waste acceptance process is generic because it is intended to accommodate any HLW form other than spent fuel. The Waste Acceptance Preliminary Specifications (WAPS) are based on the waste form and canister description (WCD). The waste acceptance process requires that OCRWM prepare WAPS for each waste form. The WAPS will be prepared for the cesium and strontium capsule waste form once the WCD is issued.

The WAPS for borosilicate glass have been drafted and are currently in review. Although these WAPS are tailored for borosilicate glass, much of their bases derive from the U.S. Code of Federal Regulations (CFR) or limitations to repository handling capability. Attachment 1 of Appendix C discusses the bases for waste acceptance requirements for borosilicate glass waste forms. Because the existing WAPS are based on regulatory requirements or repository design constraints, they provide useful guidance for the development of nonborosilicate glass waste forms. The WAPS for the Defense Waste Processing Facility (DWPF) are summarized in Table 3-1 (from DOE 1989). Of the 21 individual specifications shown in Table 3-1, all but three can be accommodated in the capsule overpack concept. A discussion of the three problematic specifications follows.

Figure 3-4. Schedule of Activities for the Tri-Party Agreement
Alternative: Storage and Disposal of Cesium Strontium
Capsules (Reference Alternative).



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Table 3-1. Waste Acceptance Preliminary Specifications for the Defense Waste Processing Facility High-Level Waste Form.

Specification	Content summary
1.1 Chemical composition	Provide projection of product elemental and phase compositions (with ranges) for life of facility for all elements in concentrations >0.5 wt%.
1.2 Radionuclide inventory	Provide canister and aggregate inventories of all radionuclides with half-life periods longer than 10 years with concentrations >0.05 Ci% at any time up to 1,100 years after production.
1.3 Radionuclide release	Production samples must show normalized releases of <1 g/m ² -d for MCC-1 tests or equivalent.
1.4 Chemical and phase stability	Report product transition temperature and time-temperature-transformation ranges. Certify that waste form has not exceeded transition temperature during storage and is at least 100 °C below transition temperature at shipment.
2.1 Canister material	Austenitic stainless steel.
2.2 Canister fabrication and closure	Leaktight according to ANSI ^a N14.5-1977.
2.3 Identification and labeling	Alphanumeric code on each canister visible from top or side of canister. Characters at least 92 points in height.
3.1 Free liquid	None allowed in canister.
3.2 Gases	None allowed in canister except helium, argon, air, or ₂ other inert. Internal pressure not to exceed 7 lb/in ² (gage) at 25 °C.
3.3 Explosiveness, pyrophoricity, and combustibility	No explosive, pyrophoric, or combustible materials allowed in canister.
3.4 Organic materials	None allowed in canister.
3.5 Free volume	Less than 20% of canister volume.
3.6 External contamination	Less than 220 alpha dpm/100 cm ² . Less than 2,200 beta-gamma dpm/100 cm ² .
3.7 Heat generation	Less than 800 W/canister reported to ±15%. Provide projections and actuals.
3.8 Maximum dose rate	Less than 10 ⁵ rem/h gamma and 10 ³ rem/h neutron at surface. Provide projections and actuals.
3.9 Chemical compatibility	Document reactivity between waste form and canister.
3.10 Subcriticality	$K_{eff} < 0.95$.
3.13 Handling features	Provide grapple and canister with lifting flange. Grapple must be remotely operable within a right-circular cylindrical cavity with diameter equal to that of the canistered waste form.
4.0 Quality assurance	Provide quality assurance program complying with OGR ^b /B-14.

^aANSI = American National Standards Institute.

^bOGR = Office of Geologic Repositories.

3.2.1 Waste Acceptance Preliminary Specification 1.3: Specification for Radionuclide Release Properties

The current reference version of the WAPS for the DWPF stipulates that the waste form must have a normalized release limit for certain matrix elements and radionuclides of less than $1 \text{ g/m}^2\text{-d}$ averaged over the 28-day MCC-1 test. Although a draft revision to this specification would change the test protocol, the changes, if adopted, would not affect the conclusion herein (see Appendix C, Section C4.2).

Most investigators agree that glass dissolution results from the associated matrix dissolution of silica. Testing with simulated NCAW glass has shown cesium release to be within a factor of two times greater than silica. Strontium release is less than silica (Goles and Nakaoka 1990). If it is assumed that waste glass has a $1 \text{ g/m}^2\text{-d}$ release rate and that this is related to the solubility of silica (i.e., release rate is proportional to solubility), then a comparison between cesium chloride, SrF_2 , and waste glass can be made by virtue of solubility. The aqueous solubility of amorphous silica, or waste glass in this case, is 50 mg of silicon/L at room temperature (Fournier and Rowe 1977). The room temperature solubility of cesium chloride is 1,300 mg of cesium/L, and for SrF_2 it is 85 mg of strontium/L (Weast 1987). Clearly, cesium chloride is much worse than glass in terms of matrix solubility and would likely not be acceptable as a waste form. The solubility of SrF_2 is much closer to that of silica, but still higher by almost a factor of two. Release rates from SrF_2 may be marginally acceptable, but a more detailed analysis will be needed.

Because of the relatively short half-life periods of ^{90}Sr and ^{137}Cs (27.7 years and 30 years, respectively), it may be possible to demonstrate that the actual radionuclide releases will be inconsequentially small by the end of the 1,000-year repository containment period. Such an analysis has not been fully pursued and would be complicated by the following considerations:

1. Long-lived impurities such as ^{135}Cs might contain enough residual radioactivity at the end of the repository containment period to invalidate the concept.
2. Complete containment of the highly soluble materials cannot be ensured for the first 1,000 years because of the high reactivity between the cesium and strontium salts and the capsule and canister materials (see discussion of WAPS 3.9 in Section 3.2.3).
3. The current specification for borosilicate glass does not permit any computation of the aging of the waste form. Durability is to be evaluated at the time of production.
4. Preliminary assessments from the geologic repository project indicate that the comparatively high solubility of cesium chloride and the uncertain chemical behavior of SrF_2 will preclude their direct use in the repository because of complications in the licensing process.

3.2.2 Waste Acceptance Preliminary Specification 1.4: Chemical and Phase Stability (10 CFR 60.135[a][2])

This specification is designed to document the thermal conditions under which the waste form will undergo significant chemical or physical changes. Although the specification is tailored specifically for borosilicate glasses, it is clear that a waste form that is not stable under expected storage or repository conditions will not be acceptable. Neither cesium chloride nor SrF_2 is subject to decomposition within the temperature ranges expected for storage or repository disposal; however, impurities can lower the decomposition temperature into this range. It will be necessary to negotiate the waste form's chemical and phase stability requirements with the repository before adopting this disposal approach.

The thermal phenomena of interest for the waste form are phase transitions, such as solid phase changes and melting. Solid phase changes that increase the waste form volume could potentially cause a rupture of the capsule. Production of a liquid phase by melting could also accelerate internal corrosion of the capsule. Strontium fluoride with sodium fluoride (NaF) impurities has a minimum melting temperature of $850 \text{ }^\circ\text{C} \pm 5 \text{ }^\circ\text{C}$ (Fullam 1977) and has no known phase transformations below this temperature (Watrous and Chen 1978). Storage of SrF_2 below $850 \text{ }^\circ\text{C}$ should be adequate to avoid liquid phase volume expansion and the resulting capsule stresses. Cesium chloride with impurities has a solid phase transformation at $465 \text{ }^\circ\text{C}$ (Fullam 1971). The minimum melting point of cesium chloride with 5 wt% NaCl and potassium chloride (KCl) is $580 \text{ }^\circ\text{C}$ (Fullam 1971), although a cesium chloride-ferric chloride (FeCl_3) mixture can melt at a temperature as low as $270 \text{ }^\circ\text{C}$. Because the cesium chloride capsules are melt-cast, the voids produced on cooling should permit the volume expansion that accompanies the solid-to-liquid transformation to occur without stressing the capsule.

3.2.3 Waste Acceptance Preliminary Specification 3.9: Chemical Compatibility (10 CFR 60.134[a][1])

The WAPS for borosilicate glass require that the contents of the canistered waste form shall not corrode the canister such that there will be an adverse effect on normal handling during storage, transportation, and repository operation. The extent of corrosiveness and chemical reactivity among the waste form, canister, and any filler materials must also be documented, as well as corrosive chemical interactions, and any reaction products generated within the canistered waste forms.

The chemical compatibilities of cesium chloride and SrF_2 with the capsule containment materials have been studied extensively (Fullam 1972, 1981; Bryan 1987), including the effects of impurities and daughter products. The principal impurities for cesium chloride are sodium chloride (NaCl) and KCl, while numerous metal fluorides, principally barium fluoride (BaF_2), calcium fluoride (CaF_2), and NaF are found with SrF_2 . Because ^{137}Cs decays to ^{137}Ba , the cesium chloride decays to BaCl , which is chemically unstable, with the reaction $2\text{BaCl} \rightarrow \text{BaCl}_2 + \text{Ba}$. The ^{90}Sr decays to ^{90}Y , which further decays to ^{90}Zr . As with BaCl , ZrF_2 is chemically unstable, with reaction $2\text{ZrF}_2 \rightarrow \text{ZrF}_4 + \text{Zr}$.

Long-term strontium corrosion studies (Fullam 1981) found that chemical attack of the Hastelloy C-276 inner capsule containment material results primarily from impurities in the SrF_2 . Once the critical impurities are consumed by the corrosion reaction (up to 12,000 h), the rate of chemical attack decreases to a low level. Fullam concludes that, for heat source applications, maintaining the temperature of the capsule wall below 800 °C would adequately contain the SrF_2 for 10 to 20 years. Because the reacting impurities are expected to be consumed by that time, additional corrosion is expected to be minimal.

Long-term cesium studies (Bryan 1987) examined cesium chloride capsules after extended storage at 350 °C to 450 °C. Linear extrapolations of initial corrosion rates suggested capsule wall failure in 25 to 35 years. Corrosion rates at lower temperatures appeared to be much slower. Moreover, because corrosion appears to result mostly from impurities, it is expected that the rate would decrease as the impurities are consumed.

The cesium and strontium capsules are assumed to be stored in the WESF basins until 2010. If required, before 2010 the capsules may be removed from WESF and placed in overpack canisters. If required, the canisters will be transported to an onsite interim storage facility.

3.3 TRI-PARTY AGREEMENT ALTERNATIVE: VITRIFICATION OF CESIUM AND STRONTIUM CAPSULES

Under this alternative (vitrification and disposal), both the cesium and strontium capsules will be dismantled and the contents removed. The ^{137}Cs and ^{90}Sr will then be combined with the HLW from the single-shell tanks and double-shell tanks. The HLW would be vitrified into borosilicate glass and placed into a lag-storage facility where it would await shipment to a permanent HLW repository for disposal.

A flowsheet for the vitrification alternative for cesium and strontium capsules is shown in Figure 3-5. The basic components of this option include continued storage of the capsules at WESF, decladding of the capsules, blending the capsule contents with planned HLW vitrification plant feed for eventual vitrification, interim storage of the glass canisters, and shipment to a permanent HLW repository for disposal.

As part of the Tri-Party Agreement, the vitrification of the cesium and strontium salts in the HWVP¹ was identified as an alternative to overpacking.

¹As of September 30, 1994, the design of and the schedule for the HWVP is on an "Administrative Hold." The data and conclusions presented in this data package are solely based on information presented in the *Feasibility Study for the Processing of Hanford Site Cesium and Strontium Isotopic Sources in the Hanford Waste Vitrification Plant*, WHC-EP-0460 (Anantatmula et al. 1991).

Several options exist for processing the capsule waste at WESF to produce a waste feed stream acceptable to HWVP: (1) do not perform any chemical separation and prepare the waste either for blending with neutralized current acid waste (NCAW) and/or complexant concentrate (CC) waste or for vitrification as a separate HWVP feed stream and (2) remove the halides from the cesium chloride and strontium fluoride before blending with NCAW and/or CC waste, or vitrify as a separate HWVP feed stream. The options are shown in flowsheet fashion in Figure 3-5.

With appropriate modification to either or both facilities, the decladding of the capsules and the processing of the contents can be performed either in the WESF or in the HWVP.

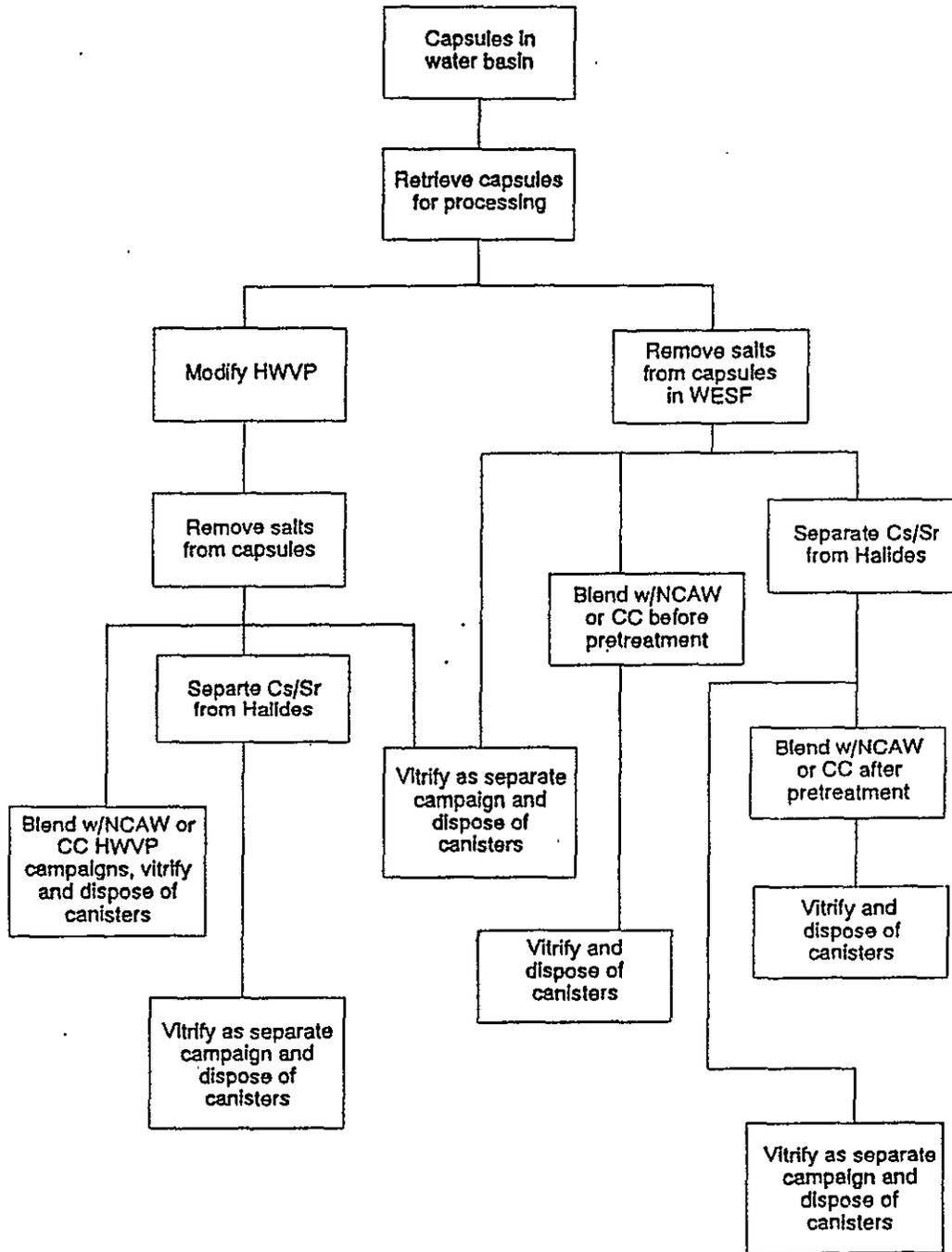
3.4 ONSITE STABILIZATION AND DISPOSAL ALTERNATIVE FOR REMEDIATING CESIUM AND STRONTIUM CAPSULES

Under this alternative, the capsules would be retrieved from the storage area, packaged into canisters and placed in a long-term storage facility for permanent disposal.

A flowsheet for the onsite stabilization and disposal alternative for cesium and strontium capsules is shown in Figure 3-6. The basic components of this option include continued storage of capsules in the WESF basins until a capsule packaging facility can be constructed to allow loading of capsules in canisters. With modification, WESF may be suitable for overpacking. (See Appendix A, Section A1.1.). This option also includes placement of the capsule canisters in dry well storage following production, and placement of surface barriers over the dry wells after all capsule canisters have been placed (see Appendix A, Sections A2.1 and A2.2).

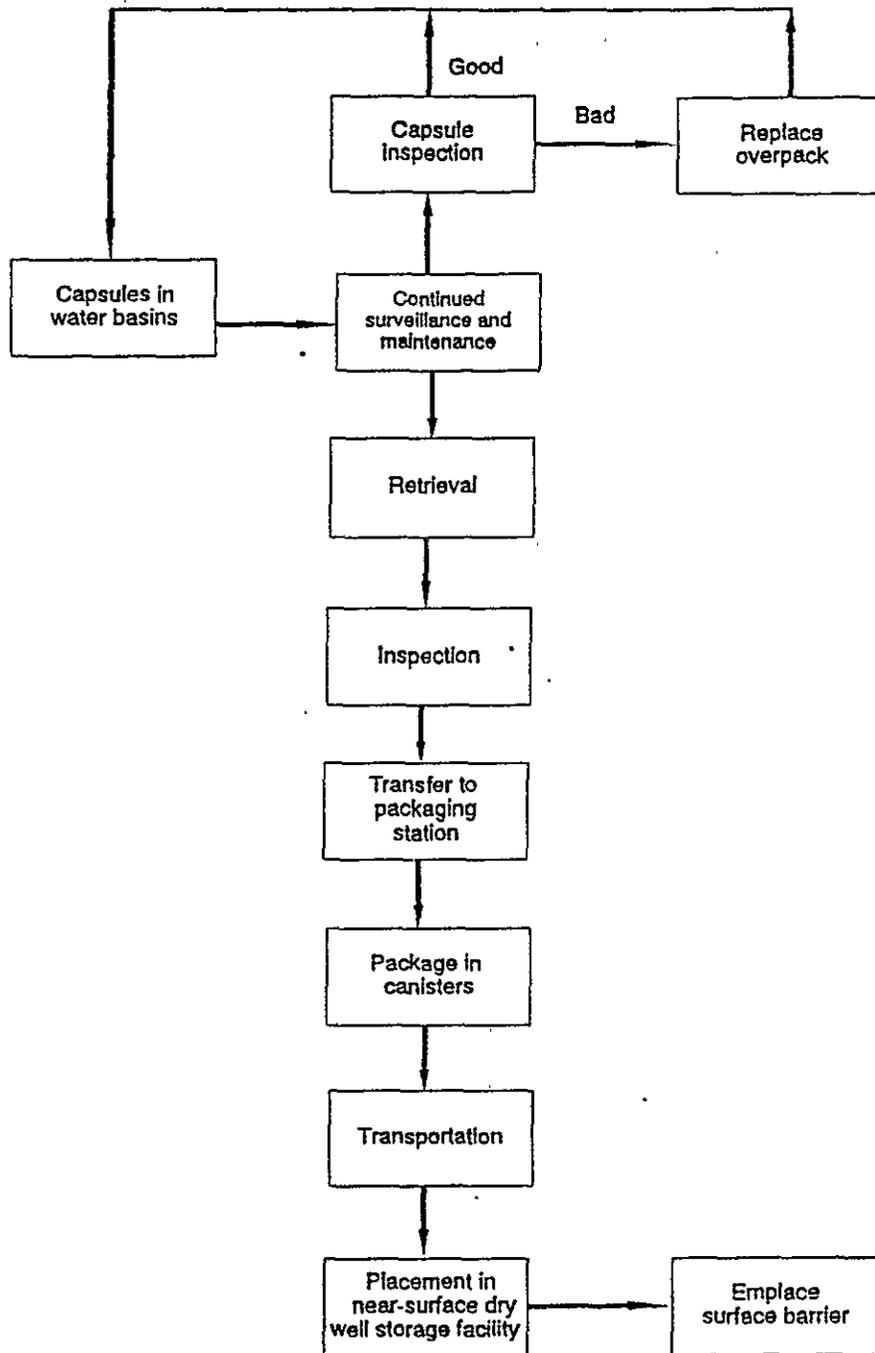
This option is identical to the no-action alternative (Section 3.1), except that surface barriers are placed over the dry wells to complete disposal. The area of the dry well site for barrier application is 3.8 ha. Figure 3-7 provides a rough schedule for the above operations.

Figure 3-5. Options for Cesium and Strontium Capsule Vitrification.



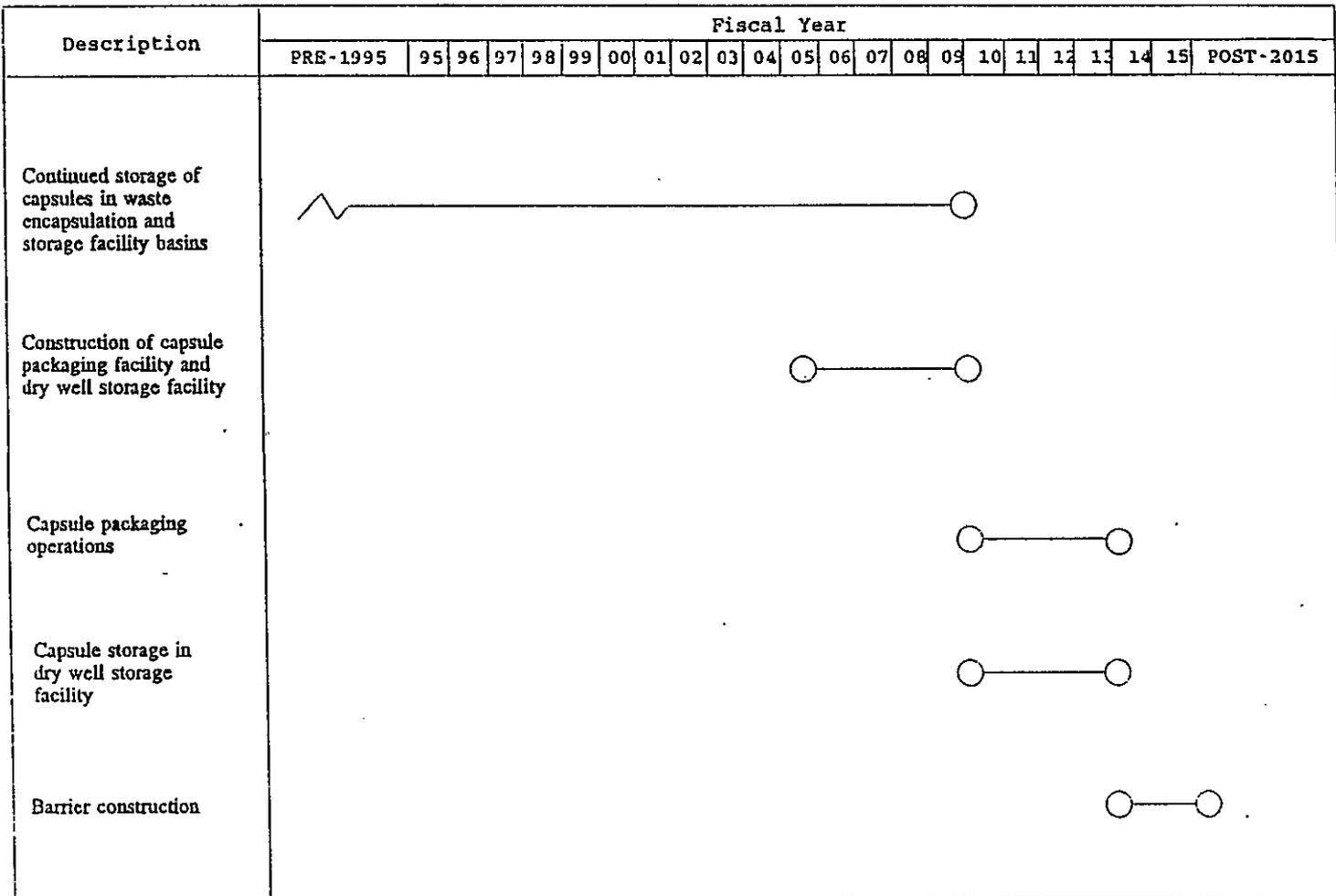
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Figure 3-6. Flowsheet for Onsite Stabilization and Disposal of Capsules.



DP097\FIG-3-6
JEB (9/21/94)

Figure 3-7. Schedule of Activities for Onsite Stabilization and Disposal Alternative for Capsules.



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4.0 ENGINEERING DATA

This chapter contains data regarding the waste disposal alternatives presented in Chapter 3.0 for disposal of cesium and strontium capsules. Data are provided on projected resource usage, personnel requirements, radiological and non-radiological environmental releases, and cost estimates.

Tables 4-1 through 4-8 reflect the estimated totals for resources, nonradioactive releases, manpower, and costs as developed for the capsule management options. Costs are in fiscal year 1995 dollars. Table 4-9 summarizes the costs for the capsule management options.

Table 4-10 shows the estimated decontamination and decommissioning (D&D) costs for the three options presented. The D&D costs are estimated to be 20 percent of the Capsule Packaging Facility (CPF) construction costs. Table 4-11 shows the estimated airborne radioactive releases from standby operation of the WESF.

Table 4-12 shows the estimates of resources, nonradioactive releases, manpower, and cost for surveillance of the Dry Well Storage Facility (DWSF) under the no-action option. The estimates cover the period between 2010 and 2090. The values representing any alternative period would be directly proportional to these estimates.

All data presented in Tables 1 through 12 were generated using information from *Hanford Defense Waste Environmental Impact Statement Engineering Data Package: Strontium and Cesium Capsules*, SD-WM-DB-003, Rev. 2 (RHO 1985). Only cost estimates were updated to reflect 1995 dollars, based on cost adjustment factors presented in Appendix B, Table B-2.

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Table 4-1. Resource Usage for Each Disposal Mode: Construction Phase for Cesium and Strontium Capsule.

Resource	Units	No Action	Onsite stabilization and disposal		Geologic disposal	Vitrification and disposal
			Barrier	Total*	Storage and containerization	
Land						
Surface committed	ha	3.3	--	5.3	1.5	
Temporarily	ha	1.5	--	1.5	1.5	
Permanently	ha	1.8	--	3.8	--	
Subsurface committed	ha	--	2.0	--	--	
Water	m ³	390	170,000	170,000	390	
Energy						
Electrical	GWh	0.08	-	0.08	0.01	
Propane	m ³	580	-	580	140	
Diesel fuel	m ³	-	1,500	1,500	0.0	
Gasoline	m ³	200	17	220	50	
Materials						
Concrete	m ³	2,300	-	2,300	2,300	
Steel	t	810	-	810	180	**
Stainless steel	t	17	-	17	15	
Iron	t	3.6	-	3.6	3.2	
Chromium	t	1.6	-	1.6	1.4	
Nickel	t					
Copper	t	4.1	-	4.1	4	
Aluminum	t	0.7	-	0.7	0.7	
Zinc	t	0.6	-	0.6	0.6	
Lead	t	25	-	25	25	
Lumber	m ³	290	-	290	290	
Argon	m ³	90	-	90	80	
Silica sand	m ³	-	-	-	-	
Soil	m ³	-	56,000	56,000	-	
Basalt riprap	m ³	-	170,000	170,000	-	
Rock filter	m ³	-	12,000	12,000	-	

*Total equals barrier values plus continued storage values except land.

**Resource usage, costs, and personnel requirements for Vitrification and Disposal option. Under this alternative, the contents of the Cs and Sr capsules are sent to an existing Vitrification Plant for processing. The mass requiring vitrification from the Cs and Sr capsules is less than 1 percent of the tank materials that require vitrification. Therefore, the processing of the capsule contents do not impose any significant additional resource (both usage and personnel) requirements on the vitrification plant. Thus no separate estimate of impacts is provided.

Table 4-2. Personnel Requirements for Each Disposal Mode:
Construction Phase for Cesium and Strontium Capsules
(worker years).

Module	No Action	Onsite stabilization and disposal		Geologic disposal	Vitrification and disposal
		Barrier	Total*	Storage and containerization	
Process					**
Radiation	0	-	-	0	
Nonradiation	100	-	100	100	
Storage/disposal					
Radiation	0	11	11	-	
Nonradiation	110	0.69	110	-	
Total					
Radiation	0	11	11	0	
Nonradiation	210	0.69	210	100	

*Total equals barrier values plus continued storage values except land.

