

SCOPING STUDIES TO REDUCE ICPP HIGH-LEVEL
RADIOACTIVE WASTE VOLUMES FOR FINAL DISPOSAL^a

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ABSTRACT

The Defense Waste Management Plan (DWMP) of June 1983 presents a reference plan for the long-term management of Idaho Chemical Processing Plant (ICPP) high-level wastes (HLW) with a goal of disposing the annual output in 500 canisters per year after FY 2007. Based on the available technology to immobilize ICPP HLW into a glass, at least 1700 canisters could be produced per year after FY 2007. Because of potentially high disposal costs, fuel dissolution, waste treatment, and waste immobilization were evaluated as possible processes to reduce immobilized high-level waste volumes. The results of these scoping studies show that modifications in the presently used fuel dissolution processes to reduce usage of inert materials could result in up to a 72% potential volume reduction when combined with a ceramic waste form. The ceramic-based waste form appears to be feasible with a potential HLW volume reduction of 60%. The neutralization process also appears to be feasible with a potential volume reduction of 80%, but may require significant development. The low-volume glass and Modified Zirflex processes may be feasible but would require significantly more process development and/or modifications and could result in only a 20 to 30% potential volume reduction; thus, they are not recommended for further development.

INTRODUCTION

The Defense Waste Management Plan (DWMP) of June, 1983, presents the President's plan for disposal of defense high-level wastes (HLW), including those generated and stored at the Idaho National Engineering Laboratory (INEL).¹ The DWMP presents a reference plan for the Idaho Chemical Processing Plant (ICPP) at the INEL, in which the annual output of high-level liquid waste (HLLW) is immobilized in 500 canisters per year and sent to a repository after FY 2007, while stored calcine is immobilized and disposed of as plant capacity permits.¹ A decision on a reference process for long-term ICPP HLW management will not be made until the early 1990s.

Based on the estimated cost of about \$95,000 per canister,² transportation and disposal of ICPP HLW could cost \$160 million per year for annually produced waste and \$95 million per year for the stored waste. Because of these potentially high disposal costs, fuel dissolution, waste treatment, and waste immobilization processes have been identified as options to reduce immobilized high-level waste volumes for repository disposal. This paper presents the results of scoping studies carried out on candidate options to reduce high-level waste volumes.

Several types of nuclear fuel are processed at ICPP, resulting in complex chemical waste compositions.³ The acidic high-level liquid waste (HLLW) includes aluminum nitrate, zirconium fluoride and sodium-bearing waste,^{3,4} all of which are solidified as granular calcined waste with compositions shown in Table I and stored in engineered storage facilities.

From October, 1963, through September, 1985, approximately 20,000 m³ of HLLW has been converted into approximately 3,000 m³ of granular calcined solids.⁵ ICPP wastes have lower fission product concentrations in the total solids content than do potentially generated commercial nuclear waste and can be stored as calcine in large bins at practical temperatures.⁴ The calcine storage bins are designed for a maximum storage temperature of 600-700°C to avoid caking or sintering of the calcine.

TABLE I

Composition of ICPP Calcines

Component	Type of Calcine and Composition, wt%		
	Alumina	Zirconia	Base-Case Fluorinel
Al ₂ O ₃	82-95	13-17	5.5
Na ₂ O	1-3	---	---
ZrO ₂	---	21-27	23.2
CaF ₂	---	50-56	56.5
CaO	---	2-4	4.1
NO ₃	5-9	0.5-2	0.5-2
B ₂ O ₃	0.5-2	3-4	4.4
CdO	---	---	6.0
Miscellaneous Fission Products and Actinides	0.5-1.5	0.5-1.5	0.5-1.5
	<1	<1	<1

The amounts of ICPP HLW projected to be generated through FY 2032 are shown in Fig. 1. The average annual rate of waste production rises until the late twentieth century because of a projected four-fold increase of fuel reprocessing throughput rate by FY 2010. Using existing technology, a glass waste form containing 33 wt% calcine⁶ would fill approximately 1700 canisters/yr (0.63 m³/canister) at the projected FY-2008 annual production rate.

