RHO-C-13

NUCLEAR SERVICES CORPORATION

INTEGRATING
DESIGN
CONSTRUCTION
OPERATION

MASTER

1700 DELL AVENUE CAMPBELL, CALIFORNIA 95008

DISCLAIMER

NSC-1-78-004

RECOMMENDATIONS FOR CODES AND STANDARDS TO BE USED FOR DESIGN AND FABRICATION OF HIGH LEVEL WASTE CANISTER

MASTER

PREPARED FOR

ROCKWELL HANFORD OPERATIONS

BY

NUCLEAR SERVICES CORPORATION

CAMPBELL, CALIFORNIA

Prepared by:

A. J. Bermingham
R. J. Booker
H. R. Booth
W. G. Ruehle
S. Shevekov
A. G. Silvester
S. W. Tagart
J. A. Thomas
R. G. West

Reviewed by: HR Booth

Approved by: D. Munsen

Revision No.	Date	NSC Job No.	Issued By	Date
0	1/26/78	RHO-0101	D. Munsen	27 JAN 78

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

1.0 INTRODUCTION

1.1 Objectives

This study is an integral part of the High-Level Waste Canister Envelope Study Program whose prime objective is to develop design criteria for the canisters that will contain the solidified high-level radioactive waste from commercial reprocessing operations. Rockwell Hanford Operations, a prime contractor of the Department of Energy, engaged in chemical processing and management of radioactive wastes, has been given the responsibility for conducting this program. The program is part of the systems study of the National Waste Terminal Storage Program managed by the Office of Waste Isolation, Union Carbide Corporation, Nuclear Division, Oak Ridge, Tennessee.

The primary objective of this study is to identify codes, standards and federal regulations that are applicable to the design and construction of HLW canisters. This task is one of a number of program elements or tasks that have been identified for study to achieve the Rockwell Hanford Operations program objectives. Up to the current time, no specific codes or standards have been identified to govern the design, fabrication, and inspection of the high-level waste (HLW) canister. Some basic regulatory requirements, applicable to processing, packaging, handling, transport, and disposal of high-level wastes do exist and will apply to waste canisters. However, there still exists a need for further development of requirements and acceptance criteria for these canisters for various phases of the anticipated canister life cycle to ensure environmental safety.

In consideration of public scrutiny and acceptance of such containers for ensuring high integrity and long-term containment of high-level waste, it is essential that specific codes, standards, and licensing and quality assurance requirements be identified and implemented for canister design and use. In cases where existing requirements of these documents are not considered satisfactory, or do not address the essential factors, additional requirements must be established and incorporated in appropriate documents for application to the above-stated objectives.

This study deals primarily with identification of those existing codes, standards, and licensing documents (or portions thereof) that we recommend be made applicable to the design, fabrication, and inspection of high-level waste canisters. This study also makes recommendations for development of additional appropriate standards and requirements to ensure safety, and the integrity of these canisters throughout the various phases of the anticipated life cycle.

1.2 Background

At the present time, no plant facilities are being operated commercially in the United States for reprocessing of spent nuclear fuel and solidifying the high-level liquid waste generated from reprocessing operations. No commercial containers have been designed and constructed for packaging these solidified wastes.

Title 10, Code of Federal Regulations Part 50 (10 CFR 50), Appendix F, requires that high-level radioactive liquid waste produced from reprocessing of nuclear fuel, be converted to a dry solid and placed in a sealed container (canister) within five years after reprocessing and transferred to a Federal repository within 10 years after processing. At the repository, the Nuclear Regulatory Commission (NRC) will take title to the radioactive waste material and emplace it in geologic storage.

The canister is considered to be part of an integrated handling and storage system. During its life cycle, the canister will be exposed to a variety of environmental conditions. The canister life cycle will include the following stages:

(a) Waste Solidification
(b) Interim Storage at the Fuel Reprocessing Plant
(c) Shipping
(d) Geologic Repository

The high-level waste canister is defined as the primary, high-integrity containment barrier between the solidified radioactive waste and the environment. As such it must -

- o Maintain its integrity while in interim storage at the fuel reprocessing facility, during shipment, and for a retrievability period of up to five years after emplacement in geologic storage.

- o Be designed and fabricated to permit a remote weld closure and remote nondestructive examination of the closure.

- o Be fabricated of a material that is decontaminable and is compatible with the waste form and storage environment.

- o Be compatible with all handling, shipping, and storage equipment and facilities.

- o Be reasonably economical to fabricate.

1.3 Scope

The scope of this study primarily encompasses the following:

(a) Identification and interpretation of applicable codes, standards, and regulations governing the design, fabrication, inspection, and licensing of high-level waste canisters;

(b) Development of preliminary canister closure design concepts for canisters which meet all applicable codes identified in (a); and,

(c) Determination of a testing program necessary for canister design verification.

To complement the above, this study includes identification of quality assurance standards to be applied to the design, fabrication, and testing of the canister.

Further, investigations have been made relative to such considerations as overpacking of the canister and materials for fabrication. It is not within the scope of this study to make specific recommendations based on these investigations. Rather, discussions of such parameters are included with particular emphasis on identifying various advantages and disadvantages of utilization of certain materials for various phases and options within the canister life cycle. Similar discussions are also provided on the necessity of overpacking the canister. These discussions are considered pertinent as a reference source for further development of canister design criteria, requirements, and standards for the high-level waste canister.

Rockwell Hanford Operations has identified certain assumptions to be utilized in this study. These assumptions are outlined in Section 3.2 of this report.

2.0 SUMMARY

A study which identifies codes, standards, and regulatory requirements for developing design criteria for high-level waste (HLW) canisters for commercial operation has been performed. This study program is part of the systems study of the National Waste Terminal Storage Program managed by the Office of Waste Isolation. The primary responsibility for conducting this program has been assigned to Rockwell Hanford Operations.

Direct application of existing codes and standards to development, design verification, manufacture, or testing of an HLW canister is limited; however, it is considered that certain provisions of existing documents will be useful as a baseline for development of canister design. Some basic regulatory requirements for solidification, packaging and disposal of high-level waste are contained in 10 CFR 50, Appendix F. In 10 CFR 71, basic packaging requirements are identified for transport of large quantities of radioactive material for both normal and accident conditions. However, further development of requirements and acceptance criteria for the canister is needed.

Based on results of this study, it has been determined that the canister should be designed as a pressure vessel without provision for any overpressure protection type devices. Section III of the ASME Boiler and Pressure Vessel Code is the most comprehensive existing code that can be applied to design, fabrication, inspection, and performance testing of the canister. Rules for the design and fabrication of three different safety classifications of nuclear-grade pressure vessels are contained in Section III. It is the recommendation of this study that the HLW canister be designed and fabricated to the requirements of the ASME Section III Code, Division 1 rules, for Code Class 3 components. As an alternative, a new subsection for Section III of the Code can be developed specifically for the HLW canister, and based principally on existing rules for Code Class 3 components. Identification of other applicable industry and regulatory guides and standards are provided in this report.

Prerequisite to the design of an acceptable HLW canister is the development of Design Criteria which will delineate the overall functional requirements and standards the canister must meet. Pursuant to the Design Criteria will be the development of a detailed Design Specification which will provide to the canister designer and fabricator minimum requirements for interface load capability, environmental qualification, material procurement, quality assurance, fabrication standards, inspection, testing, shipping, etc. Requirements for the Design Specification are found in the ASME Section III Code.

A comprehensive design verification program for the HLW canisters is considered imperative. It is the recommendation of this study that design verification be conducted principally with prototype testing which will encompass normal and accident service conditions during all phases of the canister life.

Adequacy of existing quality assurance and licensing standards for the canister ~~were~~ was investigated. One of the recommendations derived from this study is a requirement that the canister be N stamped. In addition, acceptance standards for the HLW waste should be established and the waste qualified to those standards before the canister is sealed.

Selection of a specific material for fabrication of the HLW canister is outside the scope of this study. However, an overview study was made assuming carbon steel, stainless steel, and Inconel and Incoloy series steels to be the most suitable materials for withstanding the required environmental conditions. Selection of a specific material will require a thorough investigation of the environmental conditions that the canister must withstand during the various phases of its life cycle.

A preliminary investigation of use of an overpack for the canister has been made. Based on this investigation, it is the opinion of Nuclear Services Corporation that the use of an overpack, as an integral part of overall canister design, is undesirable, both from a design and economics standpoint. Use of an overpack may also demonstrate lack of capability to attain a reasonable set of tradeoffs between facility requirements and canister design. However, use of shipping cask liners and overpack type containers at the Federal repository may make the canister and HLW management safer and more cost effective.

There are several possible concepts for canister closure design. These concepts can be adapted to the canister with or without an overpack. A remote seal weld closure is considered to be one of the most suitable closure methods; however, mechanical seals should also be investigated.

3.0 DESIGN BASES AND CONSIDERATIONS

3.1 Study Bases

Outlined below are the study bases and alternatives within the canister life cycle as defined by Rockwell Hanford Operations for this study.

3.1.1 Waste Management System

The system associated with management of commercial waste is composed of:

(a) Fuel reprocessing plant (FRP)
(b) Post-reprocessing waste solidification
(c) High-level waste canister
(d) Interim storage facility associated with FRP
(e) Transportation
(f) Geologic storage at the Federal repository

The above waste management components have been incorporated by Nuclear Services Corporation into three phases, as shown in Figure 3-1. Phase one concerns those activities associated with the licensed reprocessing facility; phase two, the shipping cask; and phase three, the Federal repository.