**Resource usage, costs, and personnel requirements for Vitrification and Disposal option. Under this alternative, the contents of the Cs and Sr capsules are sent to an existing Vitrification Plant for processing. The mass requiring vitrification from the Cs and Sr capsules is less than 1 percent of the tank materials that require vitrification. Therefore, the processing of the capsule contents do not impose any significant additional resource (both usage and personnel) requirements on the vitrification plant. Thus no separate estimate of impacts is provided.

Table 4-3. Costs for Each Disposal Mode: Construction
Phase for Cesium and Strontium Capsules (millions of 1995 dollars).

Resource	No Action	Onsite stabilization and disposal		Geologic disposal	Vitrification and disposal
		Barrier	Total*	Storage and containerization	
Capital	64	11.1	75	32	**
Research and development	19	-	19	14	
Total	81	11.1	94	46	

*Total equals barrier values plus continued storage values except land.

**Resource usage, costs, and personnel requirements for Vitrification and Disposal option. Under this alternative, the contents of the Cs and Sr capsules are sent to an existing Vitrification Plant for processing. The mass requiring vitrification from the Cs and Sr capsules is less than 1 percent of the tank materials that require vitrification. Therefore, the processing of the capsule contents do not impose any significant additional resource (both usage and personnel) requirements on the vitrification plant. Thus no separate estimate of impacts is provided.

Table 4-4. Resource Usage for Each Disposal Mode: Operational Phase for Cesium and Strontium Capsules.

Resource	Units	No Action	Onsite stabilization and disposal		Geologic disposal	Vitrification and disposal
			Barrier	Total*	Storage and containerization	
Water	m ³	6,000,000	-	6,000,000	15,000,000	**
Energy						
Electrical	GWh	120	-	110	43	
Diesel fuel	m ³	2.6	-	2.6	0	
Coal	t	43,000	-	43,000	18,000	
Materials						
Steel	t	220	-	220	160	
Silica sand	m ³	530	-	530	-	

**Resource usage, costs, and personnel requirements for Vitrification and Disposal option. Under this alternative, the contents of the Cs and Sr capsules are sent to an existing Vitrification Plant for processing. The mass requiring vitrification from the Cs and Sr capsules is less than 1 percent of the tank materials that require vitrification. Therefore, the processing of the capsule contents do not impose any significant additional resource (both usage and personnel) requirements on the vitrification plant. Thus no separate estimate of impacts is provided.

Table 4-5. Personnel Requirements for Each Disposal Mode: Operational Phase for Cesium and Strontium Capsules (worker years).

Module	No Action	Onsite stabilization and disposal		Geologic disposal	Vitrification and disposal
		Barrier	Total*	Storage and containerization	
Process					**
Radiation	140	-	140	17	
Nonradiation	54	-	54	34	
Storage/disposal					
Radiation	700	-	260	65	
Nonradiation	190	-	100	25	
Transport					
Radiation	-	-	-	-	
Nonradiation	-	-	-	-	
Total					
Radiation	840	-	400	140	
Nonradiation	240	-	150	59	

Note: Surveillance of the Dry Well Storage Facility under the continued storage option includes 440 worker years of radiation personnel and 87 worker years for nonradiation personnel.

*Total equals barrier values plus continued storage values except land.

**Resource usage, costs, and personnel requirements for Vitrification and Disposal option. Under this alternative, the contents of the Cs and Sr capsules are sent to an existing Vitrification Plant for processing. The mass requiring vitrification from the Cs and Sr capsules is less than 1 percent of the tank materials that require vitrification. Therefore, the processing of the capsule contents do not impose any significant additional resource (both usage and personnel) requirements on the vitrification plant. Thus no separate estimate of impacts is provided.

Table 4-6. Costs for Each Disposal Mode: Operational Phase for Cesium and Strontium Capsules (millions of 1995 dollars).

Module	No Action	Onsite stabilization and disposal		Geologic disposal	Vitrification and disposal
		Barrier	Total*	Storage and containerization	
Processing	46	-	35	24	**
Transportation	125	-	-	-	
Storage/disposal	55	-	95	24	
Surveillance	226	-	130	48	

*For surveillance of the DWSF from 2010 to 2090.

**Resource usage, costs, and personnel requirements for Vitrification and Disposal option. Under this alternative, the contents of the Cs and Sr capsules are sent to an existing Vitrification Plant for processing. The mass requiring vitrification from the Cs and Sr capsules is less than 1 percent of the tank materials that require vitrification. Therefore, the processing of the capsule contents do not impose any significant additional resource (both usage and personnel) requirements on the vitrification plant. Thus no separate estimate of impacts is provided.

Table 4-7. Nonradioactive Emissions for the Construction Phase for Cesium and Strontium Capsules.

Emission	Units	No Action	Onsite stabilization and disposal		Geologic disposal	Vitrification and disposal
			Barrier	Total*	Storage and containerization	
CO	t	96	46	140	24	**
Hydrocarbons	t	3.5	5.6	9.1	0.9	
NOx	t	3.6	23	27	0.9	
SOx	t	0.1	5.4	5.5	.0	
Particulates	t	0.3	3.0	3.3	0.1	
Thermal	TJ	23	59	82	5.5	
Aldehydes	t	-	0.76	0.8	-	
Dust	t	-	120	120	-	
Organic acids	t	-	0.5	0.5	-	

*Total equals barrier values plus continued storage value.

**Resource usage, costs, and personnel requirements for Vitrification and Disposal option. Under this alternative, the contents of the Cs and Sr capsules are sent to an existing Vitrification Plant for processing. The mass requiring vitrification from the Cs and Sr capsules is less than 1 percent of the tank materials that require vitrification. Therefore, the processing of the capsule contents do not impose any significant additional resource (both usage and personnel) requirements on the vitrification plant. Thus no separate estimate of impacts is provided.

Table 4-8. Nonradioactive Emissions for the Operational Phase for Cesium and Strontium Capsules.

Emission	Units	No Action			Onsite stabilization and disposal						Geologic disposal	Vitrification and disposal
					Barrier			Total*			Storage and containerization	
		1990 to 2010	2010 to 2014	2014 to 2090	1990 to 2010	2010 to 2014	2014 to 2090	1990 to 2010	2010 to 2014	2014 to 2090	1990 to 1995	
CO	t	15	6.3	-	-	-	-	15	6.3	-	8.8	***
Hydrocarbons	t	4.5	1.9	-	-	-	-	4.5	1.9	-	2.6	
NOx	t	75	32	-	-	-	-	75	32	-	44	
SOx	t	230	100	-	-	-	-	230	100	-	1,140	
Particulates	t	11	4.8	-	-	-	-	11	4.8	-	6.7	
Thermal	TJ**	1,110	420	32	-	-	-	1,100	420	-	610	
Aldehydes	t	-	-	-	-	-	-	-	-	-	-	
Dust	t	-	-	-	-	-	-	-	-	-	-	

*Total equals barrier values plus continued storage value.

**Terajoules (1E+12 joules).

***Resource usage, costs, and personnel requirements for Vitrification and Disposal option.

Under this alternative, the contents of the Cs and Sr capsules are sent to an existing Vitrification Plant for processing. The mass requiring vitrification from the Cs and Sr capsules is less than 1 percent of the tank materials that require vitrification. Therefore, the processing of the capsule contents do not impose any significant additional resource (both usage and personnel) requirements on the vitrification plant. Thus no separate estimate of impacts is provided.

Table 4-9. Total Costs for Cesium and Strontium Capsule Management Options (millions of 1995 dollars).

Cost category	No Action	Onsite stabilization and disposal		Geologic disposal	Vitrification and disposal
		Barrier	Total*	Storage and containerization	
Construction phase	83	11	94	46	**
Operational phase	226	-	128	49	
D&D	6.3	-	6.3	6.3	
Total	315.3	11	228.3	101.3	

*Total equals barrier values plus continued storage values except land.
 **Resource usage, costs, and personnel requirements for Vitrification and Disposal option. Under this alternative, the contents of the Cs and Sr capsules are sent to an existing Vitrification Plant for processing. The mass requiring vitrification from the Cs and Sr capsules is less than 1 percent of the tank materials that require vitrification. Therefore, the processing of the capsule contents do not impose any significant additional resource (both usage and personnel) requirements on the vitrification plant. Thus no separate estimate of impacts is provided.

Table 4-10. Decontamination and Decommissioning Costs for Cesium and Strontium Capsules (millions of 1995 dollars).

Option	Estimated cost* (millions of 1995 dollars)
No Action	6.3
Onsite stabilization and disposal	6.3
Geologic disposal (storage and containerization portion)	6.3
Vitrification and disposal	**

*20% of CPF construction costs.
 **Resource usage, costs, and personnel requirements for Vitrification and Disposal option. Under this alternative, the contents of the Cs and Sr capsules are sent to an existing Vitrification Plant for processing. The mass requiring vitrification from the Cs and Sr capsules is less than 1 percent of the tank materials that require vitrification. Therefore, the processing of the capsule contents do not impose any significant additional resource (both usage and personnel) requirements on the vitrification plant. Thus no separate estimate of impacts is provided.

Table 4-11. Airborne Radioactive Release from Standby Operation of the WESF: Cesium and Strontium Capsule Data.

Release category	Quantity
Alpha concentration in effluent	4 E-15 μ Ci/mL
Total yearly alpha release	1 E-06 Ci
Beta concentration in effluent	4 E-14 μ Ci/mL
Total yearly beta release	1 E-05 Ci

Note: The airborne radioactive release data include the activities of both parent and daughters.

Table 4-12. Resources, Manpower, Cost, and Nonradioactive Emissions of the Dry Well Storage Facility No Action Option, 2010-2090, Cesium and Strontium Capsule Data.

Category	Amount
Resources-electrical, GWh	9.4
Workforce, workeryears	
Radiation	440
Nonradiation	87
Cost, millions of dollars	55
Emissions-thermal, TJ	34

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APPENDIX A

ENGINEERING UNIT PROCESSES

Source: RHO, 1985, *HDW-EIS Engineering Data Package: Strontium and Cesium Capsules*, SD-WM-DB-003, Rev. 2, Rockwell Hanford Company, Richland, Washington.

Where appropriate, information was updated to reflect current estimates of cesium chloride and strontium fluoride capsules and cost estimates recalculated to reflect 1995 dollars.

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LIST OF TERMS

CPF	Capsule Packaging Facility
D&D	Decontamination and decommissioning
DOE	U.S. Department of Energy
DWSF	Dry Well Storage Facility
HWVP	Hanford Waste Vitrification Plant
OCRWM	Office of Civilian Radioactive Waste Management
WAPS	Waste Acceptance Preliminary Specifications
WCD	Waste form and canister description
WESF	Waste Encapsulation and Storage Facility

APPENDIX A

ENGINEERING UNIT PROCESSES

A1.0 PROCESSING WASTES

The preferred option for disposal calls for the capsules to be overpacked into canisters suitable for placement in the high-level waste repository. In this plan, up to four intact capsules would be placed in a canister of the same external dimensions as used by the Hanford Waste Vitrification Plant (HWVP). The characteristics of the HWVP canister are documented in the *Hanford Waste Vitrification Plant Preliminary Waste Form and Canister Description--Fiscal Year 1990 Update* (Colburn 1990). The internal structure of the canister would be modified to restrain the capsules and to provide heat transfer to the canister walls. Modifications to existing facilities (such as Waste Encapsulation and Storage Facility [WESF]) on the Hanford Site may be required to overpack the capsules.

In July 1985, the U.S. Department of Energy (DOE) Office of Civilian Radioactive Waste Management (OCRWM) established a waste acceptance process as the means by which high-level waste producers would be allowed to dispose of wastes in the commercial repository. The waste acceptance process is generic because it is intended to accommodate any high-level waste form other than spent fuel. The Waste Acceptance Preliminary Specifications (WAPS) are based on the waste form and canister description (WCD). The waste acceptance process requires that OCRWM prepare WAPS for each waste form. The WAPS will be prepared for the cesium and strontium capsule waste form once the WCD is issued.

A1.1 CAPSULE OVERPACKAGING

As stated in Chapter 3.0, capsule overpackaging, if any, would likely be accomplished in a modified WESF instead of building a new Capsule Packaging Facility (CPF). However, to be conservative and until planned engineering studies can show a more representative configuration, an CPF with a one canister per day throughput was envisioned for this study. The following key equipment pieces for loading one canister would be operated in essentially the order listed below:

1. Seven-capsule holding vaults.
2. Three-station load/weld machines:
 - At the "exit" station, a canister containing an empty capsule rack is placed on the machine.
 - At the "load" station, an actuator, arm, and grapple connects to the capsule rack and withdraws it vertically at a distance of approximately 2.1 m from the canister. Capsules, handled by

conventional hot cell, manipulators, are loaded onto the rack as it is lowered back into the canister.

- At the "weld" station, a lid is placed on the canister by a manipulator or an in-cell crane. A rotating weld head makes the weld closure.
3. Helium leak test unit.
 4. Ultrasonic weld penetration test unit.
 5. Electro-polishing decontamination tank.
 6. Canister storage pods: an array of twelve 38-cm ID steel sleeves (with lids) that penetrate the cell floor. A wind tunnel below the pods provides forced or natural convection air cooling.

The CPF is envisioned as a series of three hot cells, each 4.9-m wide and 3-m, 14-m, and 7-m long. The cells would be housed in a facility that is 43-m long, 14-m high, and occupies approximately 1,100 m² surface area. The high density, concrete shielding walls of the hot cells would be thick. It is estimated that eight viewing windows and four manipulator pairs will be required.

The number of capsules loaded into each canister varies for different alternatives due to changing (decay induced) allowable heat loads, and is listed in Chapter 2.0, Table 2-2. A process flowsheet is given in Figure A1-1, and the relationship between the CPF and the Dry Well Storage Facility (DWSF) (see Section 3.4) is shown in Figure A1-2.

The final overpack concept is subject to refinement during further analysis. The repository waste acceptance criteria may require improved waste forms and/or multibarrier containment, if the proposed design proves to be inadequate. Similarly, optimization for use in an DWSF may also be necessary during final design stages.

A1.2 FACILITY DECONTAMINATION AND DECOMMISSIONING

The CPF will require some form of final decontamination and decommissioning (D&D) after outliving its usefulness or design life. The D&D could range from fixing contaminants in place followed by entombment, to washing or abrading exposed surfaces, with the resultant wastes undergoing further treatment and/or burial. Since the CPF exists only as a conceptual design, any detailed description of its eventual D&D would be purely speculative. However, to promote an unbiased analysis, some of the costs for D&D must be charged to those alternatives that make extensive use of the hardware. A conservative factor of 20 percent of the original construction cost estimate was added for D&D. This rough estimate has been demonstrated in recent D&D activities, when inflation is correctly taken into account, and hopefully will decrease since newer facilities should be designed to promote cheaper and safer D&D. Indeed, functional design criteria are required to address eventual D&D.

Figure A1-1. Process Flowsheet for the Capsule Packing Facility.

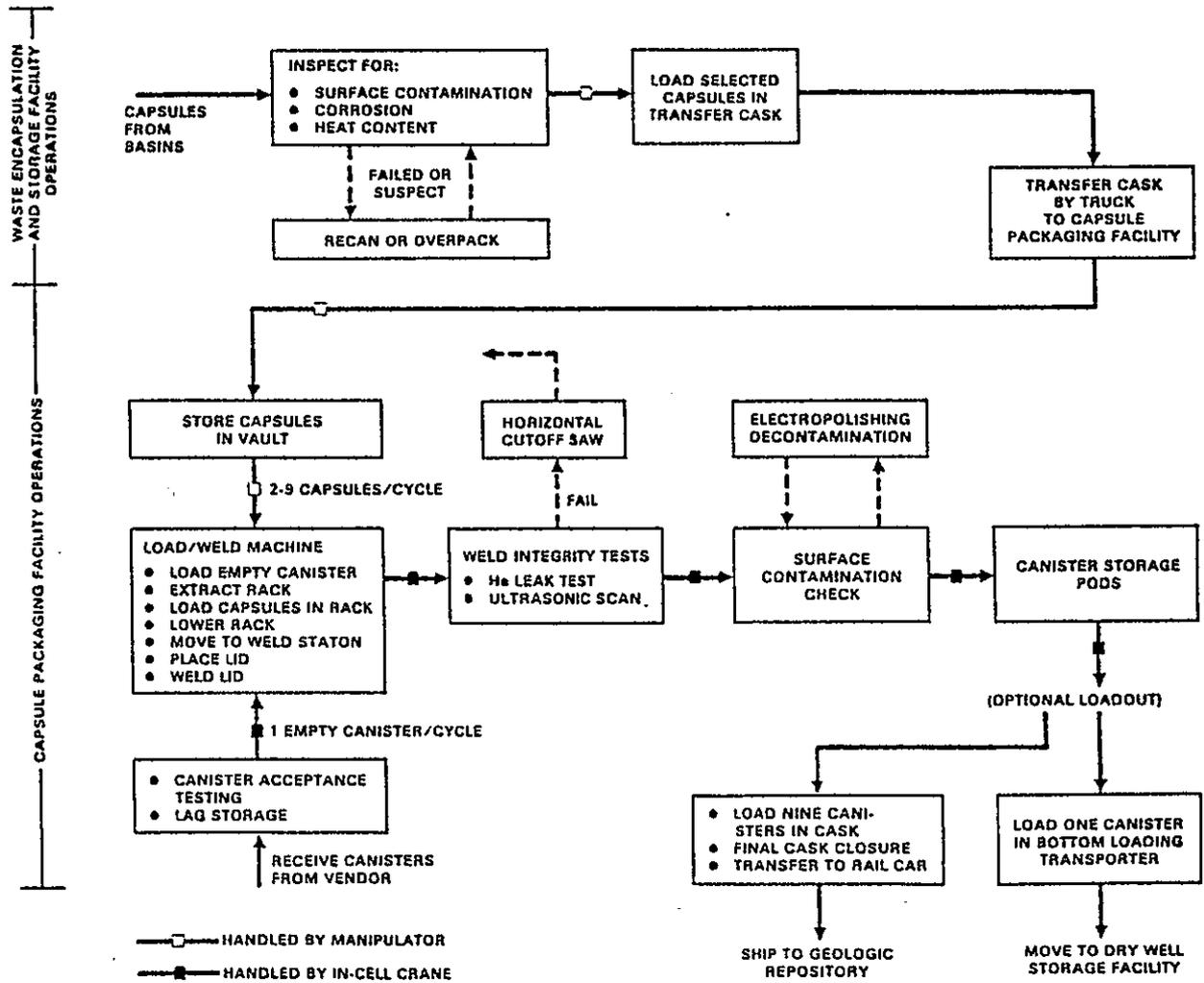
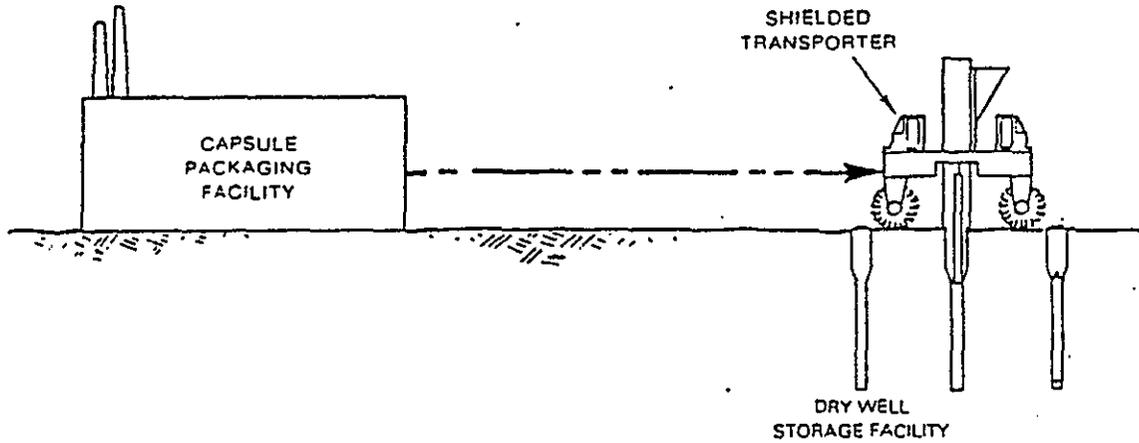


Figure A1-2. Profile View of the Capsule Packing Operations in Relation to the Dry Well Storage.



A2.0 SITE STABILIZATION AND WASTE DISPOSAL

In all cases, site stabilization activities are preceded by radiological surveys, any necessary physical surveying, and site preparation (road building and utilities).

A2.1 BARRIER CONSTRUCTION AND MARKER PLACEMENT

A2.1.1 Protective--Multipurpose "Reference" Barrier

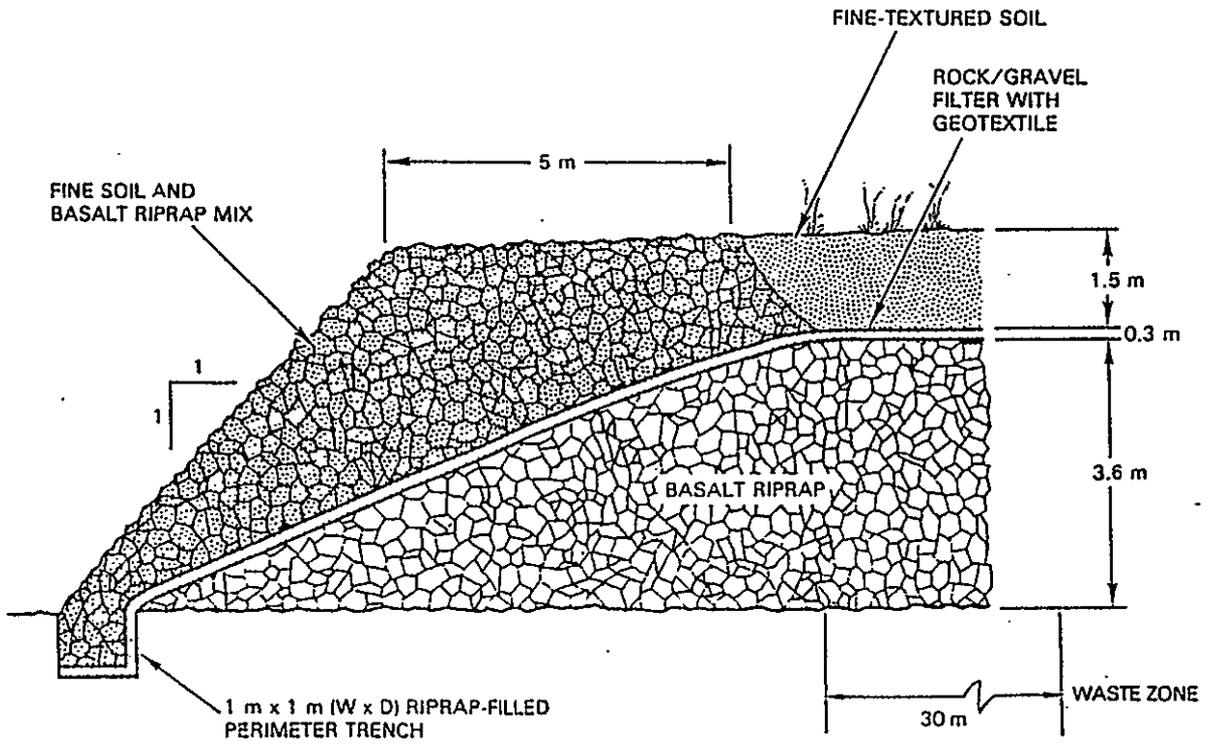
A generic or "reference" barrier (not to be confused with the reference alternative) was designed for application on transuranic (TRU) or high-level waste sites designed for onsite stabilization. Several barrier designs are currently undergoing field tests and, in practice, a range of designs might be employed on a site-specific basis. However, for the HDW-EIS this best-judgment barrier was chosen, subject to future modification once results of field tests and EIS consequences analyses are obtained.

The multilayer barrier consists of a 3.6-m thick basalt riprap layer with a 1.5-m deep basin of revegetated soil placed on top and bounded by a riprap berm. The mound is constructed of 12- to 25-cm diameter basalt riprap. A 30-cm thick layer of fine basalt gravel separates the soil from the riprap. The sides of the barrier are constructed on a 1-to-1 slope providing a 5-m wide berm of riprap for slope protection. A riprap-filled trench is provided at the toe of the barrier to deter lateral animal intrusion. Figure A2-1 illustrates the basic barrier cross section. Soil is mixed with the riprap in the "shoulder" and "toe" of the barrier.

In addition to precluding animal intrusion, the barrier is also designed to inhibit root penetration, water infiltration, wind erosion, and casual human intrusion. The upper revegetated soil layer is credited with retaining precipitation until it can be removed by evapotranspiration. The particle size interface, created with the rock filter, will prevent flow of water until saturation is reached. The soil layer should accommodate up to 50 cm of precipitation, assuming most precipitation falls during seasons with low evapotranspiration rates. Fifty centimeters of precipitation is three times the average and almost twice the observed maximum annual values. Hanford evapotranspiration ranges from 63 cm to 116 cm/yr. The vegetation selected for growth over the barrier is native shallow rooted species, which should be able to recover from periodic impacts such as drought, fire, or grazing. The underlying soil layer is stabilized by the root mass as well as a thin rock layer placed on top that armors the soil against wind erosion. The riprap below serves as a final protection against erosion, as well as inhibiting deep root penetration and casual human intrusion.

Markers, discussed in the next section, will assist with deterring human activity such as farming, excavation, mining, or drilling.

Figure A2-1. Hanford Site Reference Protective Barrier.



DETAIL OF BARRIER PERIMETER

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The depiction of the multilayer barrier as applied over the Near-Surface Dry Well Facility for capsule storage and disposal is provided in Figure A2-2. The soil/basin layer is laterally extended 30 m beyond the surface projection of the waste zone or plume.

A2.1.2 Markers

Despite the concern by some that marking waste disposal sites may only promote interference by inquisitive humans, the generally accepted requirement of attempting to warn future generations has been adopted for this study. No reasonable designs for barriers over near-surface sites can be expected to preclude a dedicated human assault on the waste. Therefore, a redundant system of surface and subsurface markers has been devised as the best available in-place warning mechanism.

During barrier placement, warning markers are installed within the soil and riprap. The markers are placed at varying depths and in a geometrical pattern so any excavation is likely to uncover at least one of the warning markers. The markers budgeted for this analysis are 25 cm ceramic discs emplaced in 1 m diameter of chipped, red granite to enhance the likelihood of detection. Figure A2-3 provides a detailed perspective of this plan. The large ceramic discs were selected for their durability and because their surface could contain more than a simple message or symbol. Other concepts under consideration included smaller clay discs that could be more cheaply produced and more extensively distributed.

