UNCLASSIFIED

i

PRELIMINARY CANISTER SIZE CRITERIA FOR
COMMERCIAL SOLIDIFIED HIGH-LEVEL WASTE

J. R. LaRiviere
E. L. Moore

Commerical Waste Engineering Section
Advanced Waste Engineering Department
Research and Engineering Division

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Energy Research and Development Administration, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

Prepared for the Energy Research and Development Administration
by Atlantic Richfield Hanford Company under Control EY-76-C-2130

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

TABLE OF CONTENTS

	Page
ABSTRACT.....	1
1.0 INTRODUCTION.....	1
2.0 SUMMARY.....	3
3.0 PRELIMINARY CANISTER SIZE CRITERIA.....	3
4.0 CONSTRAINTS UPON WASTE CANISTER DIMENSIONS.....	4
4.1 GENERAL.....	6
4.2 COMPATIBILITY WITH EXISTING FACILITIES.....	7
4.2.1 STATUS OF COMMERCIAL FUEL REPROCESSING PLANTS.....	7
4.2.2 HIGH-LEVEL WASTE SHIPPING CASKS.....	9
4.2.3 TERMINAL STORAGE FACILITIES FOR HIGH-LEVEL WASTE.....	9
4.3 COMPATIBILITY WITH PLANNED FACILITIES.....	11
4.3.1 HIGH-LEVEL WASTE SOLIDIFICATION.....	11
4.3.1.1 CALCINATION.....	11
4.3.1.2 VITRIFICATION.....	12
4.3.1.3 OTHER SOLIDIFICATION CONCEPTS.....	14
4.3.2 INTERIM STORAGE OF HIGH-LEVEL WASTE CANISTERS AT THE FUELS REPROCESSING PLANT (FRP).....	15
4.3.3 TRANSPORTATION.....	16
4.3.4 HANDLING.....	17
4.3.5 TERMINAL STORAGE.....	17
4.3.5.1 CONSTRAINTS ON CANISTER LENGTH.....	18
4.3.5.2 CONSTRAINTS ON CANISTER DIAMETER.....	18
4.3.5.3 CONSTRAINTS ON CANISTER WEIGHT.....	19
4.3.5.4 CONSTRAINTS ON CANISTER SHAPE.....	19
4.3.6 WASTE ISOLATION PILOT PLANT (WIPP).....	20
4.3.7 THE RETRIEVABLE SURFACE STORAGE FACILITY (RSSF).....	21

5.0	ADDITIONAL WORK REQUIRED TO FURTHER DEFINE WASTE CANISTER ACCEPTANCE CRITERIA.....	21
6.0	REFERENCES.....	22

FIGURES AND TABLES

FIGURE 1	PRELIMINARY CANISTER SIZE CRITERIA.....	5
TABLE 1	AVAILABLE SHIPPING CASKS FOR CURRENT GENERATION LWR SPENT FUEL.....	10

UNCLASSIFIED

PRELIMINARY CANISTER SIZE CRITERIA FOR
COMMERCIAL SOLIDIFIED HIGH-LEVEL WASTE

ABSTRACT

Preliminary high-level waste canister size criteria are presented which will be used in the conceptual design studies (currently in progress) for the initial phase geological repositories. Two such repositories, located in deep salt formations, are planned by the Office of Waste Isolation for fiscal year 1985 operations. The preliminary canister size criteria define the high-level waste canister as being cylindrical in shape with a convex bottom. The canister diameter lies in the range 6.625-inches O.D. to 24-inches O.D. The maximum active length will be 16-feet with an additional 1-foot in length allowed for a lifting attachment and the maximum allowable weight will be 12,000 pounds. Presently known constraints on canister size and weight are identified and discussed. Additional work required to further define waste canister acceptance criteria are identified.

1.0 INTRODUCTION

In 1976, ERDA initiated a greatly expanded waste management program for defense and commercial radioactive waste. The Oak Ridge Operations Office (ORO) of ERDA was given the lead responsibility for overall coordination of the expanded commercial geologic disposal program and the Office of Waste Isolation (OWI) was created within Union Carbide Corporation - Nuclear Division (UCC-ND) with the responsibility for program management of that activity.

The commercial geological disposal program was named the National Waste Terminal Storage (NWTs) Program. It represents the principal programmatic effort of ERDA for ultimate disposal of commercial nuclear waste.

UNCLASSIFIED

The NWTS Program objective is to provide facilities in various deep geologic formations at multiple locations in the United States which will safely of the commercial radioactive waste which must be delivered to a Federal repository for terminal storage. ⁽¹⁾

Within the OWI organization, A Systems Studies Section, which is part of the Planning and Analysis Department, provides information, data, and services needed to provide guidance and direction to the NWTS Program. One such study is the High-Level Waste Canister Envelope Study which is being conducted for OWI by Atlantic Richfield Hanford Company (ARHCO). In this study definitive waste canister acceptance criteria will be developed. Such criteria are needed so that firm plans, designs, and cost evaluations for both reprocessing of spent fuels and for waste disposal can be prepared. The established of waste canister acceptance criteria will ensure that the canister is compatible with the environments, from the time of waste solidification until emplacement in a Federal repository including possible retrievable operation for an extended period of time.