3.1.2 Waste Conversion and Transportation

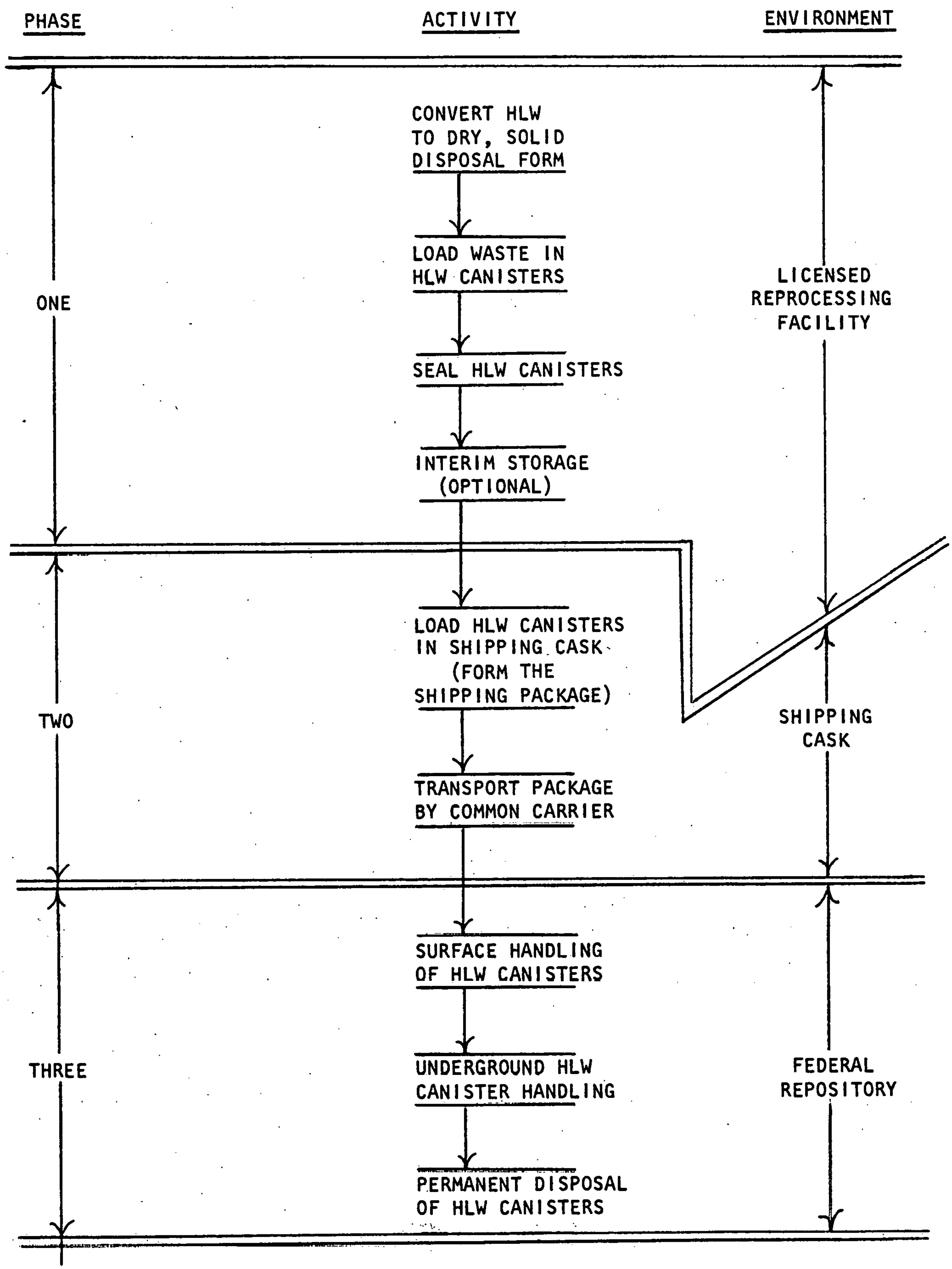
(a) Waste to be converted to a dry solid and placed within a sealed canister within five years after reprocessing.

(b) Canister (with solid waste) to be transferred to Federal repository within ten years after reprocessing.

3.1.3 Waste solidification procedures

(a) In-can zone melting to form a glass-type waste, or
(b) Continuous melting to form a glass-type waste, or
(c) Calcination and post-stabilization to form a calcine.

FIGURE 3-1: THE THREE PHASES OF HIGH LEVEL WASTE (HLW) MANAGEMENT



3.1.4 Interim storage at Fuel Reprocessing Plant
- (a) Water storage or
- (b) Air storage

3.1.5 Transportation Mode
- (a) Air cooled shipping cask

3.1.6 Geologic Repository
- (a) Salt, or
- (b) Shale, or
- (c) Basalt
- (d) Granite

3.1.7 Other Design Considerations
- (a) Canister will require remote handling for operations including closure welding and decontamination.
- (b) Canister should be reasonably economical to fabricate.
- (c) Overpacking should be investigated as a means of ensuring safety or facilitating handling.
- (d) Canister should be retrievable (with or without overpack) up to five years after emplacement in geologic storage.

3.2 Assumptions

Rockwell Hanford Operations has identified certain assumptions for this HLW canister study. These assumptions, outlined below, have been utilized by Nuclear Services Corporation in performing the study.

3.2.1. Material - 304L stainless steel is the reference material.

3.2.2 Thermal equilibrium -
350^{o}C (canister wall)
800^{o}C (centerline of glass-type waste)
700^{o}C (centerline of calcine waste)

3.2.3 Canister Dimensions
General range of canister dimensions:
6" - 24" (diameter);
10' - 16' (length)

3.2.4 Waste Density

Solidified high-level radioactive waste will have approximately the same density as aluminum (3 g/cm^3)

3.2.5 Canister Fabrication

(a) Roll and longitudinally weld a right cylinder

(b) Form and circumferentially weld a lower end cap on the cylinder

(c) Form and circumferentially weld an upper end cap containing an opening for waste filling and including a fitting for mechanically locking and weld sealing a closure cap

(d) Seal weld the closure fitting after filling

3.2.6 Canister Filling

Filled to 80% with high-level solidified waste

3.2.7 Canister Closure

The canister should be designed to permit remote weld closure and remote NDE of the closure weld. The weld closure should be assumed to be made in air.

3.3 Canister Pressure Retaining Requirements

3.3.1 Pressure Retaining Requirements

Early in the study, special interest was expressed by Rockwell Hanford Operations for a determination of the necessity and/or desirability of designing the canister as a pressure vessel. Existing regulatory requirements were reviewed to determine if there were any licensing standards that would require that the canister be designed to hold internal pressure. Such pressure could result from gases released by the waste with time, or expansion of air within the canister during heating and cooling.

Based on regulatory provisions of 10 CFR 50, Appendix F, the high-level waste must be "chemically, thermally, and radiolytically stable to the extent that the equilibrium pressure in the sealed container will not exceed the safe operating pressure for that container during the period from canning through a minimum of 90 days after receipt at the Federal repository." (For this study, per Section 3.1.7, a canister retrievability period of five years was used.) Thus, this regulation implies that the canister should be designed and constructed as a primary container to withstand internal pressure. Stabilization of the waste material, based on requirements of 10 CFR 50, Appendix F, is further discussed in Sections 5.1.3 and 6.1(d).

The desirability of designing the canister as a pressure vessel, from a technical basis, was also evaluated. If the canister were not designed to retain pressure, airborne radioactivity could be released at the Fuel Reprocessing Plant, during transportation, or at the Federal repository. At the FRP and Federal repository, airborne radioactivity could endanger site personnel and place strong requirements on the plant ventilating systems to filter and remove particulate and gaseous activity. Both of these consequences were considered highly objectionable. During transportation, release of airborne particulates and gases from the canister would place additional restrictions on the shipping cask and pose hazards to the public. Thus, from a technical basis, it was also considered desirable to design the canister as a pressure vessel.

It is, therefore, the recommendation of this study that the canister be designed as a pressure vessel.

3.3.2 Overpressure Protection

Subsequent to the decision that the canister should be designed as a pressure vessel, the necessity for overpressure protection was also examined. That is, the canister will be sized and fabricated for a specific design pressure.

Section 5.2 of this report recommends that the ASME Section III Code* be applied to design and construction of the canisters. Part ND7110(b)

*American Society of Mechanical Engineers Boiler and Pressure Vessel Code.

of Section III specifies that "individual components which are isolable from system overpressure protection shall be reviewed to determine whether additional individual overpressure protection is necessary (NB-7155)."

Overpressure protection could be designed into the canister, which would prevent it from rupturing. Rupture discs and relief valves are two such design mechanisms. Thus, if high pressures were encountered in the canister, a rupture disc or relief valve would operate, relieve the pressure, and prevent the canister from rupturing. However, radioactive material would be released to the environment.

Possible sources of overpressure were then examined to determine if canister rupture is a credible accident. Kaiser Engineers' studies (Reference 1) examined internal canister pressure that would result from entrapped air expanding during canister heating. These studies indicated a calculated pressure of 30 psia resulting from air at 14.7 psia and 80^{O}F (during closure) being heated by the waste to 650^{O}F. Kaiser Engineers subsequently selected 100 psia as a conservative preliminary design pressure.

The other source of canister internal pressure would be gases released from the waste. However, appropriate stabilization of waste material per requirements of 10 CFR 50, Appendix F (as described in Section 3.3.1 and 5.1.3) precludes waste gas release from being a source of internal pressure.

It should also be noted that in the unlikely event the canister were to rupture at the FRP or Federal repository, the radioactive material would be contained within the facility containment structures and the filter exhaust systems would prevent release of this material to the environment. If the canister were to rupture inside the shipping container, the container could be designed to retain the waste and prevent it from endangering the public.

It is thus the recommendation of this study that the canister be designed without incorporation of any overpressure protection type devices such as relief valves or rupture discs. Emphasis should be placed on stabilizing the waste before sealing it in the canisters and thus satisfying the provisions of 10 CFR 50, Appendix F. Provision should also be made for monitoring gas release from the unsealed canister for a finite period, and for leak testing the sealed canister. Maximum gas release and leak rates should be established as part of the acceptance criteria. Testing of the canister is discussed in Sections 3.6 and 5.3.

As an additional means to ensure canister quality and, as described in Section 5.1 of this report, the canister should be designed and manufactured in compliance with the ASME Code (e.g., N-stamped).

3.4 Materials

Selection of a material that would provide an acceptable level of performance within expected environments is among the more significant problem areas to be resolved in the design and fabrication of HLW canisters. A thorough investigation of factors affecting the selection of canister materials is outside the scope of this report. However, an overview of available information, which provides a framework for code and standard studies, is presented herein.

Optimally, a single material will be selected from candidate materials that exhibit acceptable levels of strength, ductility, corrosion resistance and ease of decontamination.

Careful consideration must be given to selection of a material that will withstand the environmental conditions to which the material is exposed during the various phases of the canister life cycle. These phases are illustrated in Figure 3-1. Section 3.1 (Study Bases) identifies the various alternatives and options for the life cycle phases. During the first phase of its design life the canister may be in this event, used as a container employed in the waste solidification process. In this event, the selection of a melt and filling process for the solidified waste should have as a goal, minimum fabrication degradation of the canister. The method of filling the canister should ensure that the seal weld area remains free from contaminants and that the external surface is easily decontaminated. The canister should also be supported to maintain design dimensions during canister filling. One concept of canister filling that accomplishes these goals is shown in Figure 3-2. The concept involves the use of a double walled funnel to fill the canister from a continuous melt furnace; a barrier to prevent contamination of the exterior surface of the canister, and the use of an inert gas cover during filling and canister cooling. It should also be pointed out that preformed and cooled waste in slug, pellet (or other form) could be loaded into canisters. This would eliminate such canister degradation problems as high temperature, stress corrosion, yield, and distortion.