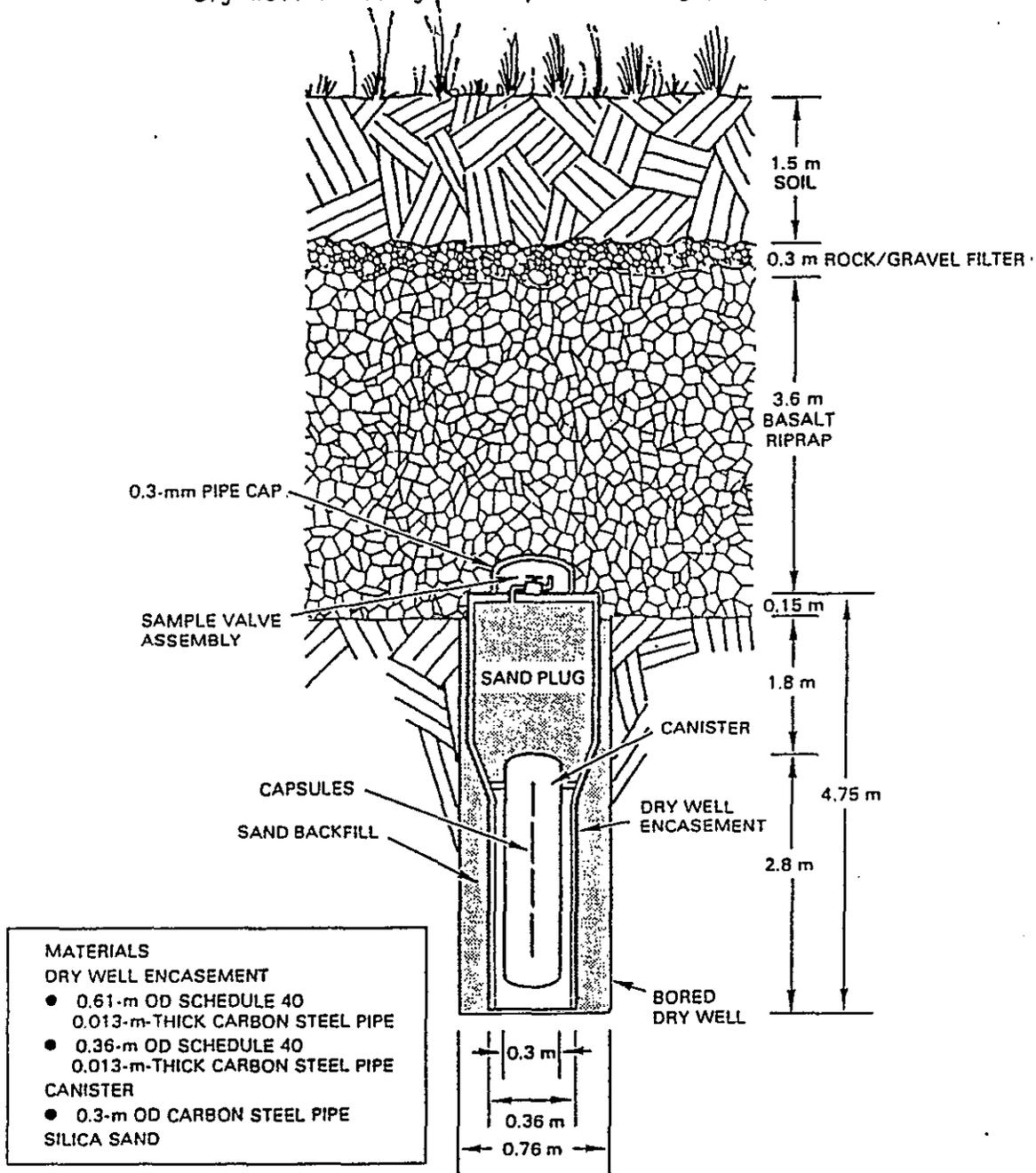
Subsurface markers, however, would not be apparent to the operators of a drilling operation; and, for the sake of preserving the barrier, it would be preferable to warn potential excavators in advance. Therefore, a system of surface markers is also proposed. Selected for cost analysis were inscribed, granite obelisks similar to the "final" design depicted in Figure A2-4. The massive monoliths were proposed to attract notice, inhibit theft, withstand erosion, and provide a large message surface. The latter is critical since proposed, messages must be repeated in various languages, and deep engraving (and therefore large letters) is needed for long-term weathering effects. The actual distribution of these markers is still in question; however, for this study two markers per site were estimated for cost purposes. They may be used to ring the entire site rather than identify specific sites since the barriers themselves serve to provide local delineation of sites (Figure A2-5).

It should be noted that Figure A2-5 serves to delineate a so-called secondary ring of surface markers, which are designed to protect the entire waste disposal area from inadvertent intruders. In addition, there will be primary site markers with a different message content.

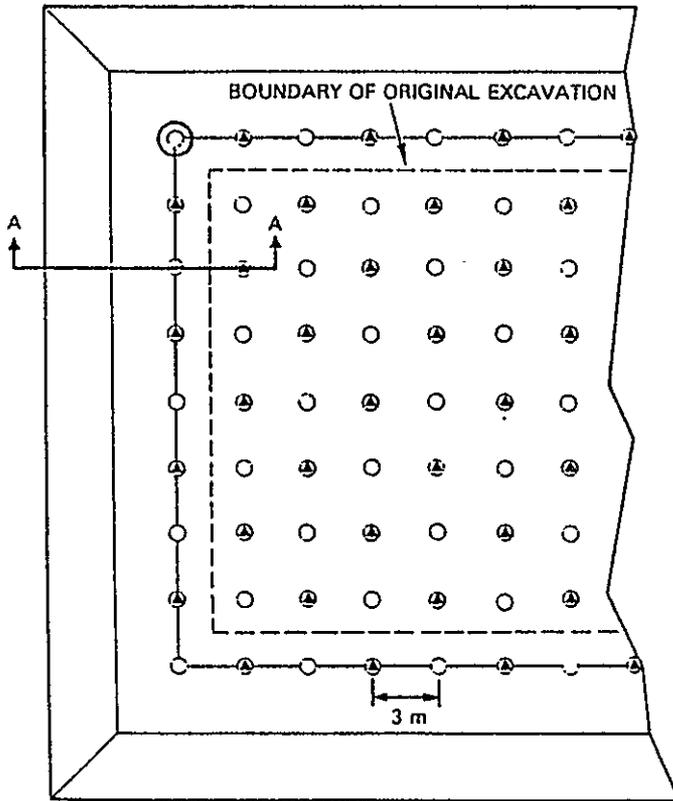
A2.2 CAPSULE STORAGE METHOD

Dry well storage will be used in two alternatives to store the capsule canisters. After the capsules are packaged and released (see Section A1.1), they will be transported by a shield-cask transporter vehicle to dry, well storage. Further analysis is required to optimize the overpack design for dry

Figure A2-2. Multilayer Barrier Applied Over the Near-Surface Dry Well Facility for Capsule Storage/Disposal.

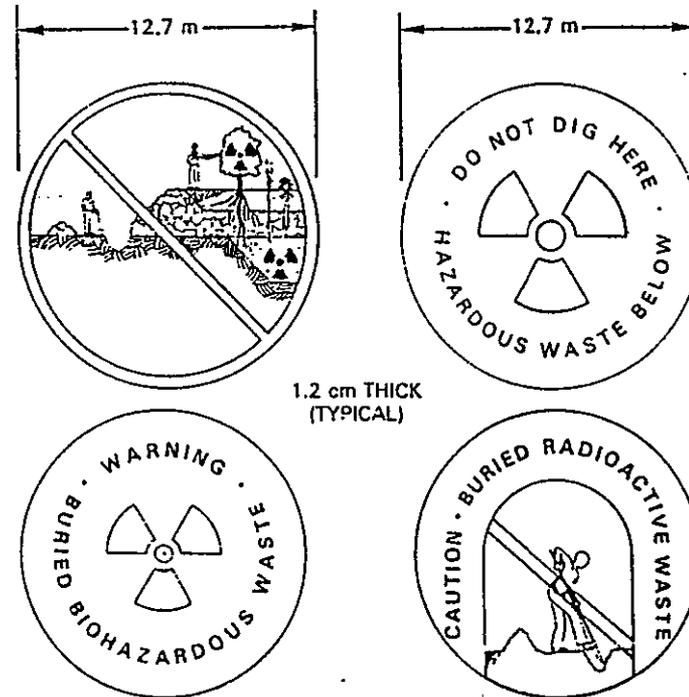
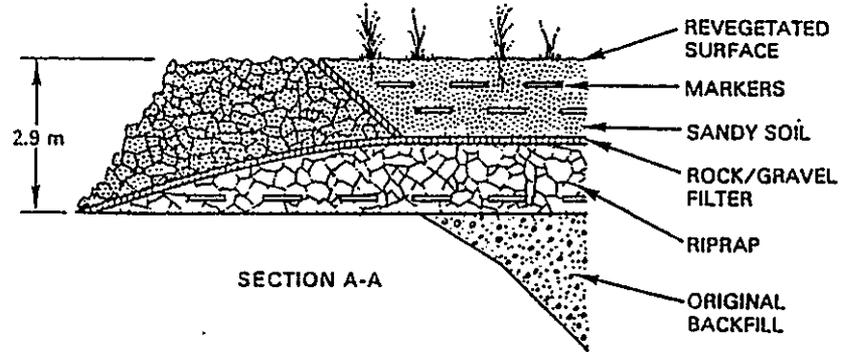


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- MARKER AT 1.8-m DEPTH
- ⊕ MARKER AT 2.5-m DEPTH

MARKERS IN RIPRAP AREA NOT SHOWN



OPTIONAL MESSAGE DISCS

PS8501-89

Figure A2-3. Subsurface Marker Distribution and Marker Detail.

Figure A2-4. Granite Surface Marker Configuration and Placement.

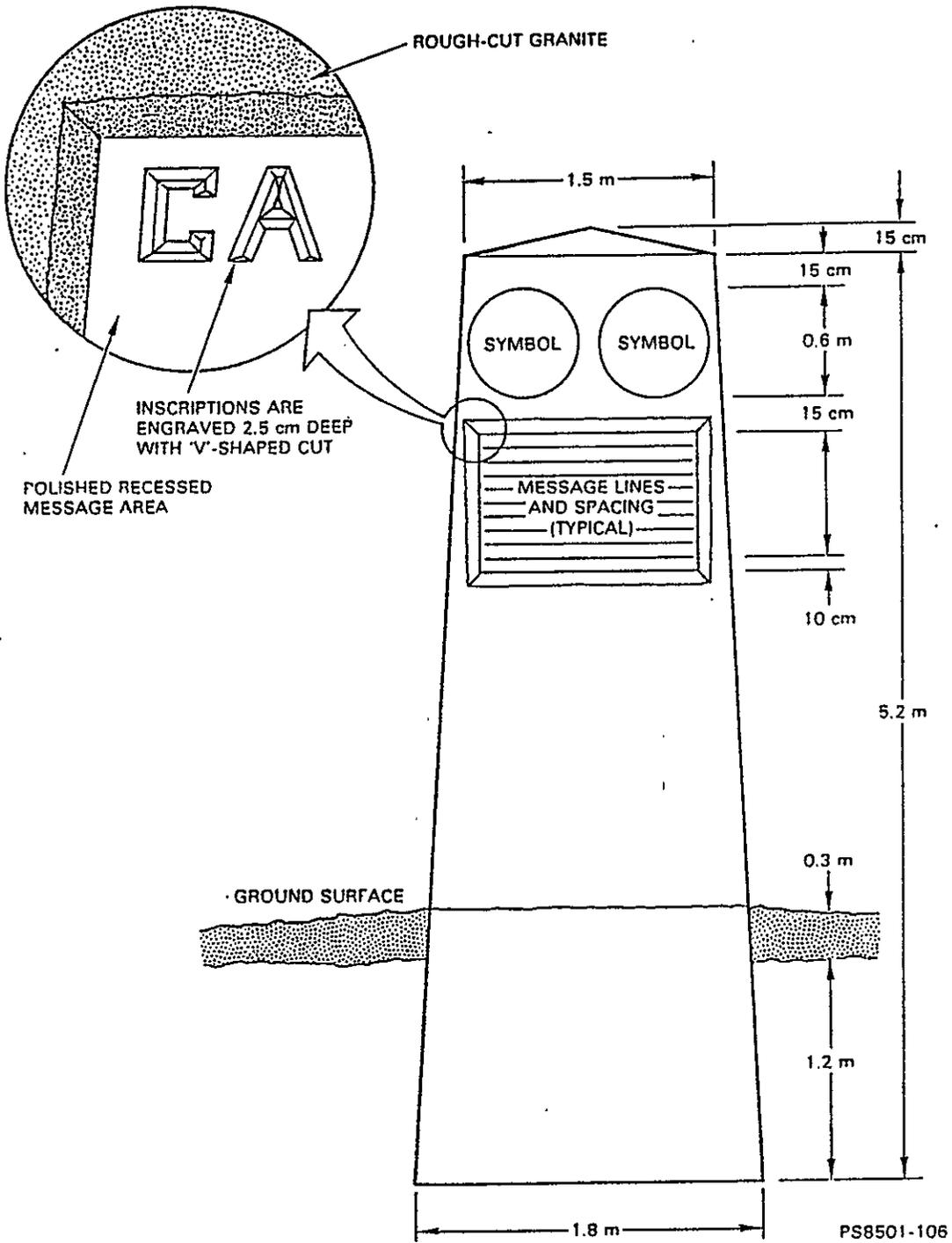
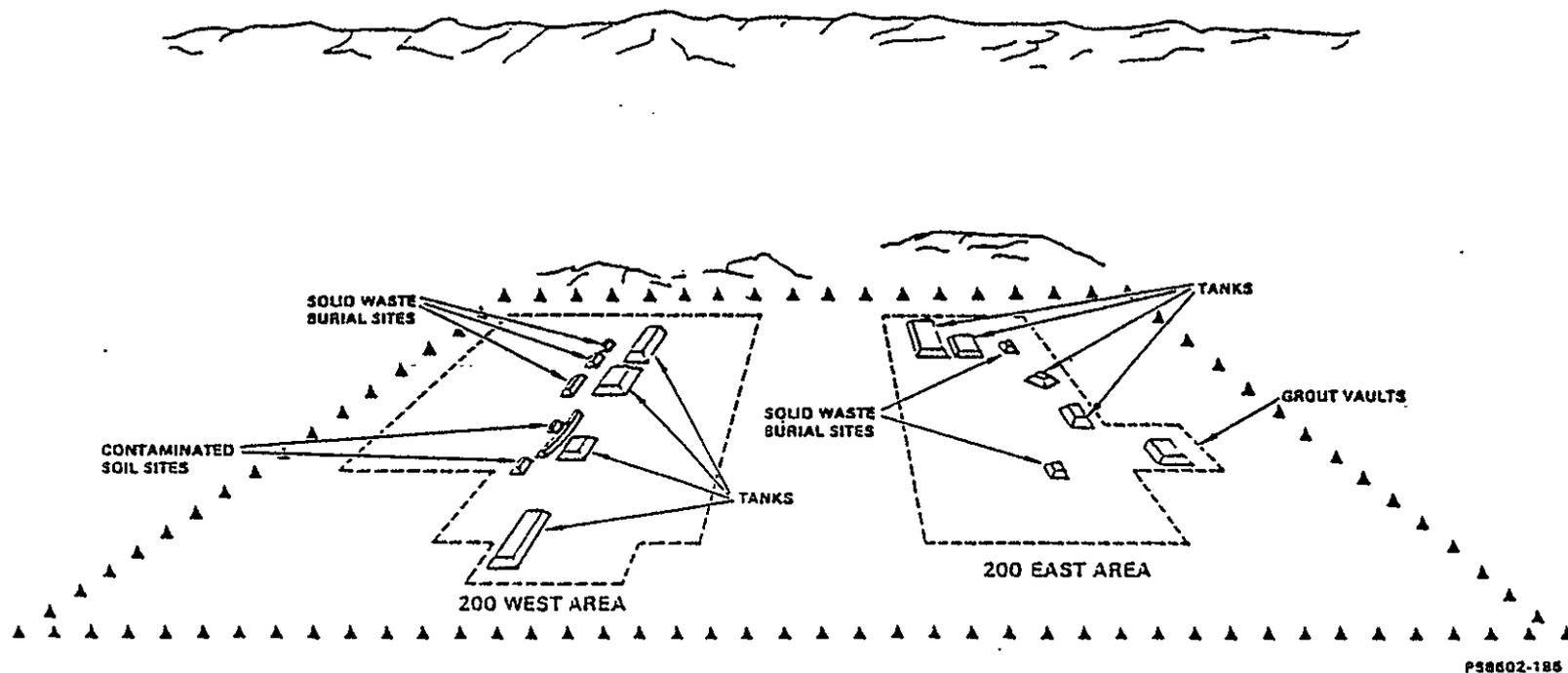


Figure A2-5. Secondary Marker Perimeter.



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The dotted lines show approximate location for surface markers while the dashed lines indicate existing fencelines for the 200 Areas.

well storage. The waste canisters will be lowered into dry wells by a transporter vehicle that also discharges sand into the space above the canister to fill the upper portion of the dry well. One canister per dry well is assumed. The transporter carries one canister per trip in a vertical cylindrical, bottom-loading cask that includes a hoist system. Figure A1-2 shows a profile view of this setup. Detailed planned engineering studies may show the feasibility of dry, overpacked surface storage, either in the open air or in a facility. However, dry well storage is assumed for this analysis in order to be consistent with current storage practices for high-activity waste.

The primary function of dry well storage is to provide long-term, passively cooled storage below grade for canisters containing encapsulated wastes. Figure A2-6 shows the assumed storage array.

Given the well configuration assumed for this analysis, a total of 672 dry wells (including 15 percent contingency) will be required for storage of the anticipated inventory of capsule canisters. Each dry well consists of a cylindrical carbon steel encasement vessel that extends approximately 0.15 m above and 4.6 m below the ground surface. The encasement vessels are shop fabricated of 0.36- and 0.61-m dia. pipe joined by a standard pipe reducer. The encasement is closed at the bottom by a pipe cap welded on the 0.36-m dia. lower section of the encasement. Each dry well encasement vessel is furnished with a carbon steel plate that is field-welded to the top of the dry well after placement of a canister and shield sand. The closure plate is furnished with lifting lugs and a sample valve assembly to obtain air samples and measure pressure inside the dry well. The sample valve is protected by a detachable weather cover, and a nameplate is provided on top of the closure plate for identification of the stored canister. A reusable metal cover is used to protect empty dry wells from the weather prior to canister placement. A stainless steel thermowell attached to the exterior of the dry well encasement protects a thermocouple that is used for periodic temperature measurements of the dry well encasement exterior surface. Figure A2-7 shows a typical dry well assembly.

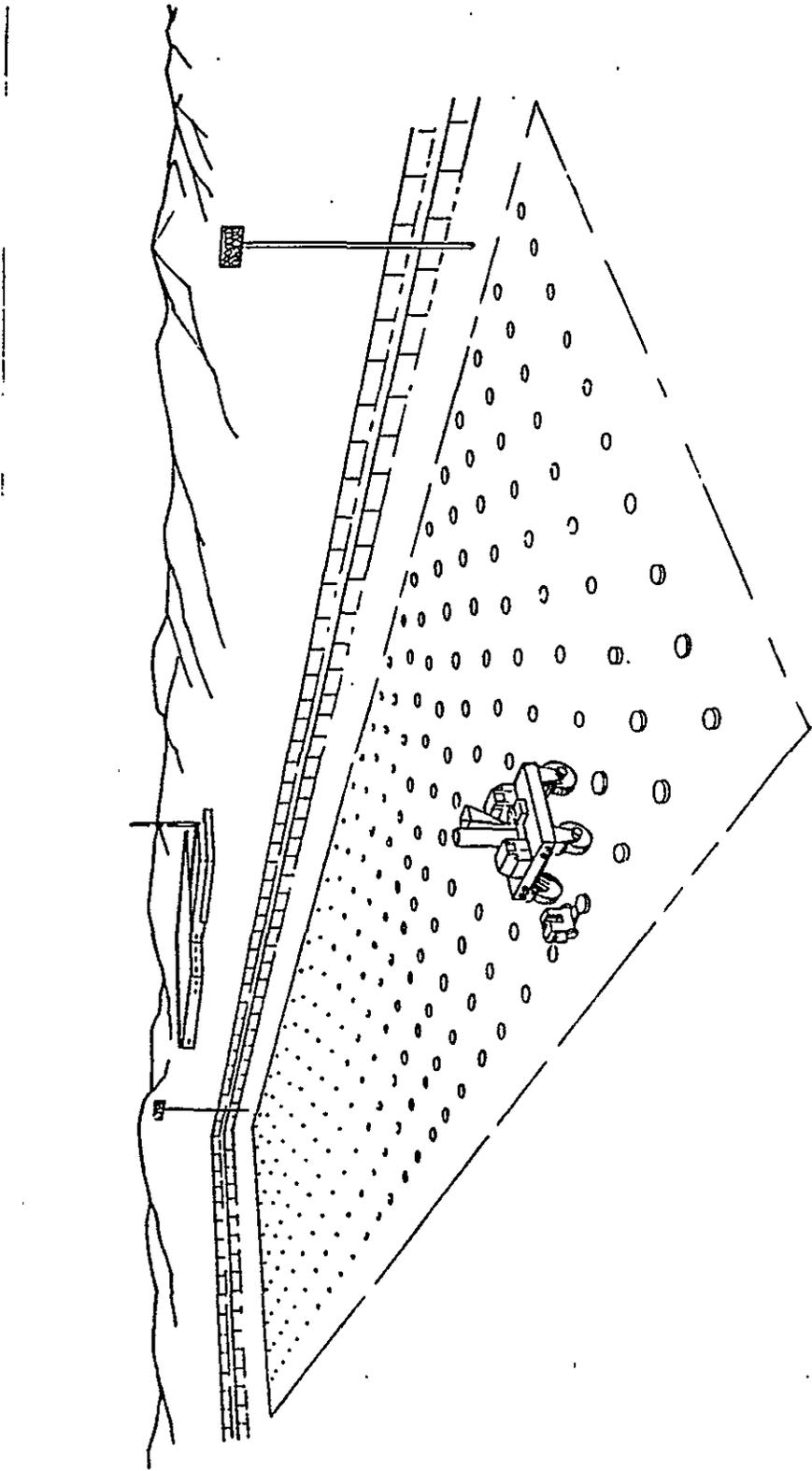
The dry wells will be arranged in a grid spaced approximately 5 m center to center. This spacing is required in order to limit capsule temperatures. Therefore, the dry storage array occupies approximately 1,800 m² of land. The array is assumed to be surrounded by chain link fencing and appropriate security systems are included. For the onsite stabilization and disposal alternative, the multilayer barrier is also added as shown in Figure A2-2. With the 30-m lateral buffer extension of the protective barrier, the total disposed dry well storage facility occupies around 38,000 m² of land.

A3.0 REFERENCES

Colburn, R. P., 1990, *Hanford Waste Vitrification Plant Preliminary Waste Form and Canister Description--Fiscal Year 1990 Update*, WHC-EP-376, Westinghouse Hanford Company, Richland, Washington.

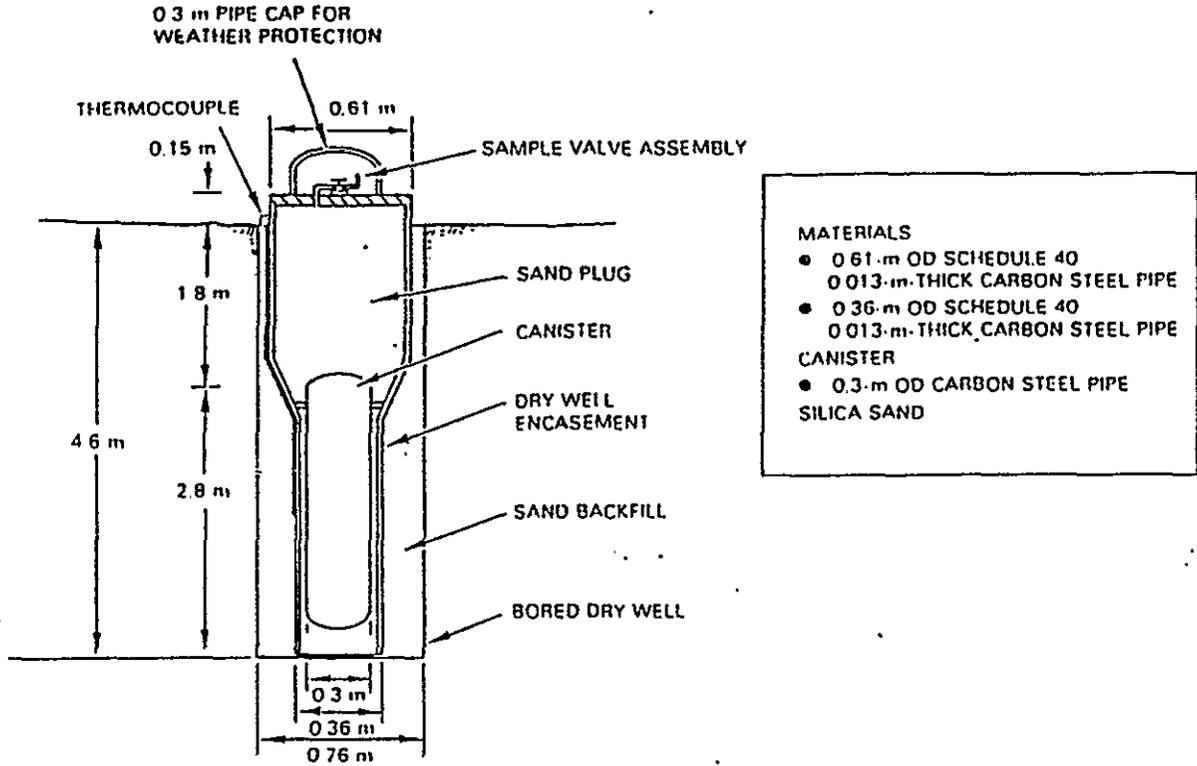
RHO, 1985, *HDW-EIS Engineering Data Package: Strontium and Cesium Capsules*, SD-WM-DB-003, Rev. 2, Rockwell Hanford Company, Richland, Washington.

Figure A2-6. Assumed Dry Well Storage Array.



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Figure A2-7. Typical Dry Well Assembly.



APPENDIX B

ENGINEERING ASSUMPTIONS

Source: RHO, 1985, *HDW-EIS Engineering Data Package: Strontium and Cesium Capsules*, SD-WM-DB-003, Rev. 2, Rockwell Hanford Company, Richland, Washington.

Where appropriate, information was updated to reflect current estimates of cesium chloride and strontium fluoride capsules and cost estimates recalculated to reflect 1995 dollars.

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LIST OF TERMS

EPA	Environmental Protection Agency
MTHM	Metric tons of heavy metal

APPENDIX B

ENGINEERING ASSUMPTIONS

B1.0 GENERAL

This appendix contains a roughly cataloged collection of detailed assumptions needed to estimate various parameters. The assumptions have been grouped under general categories, unless they pertain to only one of the alternatives.

B1.1 SITE SELECTION AND WASTE CATEGORIZATION

1. The following numbers reflect the assumed number of strontium and cesium capsules:

1,328 - existing ¹³⁷Cs capsules
601 - existing ⁹⁰Sr capsules
1,929 - total number of capsules

These numbers are based on capsule inventory estimates as of July 15, 1994. The actual number of capsules will vary as a result of inventory updates, capsules used for commercial or research, and changes in management policy. The assumed inventory of 1,929 capsules is used to generate the data tables included in this document. The expected variance in the number of capsules will not have significant impact, on the data table values.

2. Waste stored and disposed of at the Hanford Site resulted from reprocessing about 94,000 metric tons of heavy metal (MTHM) through 1972. Reprocessing facilities were idle from 1972 to 1983. Waste through 1994 resulted from reprocessing an estimated 12,000 MTHM.

B1.2 RELEASE CALCULATIONS

1. Dust estimates for site activities are from various empirical studies that encompass a broad range of particle sizes and a certain degree of resuspension. For this report, a total dust emission loading is derived based on direct addition of the various estimates, and is assumed to be primary emissions. The following formula was used:

$$\text{dust (metric tons)} = 2.7 \times \text{area (hectares)} \times \text{time (months)}$$

For activities where water trucks were used for dust control, a factor of $\frac{1}{2}$ was applied to the above formula.

2. Release factors for various engine categories are provided in

Table B-1, and were derived from two Environmental Protection Agency (EPA) sources TBD (EPA-RTP AP-42 and EPA-400/9-78-005). Factors for grams pollutant per liter of fuel were chosen to simplify estimation, despite the availability of assumptions made for engine size, load factors, and hours of use, since, fuel use estimates incorporate these factors. Factors for the types or equipment listed in Note (*) of Table B-1 were averaged on an equally weighted basis. Vehicle factors for HC, NO_x, and CO were averaged from numbers in tables for low altitude, non-California trucks without applying correction factors for speed, weight, temperature, or humidity. Calendar year 1990 factors were used, and equally weighted averages were made for model years 1985 to 1990. Factors for SO_x and particulates were based on pre-1973 models. To obtain emission factors based on fuel consumed, the following mileage estimates were used: light-duty gasoline trucks, 4.3 km/L; heavyduty diesel vehicles, 2.0 km/L.

Table B-1. Air Pollutant Emission Factors for Internal Combustion Engines.