Two initial phase repositories are planned by OWI for fiscal year 1985 operations. Conceptual design studies are in progress and will continue through fiscal year 1978. Two architect-engineering firms, Stearns-Roger Engineering Company and Kaiser Engineering, are conducting the conceptual design studies. To assist in the timely completion of these studies, canister sizing criteria are needed as soon as possible.

The purpose of this document is to furnish preliminary canister size criteria for the commercial solidified high-level waste, primarily for use in the preliminary design studies of the pilot plant repositories. These criteria consist of canister shape, diameter, length, and weight. They primarily affect the canister handling systems to be provided at the repository.

The selection of these preliminary canister size criteria is the first step in the overall objective of the High-Level Waste Canister Envelope Study. In the future these will continue to be evaluated, and subsequently refined. Other waste canister criteria will be selected, and they too will be reevaluated and refined, as needed. The progress of the high-level waste solidification development work, and the geologic pilot plants will be

UNCLASSIFIED

continually reviewed for determining any or all of the impacts upon the canister. Ultimately, a waste canister acceptance criteria will be selected and broadly issued for use.

2.0 SUMMARY

Preliminary canister size criteria for commercial solidified high-level waste are presented. Initially considered is a cylindrical canister 6.626-inches to 24-inches outside diameter, up to 17-feet in length, and weighting a maximum of 12,00 pounds when filled with waste. The criteria will ultimately be included in the canister acceptance criteria at the Federal repository. These criteria are considered preliminary because it is recognized that a considerable technical effort is still needed to further defined these, and other criteria not listed here, which will comprise the waste canister acceptance criteria. The purpose of this document is to furnish information for use in the conceptual design studies of the initial phase repositories, primarily in the area of canister handling systems to be provided at the repository.

In the establishment of waste canister acceptance criteria, a major goal is to develop criteria which are compatible with waste solidification and handling techniques at the fuel reprocessing plants and with transportation equipment as well as with handling techniques at the repository. Since there are no commercial reprocessing plants operations, the criteria are based on those plants which have been planned or proposed in sufficient detail to include a high-level waste canister concept (Allied General Nuclear Services, Incorporated and Exxon) and on the waste solidification development work underway at Battelle Northwest Laboratories.

3.0 PRELIMINARY CANISTER SIZE CRITERIA (See Figure 1)

The preliminary canister size criteria and pertinent comments regarding each criterion are as follows:

1. Canister Shape

a. Cylindrical

- Favorable for decontamination
- Favorable for heat transfer in geologic storage
- More resistant to crushing than rectangular shapes

b. Convex bottom head
with a skirt

- Minimizes stresses due to canister drop
- Provides stability for verticle positioning in storage placement hole.

2. Canister Diameter

6.625" O.D. to
24" O.D.

- Based on standard pipe sizes
- Full-scale cold vitrification tests planned up to 20", possibly to 24"
- WIPP proposes to use 16"
- Early bedded salt work based on 6" and 12"
- Retrievable Surface Storage Facilities designed for 6" to 24"

3. Canister Length

Up to 16' excluding
canister handling
device

- Allow up to 1' for canister handling device
- Full-scale cold vitrification tests planned at 10'
- Technical capability to 16' needed if spent fuels are to be placed in geologic storage
- Retrievable Surface Storage Facilities designed for 15'

4. Canister Weight

Up to 12,000 pounds

- Maximum canister size, waste in glass form.

4.0 CONSTRAINTS UPON WASTE CANISTER DIMENSIONS

The various constraints upon waste canister dimensions are discussed in this section. Canister compatibility with existing and planned facilities in the overall canister cycle, from time of waste solidification until final geologic storage, is reviewed.

UNCLASSIFIED

V7703-13

Figure 1

Preliminary Canister Size Criteria

4.1 GENERAL

Two criteria pertaining to the shape of the high-level waste canister are selected. The canister is cylindrical in shape and the bottom head is a convex head. The shape of the top head of the canister, which is the closure end, is not defined as yet.

Cylindrical Canister

It is highly desirable to minimize the use of square corners in the design of the canister. It is an accepted fact in the nuclear waste management business that rounded surfaces are easier to decontaminate than surfaces with sharp edges or square corners. All through the canister cycle of operations, the potential will exist for pickup of contamination by the canister exterior surface. To minimize the spread of contamination, both from normal and abnormal operating conditions, the canister surface will be decontaminated to acceptable levels whenever it is known to exceed the maximum permissible amount. The maximum permissible amount of contamination on the canister at the time of acceptance at the Federal repository will be set by a future high-level waste canister acceptance criterion.