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The unique nature of fuels processed at ICPP³ and the process chemicals required to dissolve the fuels results in a large amount of inert material in the waste calcine.⁴ In the future fuel dissolution process, zirconium is present from dissolution of the

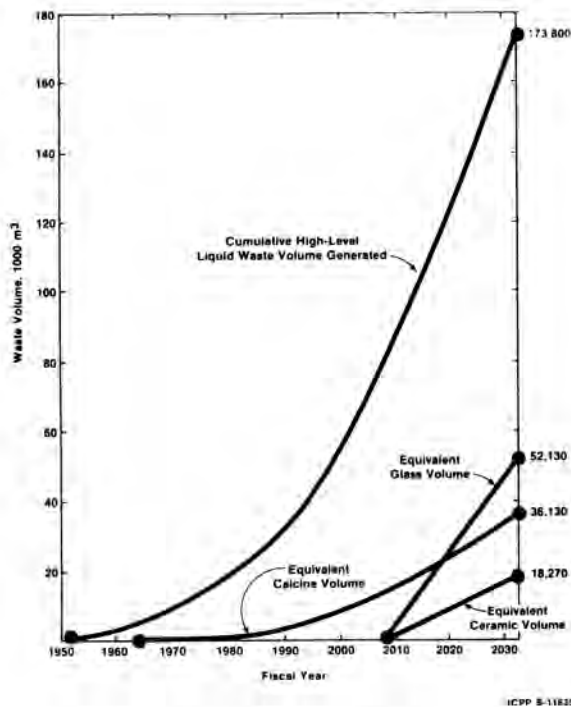


Fig. 1. Projected Accumulated HLW Volume at ICPP from FY 1951 to 2032.

fuel in a hydrofluoric-nitric acid solution containing cadmium and boron as neutron poisons. Aluminum is added to complex the fluoride in order to prevent corrosion during extraction and HLLW storage prior to solidifying the waste as a calcine. Calcium is added in excess to react with the fluoride to form an inert material, CaF_2 . The fluoride required for fuel dissolution plus additives required to complex fluorides to prevent process corrosion accounts for 64 wt% of the calcined waste as CaF_2 , CaO , and Al_2O_3 . Neutron poison requirements account for 10 wt% of the calcine as CdO and B_2O_3 , and zirconium accounts for 23 wt% of the calcine as ZrO_2 . Other materials account for the remaining 3 wt%, with fission products and actinides comprising less than 1 wt% of the calcine.

The scoping studies were conducted to evaluate feasibility of the following products and processes:

1. Modified Fluorinel fuel dissolution process
 - o Laboratory, pilot-plant, and production-scale tests to determine acceptable fluoride limits during dissolution
 - o Evaluation of conceptual waste volume reduction methods to reduce requirements for B and Cd through improved criticality control.
2. The ceramic-based waste immobilization process using a hot isostatic press (HIP)

- o Waste form development using the HIP
 - o Ceramic product characterization for phase composition and durability
3. The process to remove inert material from the HLLW by neutralization
 - o Nonradioactive and radioactive laboratory-scale neutralization studies
 - o Evaluations of conceptual process flowsheet
 4. The Modified Zirflex fuel dissolution process with HF and recyclable NH_4F
 - o Zircaloy and unirradiated fuel laboratory-scale dissolution studies
 - o Precipitation and separation of dissolved fuel
 5. The low-fluoride, low-volume glass process
 - o Low-fluoride glass frit development and
 - o Fluoride volatility and recovery from HLLW off-gas.

RESULTS AND CONCLUSIONS OF SCOPING STUDIES TO REDUCE ICPP HLLW VOLUME

Modified Fuel Dissolution Process to Reduce HF Usage

A potential reduction of hydrofluoric acid (HF) during dissolution by 15% may reduce HLLW volume by about 10%. If this reduction can be implemented, it could be operated for 20 years before any other option is ready and it would result in less calcine stored in the bins when a new immobilization technology is implemented in 2007.