Pollutant	Engine category (all units in grams pollutant/L fuel)			
	Diesel Construction equipment*	Heavy-duty diesel vehicles**	Light-duty gasoline trucks ⁺	Small gasoline utility engines ⁺⁺
Particulates	3.4	1.6	0.90	0.73
SO _x	3.7	3.4	0.47	0.64
CO	12.0	34.0	32.0	415.0
HC (exhaust)	3.8	3.5	3.3	25.0
NO _x (as NO ₂)	48.0	6.6	3.5	8.2
Aldehydes	0.85	0.4	-	0.78
Organic acids	-	0.4	-	-

- * Including loaders, bulldozers, rollers, off-highway trucks, cranes, forklift, tractors
 ** Including 18-wheelers, semi-trucks, dump trucks, buses, water trucks.
 + Trucks less than 3,860 kg (8,500 lb).
 ++ Four-stroke engines such as small electric generators and air compressors, pumps.
 - Not available.

3. Emissions relating to the production of electrical power (unless from portable generators), refining and transport of petroleum, mining of coal, manufacture of cement, and fabrication of equipment and other material (unless performed onsite) will not be addressed.
4. Transport of materials for construction and operation, as well as transport of waste (as retrieved and processed), will be included for emission estimates only if transportation occurs within the Hanford Site boundaries. Transportation of workers from town to the staging area will not be included in emission estimates.
5. Conservative estimates were made by assuming all fuel and electrical power calculated for resources were completely converted into thermal releases. Efficiencies of internal combustion engines and electrical motors were not factored in. This assumption is intended to account for any minor omissions in projected energy requirements, as well as to provide a worst case estimate. The following factors were used:
 - Gasoline - 3.37×10^7 J/L
 - Diesel - 3.84×10^7 J/L
 - Electrical power - 3.6×10^6 J/kWh
6. All traffic is assumed to travel on paved surfaces to simplify dust estimates. The dust released from road travel is estimated to be 3.5 g/km traveled.
7. General construction and earthwork dust releases are estimated by the following: 1.2 tons/acre/month (2.7 tonnes/hectare/month) x area x length activity = tons dust x 0.5 = tons dust with water control. The area used is assumed equal to the actual area of the roadway, building, or site. The length of activity is estimated by adding time spent in excavating, building construction, road building, soil mounding, and related operations. The duration of activity in units of months is obtained by assuming 170 work hours per month. The calculated dust emission derived from this equation is reduced by 50 percent to account for water-spray dust control.
8. Dust estimates are from various empirical studies that encompass a broad range of particle sizes and a certain degree of resuspension. For this report, a total dust emission loading based on direct addition of the various estimates is derived and assumed to be primary emissions.
9. Liquid wastes from personal decontamination showers, process equipment washdown, drum cleaning, and laundry discharges are assumed to be monitored and either discharged to the sanitary sewer or to a low-level waste disposal or treatment facility.

B1.3 COST ESTIMATES

1. Hanford Site overhead costs for the no-action alternative, which

includes road and fence maintenance, security, administration, and utilities, are addressed by factoring in a 100 percent surcharge to operating costs. Each waste class has waste costs directly factored in specific no-actions, such as double-shell tank replacement, subsidence maintenance, capsule inspection, and vegetation control. The 100 percent surcharge is applied to these costs. It assumed that these defense-waste related operations will provide one-third of the Hanford Site service costs. The balance will come from low-level waste and surplus facility programs.

2. A 50 percent overhead factor was used for all other alternatives to account for administrative, common service, and functional overhead costs. This factor was used on direct operating costs. A nominal 40 percent contingency was used on direct capital cost estimates. Construction costs include contingencies ranging from 20 percent for low complexity, well-known processes to 40 percent for more difficult, conceptual processes.
3. Costs are presented in 1995 dollars unless noted otherwise. All costs figures included in this data package were derived by adjusting the costs reported in existing engineering studies. Table B-2 provides the correction factors used to adjust the costs from prior years to 1995. These factors are generated for the Department of Energy, by Kaiser Engineers Hanford, and released under the title "Material and Labor Escalation Study".
4. Decontamination and decommissioning (D&D) costs were estimated at a conservative 20 percent of capital costs for new facilities only. Past experience with facility D&D, when corrected for inflation, shows numbers in the 10 to 20 percent range. In addition, it is expected that future facilities are designed considering the eventual D&D of the facility, which should lower costs. Decontamination and decommissioning of existing facilities is not addressed in this data package.
5. Costs for geologic disposal do not include transportation costs to a repository, or repository construction, operation, or emplacement costs. These costs are to be supplied later by Pacific Northwest Laboratory (PNL).
6. When existing documents were used as a basis for cost estimates, one of two scaling methods was used.

One method is simple linear scaling. This method was justifiable if original numbers were developed based on average unit factors, or if the scaling involved some process (such as surface area for disposal), which is essentially linear. Other processes (such as treatment facilities) include significant basic, one-time costs that do not depend on volume. Preferably, unit factors can be obtained or derived that account for economies of scale. Therefore, the second method used to account for this is the "6/10 rule," as follows:

Table B-2. Cost Adjustment Factors.

Fiscal Year	Annual percent of increase (mix of 55% labor and 45% material)		
	Material	Labor	Composite
1983	1.2	2.7	2.0
1984	3.0	3.4	3.2
1985	0.7	-2.5	-1.1
1986	0.7	0.1	0.4
1987	0.6	-0.1	0.2
1988	5.2	1.9	3.4
1989	5.3	1.6	3.3
1990	1.2	1.5	1.4
1991	0.6	6.5	3.8
1992	-0.4	3.3	1.6
1993	1.5	3.3	2.5
1994	2.8	3.0	2.9
1995	3.3	3.5	3.4
1996	3.4	3.5	3.5
1997	2.9	3.7	3.3
1998	2.9	4.3	3.7
1999	2.8	4.2	3.6
2000	2.9	3.8	3.4
2001	2.9	3.2	3.1

Note: Labor, material, and composite columns for fiscal years 1983 through 1993 are actual.

$$\text{Scaling factor} = (C_2/C_1)^{0.6}$$

where

C_2 = new capacity
 C_1 = existing capacity

This is more conservative when scaling down, while the linear method is more conservative when scaling up.

In this data package, the "6/10 rule" is applied to Dry Well Storage Facility (DWSF) construction and DWSF storage/disposal. The values for " C_2 " and " C_1 " apply to the number of capsule canisters.

7. Following are examples of various workers and resource costs (1995 dollars):

Supervisors - \$39.91/h
Operators - \$36.32/h
Construction crew - \$33.88/h
Diesel fuel - \$0.57/L
Electricity - \$34.86/MWh
Water - \$0.15/m³
Compactor - \$46.53/h
Bulldozer - \$81.92/h
Light-duty truck - \$9.54/h
Surface marker - \$18,700/each (not including installation).

B1.4 OTHER ASSUMPTIONS COMMON TO EACH ALTERNATIVE

1. To assess workforce requirements from the combined activities associated with each alternative, approximate start and stop dates must be assumed for each construction and operations segment presented. This information also permits detection of any periods of possible peak release loadings on the environment. To accommodate this process, a time line or schedule for each waste class, by alternative, has been provided in Chapter 3.0. The start dates are assumed but are not critical.
2. All waste disposal actions will be completed by the year 2016.
3. A radiation worker exposure rate can be conservatively assumed as 490 mrem/yr, based on recent tank farm worker dose records.
4. In all the capsule management options, it is assumed that the capsules are placed in overpack canisters. The number of capsules placed in each canister is limited by the total heat content. The following limits were used:

DWSF Storage/Disposal: 0.55 kW/canister

Geologic disposal: 1.17 kW/canister (⁹⁰Sr)
0.80 kW/canister (¹³⁷Cs)

Using the canister load limits and the average capsule heat loadings at the time of disposal, the following average numbers of capsules will be placed in the canisters:

DWSF storage/disposal: 2.8 ⁹⁰Sr capsules
3.8 ¹³⁷Cs capsules

Geologic disposal: 6.5 ⁹⁰Sr capsules
6.2 ¹³⁷Cs capsules

5. For short-lived fission product daughters, the parent nuclide activity does not include the daughter's activity (¹³⁷Cs-Ba, ⁹⁰Sr-Y, ¹⁰⁶Ru-Rh), unless otherwise noted.
6. Tritium follows water through the waste processes and is emitted with water or retained in the final waste form in proportion to the water content of the waste form.
7. In some cases, the proposed engineering method is representative of its class but is clearly not optimum. In those cases, or if little is known about other possible methods, further research and development will be required if the associated alternative is selected. In most cases, theoretical or laboratory studies support the selection of these as representative methods. When more options were available, representative and conservative (on the higher end of cost, resource, and release range) methods were selected without attempting to optimize the details.

B2.0 GEOLOGIC DISPOSAL ALTERNATIVE

B2.1 PROCESSING ASSUMPTIONS

1. For purposes of this analysis, the 200 East and 200 West Areas have been assumed as the location of the processing facilities.
2. Except for canister materials (stainless steel and silica sand), the cost, resource, and personnel requirements for operating the CPF were based on a canister production rate of 195 canisters per year, plus a period of 0.75 years for start up of the CPF. Stainless steel and silica sand resources were proportional to the number of canisters. The number of canisters and the CPF operating period are 509 canisters (including contingency) and 3.4 years, respectively.

B2.2 TRANSPORT AND DISPOSAL ASSUMPTIONS

1. Packages (glass, slag, canisters) to be sent to the high-level waste repository will meet an assumed 1.3 kW/package limit.

2. The number of repository canisters used to emplace the strontium and cesium capsules is listed below for the geologic disposal and the reference alternatives. The capsules are not converted to glass.

Geologic disposal alternative: 509 canisters

Reference alternative: 509 canisters.

B3.0 ONSITE STABILIZATION AND DISPOSAL ASSUMPTIONS

B3.1 METHODOLOGY AND SCHEDULING ASSUMPTIONS

1. The past work that resulted in the physical arrangement shown for the DWSF was preliminary. A study would be needed to optimize these physical parameters, but such a study is not warranted at this time. If corrosion problems are anticipated due to canister temperatures that are too high, a safe and reliable design would be provided by increasing the dry well spacing or decreasing the heat load (either longer storage at the WESF, placing fewer capsules in each canister, or adding thermal siphons for greater passive heat exchange).
2. The surface barrier does not have any costs, resource, or workforce requirements for the operational phase. These requirements are calculated only for the construction activities.
3. The cost bases for the surface and subsurface markers differ slightly from the final designs and spacings shown in various figures. Surface markers were assumed to be obelisks factored at a rate of two per site, as opposed to the rectangular monoliths encircling an array of sites. Subsurface markers were to be 25 cm ceramic discs placed in a geometric matrix rather than the 13 cm porcelain discs randomly dispersed throughout the barrier. It was assumed that the estimates adequately account for the revised marker system design.

B3.2 STABILIZATION ASSUMPTIONS

1. One standard barrier will be applied to sites requiring barriers. Subsidence control will be applied to compressible waste sites before barrier placement.
2. Soil, rock filter, and riprap volumes were calculated from the estimated area over the contaminated soil and waste. The formula of a prismatoid was used on the barrier in Appendix A, Figure A2-1, to obtain volumes of the various materials.
3. The DWSF dimensions are 135 m by 135 m for barrier calculation purposes. The dimensions are based on arranging the canisters in a

square grid, with each canister placed 5-m off center. The barrier then factors in a 30-m perimeter buffer.

4. The CPF operates for 4.2 years and produces 672 canisters.
5. Placement of riprap and soil during barrier construction will require bulldozers, compactors, a water truck, dump trucks, and light trucks. A crew of ten operators, two grade checkers, and one supervisor is used. Soil is placed at a rate of 240 yd³/h (184 m³/h).
6. The barrier design shown in Appendix A, Figure A2-1, is slightly different from the one used for calculation purposes. There was no soil mixed into the "shoulder" riprap, there was no geotextile in the rock filter, and the buffer extension of the barrier beyond the waste edge was not as large.

B4.0 NO-ACTION ALTERNATIVE

B4.1 METHODOLOGY AND SCHEDULING ASSUMPTIONS

1. Control for 100 years will be interpreted as through the year 2090 (100 years after disposal activities would begin for other alternatives). Active control for the continued storage alternative includes active maintenance and surveillance. Examples include maintaining fences and patrols. (Passive control will apply to other alternatives, and will consist of continued site ownership and/or zoning restrictions.)
2. Due to the high cost of continued capsule storage in water basins, including eventual repair and replacement of basins, capsules are placed in the DWSF in 2010. The CPF will operate 4.2 years and will produce 672 canisters.
3. Unless stated otherwise, it is assumed that data provided for the first 100 years can be linearly extrapolated into subsequent 100-year segments, if extended storage is postulated. Implicit in this assumption is a static state of technological advancement, unrealistic but conservative (i.e., if research and development expenses were not charged to the program).

B4.2 ENGINEERING ASSUMPTIONS

1. The vegetation control program will require four spray rigs and trucks, a water truck, and a mixing truck. Twelve personnel will be required to operate these vehicles, one of whom must be a licensed commercial pesticide applicator (LCPA). Once stabilization is achieved, a reduced control program is initiated that will require only one spray rig, one truck, and four personnel on a part-time

basis. One of the four personnel must be an LCPA. Herbicide spray rigs cover 20,000 m²/h, at a coverage rate of 2.5E⁻⁴ L/m² (1 qt/acre).

Additional herbicide application to control broadleaf intrusion will be required at irregular intervals on an average of once every three years per site.

2. Fencing and posting maintenance will be handled on an incidental, as-needed basis.
3. Environmental monitoring teams perform direct radiation monitoring, surface water sampling, groundwater sampling, atmospheric sampling, soil/sediment sampling, biota sampling, and observation. Environmental surveillance teams perform investigations and inspections, evaluations, recording and interpretation of data, procedure development, and revision. Their time averages 8 worker hours per month per site. Sites will require two monitoring and surveillance vehicle trips per month, using a light-duty truck.

B5.0 REFERENCES

RHO, 1985, *HDW-EIS Engineering Data Package: Strontium and Cesium Capsules*, SD-WM-DB-003, Rev. 2, Rockwell Hanford Company, Richland, Washington.

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APPENDIX C

FEASIBILITY STUDY FOR THE PROCESSING OF HANFORD SITE
CESIUM AND STRONTIUM ISOTOPIC SOURCES IN THE
HANFORD WASTE VITRIFICATION PLANT

Source: Anantatmula, R. P., R. A. Watrous, and
J. L. Nelson, (Westinghouse Hanford Company), and
J. M. Perez, R. D. Peters, and M. E. Peterson (Pacific
Northwest Laboratory), 1991, *Feasibility Study for the
Processing of Hanford Site Cesium and Strontium
Isotopic Sources in the Hanford Waste Vitrification
Plant*, WHC-EP-0460, Westinghouse Hanford Company,
Richland, Washington.

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LIST OF TERMS

ANSI	American National Standards Institute
CC	Complexant concentrate
CFR	U.S. Code of Federal Regulations
DF	Decontamination factor
DOE	U.S. Department of Energy
DWPF	Defense Waste Processing Facility
EA	Environmental Assessment
FRG	Federal Republic of Germany
FY	Fiscal year
HEPA	High-efficiency particulate air
HDW-EIS	Hanford Defense Waste-Environmental Impact Statement
HWVP	Hanford Waste Vittrification Plant
NCAW	Neutralized current acid waste
NCRW	Neutralized cladding removal waste
NEPA	<i>National Environmental Policy Act of 1969</i>
OCRWM	Office of Civilian Radioactive Waste Management
OGR	Office of Geologic Repositories
ORNL	Oak Ridge National Laboratory
PCT	Product Consistency Test
PFP	Plutonium Finishing Plant
PHP	Pneumatic hydropulse
PNL	Pacific Northwest Laboratory
OCRWM	Office of Civilian Radioactive Waste Management
RLFCM	Radioactive liquid-fed ceramic melter
RLST	Receipt and lag storage
ROD	Record of Decision
SAR	Safety Analysis Report
SARP	Safety Analysis Report for Packaging
SRAT	Slurry receipt and adjustment tank
TDP	Technical Data Package
TRUEX	Transuranic Extraction
WAC	Washington Administrative Code
WAPS	Waste Acceptance Preliminary Specifications
WCD	Waste form and canister description
WESF	Waste Encapsulation and Storage Facility

APPENDIX C

FEASIBILITY STUDY FOR THE PROCESSING OF HANFORD SITE
CESIUM AND STRONTIUM ISOTOPIC SOURCES IN THE
HANFORD WASTE VITRIFICATION PLANT

C1.0 INTRODUCTION AND SCOPE

C1.1 INTRODUCTION

This study assesses the feasibility of treating and disposing of encapsulated cesium and strontium waste by incorporating the capsule contents into wastes slated to be processed in the Hanford Waste Vitrification Plant (HWVP) or by vitrifying the capsules as a new HWVP feed stream. In making this assessment, alternative options for storage, treatment, and disposal have been considered briefly. Subsequent analysis will be required to determine whether the recommended disposal option falls within the scope of the existing Record of Decision (ROD) and to document any potential changes from the ROD.

All material information, both technical and economic, included in this appendix are wholly derived based on information from the *Feasibility Study for the Processing of Hanford Site Cesium and Strontium Isotopic Sources in the Hanford Waste Vitrification Plant*, WHC-EP-0460 (Anantamula et al. 1991). Where applicable, all cost values reported have been adjusted to reflect 1995 dollars. Cost adjustment factors are presented in Appendix B, Table B-2.

C1.2 OBJECTIVE

The objective of this study is to determine whether significant technical issues would preclude the incorporation of encapsulated cesium and strontium wastes into the waste streams to be processed by the HWVP. To make this determination, it is necessary to evaluate the impact of the capsule contents on the major process steps within the HWVP and to define the capsule-handling and potential pretreatment steps required to blend the cesium chloride and strontium fluoride (SrF_2) with the HWVP wastes.

The study intends to establish the feasibility of vitrifying the cesium chloride and SrF_2 in the HWVP. The impacts of cesium chloride and SrF_2 addition on the HWVP process conditions and waste form characteristics have been assessed.

C2.0 ASSUMPTIONS AND REQUIREMENTS

The assumptions and technical requirements used for this study are described in this section. They have been derived from the *Hanford Waste Vitrification Plant Technical Data Package* (WHC-1989), waste acceptance specifications for repository waste forms, and knowledge of the vitrification process and waste management practices.

C2.1 ASSUMPTIONS

All capsules are assumed to be on-site by the time the HWVP begins operations with radioactive waste. Cells and capabilities within the Waste Encapsulation and Storage Facility (WESF)/B Plant facility or the HWVP are assumed to be available for initial processing of the capsules. It has also been assumed that, for whatever steps are taken, the final waste form must meet requirements from 10 CFR 60 and 40 CFR 191 for disposal of the capsule waste in the federal high-level waste repository. Existing facilities on the Hanford Site or the HWVP itself must be capable of handling the capsules and removing the cesium and strontium for blending with HWVP feedstreams. This capability may require modifications to existing facilities, which will be described in this report.

C2.2 REQUIREMENTS

For the cesium and strontium to be processed within the HWVP, the following requirements must be satisfied:

- The final waste form must still meet the repository waste acceptance specifications for borosilicate glass.
- Processing in the HWVP should not be impacted such that equipment failures and process rates below the design criteria of 100 kg/h glass production occur.
- Environmental releases of radioactive or hazardous materials should not exceed allowable standards set for the HWVP.

C3.0 BACKGROUND

This study focused on the feasibility of vitrifying the capsule waste in the HWVP. All impacts evaluated here were based on the conceptual design of the HWVP at the time of the study (FY1991).

Programmatic changes, after FY1991, have significantly impacted the production capacity and the schedule of HWVP. Impacts of these programmatic changes, if any, on the HWVP design are not reflected in this study.

C3.1 OVERPACK CAPSULES AND SEND TO REPOSITORY

The HDW-EIS (DOE 1987) presents the scenario of overpacking the capsules and shipping to an offsite geologic repository as the preferred alternative for disposal of cesium and strontium capsules. The same alternative is also described as the preferred plan in the *Hanford Waste Management Plan* (DOE-RL 1987). This preferred option for disposal calls for the capsules to be overpacked into canisters suitable for placement in the high-level waste repository. In this plan, up to four intact capsules would be placed in a canister of the same external dimensions as used by the HWVP. The characteristics of the HWVP canister are documented in the *Hanford Waste Vitrification Plant Preliminary Waste Form and Canister Description--Fiscal Year 1990 Update* (Colburn 1990). The internal structure of the canister would be modified to restrain the capsules and to provide heat transfer to the canister walls. Modifications to existing facilities (such as WESF) on the Hanford Site may be required to overpack the capsules.

In July 1985, the U.S. Department of Energy's (DOE) Office of Civilian Radioactive Waste Management (OCRWM) established a waste acceptance process as the means by which high-level waste producers would be allowed to dispose of wastes in the commercial repository. The waste acceptance process is generic because it is intended to accommodate any high-level waste form other than spent fuel. The Waste Acceptance Preliminary Specifications (WAPS) are based on the waste form and canister description (WCD). The waste acceptance process requires that OCRWM prepare WAPS for each waste form. The WAPS will be prepared for the cesium and strontium capsule waste form once the WCD is issued.

The WAPS for borosilicate glass have been drafted and are currently in review. Although these WAPS are tailored for borosilicate glass, much of their bases derive from the U.S. Code of Federal Regulations (CFR) or limitations to repository handling capability. Attachment 1 discusses the bases for waste acceptance requirements for borosilicate glass waste forms. Because the existing WAPS are based on regulatory requirements or repository design constraints, they provide useful guidance for the development of nonborosilicate glass waste forms. The WAPS for the Defense Waste Processing Facility (DWPF) are summarized in Table C3-1 (from DOE 1989). Of the 21 individual specifications shown in Table C3-1, all but three can be accommodated in the capsule overpack concept. A discussion of the three problematic specifications follows.

Table C3-1. Waste Acceptance Preliminary Specifications for the Defense Waste Processing Facility High-Level Waste Form. (Sheet 1 of 2)

Specification	Content summary
1.1 Chemical composition	Provide projection of product elemental and phase compositions (with ranges) for life of facility for all elements in concentrations >0.5 wt%.
1.2 Radionuclide inventory	Provide canister and aggregate inventories of all radionuclides with half-life periods longer than 10 years with concentrations >0.05 Ci% at any time up to 1,100 years after production.
1.3 Radionuclide release	Production samples must show normalized releases of <1 g/m ² -d for MCC-1 tests or equivalent.
1.4 Chemical and phase stability	Report product transition temperature and time-temperature-transformation ranges. Certify that waste form has not exceeded transition temperature during storage and is at least 100 °C below transition temperature at shipment.
2.1 Canister material	Austenitic stainless steel.
2.2 Canister fabrication and closure	Leaktight according to ANSI ^a N14.5-1977.
2.3 Identification and labeling	Alphanumeric code on each canister visible from top or side of canister. Characters at least 92 points in height.
3.1 Free liquid	None allowed in canister.
3.2 Gases	None allowed in canister except helium, argon, air, or other inert. Internal pressure not to exceed 7 lb/in ² (gage) at 25 °C.
3.3 Explosiveness, pyrophoricity, and combustibility	No explosive, pyrophoric, or combustible materials allowed in canister.
3.4 Organic materials	None allowed in canister.
3.5 Free volume	Less than 20 percent of canister volume.
3.6 External contamination	Less than 320 alpha dpm/100 cm ² . Less than 2,200 beta-gamma dpm/100 cm ² .
3.7 Heat generation	Less than 800 W/canister reported to ±15%. Provide projections and actuals.
3.8 Maximum dose rate	Less than 10 ³ rem/h gamma and 10 ³ rem/h neutron at surface. Provide projections and actuals.
3.9 Chemical compatibility	Document reactivity between waste form and canister.
3.10 Subcriticality	K _{eff} < 0.95.
3.13 Handling features	Provide grapple and canister with lifting flange. Grapple must be remotely operable within a right-circular cylindrical cavity with diameter equal to that of the canistered waste form.
4.0 Quality assurance	Provide quality assurance program complying with TBD OGR ^b /8-14.

^aANSI = American National Standards Institute.

^bOGR = Office of Geologic Repositories.

C3.1.1 Waste Acceptance Preliminary Specification 1.3: Specification for Radionuclide Release Properties

The current reference version of the WAPS for the DWPF stipulates that the waste form must have a normalized release limit for certain matrix elements and radionuclides of less than 1 g/m²-d averaged over the 28-d MCC-1 test. Although a draft revision to this specification would change

the test protocol, the changes, if adopted, would not affect the conclusion herein (see Section C4.2, Glass Durability).

Most investigators agree that glass dissolution results from the associated matrix dissolution of silica. Testing with simulated NCAW glass has shown cesium release to be within a factor of two times greater than silica. Strontium release is less than silica (Goles and Nakaoka 1990). If it is assumed that waste glass has a $1 \text{ g/m}^2\text{-d}$ release rate and that this is related to the solubility of silica (i.e., release rate is proportional to solubility), then a comparison between cesium chloride, SrF_2 , and waste glass can be made by virtue of solubility. The aqueous solubility of amorphous silica, or waste glass in this case, is 50 mg of silicon/L at room temperature (Fournier and Rowe 1977). The room temperature solubility of cesium chloride is 1,300 mg of cesium/L, and for SrF_2 it is 85 mg of strontium/L (Weast 1987). Clearly, cesium chloride is much worse than glass in terms of matrix solubility and would likely not be acceptable as a waste form. The solubility of SrF_2 is much closer to that of silica, but still higher by almost a factor of two. Release rates from SrF_2 may be marginally acceptable, but a more detailed analysis will be needed.

Because of the relatively short half-life periods of ^{90}Sr and ^{137}Cs (27.7 years and 30 years, respectively), it may be possible to demonstrate that the actual radionuclide releases will be inconsequentially small by the end of the 1,000-year repository containment period. Such an analysis has not been fully pursued and would be complicated by the following considerations:

1. Long-lived impurities such as ^{135}Cs might contain enough residual radioactivity at the end of the repository containment period to invalidate the concept.
2. Complete containment of the highly soluble materials cannot be ensured for the first 1,000 years because of the high reactivity between the cesium and strontium salts and the capsule and canister materials (see discussion of WAPS 3.9 in Section 3.1.3).
3. The current specification for borosilicate glass does not permit any computation of the aging of the waste form. Durability is to be evaluated at the time of production.
4. Preliminary assessments from the geologic repository project indicate that the comparatively high solubility of cesium chloride and the uncertain chemical behavior of SrF_2 will preclude their direct use in the repository because of complications in the licensing process.¹

¹Based on technical discussions with J. L. Nelson to V. M. Oversby and to D. J. Harrison-Giesler, August 22, 1990.

C3.1.2 Waste Acceptance Preliminary Specification 1.4:
Chemical and Phase Stability (10 CFR 60.135[a][2])

This specification is designed to document the thermal conditions under which the waste form will undergo significant chemical or physical changes. Although the specification is tailored specifically for borosilicate glasses, it is clear that a waste form that is not stable under expected storage or repository conditions will not be acceptable. Neither cesium chloride nor SrF_2 is subject to decomposition within the temperature ranges expected for storage or repository disposal; however, impurities can lower the decomposition temperature into this range. It will be necessary to negotiate the waste form's chemical and phase stability requirements with the repository before adopting this disposal approach.

The thermal phenomena of interest for the waste form are phase transitions, such as solid phase changes and melting. Solid phase changes that increase the waste form volume could potentially cause a rupture of the capsule. Production of a liquid phase by melting could also accelerate internal corrosion of the capsule. Strontium fluoride with sodium fluoride (NaF) impurities has a minimum melting temperature of $850\text{ }^\circ\text{C} \pm 5\text{ }^\circ\text{C}$ (Fullam 1977) and has no known phase transformations below this temperature (Watrous and Chen 1978). Storage of SrF_2 below $850\text{ }^\circ\text{C}$ should be adequate to avoid liquid phase volume expansion and the resulting capsule stresses. Cesium chloride with impurities has a solid phase transformation at $465\text{ }^\circ\text{C}$ (Fullam 1971). The minimum melting point of cesium chloride with 5 wt% NaCl and potassium chloride (KCl) is $580\text{ }^\circ\text{C}$ (Fullam 1971), although a cesium chloride-ferric chloride (FeCl_3) mixture can melt at a temperature as low as $270\text{ }^\circ\text{C}$. Because the cesium chloride capsules are melt-cast, the voids produced on cooling should permit the volume expansion that accompanies the solid-to-liquid transformation to occur without stressing the capsule.