Although the British are currently using cylindrical containers in their development work, they still are considering the possibility of using annular shaped canisters for storing the solidified high-level waste. The annular shape provides more surface area for heat removal per unit volume of waste than does a cylindrical canister. This is of greater importance during storage at the fuels reprocessing plant where the requirements for heat removal is the greatest. However, the design presents fabrication difficulties and the resulting stress problems need to be investigated further to verify design feasibility.⁽²⁾ In addition, their increased size may create handling problems at the Federal repository.

Convex Bottom Head

A convex shaped bottom head is preferred over either a flat or a concave shaped head. The convex configuration minimized the creation high stress areas that would be caused by bumping or dropping during handling. A

UNCLASSIFIED

bottom skirt would be attached as shown in Figure 1 to provide stability during vertical implacement at the Federal repository.

4.2 COMPATIBILITY WITH EXISTING FACILITIES

In developing canister acceptance criteria, one goal has been to select criteria which are compatible with existing and planned facilities, as much as possible. The status of existing fuels reprocessing plants, high-level waste shipping casks, and high-level waste storage repositories is discussed below.

4.2.1 STATUS OF COMMERCIAL FUEL REPROCESSING PLANTS

Three commercial fuels reprocessing plants have been constructed in the United States. These are: (1) The Nuclear Fuels Services Plant (NFS) in West Valley, New York; (2) the Midwest Fuels Reprocessing Plant (MFRP) in Morris, Illinois, and (3) the Allied-General Nuclear Services (AGNS) Plant in Barnwell, South Carolina. Their status and history are discussed below.

Nuclear Fuels Services, Incorporated (NFS)

The first commercial fuels reprocessing plant in this country was Nuclear Fuel Services, Incorporated (NFS). In the late 1950s NFS embarked on a program which culminated in a multipurpose plant located in West Valley, New York. The NFS plant operated for six years, and during that time processed more than 630 tonnes of spent fuel, of which some 240 tonnes were discharged by light water power reactors. In 1972, NFS ceased reprocessing to initiate an expansion program. In that same year NFS was advised by the Atomic Energy Commission (AEC) that before the expansion program could proceed, it would be necessary to obtain a construction permit. In the fall of 1976, NFS announced its withdrawal from the fuel reprocessing business.

Midwest Fuels Reprocessing Plant (MFRP)
General Electric Company (3)

The General Electric Company (GE) initiated construction in late 1967 of the MFRP at Morris, Illinois. Construction was completed in the early 1970s and in July 1974, GE announced that the plant was not operable in its present form. There were fundamental plant technical problems, and substantial modification would be required if the plant were to have an opportunity for commercial operation. No fuels have been processed at the MFRP, but GE uses the Morris facility for the storage of spent fuel.

Allied-General Nuclear Services (AGNS) (3)

Allied Chemical became interested in spent fuel reprocessing about 1960, and in 1968 filed for a construction permit for a 1500-tonne-per-year plant at Barnell, South Carolina. Gulf Oil and Royal Dutch Shell participate with Allied Chemical in the Allied-General Nuclear Services (AGNS) venture. Construction began at Barnwell in 1971, and the separations and UF₆ facilities were complete at the end of 1975.

The regulatory uncertainties related to plutonium recycle, safety, safeguards, waste packaging, waste form, and ultimate waste disposal criteria made it impossible for AGNS to continue with the design of the plutonium processing facility and waste solidification facility. Therefore, the engineering effort on these facilities was suspended. The licensing efforts for the fuel receiving and storage system and the operating license for separations and UF₆ facilities are being pursued.

In summarizing the status of existing reprocessing plants, we find that two plants are not operating, hence, they are not producing any solidified high-level waste. The third plant has been built but has not yet operated. Furthermore, no facilities have yet been constructed for the solidification of high-level waste.

UNCLASSIFIED

ARH-ST-151

4.2.2 High-Level Waste Shipping Casks

High-level waste is not currently being commercially solidified. Casks designed specifically for shipment of solidified high-level waste have not been built.

High-level waste shipping containers are expected to resemble the casks currently available for rail shipment of spent fuel from a reactor to a reprocessing plant. Heat dissipation and protection from high radioactive dose rates, which are major considerations in the shipment of high-level waste, are also prime considerations in the shipment of spent fuel.⁽⁴⁾

Table I lists the available shipping casks for current generation Light-Water Reactor (LWR) spent fuel. The small truck casks (NFS-4, NLI-1/2, and BCL-6) and the rail casks (IF-300, TN-12, and NLI-10/24) could potentially be used for shipping canisters of solidified high-level waste. The larger truck casks (NFS-5, TN-8, and TN-9) will be governed by special overweight requirements which will be so restrictive that it appears impractical to consider them for extensive use in shipping waste canisters.