Further volume reduction could also be achieved if the amounts of Cd and B used for criticality control could be reduced in the dissolution process during the mid-FY 1990s when a process modification is scheduled. Such a modification could nearly eliminate the HLLW volumes associated with the neutron poisons, B and Cd, and the additional volume of HF normally complexed by the B. If critically safe dissolvers could be used to eliminate the need for Cd and most of the B, a 72% reduction in immobilized ceramic waste volume compared with base-case glass could potentially be realized.

Dissolution tests were carried out as part of the start-up of the new dissolution plant to verify the process for uranium recovery and dissolution characteristics. Laboratory tests with unirradiated and irradiated fuel and pilot-plant and production plant tests with unirradiated fuel show that the dissolution with a lower amount of HF will still result in the same uranium recovery as with the current flowsheet. Further tests will be run to evaluate more corrosive dissolvents which could be used in future vessels constructed with more corrosion-resistant alloys and possibly further reduce HLLW volumes. The modification to reduce HF usage by 15% and the resulting calcine volume by 10% appears feasible and will be implemented if start-up tests indicate that no problems exist with the process.

Methods of reducing the amounts of Cd and B for criticality control, such as use of enriched boron-10 or critically-safe dissolvers, need to be examined, tested, and operated to determine actual practicality of their use. Until such tests, the projected upper limit of volume reduction by eliminating Cd and B of 75% for a ceramic waste product is largely conceptual.

Thus, based on the laboratory tests, the reduction in HF usage should be implemented in the new dissolution process to achieve about a 10% reduction in waste volume. If operationally feasible, reduced Cd and B usage should be evaluated further.

Ceramic Waste Immobilization Processes

Ceramic products are crystalline-phase assemblages with the same or similar phases as natural minerals that have endured geologic conditions for millions of years, often under high temperatures and high pressures. Ceramic products were first proposed by Hatch⁷ to immobilize HLW and further developed by others, including Ringwood⁸ and McCarthy.⁹ Recently, ceramic forms with high waste loading of 50-70 wt%¹⁰ were developed for defense wastes. Ceramic waste forms appear potentially capable of immobilizing ICPP HLW at reduced volume and with good product durability for permanent isolation. Because of the presence of alkali elements in the waste, the ceramic products may have to include an amorphous phase with product durability similar to that of HLW glass.

A number of candidate ceramic phases have been formed by hot isostatic pressing (HIPing) and evaluated for ICPP wastes, which show that ceramic waste forms can have leach resistance comparable to that of glass, but permit much higher waste loadings and have higher densities than the standard borosilicate glass waste forms.^{11,12} Ceramic waste forms also appear promising because they may require no modifications to current calcining operations at ICPP to prepare the product. These high waste loadings potentially make it possible to reduce the ICPP waste volume and the overall cost of disposing of HLW.

Preliminary tests have shown that simulated ICPP HLW can be formed into ceramic or glass-ceramic forms with waste loadings of at least 70 wt% at densities ranging from 3.1 to 3.4 g/cm³. Such a product can potentially reduce ICPP HLW volume by up to 60%. A conceptual flowsheet for the ceramic process is shown in Fig. 2.

Based on initial studies summarized in Table II, all forms produce a significant volume reduction of up to 60% over base-case glass by achieving a high waste loading and density. The high-waste loading Frit 127 and Si-Li, Si-Ti-Ca, and Y-Si ceramic forms shown in Table II all have a volume reduction near 60%, give promising leaching results, and appear to be feasible based on initial data. The pyrochlore ceramic could not be loaded at as high a waste loading as could the other forms and thus does not appear as promising for volume reduction.

Based on the preliminary laboratory work to date, the glass-ceramic and ceramic waste forms appear feasible to produce high-waste loading solids which can considerably reduce waste volumes compared with the base-case glass.