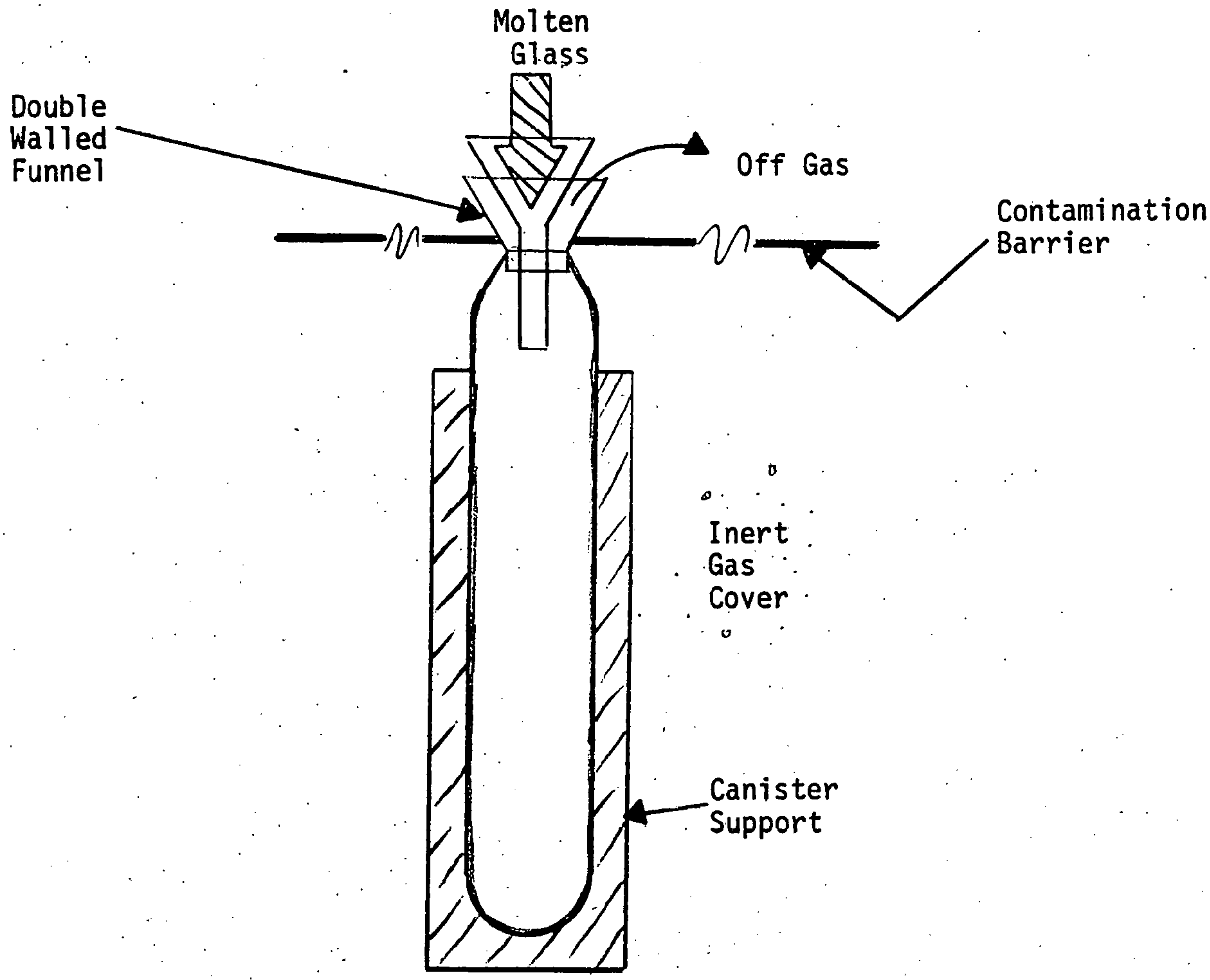
If suitable environmental protection can be provided, during canister filling and a material selected to withstand the filling environment, then the canister may be produced without reliance on an overpack canister. However, subsequent phases of the canister life require suitable material performance during interim storage and long term geologic storage, which impose additional constraints on the choice of canister material.

Candidate materials that have been assumed for this study to be most suitable include carbon steel, stainless steel, and Inconel and Incoloy series steels.

Carbon steel, although economical and easy to fabricate, exhibits relatively poor resistance to corrosion. The most widely used material in the nuclear industry for corrosive environments, and which requires a high degree of cleanliness, is low carbon content austenitic stainless steel. This material, however, becomes susceptible to stress corrosion cracking if used in a sensitized condition. High temperatures associated with the waste solidification and canister filling processes (e.g., in-can zone melting could result in sensitization of stainless steel materials.

FIGURE 3-2

CANISTER FILLING CONCEPT



Regulatory requirements which preclude use of sensitized steel are provided in Regulatory Guide 3.37. Austenitic stainless steel exhibits poor corrosion resistance upon emplacement in a salt bed environment.

Alloys such as Inconel and Incoloy have also been proposed because of their resistance to corrosion and apparent lack of susceptibility to stress corrosion cracking.

It is the recommendation of this study that material selection be based on the use of a material that provides the most appropriate balance of the following characteristics and facility tradeoffs:

a. HLW solidification process and loading plus processing quality assurance
b. corrosion resistance
c. adequate strength and ductility for canister design pressure
d. relatively easy to procure and fabricate
e. weldability or mechanical seal
f. ease of decontamination
g. cost economy

If possible, the material should be selected from a list of materials presently accepted by the Nuclear Regulatory Commission for use in ASME components (see Regulatory Guide 1.85), and confirmed by a suitable qualification program (see Section 3.6).

3.5 Overpack

The use of an overpack has been proposed (Reference 2) as an acceptable method for maintaining the primary barrier function of the canister for conditions in which degradation of the canister material cannot be controlled to an acceptable degree.

Nuclear Services Corporation has made a preliminary investigation of the use of a canister overpack as an integral part of the design. The three phases of the canister life cycle (referred to in Section 3.1.1) with provisions for overpack were considered. Based on the results of this investigation, it is the opinion of Nuclear Services Corporation that the use of an overpack, as an integral part of design, results in an undesirable design and economic constraint on the waste management system. This determination is primarily based on our rationale that the overpack would:

a. Need to be constructed to standards equivalent to the canister.
b. Provide increased thermal insulation of the HLW material.
c. Complicate design, testing, and handling of the container system.

Certain advantages of using an overpack have previously been discussed (References 2, 3, and 4). These advantages are outlined below and are related to the above-mentioned three phases of the canister life cycle. Nuclear Services Corporation has identified possible alternatives that would eliminate the need for overpack for these cases.

A. Phase 1 - Activities Associated with Licensed Processing Facility

(1) Facilitate decontamination
Alternatives

- o Use preformed, solid glass slugs, pellets, buttons, etc. to fill the canister and eliminate use of the canister in the solidification process.

- o Provide a barrier and an inert gas cover to preclude contamination of the exterior surfaces of the canister (See Figure 3-2).

(2) Replace canister due to degradation during interim storage

Alternative

- o Select a canister material capable of withstanding the environment and control storage environment to prevent unacceptable material degradation.

(3) Act as a primary barrier in the event that the canister is breached as the reprocessing facility.

Alternative

- o The FRP will be a licensed facility which is provided with containment structures and high efficiency filtering systems to contain and remove particulates and gases released from the canister. Facility operating procedures can be implemented to ensure proper control of equipment and systems.

B. Phase 2 - Shipping Cask

(1) Act as a primary barrier in the event that the canister is breached during shipping.

Alternative

- o The canisters will be transported in a shipping cask under a Nuclear Regulatory Commission license. The shipping cask will be designed and tested for normal transportation environments and design accidents. The shipping cask can be designed to serve as a sealed secondary barrier to prevent release of gases and particulates from the canister to the environment. The damaged canister can be returned to the FRP for rework and cleanup. The shipping cask can be designed with a removable liner to facilitate decontamination.

C. Phase 3 - Federal Repository

(1) Act as a primary barrier in the event that the canister is breached at the Federal repository.

Alternative

- o The Federal repository can be designed with suitable containment structures and filter systems and equipment to prevent release of particulate or gaseous activity from the canister to the environment. Facility operating procedures can be implemented to ensure proper control of equipment and systems.

(2) Provide additional resistance to corrosion during near-term geologic storage.

Alternatives

- o Provide adequate wall thickness for the canister to allow for corrosion.
- o Sleeve the geologic storage media to separate the canister from the corrosive environment.

Although the planned use of an overpack canister as an integral part of the waste management system is considered undesirable, some means of overpacking may be useful for accident situations. For example, if the canister were breached at the Federal repository, an overpack may provide a convenient containment package for transporting the damaged canister back to the FRP for recanning. Use of shipping cask liners and provision for an overpack type container for canisters at the Federal repository may appreciably increase the safety and cost effectiveness of the HLW management program.

3.6 Testing - Design Development & Design Verification

A thorough testing program, properly integrated within the overall design process, is recommended for the canister. The canister functions as a transportable, pressure-retaining barrier during all three phases of its design life. Four main concerns related to the design of the canister are described below. The relationship of test programs and the design code as tools in providing a canister that meets design objectives satisfying these concerns is also provided. Of primary concern in the design of an HLW canister is the provision of the following:

a. an acceptable level of material compatibility with design environments (waste processing and solidification, interim storage, transportation and long term storage).
b. adequacy for all design loads (e.g., internal pressure, handling loads, thermal gradients and vibration during transportation)
c. adequacy for low probability accident conditions
d. assurance of proper function or operation.

3.6.1 Environmental Testing

Testing of canister materials to determine effects of chemical and radiological environments is recommended for all stages of the canister life. This environmental testing of candidate materials should be performed to insure that the material selected for the canister is metallurgically compatible with the waste and all anticipated fabrication and design environments. Sample testing should specifically include the effects of high residual stresses that are likely to be present during the canister lifetime (due to such events as welding and differential expansion between the canister and waste). Testing of material samples that have been exposed to representative fabrication and design processes can be performed using accelerated environmental conditions. Typical design rules, such as those of the ASME III code, assume that the canister material remains ductile throughout its lifetime and this should be a verification objective of the material environmental testing. Environments considered in developing test requirements should include:

a. waste processing and solidification
b. post-fill heat treatment
c. closure and seal welding
d. interim storage
e. transportation
f. geologic storage

Special testing may be required to evaluate and control possible sensitization of some candidate materials. The material sample environmental testing should have as a goal, demonstration of adequate canister material properties at the end of the canister retrievability period.

3.6.2 Design Qualification Testing

The suitability of the canister to perform under specified design conditions such as internal and external pressure, thermal excursions, vibration, and impact situations can be demonstrated by test or analysis. Testing of the canister for most design conditions would probably be unnecessary since pressure, temperature, and mechanical load effects are readily evaluated by analysis for a ductile canister material.

3.6.3 Accident Condition Tests

During the transportation phase of the canister design life, the canister becomes part of the canister transportation package. However, during the initial and final stage of its design life the canister may be subject to accident conditions during handling and storage. Of primary concern are severe impact (e.g., canister drop) accidents. The objective of such testing is not to qualify the HLW canister for all possible accidents that the canister might experience; but rather for a level of accidents for which it need not be repaired or replaced.