The feasibility of using spent fuel elements casks for shipping high-level waste has not been extensively studied. In order to build upon existing technology, it appears that conducting such studies in the near future would be practical.

4.2.3 Terminal Storage Facilities for High-Level Waste

Currently, there are no terminal storage facilities for high-level waste in the U. S. The NWTs Program will establish initial phase repositories for the storage of commercially generated high-level waste commencing in FY 1985. These will be discussed below under Compatibility with Planned Facilities.

TABLE 1
AVAILABLE SHIPPING CASKS FOR
CURRENT GENERATION LWR SPENT FUEL (4)

<u>Common Name</u>	<u>Assemblies PWR/BWR</u>	<u>Shielding</u>	<u>Maximum Heat Removal (kw)</u>	<u>Approximate Loaded Cask Weight (MT)</u>	<u>Usual Transport Mode</u>	<u>Number Available Current/Under Construction</u>
NFS-4	1/2 ^(b)	lead	11.5	24	truck	6/0 ^(c)
NFS-5	2/4	uranium	20	25	truck	0/0
IF-300	7/18	uranium	76	74	rail	4/0
TN-8	3/0	lead	35.5	36	overweight truck	0/2
TN-9	0/7	lead	24.5	36	overweight truck	0/3
TN-12 ^(a)	12/32	steel	95	97	rail	0/1
NLI-1/2	1/2	lead	10.6	22	truck	3/2
NLI-10/24	10/24	lead	100	88	rail	0/12
BCL-6	1/1	lead	12.0	22	truck	0/1

a. Cask under construction for use in Europe. Licensing proceedings soon to be initiated in U.S. Heat removal is an estimate.

b. Means one (1) PWR assembly or two (2) BWR assemblies.

c. Means six (6) such casks are currently available and none (0) are under construction.

UNCLASSIFIED

4.3 COMPATIBILITY WITH PLANNED FACILITIES

Since there are no existing facilities in the fuel cycle which impose constraints upon the canister, the high-level waste canister acceptance criteria will be written to be compatible with the planned facilities wherever practical. These include the waste solidification facilities, interim storage of waste canisters at the FRP, transportation and handling of waste canisters, a second interim storage of waste canisters and terminal storage of the waste canisters.

4.3.1 High-Level Waste Solidification

The solidification process used and waste form which is produced are important factors in determining waste canister acceptance criteria. Considerable development work has been performed, both in this country and abroad, on various processes for solidifying high-level waste. Of these, the calcination and the vitrification processes are deemed to be essentially ready for commercial use. There are other solidification processes which are being studied or developed and these could come into use sometime in the future.⁽⁵⁾

4.3.1.1 Calcination

Liquid high-level waste consisting mostly of fission product nitrates is converted to oxides by thermal decomposition at temperatures from 500° C to 700° C. The form of the calcined oxide is granular or a powder. Residual nitrates and water can be further decomposed by batch fluidized-bed denitration at temperatures in the range of 600° C - 700° C to stabilize the calcine.⁽⁶⁾ The calcination processes which have been developed the most, and which have been considered by the commercial reprocessors for use, are fluidized-bed and spray calcination.^(7,8,9)

Fluidized-bed calcination was developed at Idaho National Engineering Laboratory (INEL) during the period 1952-1959. This process has been in use at INEL since 1963 calcining government defense waste. The fluidized bed operated in a temperature range to 500° C to 600° C using heat provided by in-bed combustion. Spray calcination has been developed at Pacific

Pacific Northwest Laboratory (PNL) over the past 15 years, and has mostly been used to feed calcine to vitrification processes. The liquid high-level waste is sprayed through nozzles into the top of a calciner barrel where the walls are held at a temperature of 700° C. The droplets are flash dried and calcined as they fall through the hot calciner barrel.

Calcines have lower thermal conductivities than glass-type wastes. Hence, the amount of heat per unit volume of calcined wastes that can be safely packaged in a canister will be less than that allowed for glass-type wastes. This is primarily based on the criteria of not allowing the solidified waste to exceed its maximum processing temperature. ⁽¹⁰⁾ Since high-level wastes may be diluted with intermediate-level wastes just prior to solidification, a wide range in the heat per unit volume of wastes can be expected. ⁽¹¹⁾ Through the use of internal fins, heat transfer from the waste calcine can be increased so that with relatively low heat per unit volume of waste calcine, canisters up to 24" diameter by 15' long appear to be feasible for use in the waste cycle and must be considered for evaluation in further studies.