C3.1.3 Waste Acceptance Preliminary Specification 3.9:
Chemical Compatibility (10 CFR 60.134[a][1])

The WAPS for borosilicate glass require that the contents of the canistered waste form shall not corrode the canister such that there will be an adverse effect on normal handling during storage, transportation, and repository operation. The extent of corrosiveness and chemical reactivity among the waste form, canister, and any filler materials must also be documented, as well as corrosive chemical interactions, and any reaction products generated within the canistered waste forms.

The chemical compatibilities of cesium chloride and SrF_2 with the capsule containment materials have been studied extensively (Fullam 1972, 1981; Bryan 1987), including the effects of impurities and daughter products. The principal impurities for cesium chloride are sodium chloride (NaCl) and KCl, while numerous metal fluorides, principally barium fluoride (BaF_2), calcium fluoride (CaF_2), and NaF are found with SrF_2 . Because ^{137}Cs decays to ^{137}Ba , the cesium chloride decays to BaCl, which is chemically unstable, with the reaction $2\text{BaCl} \rightarrow \text{BaCl}_2 + \text{Ba}$. The ^{90}Sr decays to ^{90}Y , which further decays to ^{90}Zr . As with BaCl, ZrF_2 is chemically unstable, with reaction $2\text{ZrF}_2 \rightarrow \text{ZrF}_4 + \text{Zr}$.

Long-term strontium corrosion studies (Fullam 1981) found that chemical attack of the Hastelloy C-276 inner capsule containment material results primarily from impurities in the SrF_2 . Once the critical impurities are consumed by the corrosion reaction (up to 12,000 hours), the rate of chemical attack decreases to a low level. Fullam concludes that, for heat source applications, maintaining the temperature of the capsule wall below 800 °C would adequately contain the SrF_2 for 10 to 20 years. Because the reacting impurities are expected to be consumed by that time, additional corrosion is expected to be minimal.

Long-term cesium studies (Bryan 1987) examined cesium chloride capsules after extended storage at 350 °C to 450 °C. Linear extrapolations of initial corrosion rates suggested capsule wall failure in 25 to 35 years. Corrosion rates at lower temperatures appeared to be much slower. Moreover, because corrosion appears to result mostly from impurities, it is expected that the rate would decrease as the impurities are consumed.

C4.0 EVALUATION OF HWVP PROCESSING CONSIDERATIONS

This study focuses on five major issues concerning the incorporation of higher cesium, strontium, chloride, and fluoride levels into existing HWVP feedstreams: (1) the effect on glass composition and processability, (2) the effect on glass release properties, (3) the effects of increased canister curie and watt levels on meeting WAPS and shipping requirements, (4) the effects on offgas volatility and treatment, and (5) the effect on vitrification equipment materials compatibility. Each of these issues is discussed in the following subsections.

For this study, it was assumed that the base case would be to blend all of the cesium chloride and SrF_2 from the capsules with neutralized current acid waste (NCAW). The NCAW stream is the first Hanford Site waste stream to be processed in the HWVP. Therefore, the timing for coprocessing of the two streams will be critical. If scheduling prevents blending with NCAW, the capsule waste could be blended with complexant concentrate (CC) waste or processed as a separate feed stream. Blending was considered only for the NCAW and CC waste streams because these are the two waste streams defined as "high-level waste." Attachment 2 summarizes calculations and estimates of NCAW composition after blending and the curie and watt loadings in the resulting glass canisters. Attachment 3 summarizes calculations and estimates of the CC composition after blending and the curie and watt loadings in the resulting glass canisters. Section C4.1 discusses the processing of the capsule waste as a separate feed stream.

C4.1 GLASS COMPOSITION AND PROCESSABILITY

Borosilicate glass has been selected as a waste form for nuclear waste immobilization because of its optimum combination of durability, processability, and acceptable waste loading characteristics. Failure of the glass to constrain the radionuclides would require failure of the glass structure itself. By definition, glass is an amorphous material forming a random structure. The glass structure has three main parts: (1) glass or network formers, primarily composed of oxides of silicon and boron, that provide the basic structure; (2) network modifiers, which include alkali metal and alkaline earth oxides and are used as fluxes to reduce the temperature required for processing; and (3) intermediates, such as aluminum oxide and other transition series oxides, which can take on the coordination of glass formers and play a role like the network formers. To a certain extent, the majority of elements in the periodic table can be accommodated in the glass structure. Therefore, the complex mixture that makes up defense high-level waste can be blended with the proper glass formers and any necessary intermediates and modifiers in the proper proportions to produce a glass waste form.

Limits have been established for volatile and nonvolatile oxides and elements in the reference glass for NCAW (WHC 1989). The weight percent limits for Cl⁻ and F⁻ are given in Table C4-1. The maximum Cl⁻ limit has been set at the nominal value and does not indicate that the HWVP can only process waste at these Cl⁻ concentration. Future evaluation is expected to result in increasing the maximum Cl⁻ concentration.

Table C4-1. Limits for Chlorine and Fluorine in the Glass Form.

Constituent	HWVP process limit (wt%)	
	Nominal	Maximum
Cl ⁻	0.08	0.08
F ⁻	0.3	1.73

The *Hanford Waste Vitrification Plant Technical Data Package* (WHC 1989) has also set a 1.25 wt% maximum for all fission products. This upper limit is considered somewhat arbitrary, because testing has not determined a maximum limit based on maintaining acceptable glass properties.

Cesium and strontium will enter the glass matrix as network modifiers. Much higher concentrations of Cs₂O and SrO have been placed in borosilicate glass than are currently in NCAW. For example, the thermal and isotopic glass canisters produced for the Federal Republic of Germany (FRG) using the Pacific Northwest Laboratory (PNL) radioactive liquid-fed ceramic melter (RLFCM) contained 5.7 wt% Cs₂O and 1.9 wt% SrO (Holton et al. 1988). Acceptable glass properties (i.e., acceptable in terms of viscosity and electrical conductivity) were maintained by reducing the concentration of the alkali metals and alkaline earths that were in the reference glass composition (WV 205) after which the FRG glass was modeled. During the RLFCM campaigns, acceptable processing rates were achieved and volatility was determined to be typical of glasses containing low levels of cesium and strontium. Inspection of glass samples taken from the pour stream and views of the glass surface at the top of the canisters confirmed that secondary cesium or strontium phases had not formed.

The 1,349 cesium capsules have a total inventory of 2,875 kg of cesium. The 636 SrF₂ capsules have a total strontium inventory of 1,198 kg. If distributed uniformly throughout the estimated 480 canisters of NCAW glass to be produced, the concentrations of Cs₂O and SrO in the glass would increase to 0.53 wt% and 0.28 wt%, respectively. These values are well within the processing envelope established by the production of FRG canisters cited above. Therefore, the cesium and strontium could be incorporated into the NCAW waste stream without impacting processing characteristics.

If evenly distributed throughout the currently estimated 650 canisters of CC waste, the concentrations of Cs₂O and SrO in the glass would increase to 0.40 and 0.13 wt%, respectively. Again, these values are well within the

limits established by previous experience. Therefore, the cesium and strontium could be successfully incorporated into the CC waste stream without impacting processing. Reconsideration of this approach would be necessary if the TRUEX process were employed for CC waste pretreatment. However, more than a tenfold reduction in the number of canisters would have to occur as a result of using the TRUEX process before the empirically established limits would be approached.

The behavior of fluorine in borosilicate glasses differs in many respects from the other halogens. The effective radius of fluorine is very close to that of oxygen. This permits fluorine to partially replace oxygen in the glass. The other halogens are larger and much less stable as components in the glass structure. Because of the ability to replace oxygen in the glass, fluorine has a higher solubility in the glass than the other halogens. Generally, most clear commercial glasses contain at most 0.6 wt% fluorine compared to 0.1 wt% chlorine (Volf 1984), in part because of the use of fluorine compounds as glass-finishing agents. Both liquid and crystalline phase separation occurs above the solubility limit, causing opacity. The presence of elements such as aluminum and boron in the glass somewhat increases the solubility of fluorine in the glass. Work performed by Bates (1987) in which concentrations as high as 4.3 wt% were tested demonstrated that a fluorine concentration of 1.73 wt% in HW-39 glass was acceptable. Crystalline phase separation at 3.0 wt% necessitated lowering the limit to the next lower concentration tested (i.e., 1.73 wt%).

Previous testing at PNL has included feeds with chlorine concentrations up to 0.3 wt% (Savannah River feeds), 0.35 wt% (West Valley feeds), and at least 0.08 wt% (HWVP feeds). Although halides can form molten metal salts that can accumulate on the glass surface, only minimal quantities of such salts have ever been observed. Chlorine and fluorine concentrations in NCAW glass will increase from reference levels of 0.075 and 0.3 wt%, respectively, to 0.17 and 0.37 wt%, respectively, with the incorporation of the capsule material. Chlorine and fluorine concentration in the CC glass will increase from reference levels of 0.25 and 0.05 wt%, respectively, to 0.35 and 0.12 wt%, respectively, with the incorporation of the capsule material. Based on results from previous tests, processing conditions and glass characteristics will be acceptable if the cesium and strontium salts are blended with pretreated NCAW and/or CC waste and processed through HWVP.

Only a small increase in the number of canisters produced results from incorporating the capsule materials. If one assumes no change to the 25 wt% waste loading limit, a total of 5,359 kg of wastes will be vitrified, an amount equivalent to 21,436 kg of glass. This will result in the production of 13 additional canisters, irrespective of the waste stream(s) (i.e., NCAW and/or CC) to which the cesium and strontium are added.

Another option would be to vitrify the capsule waste in a separate HWVP campaign. This option might be necessary in the event of scheduling conflicts, or some other technical limitation that would prevent blending the capsule waste with planned HWVP feed streams. For this option, the cesium chloride and SrF_2 would be dissolved or slurried in one of the HWVP tanks, blended with glass formers, and fed to the melter. Additional canisters would be produced, the number being determined by glass composition limits.

Assuming the maximum concentrations of the four capsule constituents are the same as the values discussed earlier, the total number of canisters of glass produced would be dictated by one of the following cases:

- 2,875 kg Cs/5.3 percent Cs limit = 54,245 kg glass = 33 canisters
- 1,198 kg Sr/1.6 percent Sr limit = 74,875 kg glass = 45 canisters
- 767 kg Cl/0.35 percent Cl limit = 219,142 kg glass = 133 canisters
- 519 kg F/1.73 percent F limit = 30,000 kg glass = 18 canisters.

Chloride is shown above to be the limiting case, resulting in 133 canisters. Separation of the halides from the cesium and strontium would significantly reduce the total number of canisters to 45.

In the year 2010, the decay heat in these 133 canisters would be about 2,000 W/canister, which is higher than the nominal HWVP heat loadings (400 W/canister) but not technically prohibitive. At this power level the centerline temperature of the glass canister does not exceed the phase transition temperature for borosilicate glass. The cost of production and repository disposal for the 133 canisters, based on \$575,000/canister, is estimated to be \$74.5 million. For 45 canisters, the cost is estimated to be \$25.9 million.

C4.2 GLASS DURABILITY

Specification 1.3 of the reference version of the WAP requires that the glass product leach at a rate of less than $1 \text{ g/m}^2\text{-d}$ based on measuring the fractional release of cesium, sodium, silicon, boron, and uranium in an MCC-1 test conducted during a 28-d period. An alternative test, such as the Product Consistency Test (PCT), may be used if the results can be related to the MCC-1 acceptance criterion. If the release of any of the five listed elements exceeds the $1 \text{ g/m}^2\text{-d}$ criterion, a more durable glass must be developed.

Because the leach rate is based on the amount of cesium lost divided by the fraction of cesium originally in the glass, the relative value should stay the same with the higher cesium-containing glasses. The reference glass composition, HW-39-4, has a leach rate for cesium of about $0.74 \text{ g/m}^2\text{-d}$ (Goles and Nakaoka 1990). This rate is quite high and is almost 75 percent of the WAPS limit. Any significant increase in the cesium concentration should require leach testing of the candidate glass to determine the cesium leach rate.

A draft revision to Specification 1.3 of the WAPS proposes that production glass samples be tested exclusively by the PCT, and the results compared to PCT results for DWPF Environmental Assessment (EA) benchmark glass for releases of lithium, sodium, and boron. The change to the PCT was made to decrease analytical uncertainties and shorten the time required for radioactive testing. Reference to the EA glass was invoked with the intent of bounding the release rate of future glasses to, as a minimum, the performance

of the glass in the EA. Unpublished data from the DWPF indicate that these changes are not restrictive.

C4.3 RADIATION THERMAL LOADING

Calculations were completed to determine the impact of blending the capsule material with NCAW and CC waste (the calculations are summarized in Attachments 2 and 3). For NCAW, it is estimated that at the time the HWVP begins operating, the curie content would increase between 89 percent (maximum curie case) and 99 percent (nominal watt case) above the current reference cases. The watts per canister would increase between 82 percent (maximum watt case) and 181 percent (nominal curie case) above the current reference cases. As stated in Section 3.3, the year 2015 is the earliest time defense wastes will begin to be shipped to the repository.

As demonstrated in Figure C4-1, as much as 50 percent of the radioactivity present in NCAW glass in the year 2000 will be gone by the time the canisters are shipped to the repository. The increase in watt loading will increase the centerline temperature of the canister. The WAPS state that the centerline temperature should remain below the glass transition temperature during interim storage and be at least 100 °C below the glass transition temperature ($T_g = 500$ °C) when the canisters are shipped to the repository. Figure C4-2 presents an estimate of the canister centerline temperature as a function of canister watt loading (Elmore 1986). It is apparent that the watt loading must increase by almost a factor of 5 above the maximum value of 1,524 W/canister being considered here before the centerline temperature criterion is violated.

The facility shielding requirements also must be considered. Section 12, Item 200 of the HWVP Technical Data Package (TDP) addresses shielding requirements. As a conservative approach, the HWVP project staff have used estimated curie levels that are 260 percent above the maximum values used in the calculations reported in Attachment 2. Therefore, the inclusion of the cesium and strontium salts in the NCAW feed should not require greater shielding requirements than what is currently planned.

Figure C4-1. Rate of Decay of Neutralized Current Acid Waste Canister Radionuclides.

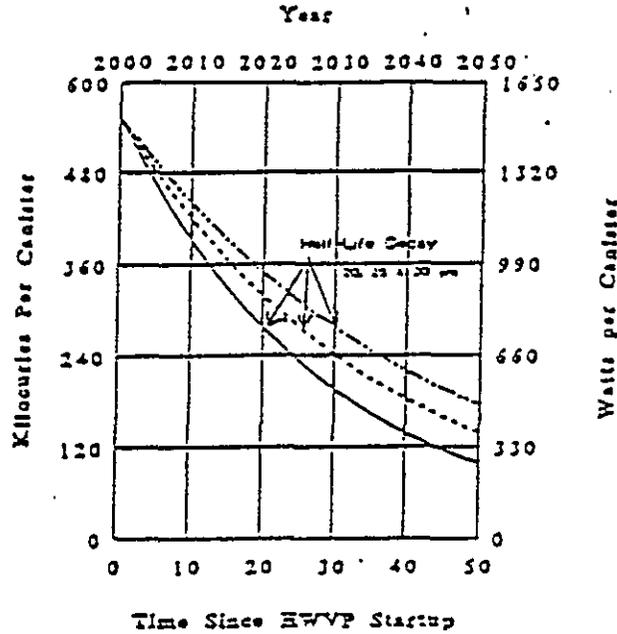
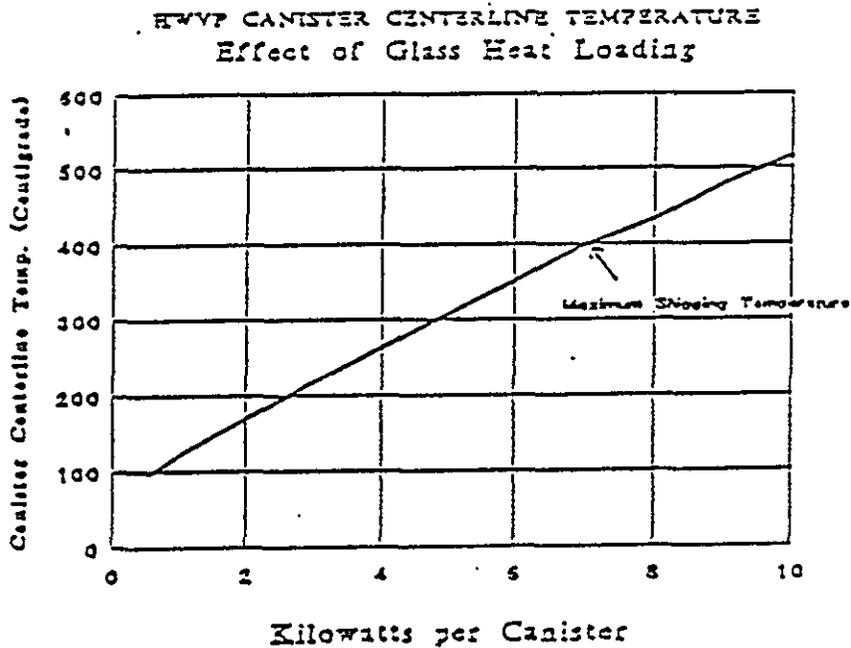


Figure C4-2. Effects of Canister Watt Loading on Canister Centerline Temperature.



C4.4 OFFGAS VOLATILITY AND TREATMENT

Higher cesium volatility may be expected as a result of the increase in halide concentration (Goles and Andersen 1986). The increase in cesium concentration itself has not been shown to increase the rate of cesium loss from the melter (Goles and Andersen 1986). Goles and Andersen (1986) reported a minimum cesium decontamination factor (DF)² of 9.5. This DF can be compared to the cesium DF of 14 for the HWVP design reference melter (Goles and Nakaoka 1990). These values can be placed into perspective by comparing them to the cesium DFs of 83 actually achieved during PSCM-23 (Goles and Nakaoka 1990) and 21 actually achieved during RLFCM operations with 5.7 wt% Cs₂O (Holton et al. 1988). It would be expected that actual ceramic melter testing would be conducted to develop offgas performance data. However, based on the data at hand, combining the NCAW and capsule salts should not have any severe impacts on the offgas treatment system.

Based on estimates made by Goles and Andersen (1987), the projected stack release of fluoride was 0.6 percent of the ambient air quality limit. Accounting for the increase in the HWVP production glass rate from 45 kg/h to 100 kg/h since the time of that study, and the added fluoride contributed from the SrF₂, the fluoride content in the offgas may increase by a factor of 3. Increasing the stack release of fluoride by a factor of 3 would result in the stack releases being just 1.8 percent of the limit value of 0.5 µg/m³.

For cesium, the projected stack release was estimated to be 0.7 percent of the limit. If cesium chloride capsule waste is vitrified, the stack release could increase to about 5.5 percent of the limit (assuming the same ratio of ¹³⁷Cs in the capsule and NCAW wastes). Goles and Andersen determined that expected strontium releases would be six orders of magnitude lower than the stack limits. Thus, increase in strontium as a result of capsule waste treatment would have no significant effect on the stack release projection. An air emission limit has not been assigned for chloride; therefore, chloride release was not included in this evaluation.

C4.5 EQUIPMENT MATERIALS COMPATIBILITY

New contaminants not presently in NCAW waste would not be introduced with the inclusion of the capsule salts. Melter DFs for chlorine and fluorine during PSCM-23 were 5 and 7.6, respectively, while HWVP design DFs are 2 and 5, respectively (Goles and Nakaoka 1990). Fluorine would increase by about 25 percent in the melter feed stream, while chlorine would increase by almost 130 percent. Impacts of higher anion concentrations on the in-cell equipment need to be evaluated. However, increases of this magnitude are not expected to increase expected corrosion rates of the equipment. Evaluations performed by Elmore and Jensen (1990) and information that they gathered during other tests for DOE indicated that applicable slurry studies have been done with chloride and fluoride concentrations up to 20,000 ppm and 2,300 ppm,

²Melter cesium DF is the ratio of the mass of cesium entering the melter to the mass of cesium escaping into the offgas treatment system. The percent retention in the glass = $(1 - 1/DF)(100)$.

respectively. Offgas condensate solutions have also been tested with up to 80,000 ppm and 9,000 ppm chloride and fluoride, respectively. Depending on pH, several alloys were identified as candidate materials for use in high halide environments. The alloy Hastelloy C-22 exhibited the lowest corrosion rates, generally less than 5 mil/yr. If the proper materials are used for the in-cell equipment, the HWVP could accommodate the halides from the cesium and strontium capsules.

Processing conditions and glass characteristics will be acceptable if the cesium and strontium salts (cesium, strontium, chlorine, and fluorine) are blended with pretreated NCAW and/or CC waste and processed through HWVP. Leach testing of glasses produced using NCAW and CC waste containing cesium and strontium salts would be required to confirm acceptable glass durability. The canister watt loading and centerline temperature will increase as a result of adding the capsule waste, but at the time of shipment to the geologic repository, the canisters will satisfy the WAPS. Ceramic melter testing would be required to obtain offgas performance data; however, it is estimated that the HWVP offgas treatment can handle the increased cesium, strontium, chlorine, and fluorine. The increases in halide concentrations should not increase the expected corrosion rates for HWVP equipment.

C5.0 PROCESS DESCRIPTIONS FOR HALIDE SEPARATION

This section describes flowsheets for separating the chloride and fluoride from the capsule waste.

C5.1 BACKGROUND

The double-shell tanks for the HWVP feedstreams are designed to accept contents in a pumpable form (i.e., slurry) only. Therefore, the direct blending of cesium chloride and SrF_2 with the HWVP feedstreams in the holding tanks is only possible if the salts are in liquid form. Blending of cesium chloride in liquid form is relatively simple because cesium chloride has a very high solubility in water. On the other hand, SrF_2 is almost insoluble in water and, therefore, the salt has to be crushed into powder and then mixed with enough water to form a slurry before transfer to tanks for the HWVP feedstreams. When halides cannot be added to the HWVP feedstreams, separation of chloride and fluoride from cesium and strontium will be required.

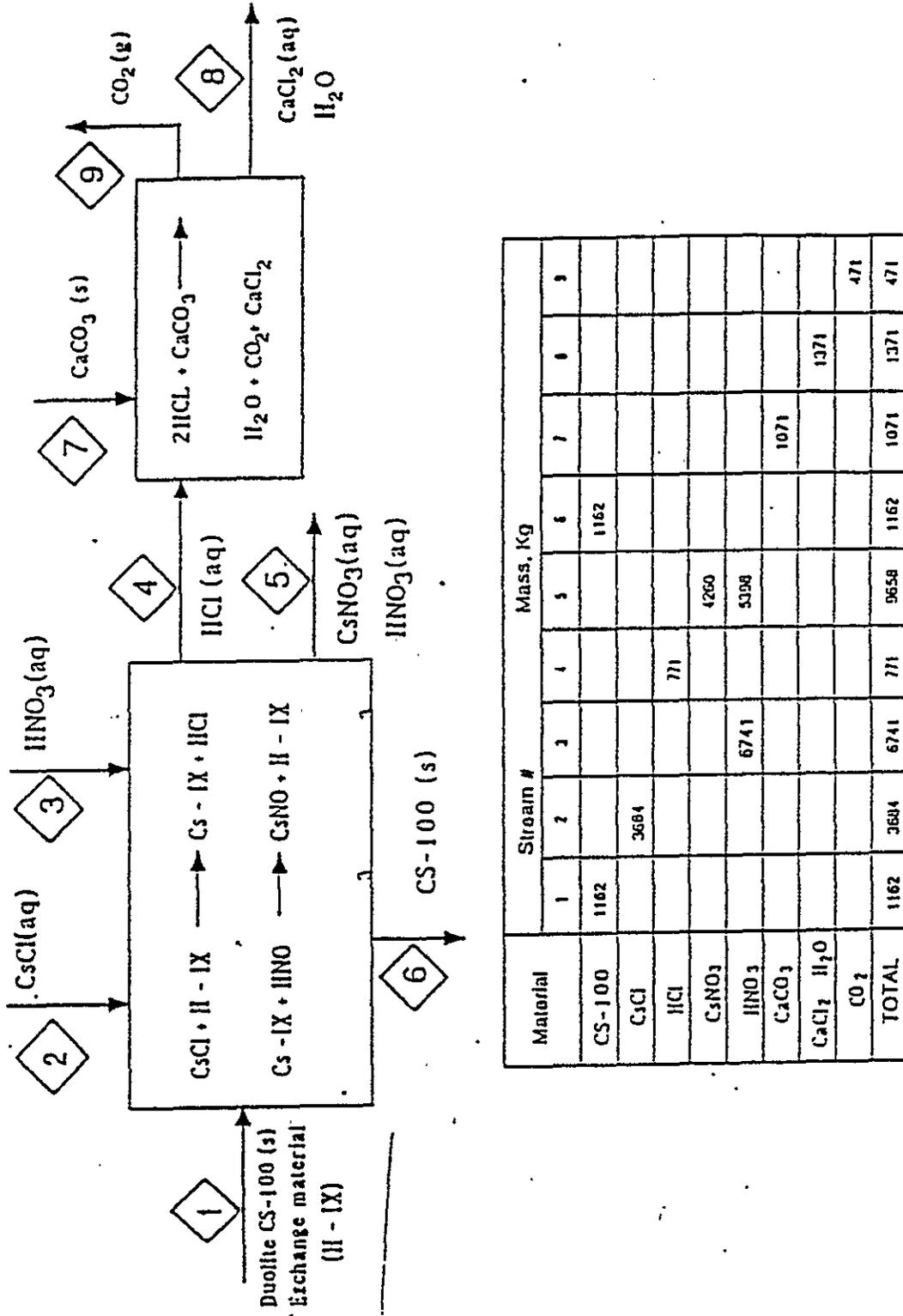
C5.2 SEPARATION OF CHLORIDE FROM CESIUM CHLORIDE

Cesium can be separated from chloride by chemical reaction or ion exchange. One chemical reaction method is the separation of Cs^+ from Cl^- by precipitation of the chloride using silver nitrate (AgNO_3). This reaction exploits solubility differences of the reaction products to achieve the final separations. Dissolving cesium chloride in water and mixing with a solution of AgNO_3 results in almost complete removal of the chloride ion as solid silver chloride (AgCl). In using ion exchange, dissolved cesium chloride is fed to a cation exchange column where cesium is held up and an electrolyte such as hydrogen chloride (HCl) is discharged. The cesium on the column could then be eluted using nitric acid (HNO_3).

C5.2.1 Ion Exchange

Ion exchange material for cesium-bearing solutions has been demonstrated for West Valley supernatant (Bray et al. 1984) and also was produced at B Plant to provide PNL with cesium for production of radioactive glass sources. Duolite Cs-100 resin was used for the work at B Plant and is assumed to be the material used for this study. The material balance is shown in Figure C5-1. This balance is based on an exchange capacity of 1.4 equivalents/L, a bed service life of 10 cycles or regenerations, and an elution volume of 5 column volumes. The elution agent is assumed to be 1 M HNO_3 . The process would involve dissolving cesium chloride in water and then feeding the solution to the top of a column. The HCl would be discharged out the bottom as Cs^+ is exchanged for H^+ . When the column is loaded, it could then be regenerated with 1 M HNO_3 , thus discharging cesium nitrate (CsNO_3) and excess acid. This eluted stream would be blended with the HWVP feed. The HCl stream that is produced could be neutralized and grouted. The spent resins could also be solidified in grout for disposal.

Figure C5-1. Flowsheet and Mass Balance for Ion Exchange
Dechlorination of Cesium Chloride.



C5.2.2 Reaction with Silver Nitrate

The flowsheet and mass balance for AgNO_3 treatment are given in Figure C5-2. To convert the 3,642-kg inventory of cesium chloride to CsNO_3 , a quantity of 3,675 kg of AgNO_3 is required, which results in the production of 4,216 kg of CsNO_3 . Initially, AgNO_3 and cesium chloride are prepared as aqueous solutions. The final solution volume can be adjusted according to the solubility limit of CsNO_3 . At 25 °C, the solubility is about 1.7M, and a final solution volume of 13,000 L is required to ensure solubility of the CsNO_3 . This volume could be reduced significantly by operating at higher temperatures because the solubility of CsNO_3 increases greatly with temperature. The solubility of AgCl is so low that an increase in temperature results in very little additional chloride ion in the solution. Equilibrium calculations show that at 25 °C, about 7 g of chloride ion is in 16,000 L of saturated AgCl solution.