Although analytical procedures are available that provide methods for evaluating severe impact accidents, acceptance criteria may be difficult to define and analytical results may be open to question. Therefore, testing is recommended to assure that the canister remains sealed as well as structurally stable following the specified accident conditions.

The types of testing that should be considered for accident conditions are:

a. Free drop of the canister onto an unyielding surface. Free drop heights and orientation should be selected to encompass all probable accident impact conditions during phase one and three of the canister design life.

b. Free drop of the canister onto an object with potential for penetrating the canister. The severity of this test should be developed considering handling and storage environments, economic constraints, etc.

c. If the canister is water cooled, it may be desirable to perform loss of coolant temperature excursion tests if considerable uncertainty exists from analysis.

d. Transportation accident tests should be carried out within a designated cask transportation package, using the hypothetical accident conditions specified in 10 CFR 71, Appendix B.

Although cask accident tests are designed to determine the adequacy of the cask containment, it is recommended that functional capability of the canister also be tested within the transportation package. The capability requirements of the canister under these circumstances should be defined in the system and design specifications.

3.6.4 Leak Test Criteria Development

Present conceptual closure designs do not rely on the closure seal weld to perform as a structural joint. However, the proper functional requirements of the seal should be established. Leak testing can then be performed by the FRP to ensure established levels of leak tightness have been met.

The establishment of leak test acceptance criteria requires studies of:

a. Potential particle size of the waste form due to breakup in the canister.
b. Waste particle escape rates as a function of canister defect hole size.
c. Surface contamination of the canister as a function of particle release rate.
d. The sensitivity of various leak rate defection methods and devices, and the effect of remote handling on equipment sensitivity.
e. The effect of canister handling and age on defect hole size.

If necessary, leak testing can be performed by purging the canister with helium before sealing, then leak testing using a helium detector. However, if an elevated internal pressure is desired as a driving force for the helium, the canister can be pierced, purged, and sealed using a small diameter laser beam. The technology to perform this operation would require adaption of test and fabrication equipment used by the nuclear fuel industry for fuel rods.

3.7 Other Design Considerations

3.7.1 Canister Stresses

If the waste is in can melted, the canister can be expected to experience secondary* type residual stresses due to the difference in the coefficient of expansion between glass and steel. However, these residual stresses are not expected to significantly degrade the design mechanical strength behavior of the candidate canister materials. As long as the canister material remains ductile and is not environmentally degraded due to such effects as stress corrosion, the residual stresses should have no significant effect on performance of ductile metals. Primary* stress loadings, which tend to cause high strains and plastic flow of the metal prior to failure, will overcome the influence of the lower order strains associated with residual stresses before any significant plastic flow occurs.

If the canister material cannot be maintained in a ductile state throughout its design lifetime, then it may be necessary to stress relieve all parts of the canister and consider mechanical devices to prevent interference stress between the glass and canister. Such devices have been examined and are briefly discussed below. (It was assumed for the basic discussion of codes and standards that a ductile metal canister material will be used).

One method of preventing the above-mentioned interference stress would be to insert a core into the canister prior to waste casting, which would crush upon waste/glass cooling and displace this filler core material. An x-shaped core dividing the glass into four segments or a liner (of this filler core) adjacent to the canister inner wall have been considered as possible configurations. If it becomes necessary to avoid such interference stresses, one candidate material would be a cera blanket (8 lbs/cu ft.) of approximately 1/2" thickness, manufactured by the Johns-Manville Corporation.

A crushable material should be capable of withstanding the casting temperature (approximately 1100°C) and be sufficiently rigid to hold its form during casting. However, the material should not crush below approximately 30 psig. In addition, the crushable material should not exhibit any gas generation properties and should have a nonporous skin to prevent saturation with liquid waste/glass mix. If the material were used as a liner in the canister, it should have high thermal conductivity.

*Primary and secondary stresses as defined by the ASME Section III code.

3.7.2 Load Combinations

It is recommended that all appropriate load combinations for the design of the canister be identified and stress limits for these conditions be specified as ASME service limits A, B, C, or D. It is suggested that the severity of load combinations and applicable ASME service limits be determined by basing it on a rational approach that evaluates the probability of occurrence of each load or load combination.

The recommendations of ANS-50 coordinating working group 2 for plant design conditions illustrates a proposed approach for power plant design conditions (Table 3-1). This approach is based on a correlation between acceptable radiation doses at a site boundary distance of 2500 meters, and probability of occurrence of design and service loads for a power plant system. It is recommended that equivalent criteria for load categorization be developed for waste canister design, based on probability of occurrence and consequences. This approach is recommended as a basis for the selection of ASME service limits for applicable loads and their combinations in the design of canisters so as to minimize the diverse and inconsistent approaches which may otherwise develop.

Loading conditions which should be considered in the design of an HLW canister should include, but not necessarily be limited to:

a. Internal/external pressure including excursions due to temperature fluctuations
b. Linear and non-linear thermal expansion effects, as applicable, due to steady state and accident transient conditions
c. Deadweight of material and contents
d. Impact loads related to earthquake, handling, transportation, and storage accident conditions
e. Significant inertial loads related to transportation

TABLE 3-1

SUMMARY OF EVENT CLASSIFICATION
SYSTEM FOR POWER PLANTS

Expected Best Estimate Frequency of Occurrence (F) Per Reactor Year	Pressure Retaining Integrity & Support Stability for ASME III Service Limit	Offsite Dose Limit
Planned Operations	A	
$F > 10^{-1}$	B	Appendix I to 10 CFR 50
$10^{-1} \geq F > 10^{-2}$	B	1% of 10 CFR 100
$10^{-2} \geq F > 10^{-2}$	C	10% of 10 CFR 100
$10^{-4} \geq F > 10^{-6}$	D	100% of 10 CFR 100

4.0 PRELIMINARY PROPOSED CLOSURE DESIGNS

In this section, some descriptions of proposed canister closure designs (with and without overpack) are presented. (Refer to Section 3.5 for discussion on canister overpack.) The wide variability of conditions (i.e., accessability, temperature, materials, environment, safeguards) which must be considered, adds complexity to these designs. As indicated in Section 1.2 of this report, the basic canister closure method specified for this study is a remote weld closure. It should be noted in further studies relative to development of canister design that mechanical type seals could also be considered.

Codes and standards that may be applied or made applicable to the canister closure welds include ASME Boiler and Pressure Vessel Code Section IX and other welding codes and standards listed in Tables 5-2 and 5-3 of this report. Some of the bases considered for the proposed closure designs are:

a. Simplicity of operations
b. Leakproofness
c. Shock resistance
d. Remote handling and installation
e. Remote decontamination
f. Remote seal welding
g. Remote viewing

Following are brief descriptions of canister closure designs with and without overpack: (Refer to Figures 4-1 and 4-2 for details.)

4.1 Closure Designs for Canister with No Overpack

Four proposed canister closure concepts without overpack are presented in Figure 4-1. Details and brief discussions of these four designs are presented in the following paragraphs:

Concept 1a — Hemispherical head covering canister with threaded coupling connection and seal welded around the coupling by remote operation. The hemispherical head on the top and probably on the bottom of the canister (where a skirt should be provided for stability purposes), is one of the most typical arrangements for this particular use. Advantages are (1) no reduction of access on top of the canister, and (2) simple installation, and (3) good shock resistance on the top and bottom.

Concept 1b — Flat cover with assumed cover thickness equal to 1/4 of the diameter of the canister. The canister shell is welded directly to the cover, tested and x-rayed prior to filling of the canister with waste. The plug is installed by remote operation and seal welded. The flat cover on top may also be used on the bottom of the canister. The 6" diameter threaded opening provides necessary access for filling. Disadvantages are: (1) decrease of access for filling, and (2) less shock resistance than 1a.

Concept 1c — Extruded cover reducing the canister diameter to the minimum opening of 6" with plug remotely installed and seal welded. This concept is similar to 1b, except it has better shock resistance. It has the same access for filling. A semi-overpack may also be provided over the cover (similar to 1d).

Concept 1d — Extruded cover reducing the canister access to a minimum opening of 6". A threaded plug is provided. This concept is a typical gas holder arrangement with reduced access for filling. Shock resistance of the cover is comparable to 1a. A semi-overpack above the cover may be provided to increase shock resistance on the top.

Concepts 1a, 1c, and 1d are considered the most suitable for the canister without overpack.

4.2 Closure Designs for Canister with Overpack

Four proposed closure concepts for the canister with overpack are presented in Figure 4-2 and the following paragraphs. These closure concepts can be applied if use of an overpack proves necessary.

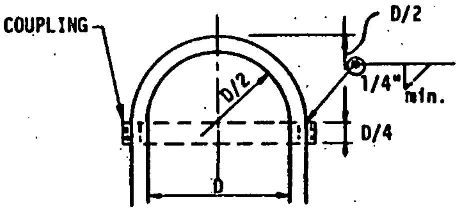
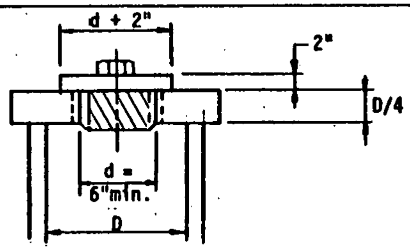
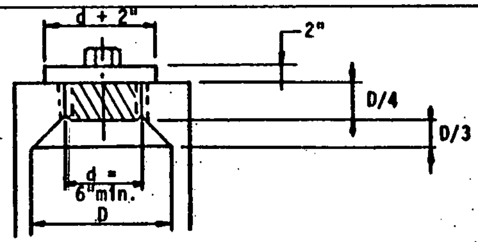
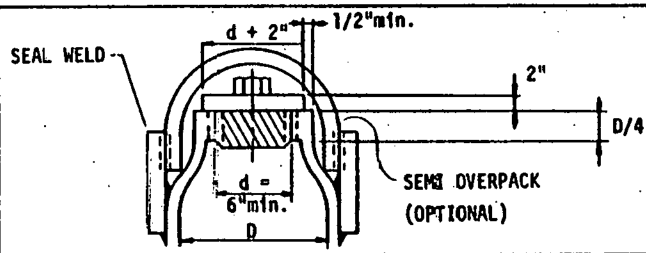
Concept 2a — This is a double shell arrangement with two identical hemispherical covers connected with couplings to the canister and overpack. A 1/4" filler seal weld is used for complete closure. Advantages are similar to those for concept 1a, which provides for full canister opening and good shock resistance.

Concept 2b — Canister similar to 1c and has an overpack with inverted cover which can be threaded and seal welded, or welded without threading by remote operation. This concept provides a heavy overpack with the canister set deep inside. Advantages include good shock protection on the ends, with the canister protected by the extended walls of the overpack, reinforced by the cover.