4.3.1.2 Vitrification

Development work on vitrifying high-level waste has been underway since the mid-1950s and has made great strides in the past decade. The product of waste vitrification is a fused substance, in which each atom of high-level waste occupies a fixed location that does not change once the glass has been solidified. Consequently, the radioactive atoms in the high-level waste assume the chemical inertness that is a quality of the glass itself. ⁽¹³⁾ The two types of vitrification processes that are currently being studied for commercial use in this country are the in-can melter (ICM) and the continuous ceramic melter (CCM).

In-Can Melter (ICM) ^(14,15)

The ICM vitrification method being developed at Battelle-Northwest (BNW), has been demonstrated there since 1970, and is being pursued most vigorously. The process starts when the waste is sprayed as a liquid into a heated chamber where moisture and other volatile materials are driven off. When the material reaches the bottom of the chamber it is in calcine form. The calcine

and frit are blended together as they are fed to the canister and heated to 1100° C. As the molten material cools at a controlled rate, it solidifies to a nonolithic block and the waste atoms are bound as integral parts of the glass, filling the inside of the canister. Experience with pilot-scale equipment indicates that no major technological breakthrough is needed before this process can be used in a full-scale plant. Pilot-scale operations with calciners and glass-making equipment have already shown that the throughput required for a commercial-scale plant can be maintained. (13) A full-scale spray calciner and in-can melter test program is scheduled to commence in April 1977, by BNW. Full-size canisters, up to 20" in diameter and possibly up to 24" in diameter, containing nonradioactive materials, will be produced in this program.

Continuous Ceramic Melter (CCM)

Development of the continuous ceramic melter (CCM) at BNW began about three years ago, and an engineering prototype was operated at temperatures 1000 to 1300° C continuously for 11 months. The CCM may be fed by a calciner or may be directly fed a slurry of liquid high-level waste and glass frit. The canisters are now filled through the use of an overflow drain, however, in the future the CCM could be used in connection with glass-shaping equipment (e.g., making marbles for use in metal matrices). As the canister is filled, it could be water cooled to help prevent overstressing. Since the CCM operates at such a high temperature, it will be useful in making glasses with high silica content (which enhances the chemical durability of glass) and for wastes with compounds of fairly high melting temperatures (e.g., iron oxide). The development of the CCM is about one year behind the ICM.

In general, vitrification will increase the allowable size of the canister significantly, as compared to calcine since the thermal conductivity of glass is much higher than that of calcines. The center line temperature limit for canisters of glass may be about 800° C, or higher, depending on the amounts of ruthenium oxide trapped in the glass matrix (the viscosity of RuO_2 is thought to change quite a bit from the viscosity of glass above 800° C). With glass, the allowable heat load for canisters in a salt repository may be higher than that for calcine. Another advantage of glass is that it is a nonolithic solid that would be safer than calcine in the event of canister

failures. Except for extremely high temperatures (about 1200° C), the potential for calcine release is 100 to 100,000 times greater than that for glass/ceramics. (16)

4.3.1.3 Other Solidification Concepts

There are several other solidification concepts that are being developed for possible future use. These concepts are mainly aimed at lowering leachability, improving heat transfer, or both. Some of the more promising processes that are being developed in the U. S. are discussed below.

Supercalcine is being developed at Penn State University and BNW. The process involves the introduction of additives like silicon to liquid high-level waste, followed by any of the calcination processes. The calcine is baked and pelletized at 900 to 1100° C, avoiding the vitrification step. The leachability of supercalcine might be as low as glass. The heat transfer properties of canisters filled with supercalcine should be better than calcine, but not as good as glass (partly due to conductivity and partly due to not being monolithic.) Therefore, canister sizes for supercalcine should be somewhere between calcine and glass.

Sintered glass-ceramics (SGC) are being developed at the Idaho National Engineering Laboratory (INEL) and BNW. The process involves calcination, then shaping into pellets or disks using either a binder material or glass frit. This is followed by a sintering process to form a product with both vitreous and crystalline phases. The final product would be put in canisters either in bulk or within another matrix (such as metal). The leachability of SGC may be comparable to glass. The heat transfer in a canister filled with pellets or disks probably would not be as good as glass (since it would not be monolithic), although the allowable centerline temperature might be higher.

Metal matrices are being developed at BNW and INEL. Some solid form of waste, such as beads, granules, or pellets, is incorporated into a metal matrix in the waste canisters. The solid form may be made of calcine, supercalcine, SGC, or glass. A variation of the metal matrix process, that is sometimes referred

to as a multiple barrier form of solidification, uses a solid of low leachability such as glass beads, coated to make them even less leachable, and incorporated into a metal matrix. There are some problems with introducing the metal and waste solids into the canister to obtain a good monolithic solid, but various techniques are being developed to overcome this. The leachability of the final metal matrix solid varies widely depending on the waste form incorporated (calcine granules in a metal matrix would be more easily leached than coated glass beads). The heat transfer in metal matrices is very good. Aluminum, zinc and lead are some of the metals being studied and their conductivities are fairly high. Therefore, larger size canisters could probably be used for metal matrices and higher heat loads in the canister might be allowable in the repository.