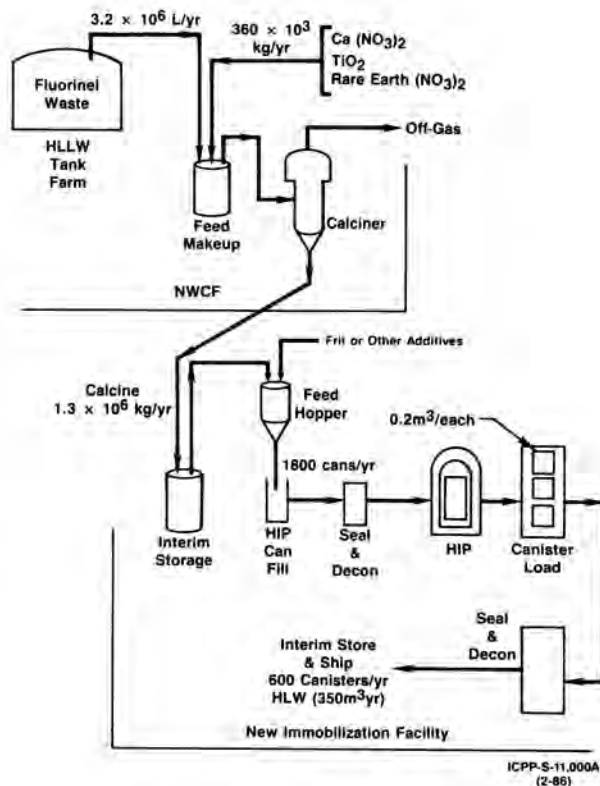


Fig. 2. Simplified Conceptual Flowsheet of the Ceramic Immobilization Process.

Studies are required to determine the effects of glass-forming additives by further investigating the waste loading and performance available with silica-lithia or other candidate frits. The loading and performance obtained using silica-calcia compared with silica-yttria additives also need to be determined.

TABLE II

Summary Characteristics of Candidate Ceramic-Based HLW Forms

Waste Type	Waste Loading, wt%	Density, g/cm ³	% Volume Reduction, from Base-Case Glass	Product Durability, 28-day MCC-1a Leaching, g/m ² -d		
				Si	Sr	Cs
Frit 127	70	3.1	55	0.2	0.2	0.1
Frit 127	80	3.2	62	0.3	0.2	0.2
Si-Li	85	3.2	65	0.7	0.3	0.7
Si-Ti-Ca	60	3.5	54	0.8	1.7	2.7
Si-Ti-Ca	70	3.5	61	0.5	2.1	1.6
Si-Ti-Ca	80	3.5	65	0.7	0.8	1.2
Y-Si	60	3.4	53	0.3	0.3	0.4
Y-Si	72	3.4	61	0.2	<0.01	0.3 ^b
Y-Si	80	3.4	65	0.7	0.02	0.8 ^b
Pyrochlore	53	3.5	48	0.1	0.1	0.7

a. MCC-1 leach test conducted at 90°C in deionized water and described in Ref. 13

b. Na results; Cs not analyzed

Processing and product characteristics need to be better established with ICPP wastes.

The developmental work should identify reference formulations for the ceramic products, which must be characterized with respect to leaching behavior and characteristics of the leached surface.

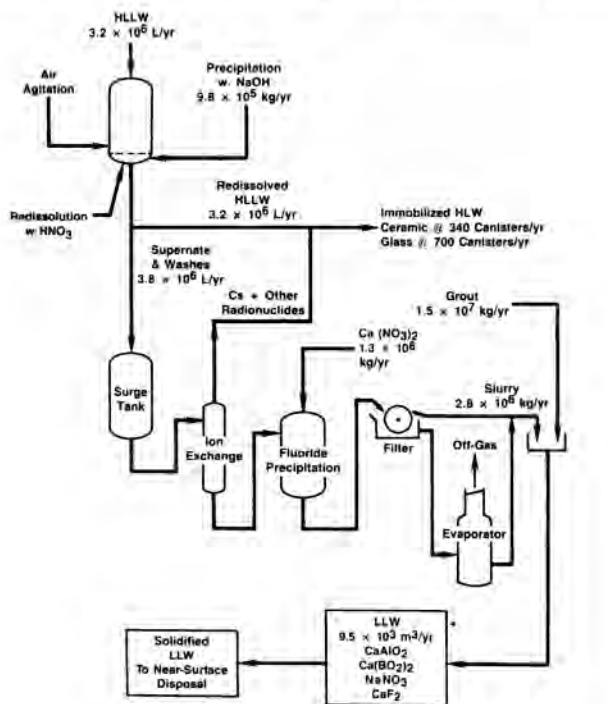
Required development work to establish product formation characteristics include:

- o Determination of required redox controls during product densification
- o Establishment of temperature--pressure consolidation versus composition curves.
- o Determination of effects of residual anions such as nitrates in the calcines on product formation and durability, and
- o Development work to identify processing stages and required equipment.

Based on experimental results, the ceramic process has a moderately high feasibility, and the products have a high potential to minimize immobilized waste volume. Thus, this technology should be developed further.

Process for the Removal of Inerts by Neutralization of HLLW

Inerts removal from the HLLW by neutralization would remove most of the fluoride as a salt, aluminum as an aluminate, and probably most of the boron as a fluoborate. In the conceptual flowsheet shown in Fig. 3, the acidic HLLW is neutralized with NaOH (or KOH not shown in Fig. 3) and then adjusted to pH 13.5.



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Fig. 3. Simplified Conceptual Flowsheet for Inerts Removal by Neutralization.

The resulting HLW precipitate contains most of the radioactivity plus the zirconium, cadmium, and iron. In the process concept, the precipitate is washed and redissolved in nitric acid prior to calcination and immobilization as a ceramic-based waste form. A glass waste form could conceivably also be used, although a ceramic would provide for a larger volume reduction. The supernatant salt solution containing Na, F, Al, B, Cs and other radionuclides is decontaminated of radionuclides by ion exchange to produce a low-level waste.

The recovered radioisotopes are recycled to the HLW fraction. The decontaminated supernatant is then reacted with calcium nitrate to form a nonhazardous calcium fluoride, calcium borate, and calcium aluminate precipitate and sodium nitrate supernate. After concentration, the solids and liquid are incorporated in a grout for final disposal as a low-level waste (LLW). Based on preliminary tests, a HLW volume reduction of up to 80 % with a ceramic waste form at 50 wt% waste loading compared with the base-case glass is possible, with approximately 9500 m³ of LLW concrete generated per year.

The neutralization process is being used for the Savannah River¹ and Hanford HLW. Preliminary experiments were run at ICPP for simulated and radioactive wastes using NaOH or KOH. The KOH could reduce the initial supernate volume because of the 16-fold higher molar solubility of the resulting KF salt compared with the solubility of NaF. The results of the test indicated that:

- o The Zr, Cd, Cr, and most of the fission products and actinides were precipitated with NaOH at pH 13.5.
- o Most of the Al, B, F, and Cs remained in the supernate salt solution.
- o Based on the laboratory-scale tests, inerts removal after neutralization could potentially reduce the immobilized ICPP HLW volume by up to 80% for a ceramic waste form at 50 wt% loading.
- o A preliminary conceptual flowsheet has been proposed specifying approximate amounts of materials and types of processing hardware required for the process.

The neutralization process appears feasible based on preliminary laboratory studies at ICPP and experience at the Savannah River and Hanford DOE sites. The ICPP studies have dealt with the neutralization step and resulting distribution of inert and radioactive components in the resulting precipitate and supernate phases, but not with the decontamination operations required to produce a LLW from the supernate. Feasibility of specific neutralization processing steps and required operating hardware has not yet been assessed for ICPP HLW, but is established for Savannah River HLW. Feasibilities of immobilized glass or ceramic waste forms have not yet been demonstrated with the precipitate fraction and could require significant development because of the relatively high level of Cd present in the HLW fraction.