Concept 2c — Canister with extruded cover. The overpack has an inverted cover and can be threaded and seal welded, or welded without threading by remote operation. Advantages similar to that of Concept 2b.

Concept 2d — Canister and overpack are assembled and tested before filling. Therefore, two plugs are installed and seal welded by remote operation. One possible disadvantage of this concept would be secondary stresses caused by differential expansion between a stainless steel, Inconel and Incoloy canister and a carbon steel overpack.

Concepts 2b, 2c, and 2d are considered to be the most suitable concepts for canister with overpack.

CONCEPT NO.	TYPE & DESCRIPTION		MATERIAL	WELDING	PLUG
1	2		3	4	5
1a		HEMISPHERICAL COVER INSTALLED & SEAL WELDED BY REMOTE OPERATION	COVER & BACKING PLATE (ASTM-304L, CARBON STEEL, INCONEL, OR INCOLOY)	REMOTE WELDING PROCEDURE	NONE
1b		FLAT COVER WITH THREADED PLUG INSTALLED & WELDED BY REMOTE OPERATION	COVER & PLUG (ASTM-304L, CARBON STEEL, INCONEL, OR INCOLOY) PLUG WITH POLY LOK SEAL	SEAL WELD AROUND THE PLUG BY REMOTE WELDING PROCEDURE	MACHINED TO TOLERANCE + 1/64" THREADED ASA 8 PITCH THREAD, 4 THREAD PER INCH
1c		COMBINED COVER WITH THREADED PLUG INSTALLED & WELDED BY REMOTE OPERATION	COVER & PLUG (ASTM-304L, CARBON STEEL, INCONEL OR INCOLOY) PLUG WITH POLY-LOK SEAL	SAME AS 1b	SAME AS 1b
1d		EXTRUDED COVER WITH THREADED PLUG INSTALLED & WELDED BY REMOTE OPERATION	COVER & PLUG (ASTM-304L, CARBON STEEL, INCONEL OR INCOLOY) PLUG WITH POLY LOK SEAL	SAME AS 1b	SAME AS 1b

NOTES:

1. PLUGS FOR CONCEPT NUMBERS 1b, 1c, and 1d SEAL WELDED WITH APPROX 1/4" FILLET WELD.

Figure 4-1. Preliminary Closure Designs
Canister Without Overpack

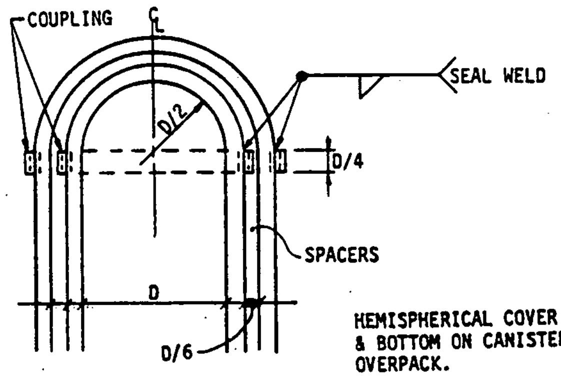
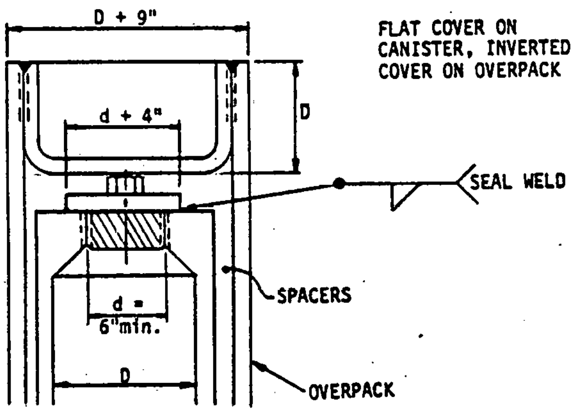
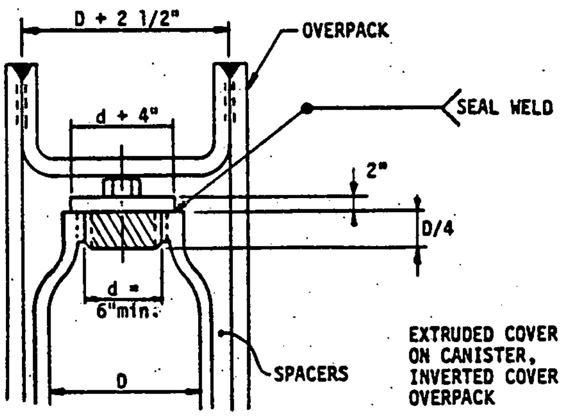
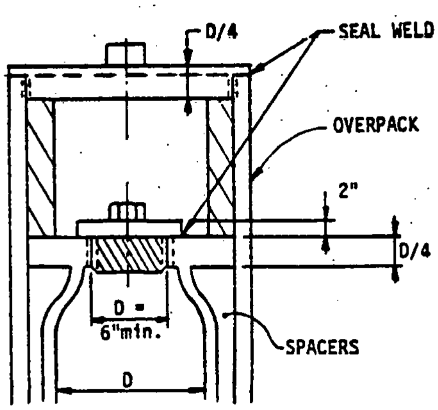
CONCEPT NO.	TYPE & DESCRIPTION	MATERIAL
2a	 HEMISPHERICAL COVER TOP & BOTTOM ON CANISTER & OVERPACK.	CANISTER, CANISTER COVER & COUPLING (ASTM-304L, CARBON STEEL, INCONEL, OR INCOLOY) OVERPACK BODY OVERPACK COVER -CARBON STEEL
2b	 FLAT COVER ON CANISTER, INVERTED COVER ON OVERPACK	CANISTER, CANISTER COVER & PLUG (ASTM-304L, CARBON STEEL, INCONEL, OR INCOLOY) OVERPACK INVERTED COVER CARBON STEEL
2c	 EXTRUDED COVER ON CANISTER, INVERTED COVER ON OVERPACK	SAME AS 2b
2d	 EXTRUDED COVER ON CANISTER FLAT COVER ON OVERPACK	CANISTER, CANISTER COVER, AND PLUG (ASTM-304L, CARBON STEEL, INCONEL, OR INCOLOY) OVERPACK & OVERPACK CYLINDER (CARBON STEEL)

Figure 4-2. Preliminary Closure Designs Canister With Overpack

5.0 CODES AND STANDARDS

5.1 Licensing for High-Level Waste Canisters

5.1.1 Licensing and Quality Assurance Programs

Radioactive material must be controlled to protect public health and safety. The Code of Federal Regulations (CFR) requires privately owned facilities that process or otherwise handle radioactive material to be licensed and to operate under a quality assurance (QA) program or the equivalent (i.e., a system of control procedures). Licensing ensures that adequate control capability is established; application of a QA program maintains proper utilization of that capability. Since licensing and QA programs work together to protect the public, they are addressed together in this section of the report.

At the present time there are some licensing and QA program requirements applicable to the waste management system. These requirements provide partial control over HLW canister applications, and have some effect on canister design. Additional licensing and QA program requirements are needed to adequately protect the public throughout the canister lifetime, and to enforce the application of verified criteria in canister design. This section describes existing requirements and indicates additional licensing that should be established for HLW canisters.

5.1.2 General Requirements for High-Level Waste Disposal

Spent fuel as removed from a nuclear reactor contains a mixture of special nuclear material* and by-product material.** A fuel reprocessing plant (FRP) processes the spent fuel to recover the special nuclear material for reuse. The by-product material residue is called "high level waste" (HLW) and is initially in liquid form.

---

* Special Nuclear Material - (a) plutonium, uranium 233, uranium enriched in the isotope 233 or in the isotope 235, and any other material which the Commission, pursuant to the provisions of section 51 of the Act, determines to be special nuclear material but does not include source material; or (b) any material artificially enriched by any of the foregoing, but does not include source material.

** By-Product Material - Radioactive material (except special nuclear material) yielded in or made radioactive by exposure to the radiation incident to the process of producing or utilizing special nuclear material.

10 CFR 50, Appendix F specifies, "...A fuel reprocessing plant's inventory of high-level liquid radioactive wastes will be limited to that produced in the prior 5 years...High-level liquid radioactive wastes shall be converted to a dry solid as required, to comply with this inventory limitation, and placed in a sealed container prior to transfer to a Federal repository in a shipping cask meeting the requirements of 10 CFR 71...All these high-level radioactive wastes shall be transferred to a Federal repository no later than 10 years following separation of fission products from the irradiated fuel..."

5.1.3 High-Level Waste Management Effects on Canister Design

Compliance with the requirements in 10 CFR 50, Appendix F, requires the waste management activities shown in Figure 3-1 and described below. Each of these activities will place constraints on the design of the HLW canister.

Phase 1. The FRP, or another licensed facility, will convert the liquid HLW to a solid form and seal it in canisters purchased from a manufacturer.

10 CFR 50, Appendix F, requires that "...The dry solid shall be chemically, thermally, and radiolytically stable to the extent that the equilibrium pressure in the sealed container will not exceed the safe operating pressure for that container during the period from canning through a minimum of 90 days after receipt (transfer of physical custody) at the Federal repository..." The methods of solidification, loading and sealing as well as chemical compatibility and gas evolution of the dry waste form and remote handling features of the reprocessing facility will have a bearing on the design and material selection for the HLW Canister.

Phase 2. The FRP, or another licensed facility, will load sealed HLW canisters in a shipping cask to form an acceptable shipping package and will arrange for transport of the shipping package to a Federal repository.

Prior to shipping, the FRP or other facility will be required to obtain a shipping package license, as prescribed in 10 CFR 71 for a "large quantity"* and based principally upon an analysis showing the capability of the shipping package to withstand the conditions defined in Paragraphs 71.32, 71.35 and 71.36 and Appendices A and B of 10 CFR 71; mark the shipping package and notify the consignee in accordance with 49 CFR 173; and transfer custody of the HLW to the Federal repository.

The design of the shipping cask and, in particular, the support structure for canisters within the cask can impose anywhere from negligible to rigid constraints on the design of the HLW canister.

Phase 3. As a minimum, the Federal repository will handle the HLW canisters to the extent necessary for removal from the shipping cask, lowering down the shaft to the mine level and locating within the storage tunnels. In addition, the Federal repository may check canisters for surface contamination and leaks.

Handling requirements at the Federal repository will have a bearing on the design of the HLW canister.

5.1.4 Existing Licensing and QA Program Requirements and Guidelines

A review of existing documents was performed to identify documented requirements applicable to licensing and QA programs in the design, manufacture, and application of HLW canisters. The types of documents reviewed included Federal regulations, Regulatory Guides, ANSI Standards (primarily the N-series), RDT Standards, and the ASME Boiler and Pressure Vessel Code (ASME Code). The design of the canister will be determined, in part, by the tradeoffs of constraints selected for the canister and its environment in each of the phases of application shown in Figure 3-1.