4.3.2 Interim Storage of High-Level Waste Canisters at the Fuels Reprocessing Plant (FRP)

Details of packaging and interim storage of solidified high-level waste have not been announced by Allied-General Nuclear Services (AGNS). Development work for vitrification and packaging are in progress. Exxon Nuclear Company has announced some of its plans regarding interim storage of solidified high-level waste and these are discussed below.

Exxon Nuclear Company has submitted a preliminary safety analysis report (PSAR) to NRC for a fuels reprocessing plant at Oak Ridge, Tennessee. ⁽¹⁷⁾ Exxon plans to solidify the liquid high-level waste immediately, using a fluid bed calcination process. Vitrification to a glass or a ceramic product would follow immediately after the calcination step. Development work on the vitrification process is currently in progress. Exxon plans to use 8" diameter canisters for waste from the uranium recycle process, and 6" diameter canisters for waste from the mixed oxide recycle process. The canisters would be stored in the spent fuel storage basin. The baskets to be used for handling PWR and BWR spent fuel assemblies can be used to handle the waste canisters. Canister length of 16' could be safely handled in the water basin.

After the interim storage period in the water basin, multiple number of canisters would be placed inside a large diameter overpack, possibly up to 24" diameter. The overpack would be sealed and shipped to the Federal repository.

For purposes of applying the waste canister acceptance criteria, the overpack in this instance would be the prime containment package of interest. It would have to meet all the waste canister acceptance criteria in order to be received at the Federal repository.

In summary, the Exxon spent fuel storage basins can be used for storage of high-level waste canisters up to 16' long, and of varying diameter. Exxon would like to use overpacks up to 24" in diameter.

4.3.3 Transportation

Transportation of waste canisters will be required between the fuel reprocessing plant (FRP) and terminal storage. This could mean transporting from the FRP to an initial phase repository and then retrieval and transporting from the initial phase repository to terminal storage, or it could be just transporting from the FRP to terminal storage.

Shipment may be made by truck, rail, barge, or by any of these combinations. The canisters will be enclosed in massive shipping casks which provide radiation protection during the containment period. Neutron shielding and heat removal are problems that also must be addressed in shipping cask design.

Shipping casks have not yet been designed for solidified high-level waste. Conceptual designs for solidified high-level waste shipping casks have been developed, and it is generally believed in the industry that existing spent fuel casks could be modified for use as solidified high-level waste casks, or that the basic spent fuel cask design could be easily adopted for use to ship solidified high-level waste. (18-21)

In the conceptual designs referenced above, canister diameter from 6" to 24" have been considered. The canister lengths are generally taken at 10', but lengths up to 16' could be accommodated since spent fuel element casks are designed to handle 16' long fuels. It is visualized that the shipping cask inner cavity can be modified or replaced to accommodate a range of high-level waste canister lengths, diameters, shapes, and characteristics up to the maximum heat dissipation and shielding capacity of the cask.

4.3.4 Handling

The waste canisters must maintain their integrity during handling operations required at the reprocessing site for interim storage and loading into the shipping cask, at the repository for unloading of the shipping cask and for placement, and during the period of retrievability in the repository.

The period of integrity required for all of these steps might total 15 years. Should the canister be overpacked during of these steps, to increase reliability or to eliminate surface contamination problems, then the overpack must meet the waste canister acceptance criteria for receipt at the Federal repository.

A remotely operated handling device is being developed by ARHCO which, hopefully, will be acceptable to both the reprocessors and the repository operators. The device latches on to a bail attached at the top of the canister. The attractive feature of the design is that the device is self-centering and self-orientating with respect to the position of the bail.

For purposes of handling, canister diameters up to 24", canister lengths to 16' and canister weights up to 12,000 pounds presents no problems insofar as the handling systems are presently envisioned. The shape of the waste canister is affected by handling constraints, however. With the selection of a cylindrical shaped canister with a convex bottom head and skirt, stresses developed through bumping and perhaps even dropping a canister during handling operations, can be minimized. Generally, the elimination of sharp corners, by using rounded edges, provides a more rugged canister, insofar as handling is concerned.

4.3.5 Terminal Storage

The geological medium used for terminal disposal can significantly affect the size of the waste canister. Each type of medium, because of the degrading affects of temperature on physical properties, will impose a unique limit on the canister surface temperature. Waste form, allowable waste temperature, waste and geological medium thermal conductivities, waste heat generation rate, etc. function in meeting canister surface temperature limits which in turn constrain the canister diameter.