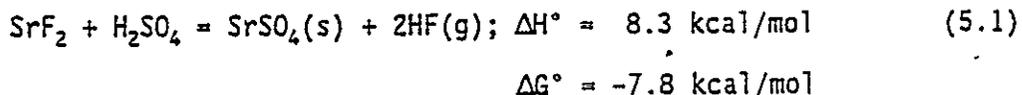
The AgCl is separated from the CsNO_3 solution and processed as solid waste. The CsNO_3 solution is blended with the HWVP feedstreams for further processing into glass.

C5.3 SEPARATION OF FLUORIDE FROM STRONTIUM FLUORIDE

Strontium can be separated from fluoride by chemical reactions. The SrF_2 is converted to strontium sulfate (SrSO_4) by reaction with sulfuric acid (H_2SO_4). The process is analogous to manufacture of hydrofluoric acid by conversion of fluorite (CaF_2). Because sulfate is undesirable in melter feed, it can be removed by metathesis of SrSO_4 with sodium carbonate (Na_2CO_3). The resulting strontium carbonate (SrCO_3) is insoluble and can be dissolved to form strontium nitrate $\text{Sr}(\text{NO}_3)_2$ by treatment with HNO_3 .

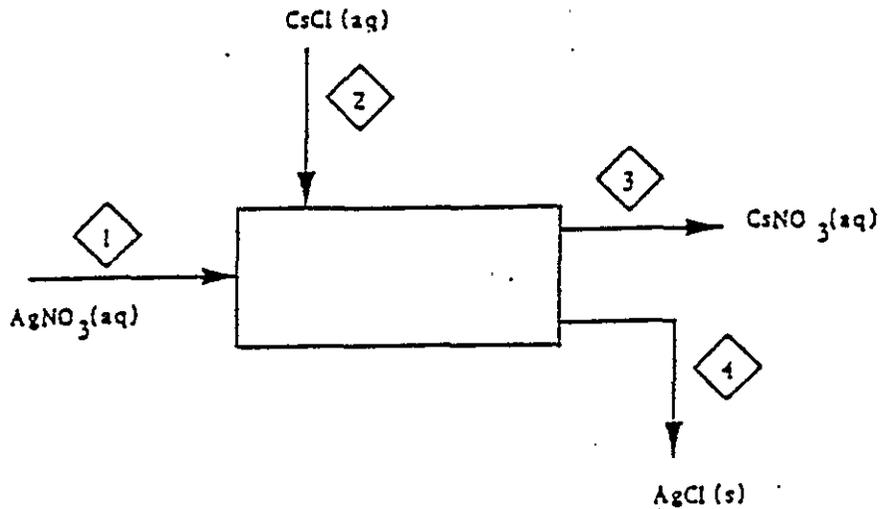
C5.3.1 Acid Decomposition of Strontium Fluoride

The yield on the acid decomposition was taken to be the same as that for CaF_2 as given in the literature. The formula for the acid decomposition reaction is:



The above reaction is analogous to the acid decomposition of CaF_2 (fluorite), which has been used industrially for the production of HF . It is assumed here that SrF_2 can be decomposed by the same process. Simons (1950) indicates that all alkaline earth fluorides undergo acid decomposition, and thermodynamic calculations from standard heats of formation show the reaction of SrF_2 to be more favorable than CaF_2 . Therefore, it could be expected that acid decomposition is feasible. The flowsheet and mass balance for this acid decomposition are given in Figure C5-3 and Table C5-1, respectively. According to Table C5-1, about 1,493 kg of H_2SO_4 is required to react with a total inventory of 1,717 kg of SrF_2 , resulting in the production of about

Figure C5-2. Flowsheet and Mass Balance for the Removal of Chloride Ion from Cesium Chloride Solution by Precipitation with Silver Nitrate.



Species	Stream 1	Stream 2	Stream 3	Stream 4
Ag ⁺	2,334			2,334
NO ₃ ⁻	1,341		1,341	
Cs ⁺		2,875	2,875	
Cl ⁻		767		767
H ₂ O	8,000	8,000	16,000	
Total	11,675	11,642	20,216	3,101

All values above are in kilograms.

No change in chloride content of the glass from the base case.

The nitrate composition of the feed would be increased by 0.7 lb/100 lb non-volatile oxides.

Figure C5-3. Flowsheet for the Acid Decomposition of Strontium Fluoride.

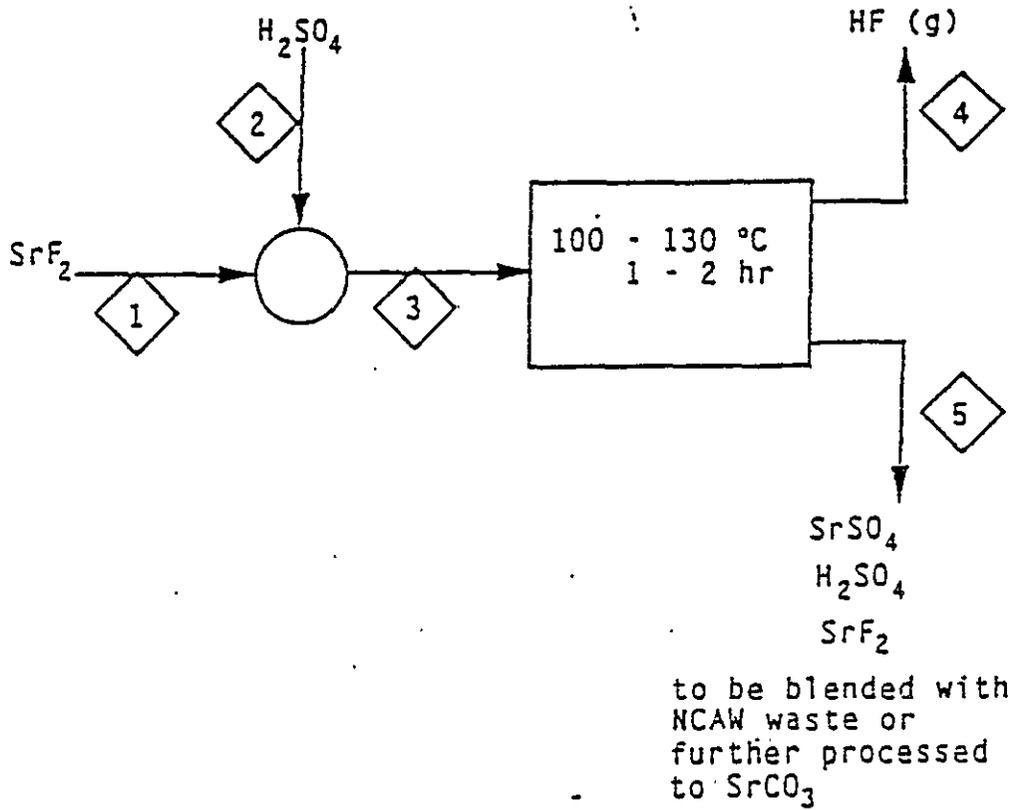


Table C5-1. Mass Balance for the Acid Decomposition of Strontium Fluoride.^a

	Stream number (kg)				
	1	2	3	4	5
SrF ₂	1,717	0	1,717	0	80.6
H ₂ SO ₄	0	1,492.6	1,492.6	0	215.1
HF	0	0	0	521.3	0
SrSO ₄	0	0	0	0	2,392.6
Total	1,717	1,492.6	3,209.6	521.3	2,688.3

^aGlass compositions

Quantity of fluorine after acid decomposition	24 kg
Quantity of fluorine in NCAW glass with treated capsules added	2,400 kg
wt% fluorine in NCAW glass with treated capsules added	0.303
wt% fluorine in reference case	0.3
Quantity of SO ₃ in NCAW glass	1,287 kg
Quantity of SO ₃ after acid decomposition	1,462 kg
Quantity of SO ₃ in NCAW glass with treated capsules added	2,749 kg
wt% of SO ₃ in NCAW glass with treated capsules added	0.35
wt% SO ₃ in reference case	0.16

2,393 kg of SrSO_4 . The process could be carried out either batchwise or continuously. The continuous process as described by Ryss (1956) is as follows: powdered SrF_2 is mixed with H_2SO_4 in a mechanical mixing device. Over a period of 1 to 2 h, the mixture passes through an externally heated revolving furnace at a temperature of 100 °C to 130 °C. The HF is driven off as a gas and SrSO_4 containing 2 percent to 3 percent SrF_2 and 7 percent to 8 percent H_2SO_4 remains. At these low temperatures, very little H_2SO_4 contaminates the HF gas stream. If this contamination is not important in this process, then higher temperatures might accelerate the reaction. Further treatment by metathesis with Na_2CO_3 , described below, could remove the sulfate, if necessary.

C5.3.2 Metathesis of Strontium Sulfate to Strontium Carbonate

Without removal of the sulfate, the weight percent sulfate in NCAW glass is estimated to be 0.35. The maximum limit for SO_4 is 0.50 wt%. The direct addition of SrSO_4 to NCAW increases the concentration of sulfate in NCAW glass close to the specified sulfate limit for the glass; therefore, the solid SrSO_4 can be converted to SrCO_3 by treatment with aqueous Na_2CO_3 to eliminate this problem. Treatment of SrSO_4 with Na_2CO_3 will result in the metathesis of SrSO_4 into SrCO_3 (both insoluble) and the sulfate will remain in solution as sodium sulfate (Na_2SO_4) per the reaction:



The equilibrium constant, K, shows this reaction to be quite favorable. If this reaction continues to completion, all the sulfate would be removed in the supernatant. The resulting carbonate has little effect on the total CO_3 in the HWVP feed, increasing the value from 17 to 17.4 kg/100 kg of nonvolatile oxides in the reference case. This reaction can be effective in reducing the amount of SO_4 in the treated capsule waste.

The flowsheet and mass balance for this metathesis are given in Figure C5-4 and Table C5-2, respectively. The metathesis of SrSO_4 generates a liquid stream consisting of 2,161 kg of Na_2SO_4 , 136 kg of H_2CO_3 , and unreacted Na_2CO_3 and H_2SO_4 . The liquid stream is processed and packaged as waste. The solid SrCO_3 can be transferred as a slurry for blending with NCAW.

C5.3.3 Dissolution of Strontium Carbonate in Nitric Acid

Because slurry transfers may lead to plugging of transfer lines, thereby affecting the processing schedule, the SrCO_3 can be converted to $\text{Sr}(\text{NO}_3)_2$ by reacting the SrCO_3 with HNO_3 as follows:

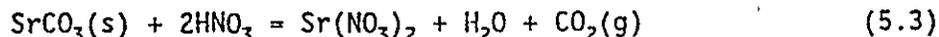


Figure C5-4. Flowsheet for the Metathesis of Strontium Sulfate to Strontium Carbonate.

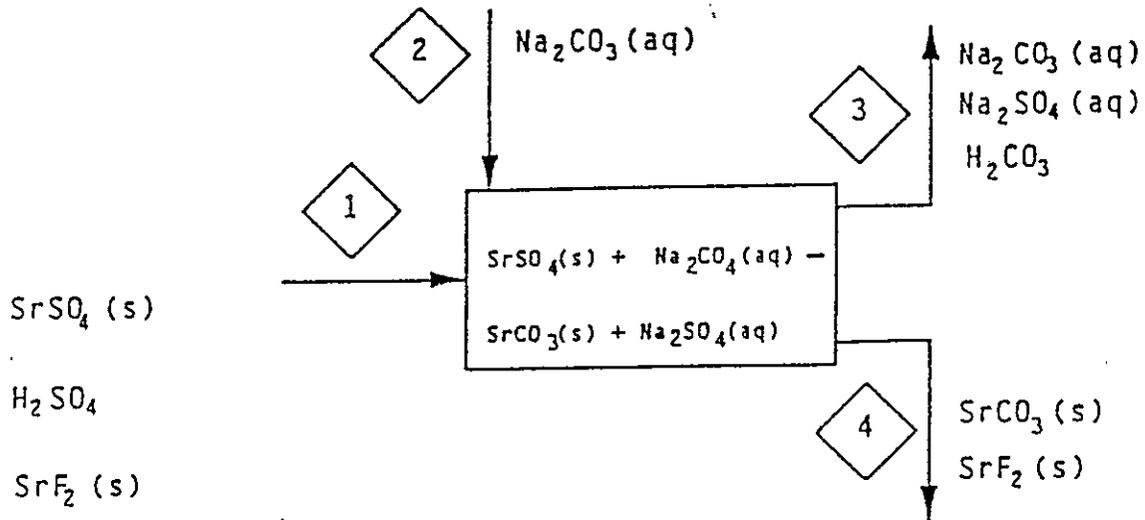


Table C5-2. Mass Balance for the Metathesis of Strontium Sulfate to Strontium Carbonate.^{a,b}

	Stream number (kg)			
	1	2	3	4
SrSO ₄	2,393	0	0	0
SrCO ₃	0	0	0	1,923
SrF ₂	81	0	0	81
Na ₂ CO ₃	0	2,120	507	0
Na ₂ SO ₄	0	0	2,161	0
H ₂ CO ₃	0	0	136	0
H ₂ SO ₄	215	0	0	0
Total	2,688	2,120	2,804	2,004

^aGlass composition changes: fluorine is unchanged from the acid decomposition case; SO₃ is unchanged from the reference case.

^bFeed composition changes:

CO ₃ in feed (kg/100 kg nonvolatile oxides)	17 kg
Quantity of nonvolatile waste oxides in glass	198,000 kg
Total quantity of CO ₃ (reference case)	33,660 kg
Quantity of CO ₃ in metathesized waste	782 kg
Total CO ₃ in feed with capsules added	34,442 kg
CO ₃ in feed with capsules added (kg/100 kg nonvolatile oxides)	17.4 kg

This will keep strontium in solution and ensure a successful and timely transfer of radioactive strontium for blending with NCAW. Equilibrium calculations show that in a 1 M acid solution, the solubility of SrCO₃ is about 0.5M (0.5M in a 1 M acid solution).

The amount of 1 M HNO₃ solution required to completely dissolve the 13 kmol of SrCO₃ is 26,000 L.

C6.0 EVALUATION OF FACILITIES AND EQUIPMENT

The evaluation of facilities and equipment for processing the capsules through HWVP are detailed in this section. The section also identifies the required modifications to the facilities and equipment.

The removal of cesium chloride and SrF_2 salts from the capsules and processing of the salts can be performed at either WESF or HWVP. If WESF is used for these operations, the processed liquids will be pumped to B Plant for blending with the waste feedstreams, and then to the double-shell tank 241-AY-102 (referred to as tank 102-AY) before finally being vitrified at the HWVP. However, if HWVP is used for the capsule processing operations, the processed capsule waste can be blended directly with the waste feedstreams in the slurry receipt and adjustment tank (SRAT) at the HWVP, thereby bypassing the B Plant, tank 102-AY, and the receipt and lag storage tank (RLST).

One decided advantage of processing the capsules directly in the HWVP is in lessening the probability of a tank "bump" in tank 102-AY. A "bump" is caused when self-heating results in accumulation and subsequent release of gas. If the wastes going into tank 102-AY are to be concentrated before their transfer to HWVP for processing into glass, tank 102-AY might experience a tank bump. This problem will be exacerbated if the capsules containing cesium and strontium have to be processed through the 241-AY and -AZ Tank Farms. The cesium will be contained in the supernatant, but the strontium will settle to the tank bottom with the sludge. The presence of strontium in the sludge increases the self-heating of the sludge and will, therefore, increase the probability of a tank bump. Thus, the potential for a tank bump in tank 102-AY must be investigated before routing the capsule waste through the 241-AY and -AZ Tank Farms to the HWVP.

Processing the capsules directly at the HWVP will also bypass the RLST. The proposed construction material for this tank is AISI 316. The waste composition in this tank is carefully controlled to avoid chloride stress corrosion cracking. The tightly controlled composition specification for this alloy sets limits on the chloride concentration of the wastes that will be treated by the tank. Presently, the liquid heel that remains in tank 102-AY is presumed to contain a high chloride concentration, which may be reduced to acceptable levels by mixing with wastes containing low chloride concentrations. However, blending the capsule waste with NCAW and/or CC waste may not reduce the chloride concentration of the waste feedstream to levels that are low enough to preserve the integrity of RLST.

The WESF will be used to perform TRUEX pilot-plant operations (under project W-153) and the HWVP bench-scale melter test simultaneously. Therefore, performing the capsule processing operations directly at the HWVP (with some modifications to the current HWVP design) will avoid any conflict with the two planned projects.

The following sections give the details for disassembly and processing of capsules at WESF. It should be noted, however, that the same process details also apply if operations are carried out in the HWVP. (See Section 6.3 for the modifications required to HWVP.)

C6.1 BLEND CESIUM CHLORIDE AND STRONTIUM FLUORIDE DIRECTLY WITH NEUTRALIZED CURRENT ACID WASTE AND/OR COMPLEXANT CONCENTRATE WASTE

The removal of cesium chloride and SrF_2 salts from the capsules can be performed at WESF. The WESF building contains seven hot cells (A, B, C, D/E, F, and G). The A Cell was previously used for solid waste loadout. The cell contains the equipment required for handling high-level solid waste from the remaining process cells. The portion of A Cell that is adjacent to B Cell contains a stainless steel hood for receiving the contaminated solid waste and a transfer drawer that is accessible from the service gallery. Both B Cell and C Cell were used for SrF_2 processing. The D/E Cell is a double cell that was used for cesium chloride processing. The F Cell was previously used for decontamination of the inner capsules and subsequent welding and helium leak testing of the outer capsules. All of the operations are performed remotely with manipulators by operators positioned behind the lead glass windows of each cell. The G Cell was primarily used for placement of the outer capsules onto the inner capsules. Pass-throughs are located between adjoining cells for passage of equipment and solid waste. The WESF floor plan is discussed in more detail elsewhere (Sewell 1986).

This option can be carried out in either B and C cells or the D/E Cell of the WESF for both cesium chloride and SrF_2 capsules.

C6.1.1 Dismantling of Cesium Chloride Capsules

The WESF was previously used to dismantle 46 cesium chloride capsules from the WESF pool cells and two cesium chloride capsules from Sandia Laboratory and to process cesium chloride for transfer to B Plant and ultimate transfer to PNL as CsNO_3 solution in support of PNL's fabrication of isotopic heat sources for shipment to the FRG.

The WESF crew modified an existing pipe cutter to cut through the stainless steel outer capsule. The operations were performed in G Cell. This activity reduced the actual time of cutting and significantly reduced the amount of in-cell solid waste that would have been generated. Because of the very low smearable contamination, the cut up outer capsules were sent to low-level waste burial grounds. After the outer capsules were removed in G Cell by cutting off the welded end caps, the inner capsules were transferred to the D/E Cell and cut into three pieces using a chopsaw. The cut pieces were dumped into 5-gal-capacity buckets of water. This operation dissolved the cesium chloride, leaving the inner capsule material unaffected. The stainless steel inner capsule materials were packaged in 0.21-m^3 (55-gal) drums and transported to the B Plant canyon. The drums with the capsule materials were stored at the B Plant canyon until there was enough contaminated equipment for disposal. The drums were then loaded into the drag-off boxes along with the equipment and buried in the 218-E-10 burial grounds in the 200 East Area. The water containing the dissolved cesium chloride was dumped into the sump of D/E Cell, which used steam jets to pump the cesium chloride solution to B Plant for further processing.

The same procedure can be used again, although new chopsaws need to be assembled in-plant or procured from an outside vendor. The volume of water needed to process all 1,349 cesium chloride capsules is conservatively estimated to be approximately 4.54 m³ (1,200 gal); resulting in approximately 5.68 m³ (1,500 gal) of the processed solution for transfer to B Plant.

C6.1.2 Dismantling of Strontium Fluoride Capsules

The method for cutting cesium chloride capsules with chopsaws at WESF has been demonstrated before. Although the same method can be used to cut the SrF₂ outer capsules, the method cannot be used to cut SrF₂ inner capsules. Based on experience at the ORNL (ORNL 1988), SrF₂ is usually stuck to the inner Hastelloy capsule wall. The ORNL routinely made two longitudinal cuts 180 degrees apart on the Hastelloy capsule using a saw with a 0.13-m (5-in.) disc-type blade, and used a hammer and chisel to break the SrF₂ salt loose from the capsule walls. This operation broke the SrF₂ salt into several pieces. Using this procedure, ORNL was able to process about six capsules in three 8-h shifts. The procedure, therefore, must be modified to include these steps. The chopsaws also require modification to make the longitudinal cuts. The solid waste (inner and outer capsule materials) generated can be disposed of in a manner similar to that of cesium chloride capsules, as described above.

Once the SrF₂ salt is separated from the capsule into several pieces, the pieces can be transferred into a jaw crusher to crush SrF₂ salt into particles <300 μm (number 50 sieve) size. Particles >300 μm (number 50 sieve) size can be recycled through the jaw crusher until the entire batch of capsules is crushed to <300 μm (number 50 sieve) size particles. Jaw crushers are available commercially for remote operations in a hot cell. The SrF₂ powder can then be transferred to B Plant as a slurry in water, because SrF₂ has negligible solubility in water.

The slurry transfer is expected to be possible only if the solids content is less than 4 vol%. This limit is imposed to eliminate plugging of the transfer lines. Therefore, the volume of water used for transfer should be about 25 times as much as that of the solids--approximately 19.49 m³ (5,150 gal) of water for processing all 636 capsules.

Because of limited space availability in the WESF hot cells, tanks with capacity of no more than about 0.38 m³ (100 gal) can be accommodated. Therefore, for the SrF₂ slurry processing, a 0.38-m³ (100-gal) holding tank can be procured and the process of dismantling the capsules, crushing, mixing with water, and pumping into a holding tank can be made into a batch operation. The size of each batch can be fixed at 9 capsules, resulting in a total of 71 batches for processing all of the 636 SrF₂ capsules. Because the sump and the steam jets cannot be used to transfer the SrF₂ slurry, unlike the cesium chloride case, a vortex pump will be used to pump the slurry to B Plant. Before transfer of the slurry to B Plant, the mixture of water and SrF₂ solids in the holding tank will be agitated by a paddle agitator (similar to the one in use at B Plant) to keep the SrF₂ solids from settling.

One primary cause for concern in transferring the slurry is the possibility that the 0.025-m (1-in.) transfer lines from WESF to B Plant could become plugged by the SrF_2 solids, especially near the pipe elbows and bends in the line. Such plugging could be avoided by crushing the SrF_2 solids to <300- μm size particles, mixing with at least 25 times as much volume of water, and sizing the vortex pump to deliver at least 3.048 m/s (10-fps) fluid velocities.

In addition to the possibility of plugged transfer lines from WESF to B Plant, problems are associated with processing the SrF_2 slurry in B Plant. The intermediate storage and transfer tanks at B Plant have a much larger capacity (18.93 m³ [5,000 gal]), and the relatively small (0.38-m³ [100-gal]) batch transfers from WESF will have to be mixed with large volumes of water to successfully transfer the majority of the highly radioactive solution at the 4 vol% limit. This will drastically increase the total volume of capsule waste going to the HWVP.

An alternative to transferring SrF_2 solids as a slurry is transporting them as solids from outside the WESF building in a truck to B Plant. However, this option would require a double-shielded cask meeting present regulations. Presently, a cask with an approved design is not available. In addition, there is also the problem of adding the SrF_2 solids (powder) to the pretreated NCAW and/or CC waste in the B Plant.

C6.1.3 Blending of Cesium Chloride and Strontium Fluoride Solutions with Neutralized Current Acid Waste and/or Complexant Concentrate Waste and Transfer to Hanford Waste Vitrification Plant

The blending of cesium chloride and SrF_2 solutions with NCAW and/or CC waste is not expected to require any modifications to B Plant. A series of jumpers exist within B Plant that would facilitate transfer of liquids from a given tank to any other tank within any cell of the B Plant. However, these jumpers may be committed for use in the processing of NCAW at B Plant, requiring the procurement and installation of new jumpers and associated piping and connections. The current pretreatment scheme requires that the NCAW supernatant be processed at B Plant to remove cesium. The NCAW solids will be processed at the 244-AR Vault and transferred to tank 102-AY. The cesium extracted from the NCAW supernatant also will be transferred to tank 102-AY for blending with the processed NCAW solids. The cesium and strontium from the capsules initially will be transferred to B Plant, and from there will be pumped to tank 102-AY for blending with the pretreated NCAW. The mixture then will be transferred to HWVP for vitrification. Instead of transferring separately, the cesium recovered from the NCAW supernatant and the cesium and strontium from the capsules could be pumped into a holding tank at B Plant, and the mixture could then be transferred to tank 102-AY.

Existing aging waste transfer systems, in conjunction with new aging waste transfer lines to be installed, will be used for transferring processed NCAW and capsule solutions from B Plant to tank 102-AY. The pretreated NCAW from tank 102-AY will be routed through Diversion Box Number 1 (located due south of the B Plant) to HWVP for vitrification. The transfer from

tank 102-AY to Diversion Box Number 1 will be made using the existing lines and those installed under Project W-028. The final transfer from Diversion Box Number 1 to the HWVP will be performed using transfer lines to be constructed by the HWVP Project.

C6.2 PRETREATMENT OF CESIUM CHLORIDE AND STRONTIUM FLUORIDE BEFORE BLENDING WITH NEUTRALIZED CURRENT ACID WASTE AND/OR COMPLEXANT CONCENTRATE WASTE

This option is similar to the direct blending option, and can also be carried out either in the WESF or the HWVP for both cesium chloride and SrF_2 capsules.

Direct blending of cesium chloride and SrF_2 salts with NCAW and/or CC waste feedstreams may produce a feed with a composition that is outside the HWVP design specifications. This would necessitate the removal of chlorine and fluorine from cesium chloride and SrF_2 before blending with the waste feedstreams. This scenario identifies the facilities and equipment needed for the (1) removal of halides from the cesium and strontium salts, (2) transfer of the cesium/strontium solutions to B Plant, and (3) blending of the solutions with the NCAW and/or CC waste feedstreams.

The removal of cesium chloride and SrF_2 salts from the capsules and disposal of solid wastes generated by these operations will be performed in the same manner as described in Section 6.3.

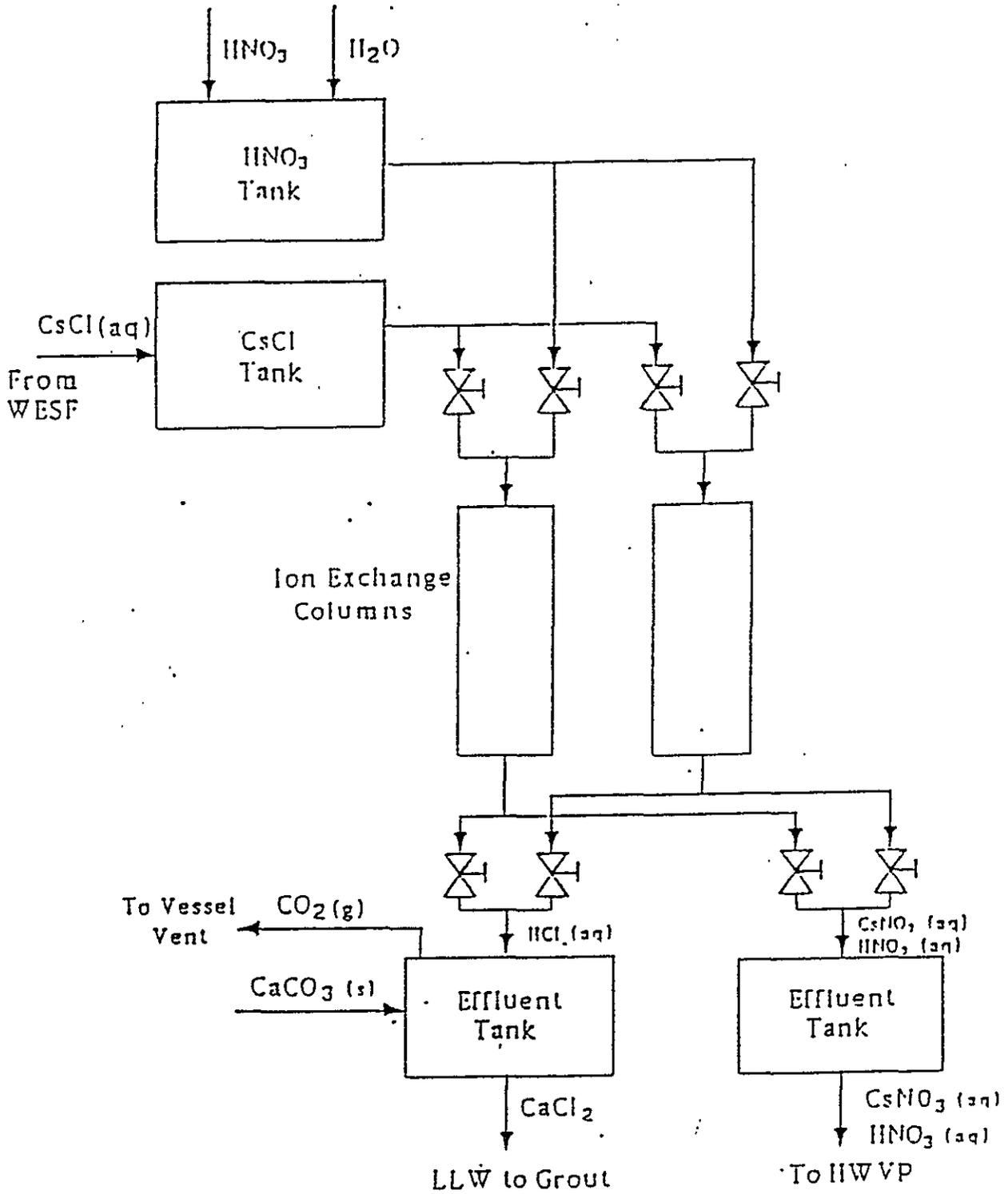
C6.2.1 Separation of Chloride from Cesium Chloride

The two methods considered for separating Cl from cesium chloride are ion exchange and reaction with silver nitrate.