* 10 CFR 71.4 (f); "Large Quantity" -..."a quantity of radioactive material, the aggregate of which [in a shipping package] exceeds... 50,000 curies...[and is in] 'special form'...

Therefore, the search for applicable documents extended to those for fuel reprocessing facilities, large shipping casks for radioactive material, and anticipated handling systems at the Federal repository. The documents of interest found by this review are shown in Tables 5-1A, 5-1B and 5-1C.

5.1.5 Applicability of Existing Documents to HLW Management

Phase 1 and 2 activities will probably be performed by the FRP under a facility license per 10 CFR 50 (see Figure 5-1 and Table 5-1A). However, since the high-level waste is by-product material and contains no special nuclear material, these activities could be performed by another facility under a 10 CFR 33 facility license.

10 CFR 50 requires the licensee to operate under a QA program that complies with the criteria in Appendix B. The scope required for a license and QA program is adequately prescribed in 10 CFR 50. However, guidelines are needed to determine how to effectively comply with 10 CFR 50. Amendments to 10 CFR 71, which upgrade requirements for quality assurance of packagings, were made effective in October of 1977. These amendments included revisions to Sections 71.24, 71.51, 71.53, and 71.54, and an addition of an Appendix E, "Quality Assurance Criteria for Shipping Packages for Radioactive Material," Appendix E, is patterned closely after Appendix B of 10 CFR Part 50. These amendments should be appropriately applied to the shipping package (Phase 2). As shown in Table 5-1B, some Division 3 Regulatory Guides have been issued for fuels and material facilities and others are in preparation. In addition, American National Standards Committee N46 has been formed to establish QA Program Standards for fuel reprocessing plants.

Early balloting on proposed ANSI N46.2-series documents indicates that this series will either be very similar to the ANSI N45.2 series for nuclear power plants or the ANSI N45.2 documents will be extended to apply to both types of plants. (See also Table 5-1C.)

10 CFR 33 defines an applicable type of license called a "Type A specific license of broad scope." For a facility other than an FRP performing the Phase 1 and 2 activities above, Paragraph 33.13 of 10 CFR 33 sets rules for the applicant's organization and control procedures which should be equivalent to a QA program required in 10 CFR 50, Appendix B, in ensuring public health and safety. The new Appendix E to 10 CFR 71 adds QA program requirements specifically applicable to the canister.

Since Phase 3 activities are performed by Federal repositories owned and controlled by the Federal Government, the facility may be exempt from licensing and QA program requirements and subject only to special requirements established within the Department of Energy.

5.1.6 Recommended QA Program Bases to Ensure HLW Canister Quality

High-level waste canisters must provide high-integrity containment for radioactive material (see 5.1.1 above and 10 CFR 50, Appendix F). As related to the need for quality in manufacturing, this function coincides with the safety-related function of nuclear power plant components for which the ASME Code, Section III was established. Consequently, as discussed in Section 5.2 of this report, Nuclear Services Corporation recommends Federal regulations requiring a special license for HLW canisters as required for shipping packages and making it mandatory that HLW canisters be N-stamped.* The special license is a means by which appropriate Phase 1 (see Figure 3-1) processing and process quality control can be enforced. Compliance with the ASME Code will establish a baseline of quality for the HLW canister, and require organizations that design or fabricate HLW canisters to operate under a Code-compliance QA program and, thereby, ensure predictable quality in delivered items.

High-level waste should be stabilized prior to placement in the canister or sealing the canister. Acceptance standards should be established for the solidified waste. It should be assured that the waste meets these standards before sealing of the canister.

* Compliance with the ASME Code, Section III, Division 1 may be required, but N-stamping specifically excluded from the requirement.

FIGURE 5-1: LICENSING & ITS USE IN HIGH LEVEL WASTE (HLW) MANAGEMENT

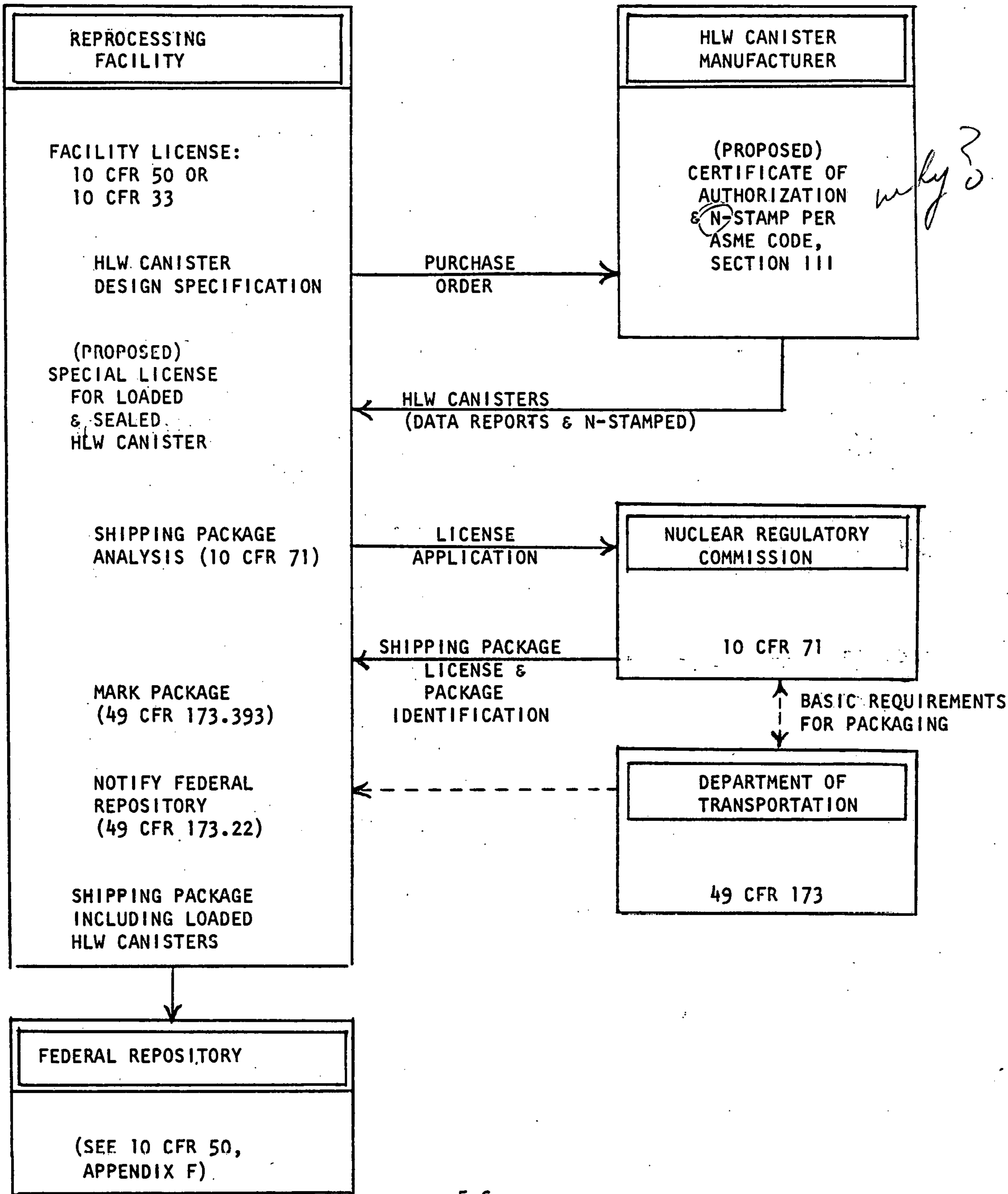


TABLE 5-1A

FEDERAL REGULATIONS
APPLICABLE TO HLW MANAGEMENT LICENSING

Requirements in Federal regulations consist of licensing and QA program criteria for a fuel reprocessing plant, general administrative controls required for other by-product material licensed facilities, special licenses for shipping packages, and shipping requirements.

10 CFR 50 Licensing of Production and Utilization Facilities (applicable to facility license for FRP complex)

10 CFR 50 App. B, Quality Assurance Program

10 CFR 50 App. F, Policy Relating to the Siting of Fuel Reprocessing Plants and Related Waste Management Facilities

Part 33.13 - Requirements for the issuance of a Type A specific license of broad scope

10 CFR 71 - Packaging of Radioactive Material for Transport....

Part 71.24	Quality Assurance
Part 71.31	General Standards for all packaging
Part 71.32	Structural standards for Type B and large quantity packaging
Part 71.34	Evaluation of a single package
Part 71.35	Standards for normal conditions of transport for a single package
Part 71.36	Standards for hypothetical accident conditions for a single package
Part 71.51	Establishment and maintenance of a quality assurance program
Part 71.53	Preliminary determinations
Part 71.54	Routine determinations

Appendix A - Normal Conditions of Transport

Appendix B - Hypothetical Accident Conditions

Appendix E - Quality Assurance Criteria for Shipping Packages for Radioactive Material

49 CFR TRANSPORTATION

Part 173.22 Shipper's Responsibility

Part 173 Shipper's General Requirements and Packagings

Part 173.389(b) Large Quantity Radioactive Material

Part 173.398 Special Tests (Same as 10 CFR 71, Appen. A&B)

TABLE 5-1B

REGULATORY GUIDES APPLICABLE TO HLW MANAGEMENT LICENSING

The Nuclear Regulatory Commission issues Regulatory Guides under ten subject headings, called divisions. The divisions of primary interest to this report, and the documents that are applicable, are listed in this Table. Also included are new Regulatory Guides of interest that are under development.

Regulatory Guides are the principal mechanism employed by the Nuclear Regulatory Commission (NRC) to implement Federal regulations. These documents state the position of the NRC, endorse more detailed documents such as ANSI Standards and the ASME Code, and generally request that licensees conform to the Guide or establish an equally effective alternate system.

Division 1 - Power Reactors

1.26 Quality Group Classifications and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants.

(Provides a basis for construction of the HLW Canister to ASME Code, Section III, Code Class 3.)

1.28 Quality Assurance Program Requirements (Design and Construction of Nuclear Power Plants). (Endorse ANSI N45.2)

Division 3 - Fuels and Materials Facilities

3.3 Quality Assurance Program Requirements for Fuel Reprocessing Plants and for Plutonium Processing and Fuel Fabrication Plants.

(Endorses a QA program for the Fuel Reprocessing facility that conforms to ANSI N45.2-1977.)

The following Regulatory Guides are under development:

a. Quality Assurance for the Design, Construction, and Operation of Fuel Reprocessing Plants
b. Guide for Design of Irradiated Fuel Receiving and Storage Facilities
c. Temporary Storage of High-Level Liquid Waste at Fuel Reprocessing Plants.

Division 7 - Transportation

7.5 Administrative Guide for Obtaining Exemptions from certain NRC Requirements Over Radioactive Material Shipments.

(Provides a mechanism for NRC licensees to obtain an exemption from DOT regulations.)