The length of the waste canister can be affected by the degree of difficulty of mining in a particular geologic medium, for example, mining out of areas to provide head room for installation of the sleeve, canister and shield plug. Of the geologic formations under consideration, the heat transfer and physical properties of salt are best known. The thermal conductivity of salt is higher than that of any of the other geologic media under consideration. (22)

4.3.5.1 Constraints on Canister Length

Canister lengths of 10 to 15' have been used in various waste management studies involving the waste canister. The head room available in a given underground storage/disposal facility will limit the acceptable length. A typical canister used in most of the conceptual design studies is 10' in length. It appears that it is a matter of economics which will determine the maximum canister length that can be handled in terminal storage. Such economic studies are planned and until they are complete, we should consider the possibility of handling waste canisters up to 16' in length.

4.3.5.2 Constraints on Canister Diameter

The maximum acceptable diameter of a waste canister will also depend upon whether retrievable geologic storage or terminal geologic disposal is contemplated. An initial period of retrievability exists during geologic storage. During this period, the holes in the mine floor would be specially prepared, by installation of a liner, and the room itself would be left open for easy access. Also, during this time it may be desirable to doubly contain the canister by over-packing with a second canister.

These provisions for retrievability (overpacking, liner) will reduce the acceptable diameter of the inner waste canister for final storage. A typical canister used in many conceptual design studies for a salt mine demonstration waste repository is 12" in diameter.

Additional studies are planned to investigate the feasibility of storing canisters up to 24" in diameter in salt. Once again it may be a matter of economics that determines the maximum canister diameter, however larger diameter canister will increase shielding requirements hence greater weight for the transporter which may be a problem in the mine. Other geologic media may impose different constraints on the canister diameter. Planned studies will also include such evaluations. However, the first initial phase repositories will be constructed in salt beds, and our first priorities will be to support salt storage.

The maximum canister size will need to be established before completion of conceptual design.

4.3.5.3 Constraints on Canister Weight

The canister will be unloaded from the shipping cask at the repository and transported down the shaft to the storage level. There it will be transferred by a shielded vehicle to the storage hole. The normal lifting and transporting operations should not constrain the canister weight. The floor loading which will mainly result from the transporter weight could possibly be limiting depending on the number of holes in the floor, the hole spacings used, and the geologic medium itself. Allowing up to 12,000 pounds for the weight of a waste canister permits using a 24" diameter by 16' long (active length) canister filled with glass type of waste.

4.3.5.4 Constraints on Canister Shape

The handling of canisters and the constraints imposed upon them due to handling are discussed above. This same discussion applies to the canister handling aspects at the repository.

Cylindrically shaped canisters are highly desirable for heat transfer reasons in the storage facility. Circular storage holes will be drilled and a minimum of backfilling around the liner will be required. Maximum heat transfer from a cylindrical canister to surroundings can be achieved because even heat distribution can be more easily attained than with other geometries.

4.3.6 Waste Isolation Pilot Plant (WIPP)

The Sandia Laboratories have been authorized by ERDA to conduct conceptual design studies of a Waste Isolation Pilot Plant (WIPP) for the deep geologic storage of ERDA defense wastes, i.e., transuranics (TRU) and intermediate-level waste (ILW). It has also been requested by ERDA that a limited amount of commercial high-level waste storage tests be conducted in the WIPP. The WIPP would be located 30 miles east of Carlsbad, New Mexico, and storage would be in salt beds.

In keeping with this latter request, Sandia has issued a waste canister definition that would be suitable for use at the WIPP. Their reference canister is defined as a double-shell right circular cylinder with an outside diameter of 16" and a length, not including handling hardware, of 10'. The thermal power of the reference canister is 3.5 kw at the time of emplacement in the salt (10-year old waste).

The dimensions of the WIPP reference waste canister and its shape are in keeping with the preliminary size criteria listed in Section 3.0 of this report. It is too early to state that the dimensions of the WIPP reference waste canister are maximum limitations. More work needs to be done to explore the possibilities of increasing the WIPP reference canister dimensions of diameter and length and also the canister heat load.

4.3.7 The Retrievable Surface Storage Facility (RSSF)

Conceptual design studies have been conducted by ARHCO on the storage of solidified high-level waste for an interim period, up to 100 years, prior to terminal disposal. The waste was to be stored in a surface facility and it was to be retrievable at all times during the storage period. Hence, it was called the RSSF. Waste canisters, up to 24" in diameter and up to 15' long, would be overpacked within a 2" thick carbon steel storage cask which would be sealed by welding. The sealed storage cask would be placed inside of a thick, 38 to 40" concrete shielding ring to form a storage unit. A 6" air gap would be left between the storage cask and the outer concrete ring.

This would provide an annulus around the storage cask and allow cooling by natural convection. Air would enter the annulus at the bottom and leave at the top. The storage units would be placed above ground in a storage area and spaced far enough apart to permit each to operate as an individual passive cooling unit.

The current ERDA waste management plan is aimed to provide geologic storage of high-level waste as soon as is possible. The RSSF is being held as a contingency plan.