Based on the evaluations to date and on the potentially favorable impact on ICPP immobilized HLW volumes, the neutralization process, as well as other inerts removal processes such as actinide removal, should be investigated further. Additional work should be carried out in the following areas:

- o Further neutralization tests should be made using both simulated and radioactive waste solutions to compare the use of KOH with NaOH and to assess the impact of a more soluble salt, KF on the distribution of radioactive and nonradioactive elements in the supernate and precipitate.
- o Neutralization tests should be carried out at different pH values from 8 to 13.5 to assess the effect of pH on the distribution of radionuclide and inert elements in the supernate and precipitate.
- o Settling and filtration or other separation of the precipitate from the supernate should be tested and scaled up in order to evaluate the feasibility of candidate methods. The precipitate resulting from neutralization appears to be gelatinous and to require long filtration times based on initial tests.
- o Literature evaluations and scoping tests should be made on supernate solution decontamination to remove Cs and other trace radioisotopes as necessary. Limitations which might be imposed by the presence of K on Cs removal should be determined in order to also determine the overall feasibility of using KOH for the initial precipitation and separation.
- o Tests on processing of the HLW precipitate, including redissolution, calcination, and formulation of ceramic and glass waste forms should be investigated further. Development of new formulations may be required. Potential problems may exist in producing a glass with desired viscosity at a practical temperature and waste loading. For some of the processes such as calcination, modifications for potentially higher heat removal requirements should be evaluated.
- o Further processing of the decontaminated supernate to produce nonhazardous CaF_2 from the hazardous NaF or KF should be evaluated. Formulations for making acceptable LLW forms for onsite disposal and LLW processing and disposal methods should be evaluated. Fluorides and boron can retard the setting of grout, and the suitability of on-site disposal should be evaluated.

Thus, based on the initial experiments, the neutralization process is potentially feasible for chemically removing most of the radionuclides as a precipitate, while separating mainly nonradioactive materials into the salt solutions. Feasibility of decontaminating the salt solution and of further immobilizing it in a grout needs to be evaluated. Immobilization of the HLW precipitate in a glass or ceramic needs to be developed.

Modified Zirflex Fuel Dissolution Process

Based on laboratory-scale studies, the conceptual Modified Zirflex fuel dissolution process flowsheet shown in Fig. 4 consists of dissolution of the spent fuel in boiling NH_4F followed by a metathesis reaction of the solution by adding NH_4OH to precipitate the Zr, U, and fission products as hydroxides or hydrous oxides and to reclaim the fluoride as NH_4F for recycle.¹⁴ The precipitate is then dissolved in HNO_3 and provides a feed for solvent extraction that contains about one-fifth of the fluoride present in the current ICPP dissolution process.

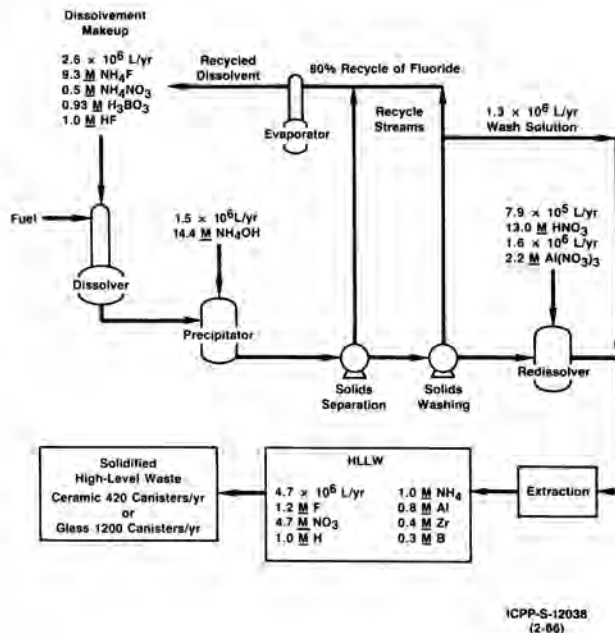


Fig. 4. Simplified Conceptual Flowsheet of the Modified Zirflex Head-End Fuel Dissolution Process.