C6.2.1.1 Ion Exchange. B Plant was used previously to dechlorinate cesium chloride and convert the cesium to a nitrate salt. A similar process could be used for the cesium chloride capsules. At WESF, the capsules would be opened and the cesium chloride dissolved in water. The dissolved cesium chloride would be transferred to B Plant and processed through ion-exchange columns. The flow diagram for the process is given in Figure C6-1. In a previous study, Cell 18 at B Plant was identified as a likely location for ion-exchange processing (PNL 1984).

The cesium chloride is first dissolved in water at WESF to a concentration of 0.5M. The total volume of cesium chloride solution would be 42,800 L. In a 380-L capacity tank, there would be about 113 batches. On transfer to B Plant, a holding tank would be needed. A 5,000-L tank would contain about 9 batches of cesium chloride solution.

Figure C6-1. Flow Diagram for Ion-Exchange Dechlorinization of Cesium Chloride.



Two ion-exchange columns would allow alternating processing and regeneration cycles. If the columns are 0.46 m in diameter x 3.7 m long (18 in. in diameter x 12 ft long), they would have an effective resin capacity of 390 L (allowing 35 percent of the bed for head space). The total amount of bed volume is based on an exchange capacity of 1.4 equivalents/L. At this rate, 15,300 L of effective resin volume would be used. If each bed is used for ten cycles before exhaustion, then each would be changed out once during the campaign, corresponding to an actual resin requirement of 1,530 L. The column is regenerated with 1 M HNO_3 . About 107,000 L of acid would be used. With a 5,000-L tank, about 21 batches of acid would be made up. The solution produced from regeneration is composed of cesium nitrate and HNO_3 . This solution has the same volume as the HNO_3 regeneration solution and could be held in another 5,000-L tank for transfer to HWVP.

During ion exchange, HCl is discharged from the column and is collected in another holding tank. After neutralization with limestone, this waste stream could be sent for grout disposal. About 12,000 L of CaCl_2 solution would be generated. With a 1,000-L tank, this low-level waste would be transferred for grout disposal 12 times.

C6.2.1.2 Reaction with Silver Nitrate. As mentioned before, the separation of chloride from cesium chloride can be carried out either in WESF or HWVP. Figure C6-2 shows the flow diagram for the separation process. The process involves chopping the outer and inner capsules, dissolving the cesium chloride in water, mixing the cesium chloride/water solution with aqueous AgNO_3 to precipitate chloride as AgCl , separating the solid AgCl from the CsNO_3 solution, and finally pumping the CsNO_3 solution to B Plant (if capsule processing operations are performed at WESF) for blending with NCAW and/or CC waste.

Chopsaws can be used for dismantling the inner and outer capsule materials. The cesium chloride salt can be dissolved in water in a 0.19-m^3 (50-gal) tank and the cesium chloride solution can then be transferred to a 0.38-m^3 (100-gal) tank for treatment with AgNO_3 . The total volume of the AgNO_3 and cesium chloride solutions involved in the chloride separation process is approximately 22.71 m^3 (6,000 gal). With a 0.38-m^3 (100-gal) process tank, the entire cesium chloride inventory can be processed in about 70 batches.

The next step is to separate the solid AgCl from aqueous CsNO_3 . This can be accomplished using a solid-liquid separation process currently being tested at the Chemical Engineering Laboratory for filtering the NCRW stimulant with a pneumatic hydropulse (PHP) filter. The filter used is a sintered stainless steel or Hastelloy metal filter of the required pore size. The filter housing consists of the metal filter through which the solution to be filtered is passed. Figure C6-3 is a schematic of a modified version of the test setup proposed for filtering the NCRW stimulant. As shown in the figure, the solution is pumped from the 0.38-m^3 (100-gal) AgNO_3 treatment tank through the filter system into a filtrate collection tank. To keep the AgCl solids suspended, the AgNO_3 treatment tank is agitated by means of a paddle agitator.

Figure C6-2. Flow Diagram for Processing Cesium Chloride to Remove Chloride.

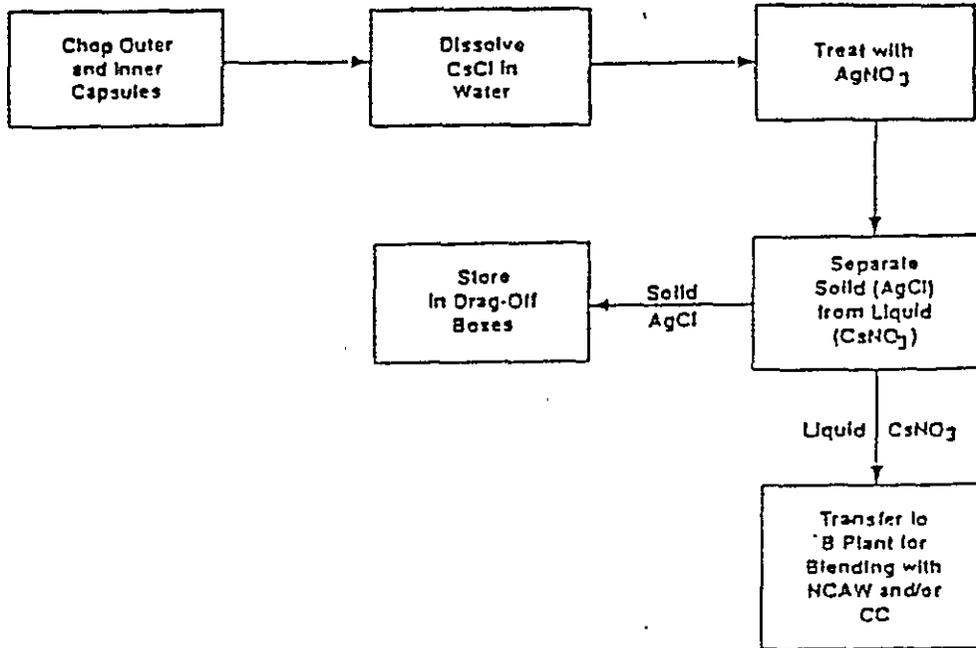
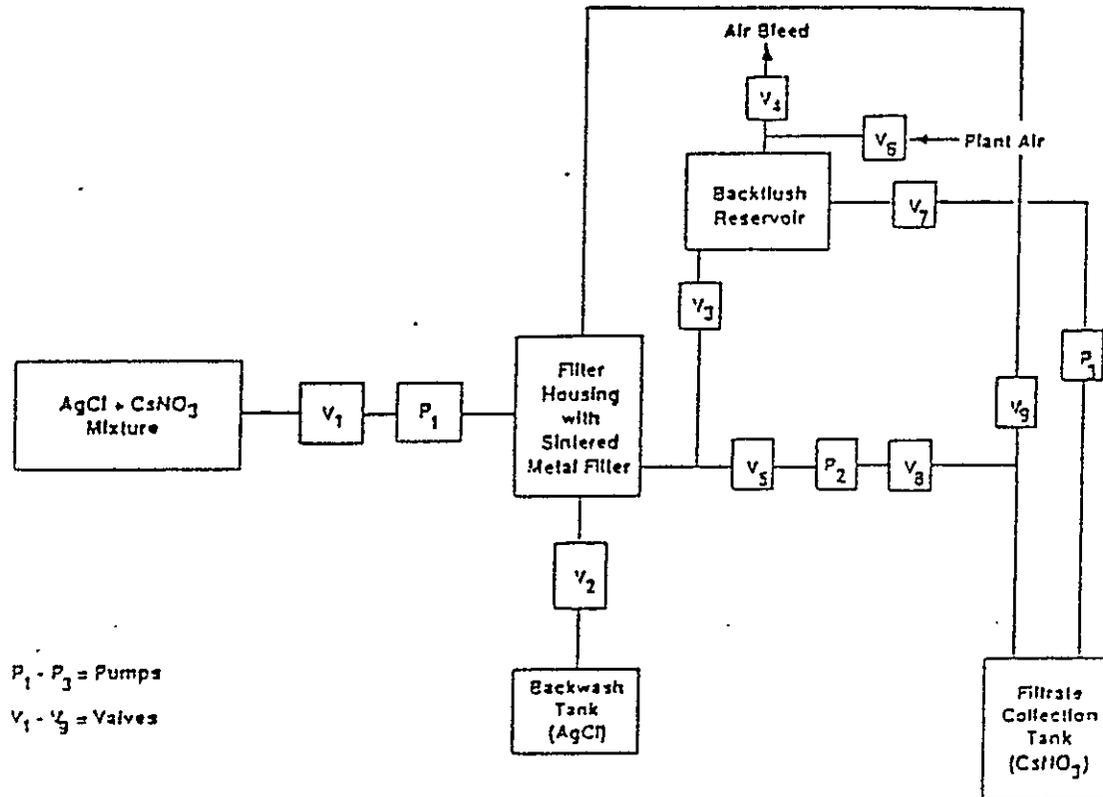


Figure C6-3. Schematic of Filter System for Separating Silver Chloride from Cesium Nitrate.



The stirred solution is pumped until a specified minimum flow rate and/or a specified maximum pressure drop is reached. When these conditions are met, the valve at the exit of the AgNO_3 treatment tank and the filtrate exhaust valve are closed.

The air inlet valve is opened to pressurize the backflush reservoir containing water to 0.55 MPa (80 psi). The blowdown outlet valve is then opened, causing the water to discharge through the filter and forcing the filter cake into the blowdown collection vessel. Before opening the blowdown outlet valve, the majority of the liquid from the filter housing is pumped out into the filtrate collection tank to ensure that very little of the CsNO_3 solution escapes into the collection vessel along with the AgCl solids. The air pressure valve and blowdown valve are closed. The feed valve and the filtrate exhaust valve are opened. The air is bled from the system and the cycle is repeated. The AgCl filter cake is dried and transported to B Plant canyon for burial as solid waste. The solid AgCl waste is considered to be mixed waste because AgCl is hazardous. The mixed waste should be packaged in accordance with appropriate regulations, transported according to DOT-49 CFR, and stored in the central waste complex for future retrieval and shipping to a mixed waste storage facility. The CsNO_3 solution is dumped in the sump and transferred to B Plant using steam jets.

C6.2.2 Separation of Fluoride from Strontium Fluoride

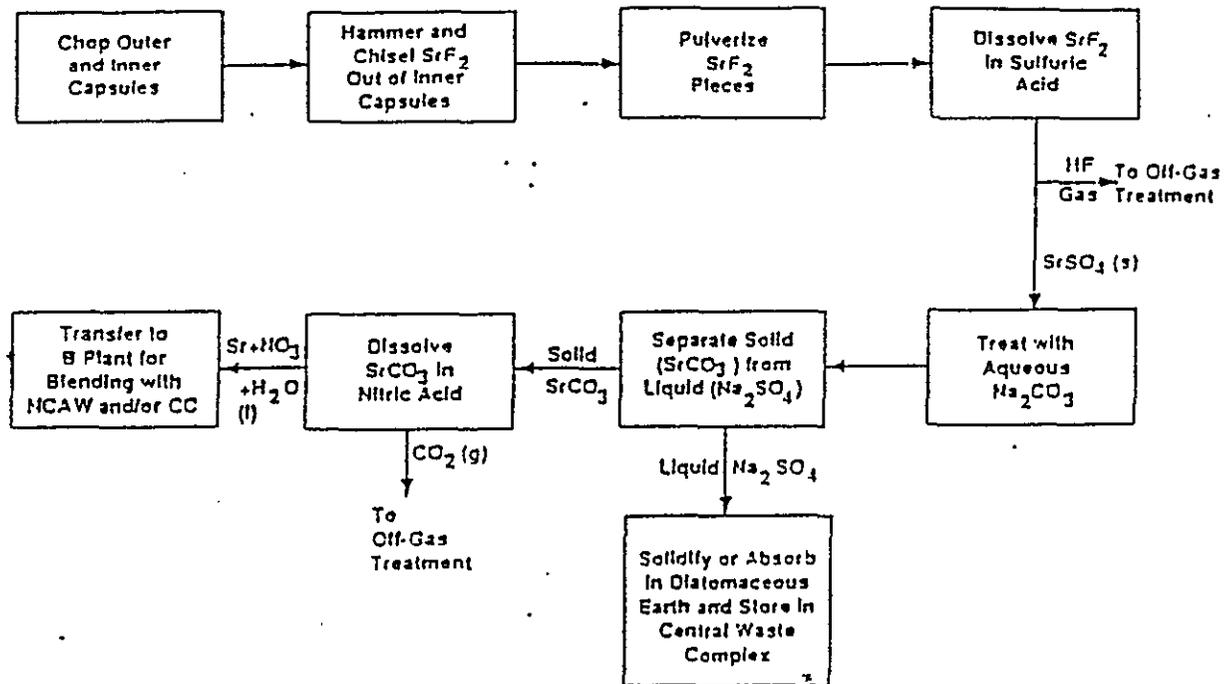
The separation of fluoride from SrF_2 can also be carried out in the chemical process D/E Cell at WESF. The flow diagram for the separation process is shown in Figure C6-4.

As schematically illustrated in the figure, the separation process involves chopping the outer and inner capsules with modified chopsaws, breaking the SrF_2 salt loose from the Hastelloy capsule into several pieces, pulverizing the SrF_2 pieces into particles of $\leq 300 \mu\text{m}$ (number 50 sieve) size, dissolving in sulfuric acid to convert to SrSO_4 , treating with aqueous sodium carbonate to convert to SrCO_3 , dissolving in HNO_3 to transform to soluble $\text{Sr}(\text{NO}_3)_2$, and finally pumping to B Plant for blending with NCAW and/or CC waste.

The steps involving chopping, hammering and chiseling, and pulverizing are the same as those described in Section 6.3.2 for processing SrF_2 , where fluoride separation is not required. These operations can be performed either in the D or E portion of the D/E cell of the WESF.

The pulverized SrF_2 powder is then dissolved in sulfuric acid for conversion to solid SrSO_4 . A stoichiometric amount of 0.81 m^3 (215 gal) of concentrated H_2SO_4 (18M) is needed to dissolve the SrF_2 . However, this amount needs to be increased by 20 percent to 0.98 m^3 (258 gal) to ensure that most of the inventory of SrF_2 is dissolved. Thus, the total volume of SrF_2 and H_2SO_4 will be about 1.61 m^3 (425 gal). Using a 0.08-m^3 (20-gal) tank, the entire operation of dissolution of SrF_2 in H_2SO_4 can be carried out in about 40 batches, assuming the effective volume of the tank to be about 75 percent. According to the flowsheet, this reaction is hastened by agitation and an increase in temperature and, therefore, the SrF_2 and H_2SO_4 initially will be

Figure C6-4. Flow Diagram for Processing Strontium Fluoride to Remove Fluoride.



mixed in the 0.08-m³ (20-gal) tank and transferred to an externally heated rotating furnace of the same capacity operated at 100 °C to 130 °C. The SrF₂ and H₂SO₄ mixture will be kept in the furnace at the specified temperature for approximately 2 hours, after which time the conversion of SrF₂ to SrSO₄ for the most part is expected to be complete. Besides SrSO₄, the other byproduct of the reaction between SrF₂ and H₂SO₄ is HF, which is in gaseous form and will be directed into the offgas treatment system. Presently, there is no offgas treatment system in WESF. Such systems have not been required since previous operations at WESF primarily involved encapsulation of cesium chloride and SrF₂, and only more recently were some cesium chloride capsules disassembled and processed. Disassembly and processing of the cesium chloride capsules did not require elaborate chemical processing steps. However, gaseous wastes were filtered through at least two stages of high-efficiency particulate air (HEPA) filters before release to the atmosphere.

The generation of gaseous HF requires an offgas treatment system that would convert the majority of the HF gas to a nonhazardous substance and eventually release vapors containing permissible levels of HF to the atmosphere. The treatment involves neutralizing the HF with KOH to generate KF. However, KF is considered to be a dangerous substance by the Washington Administrative Code (WAC). Therefore, KF will be treated with MgO (and H₂O), which will convert most of the KF to MgF₂ and will regenerate KOH. The low solubility of MgF₂ and MgO facilitates filtering from the liquor, leaving a regenerated KOH. The MgF₂ and MgO are not currently regulated waste according to WAC and possibly can be disposed of in drag-off boxes with solid capsule materials waste. The remaining KOH can be used again for further treatment of HF. The quantities of KOH and MgO required for treatment of the entire inventory of HF gas are 1,458.02 kg and 1,046.34 kg, respectively. Monel 400 can be used effectively as a containment material to resist corrosion by the offgas condensate.

Conversion of SrSO₄ to SrCO₃ by reaction with Na₂CO₃ is an effective way to reduce the total SO₄ going into the HWVP feedstreams. Any unreacted solid SrF₂ remaining from treatment with H₂SO₄ will be unaffected by the metathesis with Na₂CO₃.

The stoichiometric amount of solid Na₂CO₃ required to convert the entire inventory of SrSO₄ is 2,120 kg. However, since aqueous Na₂CO₃ is needed for this reaction, this translates to 10.61 m³ (2,804 gal) of 2M liquid Na₂CO₃. Taking into account the effective volume of the tank, the total adjusted amount of the SrSO₄ + Na₂CO₃ mixture is about 15.14 m³ (4,000 gal). By making use of a 0.38-m³ (100-gal) tank, the entire SrSO₄ inventory can be processed in 40 batches. Because of the limited solubility of SrSO₄, the SrSO₄/Na₂CO₃ mixture will be agitated to ensure completion of the metathesis in a matter of hours. As a byproduct of this metathesis, 2,161 kg of Na₂SO₄ is produced as liquid waste along with a small amount of Na₂CO₃. The separation of solid SrCO₃ (and any unreacted SrF₂) from the liquid waste can be accomplished using the PHP filter system described in Section 6.4.1. The liquid waste will be solidified or absorbed in diatomaceous earth for transportation and disposal as solid waste. However, the hazardous nature of the waste must be assessed. If the waste is hazardous it will be packaged with enough shielding to reduce the dose rate to <200 mR/h and stored in the central waste complex for future retrieval. If the waste is classified as nonhazardous, it will be transported

in 0.21-m^3 (55-gal) drums to the B Plant canyon for storage in drag-off boxes for eventual burial with contaminated equipment.

The SrCO_3 produced by the metathesis of SrSO_4 with aqueous Na_2CO_3 can be directly transferred to B Plant as a slurry in water for blending with the NCAW and/or CC feedstreams. This transfer is similar to the slurry transfer of SrF_2 described in Section 6.3.2. Therefore, the entire inventory of SrCO_3 solids will require approximately 15.14 m^3 (4,000 gal) of water. The SrCO_3 solids will be mixed with water in a 0.38-m^3 (100-gal) tank and transferred to the B Plant for further processing in 53 batches. As before, the success of this operation strongly depends on the capability of the vortex pump, which will be used to pump the slurry without any plugging of the transfer lines.

The uncertainty of plugged transfer lines can be eliminated by transferring strontium in liquid form rather than solid form. The SrCO_3 can be dissolved in HNO_3 to yield a $\text{Sr}(\text{NO}_3)_2$ solution to be mixed with the reference feed. Equilibrium calculations indicate that the entire inventory of SrCO_3 would require 26 m^3 (6,868 gal) of 1 M HNO_3 . Assuming the actual amount of HNO_3 used to be 25 percent more than the stoichiometric amount yields 32.5 m^3 (8,586 gal) of HNO_3 . Considering that only 75 percent of the volume of any given tank can be used effectively, the entire inventory would require 117 batches for transfer to B Plant using a 0.38-m^3 (100-gal) tank.

C6.2.3 Blending of Cesium Nitrate and Sodium Nitrate Solutions with Neutralized Current Acid Waste and/or Complexant Concentrate Waste and Transfer to the Hanford Waste Vitrification Plant

The blending of CsNO_3 and $\text{Sr}(\text{NO}_3)_2$ solutions with NCAW and/or CC waste is not expected to require any modifications to the B Plant. However, processing of the capsule waste at WESF to remove the chloride and fluoride will make the resulting solutions quite acidic. These solutions will have to be neutralized either in tank TK-25-1 or before transfer to tank TK-25-1 before blending with the NCAW waste. The combined presence of free fluoride and acid solutions is normally a cause for concern for the integrity of the storage and treatment tanks. However, in the present case, the B Plant tanks will be exposed to acid capsule wastes with free fluoride for a relatively short period of time. The acid wastes are quickly neutralized to reduce their corrosion potential. In addition, the fluoride concentration of the HWVP waste feed is expected to increase from 0.3 to only 0.37 by blending the capsule waste.

As discussed in Section 6.1.3, the processed capsule solutions will be blended with the pretreated NCAW solutions in tank 102-AY and eventually transferred to HWVP for vitrification.

C6.3 HANFORD WASTE VITRIFICATION PLANT FACILITIES AND REQUIRED MODIFICATIONS

The removal of the cesium chloride and SrF_2 salts from the capsules and processing of the capsules can be performed in the HWVP. One of the laydown areas in HWVP can be modified to provide the necessary capabilities. Laydown space would have to be identified elsewhere in the facility. The same

equipment described in Section 6.1 would be required for this work. Additional HWVP-specific equipment is (1) two shielded windows, (2) a remote camera, (3) lining and coverblocks for the cell, (4) four master-slave manipulators, (5) a 2.5-ton hoist, (6) in-cell lighting, (7) process water, stream, and air service, (8) a sump, jet and regulated drain to the decontamination treatment tank, (9) two doubly encased process lines to the SRAT, and (10) three stainless steel vessels and associated agitators, transfer jets and piping.

C7.0 COST ESTIMATES

This section describes the cost estimates for (1) storing the capsules indefinitely, (2) overpacking the capsules and shipping them to the repository, and (3) processing the capsules through HWVP.

C7.1 PROCESS CAPSULES THROUGH THE HANFORD WASTE VITRIFICATION PLANT

C7.1.1 Blend Cesium Chloride and Strontium Fluoride Directly
with Neutralized Current Acid Waste and/or Complexant
Concentrate Waste

This option requires the removal of cesium chloride and SrF_2 salts from the capsules and blending directly with NCAW and/or CC waste without processing to remove the halides. As mentioned in Section C6.3, this work can be carried out in the WESF. The outer capsules will be removed in G Cell by cutting off the welded end caps with chopsaws. Then, the inner capsules will be transferred to E Cell for cutting off the welded end caps with chopsaws and processing the cesium chloride and SrF_2 salts. The cutting and processing operations for cesium chloride are relatively simple compared to those for SrF_2 . Chopsaws similar to those used previously at WESF for cutting the cesium chloride capsules will be used again for cutting the cesium chloride capsules. The design will be improved, however, for more efficient operations. The chopsaw design needs to be modified for SrF_2 to make the longitudinal cuts. It is proposed in Section C6.4.2 to use a 0.38-m^3 (100-gal) tank for processing the SrF_2 slurry; the same tank also can be used for dissolving cesium chloride in water. Therefore, the equipment needed to carry out this option at WESF includes chopsaws, pulverizers (jaw crushers), a 0.38-m^3 (100-gal) tank with agitator, and a vortex pump. The itemized list of equipment and material costs is given in Table C7-1. The cost for equipment and materials for this option is \$173,000. This cost includes piping and installation, and minor decontamination of cells. Extensive decontamination is not required since the installation is performed remotely.

The only byproduct waste that is generated for this option is the solid waste associated with the disassembly of the inner and outer capsule materials. The cut up inner capsule materials will be disposed of in drag-off boxes from the B Plant canyon. The cut up outer capsule materials will be disposed of in low-level waste burial grounds. The capsules that were presumed to be leaking will have a third containment capsule. In this case, all three containment materials will be disposed of in drag-off boxes. The cost for solid waste disposal is estimated to be \$207,000 assuming the drag-off box costs \$173,000.

In addition to the costs above, costs are incurred for the operation of WESF, B Plant, 244-AR Vault, tank farms, and HWVP for processing the capsule wastes.

Table C7-1. Cost Estimate (1995 dollars) for the Equipment Required for Processing Cesium Chloride and Strontium Fluoride at the Waste Encapsulation and Storage Facility for Direct Blending with Neutralized Current Acid Waste and/or Complexant Concentrate Waste.

Item	Location	Quantity	Capacity	Material	Cost (\$1,000)
Chopsaw	WESF G Cell and E Cell	4	--	Carbide steel blade	69
Pulverizer	WESF E Cell	2	318-364 kg/h	Ni-hard crushing plates	35
Mixing/dissolving tank	WESF E Cell	1	0.38 m ³ (100 gal)	316 L	23
Vortex pump	WESF E Cell	1	1 HP/20 gal/min	316 L	6
Piping and installation					29
Decontamination of cells					12
Total					174

The WESF operations require five operators per 8-hour shift. There are 4 shifts per week for around-the-clock coverage, which requires a total of 20 full-time operators. In addition, 10 surveillance support personnel are needed. Therefore, total manpower required for WESF operations is 30.

Assuming that it takes 1 wk to process 48 cesium chloride capsules (estimate based on previous WESF experience), the time to process all 1,332 cesium chloride capsules will be 28 wk. However, about 40 percent downtime will be needed for activities such as general maintenance, manipulator replacement, and operator training. Therefore, the estimated total time for cesium chloride capsule processing is about 1 year. The time required to process SrF₂ capsules can be calculated in a similar way using ORNL experience as a basis. The result is 167 days to process all 601 capsules. However, since slurry transfers may incur plugging problems, this time should be increased by 25 percent. Therefore, the total time to process all 601 SrF₂ capsules will be approximately 0.71 years.

Thus, the WESF will be operating for a total of 1.71 years. Assuming a workforce cost of \$103,000/workyear, the cost of operating the WESF to process all the cesium chloride and SrF₂ capsules for blending directly into NCAW and/or CC waste at B Plant will be \$5.1 million in FY 1995 dollars.

The existing safety analysis report (SAR) for the WESF may have to be modified because disassembly and processing of capsules for cesium and strontium disposal represents a mission change. In addition to a mission change, the processing of capsules at WESF may lead to generation of gases and/or airborne contamination containing radioactive substances. The cost to do the modification to the WESF SAR is estimated at \$460,056.

Therefore, the total cost for the option of processing capsules at WESF for direct blending with NCAW and/or CC waste is \$5.6 million in FY 1995 dollars.

The incremental cost of operating B Plant and other facilities such as the 244-AR Vault and tank farms for the processing of the capsule waste is not expected to be significant compared to the total cost of the capsule disposal mission. This is especially true for B Plant because the B Plant will be in operation for the processing of NCAW and TRUEX processing of other wastes, which includes several million gallons of liquid waste. However, a vortex pump is needed to transfer SrF_2 slurry between tanks at B Plant, which will require minor modification of the B Plant. This extra cost is estimated to be \$1.2 million when considering the increase in the volume of waste after diluting the waste with water. The waste would be diluted so that a liquid heel of high radioactivity does not remain in the rather large tanks at B Plant. Another additional cost may be modification of the SAR for B Plant, 244-AR Vault, and tank farms. The cost of SAR modification is estimated to be a total of approximately \$1.4 million for all three facilities.