7.7 Administrative Guide for Verifying Compliance with Packaging Requirements for Shipments of Radioactive Materials.

(Endorses ANSI N14.3-1975.)

TABLE 5-1C

STANDARDS APPLICABLE TO HLW MANAGEMENT LICENSING

The American National Standards Institute issues administrative and technical standards in a number of fields. The ANSI N-series, for the nuclear power industry, is of interest for this report. The applicable issued ANSI N-Standards, and others under development and of interest, are listed. There are no RDT Standards applicable to HLW Management.

5.8-1967	Radioactive Waste Categories, Definition of
14.5 (being developed)	Leakage Tests on Packages for Shipment of Radioactive Materials
14.10.1-1973	Administrative Guide for Packaging and Transporting Radioactive Material
14.10.3-1975	Administrative Guide for Verifying Compliance with Packaging Requirements for Shipments of Radioactive Materials
15.13-1974	Fuel Reprocessing Facilities, Nuclear Material Control System (a guide to practice)
45.2 & 45.2.X Series	QA Program Requirements for Nuclear Plant (may be made applicable to Fuel Reprocessing Plants, or else 46.2 and 46.2.X series now being developed will be issued)
101.3-1972	Guide to Principal Design Criteria for Nuclear Fuel Reprocessing Facilities.
305-1974	Design Objectives for Highly Radioactive Solid Material Handling and Storage Facilities in a Reprocessing Plant.

5.2 Development and Design Verification of the HLW Canister

5.2.1 Current Status of Requirements

At the present time, as shown in Section 5.1 of this report, facility license requirements (administrative controls) exist for waste processing facilities. Shipping package license requirements are shown to be adequate for HLW canister transport. Section 3.0 shows that specific concepts are under study for the Phase 1 (see Figure 3-1) operations of high-level waste forming and canister loading and sealing, and limiting dimensions have been selected for canisters. However, since design and application criteria for HLW canisters are meager at this time, directly applicable codes and standards are limited. Those that may be applied or made applicable to the development and design verification of the HLW canister are presented in Table 5-2. This table includes some codes and standards which can be applied to fabrication and testing for the canister development and design verification phases.

The limited number of applicable codes and standards available now in no way detracts from the projected usefulness for these types of rules and guidelines. Such documents will play a major role in defining quality characteristics and providing for verification of quality achievement in HLW canisters to ensure public health and safety.

In the following paragraphs, codes and standards are briefly described and applications are recommended.

5.2.2 Codes

5.2.2.1 Purpose and Scope

Codes are intended to provide a means for ensuring a baseline of quality (i.e., safety and reliability) in all items of a type or for a given function. The scope of codes varies from a single aspect such as design to a complete control system over all elements of a specified item.

5.2.2.2 The ASME Code

The ASME Boiler and Pressure Vessel Code (ASME Code) is the most comprehensive code applicable to the design, fabrication, inspection and testing of pressure retaining components and component supports whose failure could, if not controlled, endanger the public safety.

The ASME Code currently includes eleven sections, of which five (Sections I, III, IV, VIII and X) address specific types of containers or containers with a common safety-related function; five (Sections II, V, VI, VII and IX) provide supporting rules for materials, design or selected operations; and one (Section XI) addresses inservice inspection at nuclear power plants.

5.2.2.3 Selection of Applicable ASME Code Section

The HLW canister will perform two safety-related functions:

a. Containment of radioactive material.

b. Containment of gas and vapor at modest but finite pressures (probably less than 100 psi) due to:
   (1) A temperature increase in the canister resulting in contained air heating or increase in the vapor pressure of volatile materials such as cesium and ruthenium compounds, or
   (2) Helium buildup by actinide (α,n) reactions.

Based on the containment functions, either Section III, "Nuclear Power Plant Components," Division 1, "Metal Components" or Section VIII, "Pressure Vessels," Division 1 or 2 could be selected.

Section III is specially tailored for nuclear application. The primary safety consideration is containment of radioactive material under pressure. Section III generally provides specific rules for vessel materials, fabrication, and inspection. In Section VIII the primary safety consideration is pressure containment and does not generally provide specific rules for vessel materials, fabrication, and inspection.

Consequently, Section III provides a better matchup with the canister's function and thus is recommended by Nuclear Services Corporation for application to canister manufacture (detail design, fabrication, and quality control inspection and testing during fabrication).
For application of the ASME Code, Section III, Division 1 to the manufacture of canisters, the scope statement in NA-1100 could be modified or a code case issued to include HLW canisters.

5.2.2.4 Applicable ASME Code Subsection

The ASME Code, Section III, Division 1 includes rules for three Code Classes (Subsections NB, NC and ND) and three special types of components (Subsections NE, NF and NG).

The internal pressure and corrosion, radiation effects, and external loading anticipated for the HLW canisters are similar to those for nuclear power plant components for which Regulatory Guide 1.26 recommends Code Class 3. On this basis of comparison, application of Code Class 3 for the canister is considered to be satisfactory. The existing design rules in Subsection ND are considered adequate with possibly two exceptions: (1) rules for fatigue in the event that vibration during transportation or temperature excursion cycles during filling become significant limitations on design; and (2) design rules (e.g., stress limits) for low probability accident occurrences (e.g., service level D occurrences for ASME components). In this event, fatigue rules could be applied from Subsection NB-3200 and service level D design limits for severe accidents could be applied from Appendix F of Section III. With these possible exceptions, it is not considered necessary to impose the more stringent rules of Code Classes 1 or 2 unless it is verified through subsequent analysis and testing that classification of Code Class 3 will result in questionable canister integrity.

Remote handling and high temperature gradients during canister loading and sealing, and long-term exposure to salt or other disposal mine media are unique considerations. One alternative would be to add a new Subsection NH, "HLW Canisters" to the Section III code to provide guidance to cope with these unique conditions. [The ASME previously did this for Class MC Components (Subsection NE), Component Supports (Subsection NF) and Core Support Structures (Subsection NG)].

If requirements do not vary too greatly among HLW canister models, the addition of a new Subsection NH may offer distinct advantages:

a. NH-2000, "Materials" - Specifications could be limited to materials proven by development and design verification testing to be applicable to a particular phase or the entire canister lifetime.

b. NH-3000, "Design" - Methods and data could be limited to those proven by development and design verification testing or analysis to be applicable to the canister.

c. NH-4000, 5000, 6000 and 8000 (Fabrication, Testing, Examination and Nameplates, Stamping and Reports) - These rules could be the same as the corresponding ND Articles.

d. NH-7000, "Protection Against Overpressure" - Rules of this type are presumed to be unnecessary.

Thus, there is a choice, either Code Class 3 could be selected and special requirements prescribed in standards and enforced by Regulatory Guides, or Subsection NH could be established to cover both common and special requirements applicable to all HLW canister models. However, the establishment of a new Subsection would require a few years to implement with a considerable time investment of the ASME. Nuclear Services Corporation recommends that the selection be deferred until further characterization of conditions and tradeoffs have been completed, and the scope of requirements is better understood. At that time, the advantages of one means of promulgation over the other will be easier to determine.

5.2.3 Standards

Standards are widely distributed documents addressed to a particular topic and published by one or more technical societies. Standards may, as examples, define administrative controls, technical methods, fabrication processes, quality assurance or quality control measures, or acceptance criteria. The flexibility and availability of standards make them very useful for creating quality base lines, or for assuring application of uniform, proven methods in critical activities. In the nuclear industry, the ANSI Standards and particularly the ANSI N-Standards have proven to be highly acceptable and effective. Adherence to or application of a standard in the nuclear industry is generally enforced by promulgation of a Regulatory Guide by the Nuclear Regulatory Commission (NRC) that endorses the standard. This places an obligation on licensees to adopt the standard or develop equivalent measures that are acceptable to the NRC, and to require in purchasing documents that lower-tier organizations also conform to the standard.

Testing requirements and procedures for quality control of the processing facility (see Section 5.3.3.5) should be established in ANSI N Standards.

5.2.4 Development Objectives

There is much work yet to be accomplished to develop and verify design requirements for the high-level waste canister. An essential part of this development must be a coordinated effort that will:

a. Request the ASME to extend the scope of Section III to include design and manufacture of high-level waste canisters.

b. Adopt the rules of Subsection ND (Class 3) of the ASME Code, Section III for the canisters, or create a new Subsection NH to establish an adequate baseline of rules for canister quality.

c. (1) Adapt 10 CFR 50 (or 10 CFR 33, see Section 5.1) to cover all requirements common to any type of facility licensed to perform the Phase 1 activities in Figure 3-1, or

(2) Create Federal regulations that include special license requirements for facilities that perform Phase 1 activities, including the provision of Design Specifications to Manufacturers for production of high-level waste canisters (see Figure 5-1).

d. Develop ANSI N-Standards for the unique requirements in the Phase 1 or 2 activities of Figure 3-1.

e. Develop Regulatory Guides to enforce conformance to the ASME Code and ANSI N-Standards for requirements not enforced by Federal regulations.

f. Develop Regulatory Guides (and, perhaps, ANSI N-Standards) to establish acceptance criteria for high-level waste canisters at the Federal repository.

5.2.5 Design Development Output

There are three options to be considered as the output or point of completion for HLW canister design development and design verification:

1. Detailed design documents for use in defining and manufacturing one or more fixed-design models of the canister.

2. Design Specifications (see ASME Code, NA-3250) for use in detailed design of canisters that reflect the requirements for one or more defined HLW management systems.

3. Design Criteria, verified to be in compliance with Federal regulations and codes and standards applicable to HLW canisters manufacture and use in the three phases of canister application, and that provide sufficient detail from which to derive canister Design Specifications for a particular HLW management system.

Option No. 1 appears unlikely to be selected, at least in the foreseeable future. Option No. 2 may eventually be selected, but probably not until considerable HLW management and disposal experience has been gained both by the Federal and the private organizations in the system. Thus, it is assumed that Option No. 3 will be selected for initial application and may remain the principal basis for some time.

5.2.6 Design Verification

Appendix B of 10 CFR 50 and all standards, such as ANSI N45.2 and ANSI N45.2.11, that provide criteria for design control require design verification. Each document permits design verification by document review, alternate calculations, or testing, as appropriate to the particular case. However, there is also the question of when design elements for the HLW canister should be verified.

It is assumed in this report that HLW canister Design Criteria must be the keystone for design control. Then, as described in Section 5.3, Design Specifications will be derived from the criteria and translated into detailed design documents for use in manufacturing.

Design verification for these activities should be limited to reviews and alternate calculations to verify that design information has been properly understood and correctly translated.

All design development and design verification of prototype canisters to prove the acceptability of design features and data must be performed to establish the HLW canister design criteria. In other words, all information accepted for inclusion in the HLW canister design criteria must have been previously verified for compliance and applicability. Organizations such as fuel reprocessing plants, that will use these criteria will not have appropriate facilities or the background data necessary for design verification associated with design development.