Laboratory studies have been conducted on Zircaloy and unirradiated fuel coupon dissolution, precipitation and solids separation, and flowsheet calculations for volumes of reagents and products.

The results of this study indicate that unoxidized Zircaloy coupons can be readily dissolved by the Modified Zirflex dissolvent and follow first order chemical kinetics. The thickness of an oxide coating of $25 \mu\text{m}$ (1 mil) such as might be formed in a reactor reduces the dissolution rate of a Zircaloy coupon, resulting in a penetration time of 1 hour to more than 9 hours, depending on the dissolvent. The use of boron as a soluble neutron poison, in general, also reduces the dissolution rate. The shortest dissolution time for oxidized Zircaloy coupons was measured for a Modified Zirflex dissolvent of $9.3 \text{ M } \text{NH}_4\text{F}$, $0.5 \text{ M } \text{NH}_4\text{NO}_3$, $0.93 \text{ M } \text{H}_3\text{BO}_3$ and $1.0 \text{ M } \text{HF}$. Other concentrations and combinations of HF and HNO_3 produced longer dissolution times, and none of these dissolvents appeared to dissolve the oxide coating at a reasonable rate.

Either ammonia gas or ammonium hydroxide precipitates more than 99% of the zirconium from dissolved Zircaloy and about 45% of the boron. After about 2 hours of settling of the precipitate, 50 to 65% of the original liquid volume can potentially be decanted for recycle preparation. The solids can also be concentrated to about 23 wt% by centrifuging at 8000 rpm for 15 minutes and about 80% of the original liquid volume can be decanted for recycle preparation. The solids can be filtered into a moist filter cake containing about 28 wt% solids. This concentration would allow about 84% of the original liquid volume to be recovered for recycle preparation. However, all of these precipitate separation methods may be difficult to implement on a practical scale.

Thus, calculations indicate that about 80% of the fluorides required for dissolution could be recovered and recycled. However, because of the use of added fluoride for dissolution, the HLW volume reduction compared with base-case glass is calculated to be about 30% instead of about 55% originally estimated, assuming a 30 wt% loaded glass waste form.

The Modified Zirflex dissolution chemistry may be feasible for ICPP fuels. However, the potential waste volume reduction of only 30% is not as large as may be achievable with other options as described in this paper. Significant scale-up development would be required to prove the process, and major changes in ICPP processing would be necessary to implement this process. Thus, the Modified Zirflex process should not be developed beyond a laboratory feasibility evaluation of irradiated fuel dissolution.

Low-Fluoride Low-Volume Glass Waste Immobilization Process

A potential method for the reduction of ICPP waste volumes for final disposal is to volatilize the fluoride present in the ICPP waste during slurry feeding of the waste to a standard glass melter. In this process concept, HLLW is fed directly to a glass melter under conditions that volatilize approximately 80% of the fluoride to produce a low-fluoride glass. This is accomplished by adding only enough calcium nitrate to form nonvolatile CaF_2 with only 20% of the fluoride in solution, rather than forming CaF_2 with all of the fluoride as is done in the calcination process. The volatilized fluoride is then captured in the off-gas system either by liquid or solid scrubbing, immobilized as CaF_2 , and disposed of as a LLW. A solidified LLW form is made by forming concrete with the residual CaF_2 - CaO mixture at a 30 wt% loading.

Feasibility studies have been carried out for: (1) a low-fluoride glass formulation, slurry feeding of the laboratory-scale melter, and (3) fluoride volatility and trapping from simulated HLLW. The results of the studies show that while a feasible frit-waste composition can be made, only 38% of the fluoride appears to be volatilized from the waste. The amount of fluoride which would be volatilized from the glass melt and the possible impacts such volatility would have on melter corrosion, or on forming high-melting aluminum fluoride compounds such as cryolite has not been assessed. While no tests have been run to evaluate the radionuclide transport and trapping during recovery of the fluoride from the off gas, significant off-gas concentration of radioactive particulates and possibly semi-volatiles would be expected based on melter tests by others¹⁵ and also by the operation of the ICPP New Waste Calcining Facility (NWCF). Recent tests with the NWCF off-gas system have shown that sufficient actinide components are loaded on to the HEPA filters to cause them to be classified as transuranic (TRU) waste. Thus, the fluoride recovered in the off gas may also contain sufficient fission and actinide products so as not to be classified as a LLW.