The capsule waste blended into NCAW and/or CC waste is estimated to increase the number of canisters produced at HWVP by a maximum of 5. At \$575,000 per canister (including HWVP operating costs), the incremental cost is \$2.88 million. Thus, the total cost for the direct blending of capsules with NCAW and/or CC waste is \$11 million in FY 1995 dollars.

C7.1.2 Pretreatment of Cesium Chloride and Strontium Fluoride Before Blending with Neutralized Current Acid Waste and/or Complexant Concentrate Waste

The procedure and equipment required to remove the cesium chloride and SrF_2 from the metal capsules are identical to those described for the option in Section C7.1. The separation of cesium chloride requires one tank for dissolving the cesium chloride and a second tank for treating the aqueous cesium chloride with AgNO_3 . A PHP filter assembly is required to separate the solid AgCl from aqueous CsNO_3 . As described in Section C6.4.2, the separation of SrF_2 is more involved and requires more equipment compared to that for cesium chloride. Initially there is treatment with H_2SO_4 to dissolve the fluoride, which results in the release of HF gas as a byproduct that should be handled with an offgas treatment system. This is followed by a treatment with aqueous Na_2CO_3 to convert the SrSO_4 to SrCO_3 . A PHP filter assembly can be used to separate solid SrCO_3 from liquid Na_2SO_4 . The final step is dissolving the SrCO_3 in 1 M HNO_3 to convert SrCO_3 into soluble $\text{Sr}(\text{NO}_3)_2$ for final transfer to B Plant. The costs of equipment and chemicals required for processing cesium chloride and SrF_2 to separate the halides at WESF are itemized in Tables C7-2 and C7-3. The cost of equipment and materials for this option is

Table C7-2. Cost Estimate (1995 dollars) for the Equipment Required to Separate the Halides from Cesium Chloride and Strontium Fluoride at the Waste Encapsulation and Storage Facility Before Blending with Neutralized Current Acid Waste and/or Complexant Concentrate Waste.^a

Item	Location	Quantity	Capacity	Material	Cost (\$000)
Chopsaw	WESF G Cell and E Cell	4	--	Carbide steel blade	69
Pulverizer	WESF E Cell	2	318-364 kg/h	Ni-hard crushing plates	35
Mixing/dissolving tank ^b	WESF E Cell	1	0.08 m ³ (20 gal)	316 L	6
		2	0.38 m ³ (100 gal)	316 L	46
Rotating furnace	WESF E Cell	1	0.08 m ³ (20 gal)	316 L	17
Offgas treatment System ^b	WESF E Cell	1	--	Monel-400	230
PHP filter system ^b	WESF D Cell	1	--	316 L	115
Piping and installation					115
Decontamination of cells					12
Total					645

^aThe difference in equipment cost between pumping the SrCO₃ slurry directly to B Plant versus dissolving in HNO₃ and pumping liquid Sr(NO₃)₂ is negligible.

^bThe same tanks, offgas system, and PHP filter system will be used to process both cesium chloride and SrF₂.

Table C7-3. Cost Estimate (1995 dollars) for the Chemicals Required to Separate the Halides from Cesium Chloride and Strontium Fluoride.

Item	Quantity	Cost (\$)
Sulfuric acid (18M) ^a	1.14 m ³ (300 gal)	690
Sodium carbonate ^a	2,120 kg	748
Nitric acid (1 M) ^a	32.18 m ³ (8,500 gal)	39,100
Potassium hydroxide (5M) ^a	9.46 m ³ (2,500 gal)	8,626
Magnesium oxide ^b	1,045 kg	840
Silver nitrate ^a	3,675 kg	533,415
Total		583,419

^aQuotation from W. H. Baddley, Chairman, Baddley Chemicals Incorporated, to Bob Hunter, Westinghouse Hanford Company (August 28, 1990).

^bQuotation from Ken Quailles, Quality Discounts, Distributor for J. T. Baker, Inc. (August 28, 1990).

\$644,000. As before, this cost also includes piping and installation and minor decontamination of cells. The cost of chemicals required for processing is \$654,000. This cost may be increased by 10 percent to cover the cost to prepare some of the solutions and to purchase chemicals in batches smaller than bulk quantities. This raises the total cost of chemicals to \$720,000.

The solid wastes generated for this option are the cut-up inner and outer capsule materials and AgCl. As described in Section C7.3, the cut-up capsule materials will be disposed of in drag-off boxes and the costs will be the same as for the direct blending option. The solid AgCl waste is considered to be mixed waste. The mixed waste should be packaged in containers recommended by the Safety Analysis Report for Packaging (SARP) for the WESF, transported according to DOT-49 CFR, and stored in the central waste complex for future retrieval. The cost for the solid AgCl waste disposal is estimated to be \$46,000. Therefore, the cost for solid waste disposal is \$276,000.

In addition to the solid wastes, the separation of the halides also results in the production of liquid wastes (Na_2SO_4 , Na_2CO_3 , and H_2CO_3). Since the volume of Na_2CO_3 in solution exceeds 10 percent, the liquid waste will be classified as mixed waste. Similar to the methods used for AgCl, the waste should be packaged in containers recommended by the SARP for the WESF, transported according to DOT-49 CFR, and stored in the central waste complex for future retrieval. The estimated cost for the liquid waste disposal is approximately \$1,219,000.

Assuming the WESF operating costs are 50 percent more than those for the direct blending option, and the costs for WESF SAR modification are the same as those for the direct blending option, the total cost for the option of processing capsules at WESF by removing the halides before blending with NCAW and/or CC waste is \$11.7 million in FY 1995 dollars.

As before (see Section C7.1), the incremental cost of operating the B Plant, 244-AR Vault, and tank farms for processing the capsule waste is assumed to be \$1.15 million. The cost of modifying the SAR to account for capsule waste processing is assumed to be the same as that for the direct blending case, viz., \$1.4 million.

The capsule waste blended into NCAW and/or CC waste is estimated to increase the number of canisters produced at HWVP by a maximum of 5. At \$575,000 per canister, the incremental cost is \$2.88 million. Thus, the total cost for the option of processing capsules by removing the halides before blending with NCAW and/or CC waste is \$17.1 million in FY 1995 dollars.

C7.1.3 Hanford Waste Vitriification Plant Campaign Specifically for Capsules

If it is impossible to blend the capsule material with either NCAW or CC waste, a special campaign could be run in the HWVP. Whether or not the halides are removed from the cesium and strontium will affect the total amount of glass produced and the resulting costs. As described in Section C4.1, if the halides are kept with the cesium and strontium, as many as TBD 133 canisters could be produced. Costs are estimated to be on the order of \$82.12 million for this case. If the halides are separated from the cesium

and strontium, approximately TBD 45 canisters would be produced at a cost of \$37.61 million. These costs are estimated based on initial WESF operations: \$5.64 million to disassemble the capsules and remove the salts and \$6.1 million for halide removal operations, followed by vitrification and disposal at \$575,000 per canister. These costs were presented in detail earlier in Section C7.0.

C7.1.4 Hanford Waste Vitrification Plant Modifications

The cost for modifying a laydown area and obtaining the equipment described in Section C6.3 is estimated to be \$10.12 million. This cost includes the engineering required to modify the current design, and reflects the costs for modifying the laydown area on the east side of the facility. Costs for modifying the laydown area on the west side would be greater because of the need for additional piping runs and other engineered features in this more complex area. The west laydown area may be preferable because of the master-slave manipulator transport paths within the vitrification facility. Modification of one of the laydown areas is not feasible without providing the laydown area space elsewhere in the facility. If the vitrification building is lengthened to provide this space, the incremental cost per foot is \$1.4 million. If the process cell layout is modified to provide this space without increasing the footprint, the additional cost will be at least \$4.6 million. Additional operating costs will be incurred if this is implemented. Life cycle cost impacts would have to be evaluated to determine the overall cost impact. The minimum cost for modifying the HWVP to vitrify the capsule waste is estimated at \$15 million.

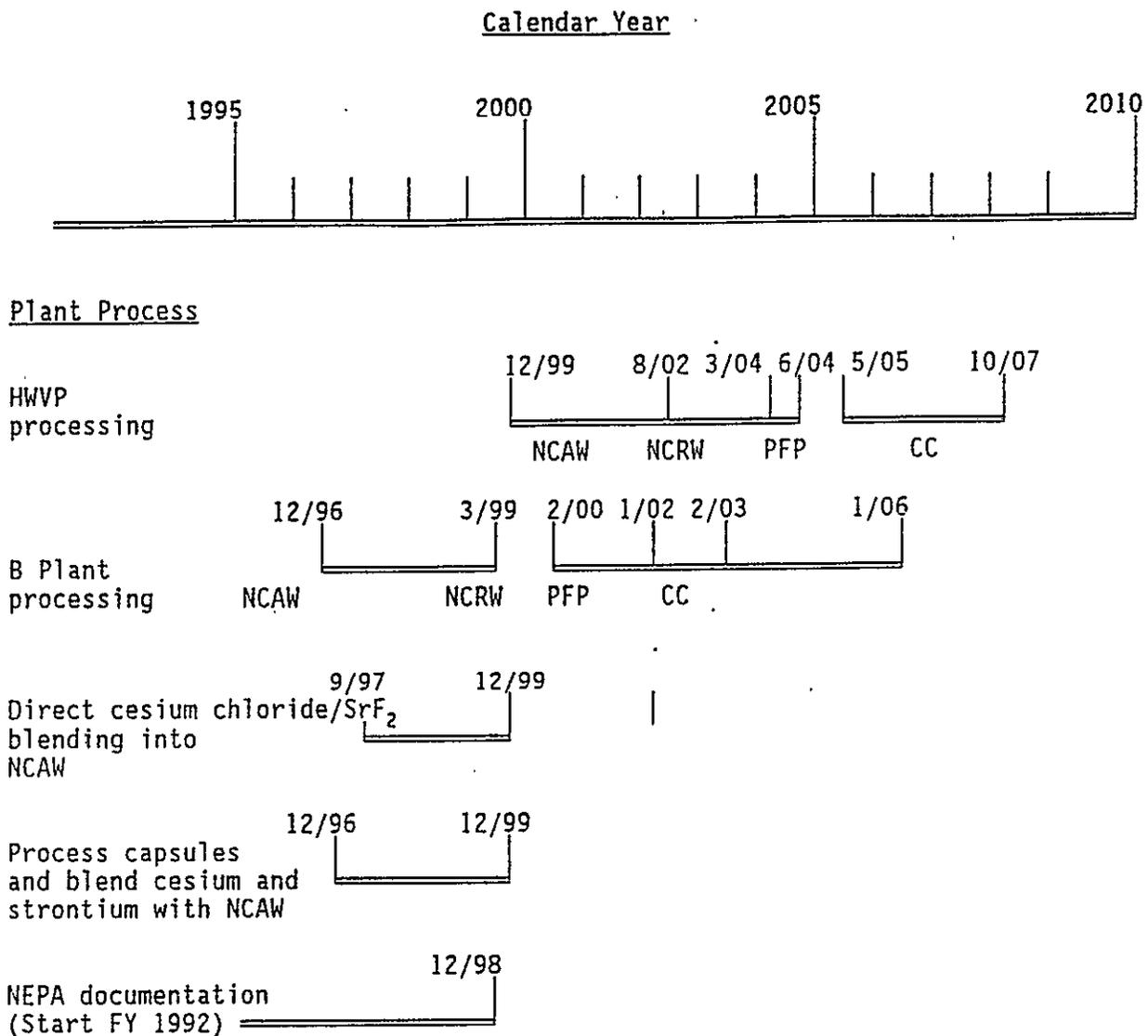
Three options exist for subsequent vitrification operations. The low-cost option is to blend the capsule contents with pretreated NCAW and/or CC waste as it is brought into the plant. This option will result in an estimated increase of five canisters and a total cost of \$23.5 million, which includes costs for the previously discussed HWVP modifications and activities to remove the salts from the capsules (\$5.6 million). Salt removal costs are considered here to be the same in HWVP as in WESF. The second option is to include the separation of the halides from the cesium and strontium in the flowsheet, followed by vitrification of the cesium and strontium in a separate campaign. Approximately 45 canisters would be produced, based on the concentration limit imposed by strontium (1.6 wt percent SrO) on the glass. Total process costs are estimated to be \$52.5 million. The third and most costly option is to vitrify the capsule material as a separate HWVP campaign without first separating the halides. Approximately 133 canisters would be produced at a total estimated cost of \$97.2 million.

C8.0 SCHEDULE

The schedule for performing the two options of blending the capsules with NCAW and/or CC waste is given in Figure C8-1. The schedule includes the time required for completing activities such as preparing the WESF cells, ordering equipment, installing equipment, ordering chemicals, and modifying the SAR. It should be pointed out that some of these activities will be carried out simultaneously.

Also illustrated in Figure C8-1 are the schedules for processing NCAW, neutralized cladding removal waste (NCRW), Plutonium Finishing Plant (PFP) waste, and CC waste in B Plant and HWVP. As shown by the figure, the schedule for processing capsule waste coincides with the processing of NCAW in the B Plant and the startup of HWVP operations with NCAW processing. The present schedule for the blending of capsule waste with NCAW assumes return of the capsules to WESF from organizations leasing some of the capsules, and preparation of any required *National Environmental Policy Act of 1969* (NEPA) documentation occurring before December 1996. This is an aggressive schedule that is driven by the HWVP schedule. To maximize the opportunity to blend the capsule waste with the planned HWVP feed streams, a decision to vitrify the capsule waste must be made soon. The engineering study will be the basis for making this decision.

Figure C8-1. Schedule for Blending the Capsules with Neutralized Current Acid Waste.



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APPENDIX C

ATTACHMENT 1

REQUIREMENTS FOR REPOSITORY DISPOSAL OF
DEFENSE HIGH-LEVEL WASTE

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Both commercial spent fuel and defense high-level wastes have been identified by the Nuclear Waste Policy Act as materials to be disposed of in a geologic repository. Chapter 10, Part 60 of the Code of Federal Regulations (10 CFR 60) identifies specific requirements for the characteristics of all wastes to be disposed of in a geologic repository. The Waste Acceptance Preliminary Specifications (WAPS) for the Defense Waste Processing Facility (DWPF) have been developed to provide guidance to the DWPF as to the legally required characteristics of the waste form, and the required characteristics to ensure that the DWPF product can be safely handled at the repository.

The WAPS contain 21 individual specifications. Of these, 12 are directly mandated by 10 CFR 60 for both spent fuel and defense high-level waste. The titles of the specifications and the applicable subpart citation from 10 CFR 60 are shown below.

1.4	Chemical and Phase Stability	135(a)(2)
2.1	Material	135(a)(1)
2.2	Fabrication and Closure	135(c)(1)
2.3	Identification and Labeling	135(b)(4)
3.1	Free Liquid	135(b)(2)
3.2	Gas	135(b)(2)
3.3	Explosiveness, Pyrophoricity, Combustibility	135(b)(1),(c)(3)
3.4	Organic Material	135(a)(1)
3.9	Chemical Compatibility	135(a)(1)
3.10	Subcriticality	131(b)(7)
3.12	Drop Test	134(c)(1),(c)(2)
4.0	Quality Assurance	150

The remaining nine specifications are not directly mandated by federal law, but are required by the candidate repository project as a part of the repository performance assessment evaluations, or to ensure that the remote handling capability at the repository will be compatible with the waste form. A discussion of these specifications is provided below.

Specifications 1.1 (Chemical) and 1.2 (Radionuclide Inventory) are required by the repository project as a part of the performance assessment. The repository is required by 10 CFR 60 Subparts 113 and 134(a)(2) to account for all chemical and radiochemical species contained by the repository and to ensure that none of these species impair the repository waste isolation capability. Additionally, the repository must maintain an accounting of all radionuclides contained in the repository in order to comply with accountability requirements implicit in 10 CFR 60 Subpart (a)(ii)(B). To achieve this, the repository project requires that documentation be provided for the radionuclide content of all wastes. For spent fuel this will be achieved by ORIGEN calculations using fuel manufacturing records, burnup

records, and storage records. Such records exist for virtually all fuels in the United States, except for the very oldest. Defense high-level waste is required to meet the same accountability requirements as spent fuel, for the same reasons.

Compliance with these requirements is more problematic for defense high-level waste producers because the waste materials are derived from more complicated chemical processes (reprocessing), blended so that all discrete identity is lost (storage) and then remanufactured into a completely new waste form (vitrification, for DWPF and Hanford Waste Vitrification Plant [HWVP]). Nonetheless, the same requirements apply to spent fuel as for defense high-level waste, even though compliance techniques will differ.

Specification 1.3 (Radionuclide Release), as currently written, is used to characterize the intrinsic interactions of the waste form with hydrothermal environments. Spent fuel undergoes hydrothermal radionuclide release testing as a part of the repository performance assessment studies. Defense high-level waste will undergo similar testing for the same reasons. Since defense high-level waste has greater potential for inhomogeneity than does spent fuel, the high-level waste producers must conduct testing on a wider base of compositions, reflecting the greater variability of the product. Although the required data for radionuclide release is greater for defense high-level waste than the spent fuel, the need for such data is applied equally for the two waste forms.

Specification 3.5 (Free Volume) was promulgated by the Basalt Waste Isolation Project (BWIP). The BWIP required that the contents of the waste package (the waste form) provide support for the waste package itself against the hydrostatic head of the repository. For spent fuel, this support will be provided by the arrangement of the fuel rods within the waste package. For defense high-level waste, this will be provided by the vitrified waste form in an essentially completely full canister. Although the need for structural support does not equate to a "free volume" specification for spent fuel, the need for this characteristic is common between the two waste forms.

Specification 3.6 (Removable Radioactive Contamination) has no corresponding constraint for spent fuel. The repository project will use separate surface handling facilities for spent fuel and defense high-level wastes. The project wishes to use contact maintenance procedures for defense waste handling equipment, but realizes that this cannot be achieved for spent fuel.

Specification 3.7 (Heat Generation) and 3.8 (Maximum Dose Rates) correspond to the spent fuel acceptance specification that the repository must be capable of accepting 5-year-young spent fuel of 33,000 MWD/MTU burnup (Generic Requirements for a Mined Geologic Disposal System). This spent fuel specification essentially defines an upper limit for heat generation and dose rate. Similarly, such specifications have been established for defense high-level waste.

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- Specification 3.11 (Dimensions) and 3.13 (Handling Features) are established to ensure that the defense high-level waste forms can be accommodated by the repository handling equipment. Similar handling envelopes have been defined for spent fuel.

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APPENDIX C

ATTACHMENT 2

EFFECTS OF CESIUM CHLORIDE AND STRONTIUM FLUORIDE
CAPSULE ADDITION TO NEUTRALIZED CURRENT ACID WASTE GLASS

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Table 1. Impact of Cesium Chloride and Strontium Fluoride Capsules Added to Neutralized Current Acid Waste/Hanford Waste Vitrification Plant Glass. (sheet 1 of 2)

Number of canisters of glass from NCAW processing	480 canisters.	per Bob Watrous (telecon 6/1/90)
wt% Cs ₂ O in NCAW glass (reference case)	0.15 wt%	HWVP TDP
kg of glass per canister	1,650 kg	HWVP TDP
Total quantity of glass	792,000 kg	Calculated
Quantity of Cs ₂ O	1,188 kg	Calculated
Quantity of cesium in NCAW glass canisters	1,121 kg	Calculated
Number of capsules of cesium chloride	1,349 capsules	WHC-EP-0195
Quantity of cesium chloride per capsule	2.7 kg	Per G. Tingey (telecon w/ R. D. Peters 6/1/90)
Total quantity of cesium chloride	3,642 kg	Calculated
Quantity of cesium in capsules	2,875 kg	Calculated
Quantity of cesium in NCAW glass with capsules added	3,996 kg	Calculated
wt% Cs ₂ O in NCAW glass with capsules added	0.53 wt%	Calculated
wt% SrO in NCAW glass (reference case)	0.1 wt%	HWVP TDP
Total quantity of SrO	792 kg	Calculated
Quantity of strontium in NCAW glass canisters	670 kg	Calculated
Number of capsules of SrF ₂	636 capsules	WHC-EP-0195
Quantity of SrF ₂ per capsule	2.7 kg	Per G. Tingey (telecon w/ R. D. Peters 6/1/90)
Total quantity of SrF ₂	1,717 kg	Calculated
Quantity of strontium in capsules	1,198 kg	Calculated
Quantity of strontium in NCAW glass with capsules added	1,867 kg	Calculated

Table 1. Impact of Cesium Chloride and Strontium Fluoride Capsules Added to Neutralized Current Acid Waste/Hanford Waste Vitrification Plant Glass. (sheet 2 of 2)

wt% SrO in NCAW glass with capsules added	0.28 wt%	Calculated
wt% chlorine in NCAW glass (reference case)	0.075 wt%	HWVP TDP
kg of glass per canister	1,650 kg	HWVP TDP
Total quantity of glass	792,000 kg	Calculated
Quantity of chlorine in NCAW glass canisters	594 kg	Calculated
Number of capsules of cesium chloride	1,349 capsules	WHC-EP-0195
Quantity of cesium chloride per capsule	2.7 kg	Per G. Tingey (telecon w/ R. D. Peters 6/1/90)
Total quantity of cesium chloride	3,642 kg	Calculated
Quantity of chlorine in capsules	767 kg	Calculated
Quantity of chlorine in NCAW glass with capsules added	1,361 kg	Calculated
wt% chlorine in NCAW glass with capsules added	0.17 wt%	Calculated
wt% fluorine in NCAW glass (reference case)	0.3 wt%	HWVP TDP
Total quantity of fluorine	2,376 kg	Calculated
Number of capsules of SrF ₂	636 capsules	WHC-EP-0195
Quantity of SrF ₂ per capsule	2.7 kg	Per G. Tingey (telecon w/ R. D. Peters 6/1/90)
Total quantity of SrF ₂	1,717 kg	Calculated
Quantity of fluorine in capsules	19 kg	Calculated
Quantity of fluorine in NCAW glass with capsules added	2,895 kg	Calculated
wt% fluorine in NCAW glass with capsules added	0.37 wt%	Calculated

Table 2. Impact of Curie and Watt Increases on Neutralized Current Acid Waste/Hanford Waste Vitrification Plant Glass.

1. Current NCAW/HWVP glass curie and watt contents (Source: WHC 1989).				
Case	Curies (Ci/gal)	Curies per canister	Watts (W/gal)	Watts per canister
Nominal	38.6	134,729	0.109	380
Maximum	84.22	93,890	0.24	838

Input data	
Gallons NCAW/canister	3,490 gal
lb waste oxides per gal NCAW	0.26 lb
kg glass per canister	1,650 kg
Reference glass waste loading	25%
Total number of NCAW canisters	480
Assumed time of Ci and W estimate	1998-2002

2. Approximate curie and watt content of cesium chloride and SrF ₂ capsules (Source: IDB: DOE/RW-0006, REV. 5).				
Case	Total curies	Total curies	Total watts	Total watts
Year	1/1/95	1/1/10	1/1/95	1/1/10
cesium chloride	90,000,000	63,000,000	222,128	157,198
SrF ₂	51,000,000	36,000,000	173,584	123,074
Number of cesium chloride capsules				1,349
Number of SrF ₂ capsules				636

3. Estimated NCAW/HWVP glass curie and watt contents following blending of cesium chloride and SrF ₂ capsules with NCAW.				
Case	Curies/can	Watts/can	Curies/can	Watts/can
Date	1/1/10	1/1/10	1/1/00	1/1/00
Nominal	<341,000	<964	267,440	1,066
Maximum	<500,000	<1,422	554,690	1,524

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APPENDIX C

ATTACHMENT 3

EFFECTS OF CESIUM CHLORIDE AND STRONTIUM FLUORIDE
CAPSULE ADDITION TO COMPLEXANT CONCENTRATE GLASS

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Table 1. Impact of Cesium Chloride and Strontium Fluoride Capsules Added to Complexant Concentrate/Hanford Waste Vitrification Plant Glass. (sheet 1 of 2)

Number of canisters of glass from CC processing	650 canisters	HWVP TDP
wt% Cs ₂ O in CC glass (reference case)	0.12 wt%	HWVP-87-V110201A
kg of glass per canister	1,650 kg	HWVP TDP
Total quantity of glass	1,072,500 kg	Calculated
Quantity of Cs ₂ O	1,287 kg	Calculated
Quantity of Cs in CC glass canisters	1,214 kg	Calculated
Number of capsules of cesium chloride	1,349 capsules	WHC-EP-0195
Quantity of cesium chloride per capsule	2.7 kg	Per G. Tingey (telecon w/ R. D. Peters 6/1/90)
Total quantity of cesium chloride	3,642 kg	Calculated
Quantity of Cs in capsules	2,875 kg	Calculated
Quantity of Cs in CC glass with capsules added	4,089 kg	Calculated
wt% Cs ₂ O in CC glass with capsules added	0.40 wt%	Calculated
wt% SrO in CC glass (reference case)	0 wt%	HWVP-87-V110201A
Total quantity of SrO	0 kg	Calculated
Quantity of strontium in CC glass canisters	0 kg	Calculated
Number of capsules of SrF ₂	636 capsules	WHC-EP-0195
Quantity of SrF ₂ per capsule	2.7 kg	Per G. Tingey (telecon w/ R. D. Peters 6/1/90)
Total quantity of SrF ₂	1,717 kg	Calculated
Quantity of strontium in capsules	1,198 kg	Calculated
Quantity of strontium in CC glass with capsules added	1,198 kg	Calculated

Table 1. Impact of Cesium Chloride and Strontium Fluoride Capsules Added to Complexant Concentrate/Hanford Waste Vitrification Plant Glass. (sheet 2 of 2)

wt% SrO in CC glass with capsules added	0.13 wt%	Calculated
wt% Cl in CC glass (reference case)	0.25 wt%	HWVP-87-V110201A
kg of glass per canister	1,650 kg	HWVP TDP
Total quantity of glass	792,000 kg	Calculated
Quantity of chlorine in CC glass canisters	1,980 kg	Calculated
Number of capsules of cesium chloride	1,349 capsules	WHC-EP-0195
Quantity of cesium chloride per capsule	2.7 kg	Per G. Tingey (telecon w/R. D. Peters 6/1/90)
Total quantity of cesium chloride	3,642 kg	Calculated
Quantity of chlorine in capsules	767 kg	Calculated
Quantity of chlorine in CC glass with capsules added	2,747 kg	Calculated
wt% chlorine in CC glass with capsules added	0.35 wt%	Calculated
wt% fluorine in CC glass (reference case)	0.05 wt%	HWVP-87-V110201A
Total quantity of fluorine	396 kg	Calculated
Number of capsules of SrF ₂	636 capsules	WHC-EP-0195
Quantity of SrF ₂ per capsule	2.7 kg	Per G. Tingey (telecon w/R. D. Peters 6/1/90)
Total quantity of SrF ₂	1,717 kg	Calculated
Quantity of fluorine in capsules	519 kg	Calculated
Quantity of fluorine in CC glass with capsules added	915 kg	Calculated
wt% fluorine in CC glass with capsules added	0.12 wt%	Calculated

Table 2. Impact of Curie and Watt Increases on Complexant Concentrate/Hanford Waste Vitrification Plant Glass.

1. Current CC/HWVP glass curie and watt contents (Source: DOE RW-0184, volume 1 of 6, Table C-3.4.5).		
	Curies per canister	Watts per canister
	230	1

kg glass per canister	1,650 kg
Reference glass waste loading	25%
Total number of CC canisters	650 Canisters
Assumed time of Ci and watt estimate	1998-2002

2. Approximate curie and watt content of cesium chloride and SrF ₂ capsules (Source: IDB: DOE/RW-0006, REV. 5).				
Case	Total curies	Total curies	Total watts	Total watts
Year	1/1/95	1/1/10	1/1/95	1/1/10
cesium chloride	90,000,000	63,000,000	222,128	157,198
SrF ₂	51,000,000	36,000,000	173,584	123,074
Number of cesium chloride capsules				1,349
Number of SrF ₂ capsules				636

3. Estimated CC/HWVP glass curie and watt contents following blending of cesium chloride and SrF ₂ capsules with CC.				
Case	Curies/can	Watts/can	Curies/can	Watts/can
Year	1/1/10	1/1/10	1/1/95	1/1/95
Nominal	152,538	431	217,153	609

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