The waste canister size criteria as developed for the RSSF are within the limits set forth in Section 3.0 of this report. However, it remains to be determined in future studies whether the geologic storage facilities will be able to safely handle and store such large canisters.

5.0 ADDITIONAL WORK REQUIRED TO FURTHER DEFINE WASTE CANISTER ACCEPTANCE CRITERIA

Criteria not covered in this report include: the heat load, surface contamination level, canister handling device, materials of construction, and fabrication code. These are being investigated by ARHCO in ongoing studies. Status reports on the progress of defining these criteria will be issued at the conclusion of significant tasks. The waste solidification development work, the geologic storage design studies, and technical backup work will be closely followed by ARHCO. Progress in these areas will be evaluated and discussed with those having an interest in waste canister criteria, especially the industrial firms who are actively participating in this area of work. Tentative criteria will be proposed and comments solicited. Sufficient technical work will be performed to adequately back up each criterion when issued. Similarly, the preliminary size criteria set forth in this report will also be reevaluated and further defined as the work discussed in the text progresses.

6.0 REFERENCES

1. Y/OWI/TM-11/1, Volume 1, "National Waste Terminal Storage Program Information Meeting," December 7 and 8, 1976
2. Proceedings of the International Symposium on the Management of Wastes from the LWR Fuel Cycle, Page 161, Denver, Colorado, July 11, 16, 1976
3. L. J. Colby, Jr., "Fuel Reprocessing in the United States: A review of Problems and Some Solutions," Nuclear News, January 1976
4. ERDA-76-43, Volume 3, May 1976, "Alternatives for Managing Wastes from Reactors and Past Fission Operations in the LWR Fuel Cycle"
5. Reference 3, Volume 2, Section 6.0
6. ICP-1087, June 1976, J. A. Rindfleisch, "Stabilization of Simulated Commercial Waste Calcine"
7. ICP-1075, June 1975, W. A. Freeby, "Interim Report: Fluidized-Bed Calcination of Simulated High-Level Commercial Wastes"
8. BNWL-2138, November 1976, W. J. Bjorklund, "Fluidized Bed Calcination Experience With Simulated Commercial High-Level Nuclear Wastes"
9. BNWL-2059, August 1976, W. J. Bonner, H. T. Blair, and L. S. Romero, "Spray Solidification of Nuclear Wastes"
10. Code of Federal Regulations, Title 10, Part 50, Appendix F (10 CFR 50, Appendix F)
11. A. K. Williams, "Analysis of Proposed Waste Management Practices at the Allied-General Nuclear Services Facilities," 1976
12. ARH-LD-104, December 1974, D. E. Larson, "High-Level Waste Canister Thermal Characterization for the Sealed Storage Cask Concept"
13. Gary J. Dau, "Nuclear Waste Management," published in Battelle Today, February 1977
14. BNWL-2092, November 1976, D. E. Larson and W. F. Bonner, "High-Level Waste Vittrification by Spray Calcination/In-Can Melting"
15. BNWL-2061, May 1976, H. T. Blair, "Vitrification of Nuclear Waste Calcines by In-Can Melting"
16. BNWL-1893, February 1975, Battelle, "Quarterly Progress Report, Research and Development Activities, Waste Fixation Program October through December 1974"

17. Docket 50-564, Exxon Nuclear Company, Nuclear Fuel Recovery and Recycle Center
18. ERDA-76-43, May 1976, "Alternatives for Managing Wastes From Reactors and Post-Fission Operations in the LWR Fuel Cycle," Volume 3, p.p. 22.7-22.13
19. ORNL-TM-2971, December 1970, J. J. Perona, R. S. Dillon, and J. O. Blomeke, "Design and Safety Considerations of Shipping High-Level Radioactive Wastes"
20. ORNL-TM-3651, February 1972, J. J. Perona and J. O. Blomeke, "A Parametric Study of Shipping Casks For Solid Radioactive Wastes"
21. BNWL-2179, January 1977, A. M. Platt, "Nuclear Waste Management and Transportation, "Quarterly Progress Report, July through September"
22. BNWL-1900, May 1974, "High-Level Radioactive Waste Management Alternatives, Section 4: Geologic Disposal," Volume 2, p.p. 4.C.1
23. ERDA-76-43, May 1976, "Alternatives for Managing Wastes from Reactors and Post Fission Operations in the LWR Fuel Cycle," Volume 3, p.p. 21.1-21.42

UNCLASSIFIED

DISTRIBUTIONNo. of
Copies

3 Energy Research and Development Administration,
Richland Operations Office

O. J. Elgert

8 Atlantic Richfield Hanford Company

D. A. Calmus
R. J. Gimera
E. L. Moore
R. A. Oliver
R. C. Roal
R. E. Smith
D. D. Wodrich

Central File

25 Office of Waste Isolation

J. M. Morrison