Therefore, the low-volume glass process has intermediate-to-low feasibility, compared with other options, in significantly reducing the amount of HLW volume. The base-case glass process should nevertheless be considered as a back-up waste immobilization technology, even though it has a large HLW volume.

Evaluation of Scoping Studies for Alternative Processes and Recommendations for Further Work.

Table III is a summary of the characteristics and evaluation of scoping studies carried out on the alternative processes for reducing ICPP immobilized HLW volume.

TABLE III

Comparison of Alternative Processes to Reduce ICPP HLW Volume

Technology	Percent HLW Volume Reduction as Ceramic	LLW Volume, m^3/yr	Recommendation
1. Modified Fluorinel:		None	o Continue Development
o by HF Reduction	65		o Evaluate Reduced B and Cd Usage for Criticality Control
o by Critically Safe Dissolvers	72		
2. Ceramic	60	None	o Continue Development
3. Inerts Removal by Neutralization	85	9,500	o Continue Development
4. Modified Zirflex	75	None	o Stop Development After Lab-Scale Test
5. Low-Volume Glass	22 ^a	2,000	o Continue with Base-Case Glass o Stop Low-Volume Glass Development

a. Glass waste form.

Volume Reduction Potential. The four alternative processes which can use a ceramic HLW form show a potential for volume reduction over base-case glass ranging from 60 to 80%. The following conclusions in volume reduction can be drawn based on these scoping studies:

- o The modified Fluorinel process is the only process evaluated which can potentially be implemented in the near term, or up to 20 years before any other option to reduce waste volume. Thus, even though the annual percent volume reduction is small, the cumulative volume reduction is significant because of the possibility for early implementation.

- o The ceramic process can potentially reduce immobilized HLW volume by 60% regardless of any other process used. A volume reduction of up to 80% may be possible when combined with other volume reduction processes, such as neutralization and modified Fluorinel processes.
- o The neutralization process results in a potential 80% volume reduction when combined with the ceramic process.

Feasibility and Practicality. Significant questions on feasibility and/or practicality need to be resolved for ceramic and neutralization processes.

- o Ceramic Process. The potential problem in process feasibility lies in radioactive solids handling for canister filling and remote sealing of the canisters prior to densification to produce a ceramic form. Other parts of the process appear to be feasible.
- o Neutralization Process. The feasibility of separation of the HLW precipitate, which is gelatinous, from the salt solution, separating the radionuclides from the LLW fraction, and immobilizing the LLW in a grout need to be evaluated. Other parts of the process appear to be relatively feasible.

Recommendations. Further research and development effort should be concentrated on the ceramic and neutralization processes. Wherever practical, continued improvements to reduce waste volumes in the Fluorinel fuel dissolution process should be examined. These processes have the greatest volume reduction potential and have a higher feasibility than the Modified Zirflex and low-volume glass processes. Ceramic process research effort should concentrate on:

- o Waste form development
- o Design of canister for densification operation,
- o Remote canister filling and sealing, and
- o Process scale-up

The neutralization process research should concentrate on

- o Supernate decontamination methods,
- o Processing and forms to immobilize the HLW precipitate,
- o Process requirements to immobilize the salt solution as a LLW, and
- o Process conditions for the neutralization, precipitation, and solids separation.

Research on a base-case glass process as a back-up technology should be continued to the extent required for ICPP HLW immobilization. Although a large volume of immobilized HLW is produced, the technology will have been developed at Savannah River, Hanford, and West Valley.

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