5.2.7 Design Criteria

As the keystone for design control, the HLW canister Design Criteria must account for the conditions imposed on the canister during each of the three phases of canister use. These conditions will include both normal and accident conditions associated with:

Phase 1 - Waste processing facility design, processing, methods applied for remote handling, decontamination, interim storage, and quality control.

Phase 2 - Transport conditions transmitted through the shipping cask to the HLW canister, such as thermal gradients and transportation vibration or the effects of incomplete support for the canister.

Phase 3 - Acceptance criteria for loaded canisters at the Federal repository, handling methods on the surface and underground.

As design development and design verification progresses, the work and results should be published in unclassified reports. Proven criteria may then be incorporated in Federal regulations for special licenses to use HLW canisters, the ASME Code for application in manufacturing and, as appropriate, in ANSI N-Standards enforced by Regulatory Guides.

5.3 Manufacture and Use of the HLW Canister

5.3.1 Scope and Status

This section covers application of the HLW canister Design Criteria after development and design verification of prototype canisters have been completed. It includes the preparation of Specifications (see ASME Code, NA-3250) for canister procurement, manufacturing in compliance with Section III, Division 1 of the ASME Code, canister loading, sealing and transport under special licenses, and final disposal at the Federal repository.

Most of the requirements and guidelines for HLW canister manufacture and use will derive from the development work covered in Section 3.6 and 5.2 of this report. The codes and standards that may be applied or made applicable to canister manufacture and use include those in Table 5-3, as well as those in Tables 5-1A, 5-1B, and 5-1C and the codes as presented in Table 5-2, which are related to fabrication and testing.

5.3.2 Proposed Control System

The control system for manufacture and use of HLW canisters that is proposed by Nuclear Services Corporation is outlined in Figure 5-1 and discussed in Sections 5.1 and 5.2. The proposed system may be summarized as follows:

a. HLW canister design criteria - Documented possibly in ANSI N-Standards.

b. Design Specifications for canister procurement - To be derived from the HLW canister Design Criteria and to be in compliance with the specified subsection in the ASME Code, i.e., Subsection ND or NH. Canister procurement is assumed to be the responsibility of the fuel reprocessing plant or other licensed, waste processing facility.

c. Canister manufacture - Required by the Design Specifications to be in compliance with the specified subsection in the ASME Code, Section III, Division 1 and to be performed by a manufacturer holding an appropriate ASME Certificate of Authorization. N-stamping each canister is recommended.

d. Phase 1 use of the canister (see Figure 3-1) - Required by Federal regulations to be performed under a special license, or by Regulatory Guides that obligate the fuel reprocessing plant or other licensed, waste processing facility to employ specified processes, quality control, canister handling procedures, and other measures to ensure public health and safety.

e. Phase 2 use of the canister - Required to control operations as stated in d. above. The FRP will obtain a special license for the shipping package that contains loaded canisters per 10 CFR 71 and to control transport of this package to the Federal repository per 49 CFR 173.

f. Phase 3 use of the canister - As required by the Department of Energy for operation of the Federal repository which (it is recommended) includes acceptance criteria for loaded canisters transported to the Federal facility.

5.3.3 Application of the System

When the HLW canister Design Criteria have been established and the proposed control system implemented, canister manufacture and use will proceed as follows.

5.3.3.1 The fuel reprocessing plant (FRP)* will derive Design Specifications from the Design Criteria for use in canister procurements. Prior to use in procurement, the Design Specifications will be required to be reviewed to ensure that criteria have been fully and correctly applied. This is a form of design verification.

5.3.3.2 The canister manufacturer will translate the Design Specifications into detailed design documents for use in manufacturing, as prescribed in his QA program. Prior to use, these design documents will be required to be reviewed to ensure that specifications have been fully and correctly applied. This, too, is a form of design verification.

---

*or other licensed, waste processing facility throughout this section.

5.3.3.3 The canister manufacturer will plan and control manufacturing, including procurement of material, items, and services, in accordance with his QA program. Manufacturing controls will include:

a. Fabrication per documented plans,
b. Performance of special processes per written, qualified procedures and by qualified personnel,
c. Performance and documentation of inprocess quality control, e.g., inspections, witnessing, and testing,
d. Independent inspections and document reviews by an Authorized Nuclear Inspector (ANI),
e. Performance of final inspection and pressure testing, witnessed by the ANI,
f. Preparation of Data Reports and certification by the ANI,
g. N-stamping each HLW canister, if required,
h. Shipment of completed canisters and documentary evidence of quality to the FRP.

5.3.3.4 The FRP will convert the high-level liquid waste into an acceptable dry solid form, and load this solid waste into canisters. The conversion and loading may be separate steps or combined. As discussed in Section 5.1.6, provisions should be included to verify that the chemical composition of the dry waste form meets acceptance criteria to ensure compatibility between the waste and canister. Waste should be stabilized prior to placement in the canister or sealing the canister.

5.3.3.5 The FRP will seal the canister, by mechanical means, welding, or both, to form a high-integrity containment barrier for the high-level waste. Prior to sealing it must be verified, by measuring the rate of gas evolution or other means, that any gases released by the waste will be within specified limits so as not to exceed maximum design pressure of the canister. After sealing, canisters must be leak-tested, by the use of a helium leak detector or other means, to ensure that radioactive constituents in the waste cannot be released to air or water in concentrations exceeding those specified in 10 CFR 20.

Sealed HLW canisters must be tested for removable surface contamination, by swipe-tests or other means, to ensure that acceptance criteria at the FRP, for insertion into the shipping cask, and for acceptance at the Federal repository are met. If excess contamination is detected, the canister should be decontaminated before any further handling is permitted.

5.3.3.6 The FRP will obtain a special license for the shipping package consisting of the shipping cask, internal canister supports and loaded, sealed canisters. The special license will be obtained from the Nuclear Regulatory Commission on the basis of an analysis of the capability of the package to withstand the Appendix A and B conditions and other loading conditions as prescribed in 10 CFR 71.

5.3.3.7 The FRP will identify the shipping package, notify the consignee (Federal repository) and arrange for transport in accordance with 49 CFR 173.

5.3.3.8 The Federal repository will probably conduct receiving inspection of loaded canisters to ensure compliance with acceptance criteria imposed by the repository. Accepted canisters will be handled within the Federal repository facility and placed in underground media for disposal, in accordance with rules to be generated by the Department of Energy.

TABLE 5-2
CODES AND STANDARDS
CANISTER DEVELOPMENT AND DESIGN VERIFICATION

CODE, STANDARD OR REGULATION (CSR)	APPLICABLE PARAGRAPH OF DOCUMENT	DESCRIPTION OF REVISIONS NEEDED IN CSR	COMMENTS ON DOCUMENTS REVIEWED
Reg. Guide 1.26, Quality Group Classifications and Standards for Water-Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants (Rev. 3, 2/76)	All (2)	New CSR required. Include applicability to HLW canister.	For nuclear power plants. However, it is considered this could be applied to canisters. Gives guidance on code classification of components. Implies canister would be in quality standard Group C (ASME Sec. III, Class 3).
Reg. Guide 1.31 Control of Ferrite Content in Stainless Steel Weld Metal	All	NA	10 CFR 50 Appendix B requires that measures be established to ensure materials control, special processes (such as welding) control, and proper testing performance. This guide describes a method acceptable to the NRC staff for implementing these requirements with regard to the control of welding in fabricating and joining austenitic stainless steel components and systems.
Reg. Guide 1.50 Control of Preheat Temperature for Welding of Low Alloy Steel	All	NA	This guide describes an acceptable method of implementing the 10 CFR 50 Appendix B requirement that measures be established to assure control of materials and of special processes such as welding and that proper process monitoring be performed, with regard to the control of welding for low-alloy steel components during initial fabrication.

NOTES:

(1) Fully Applicable
(2) Partially Applicable
NA Not Applicable

TABLE 5-2 (Cont)

CODE, STANDARD OR REGULATION (CSR)	APPLICABLE PARAGRAPH OF DOCUMENT	DESCRIPTION OF REVISIONS NEEDED IN CSR	COMMENTS ON DOCUMENTS REVIEWED
Reg. Guide 3.28 Welder Qualification for Welding in Areas of Limited Accessibility in Fuel Reprocessing Plants and in Plutonium Processing and Fuel Fabrication Plants	All	NA	This document is similar to Reg. Guide 1.71 and could apply to the seal weld of the filled canister at the fuel reprocessing plant and to the seal weld of the overpack of one is used.
Reg. Guide 3.29 Preheat and Interpass Temperature Control for the Welding of Low-Alloy Steel for Use in Fuel Reprocessing Plants and in Plutonium Processing and Fuel Fabrication Plants	All	NA	This guide describes a method acceptable to the NRC staff for meeting the requirements of 10 CFR 50 Appendix B and 10 CFR 70 Paragraphs 70.22(f) and 70.23(b) with regard to the control of welding of low-alloy steel components for fuel reprocessing plants and for plutonium processing and fuel fabrication plants. This document could apply to the canister as part of the fuel reprocessing plant.
Reg. Guide 3.36 Nondestructive Examination of Tubular Products for Use in Fuel Reprocessing Plants and in Plutonium Processing and Fuel Fabrication Plants	All	NA	This guide describes a method acceptable to the NRC staff for meeting the requirements of 10 CFR 50 Appendix B and 10 CFR 70 Paragraphs 70.22(f) and 70.23(b) with regard to specifying, in the interest of standardization, procedures acceptable to the NRC staff for the nondestructive examination of high-integrity tubular products.

NOTES:

(1) Fully Applicable
(2) Partially Applicable
NA Not Applicable

TABLE 5-2 (Cont)

CODE, STANDARD OR REGULATION (CSR)	APPLICABLE PARAGRAPH OF DOCUMENT	DESCRIPTION OF REVISIONS NEEDED IN CSR	COMMENTS ON DOCUMENTS REVIEWED
Reg. Guide 3.37, Guidance for Avoiding Intergranular Corrosion and Stress Corrosion in Austenitic Stainless Steel Components of Fuel Reprocessing Plants	C2 through C7(2)	NA	Provides guidance on qualification testing and other guidance for avoiding integranular corrosion and stress corrosion in austenitic stainless steel components of fuel reprocessing plants. Precludes the use of sensitized steel components.
Reg. Guide 7.6 Stress Allowables for the Design of Shipping Cask Containment Vessels	All	NA	Provide stress allowables for the design of shipping cask containment vessels. Reg. Guide 7.6 states that there are no design standards for evaluation of the structural integrity of cask containment vessels. Therefore, the staff has adapted portions of Section III of the ASME Code to form acceptable design criteria. Normal conditions → service limit A (as adapted) Accident conditions → service limit D (as adapted) NOTE: Containment vessel is defined as the receptacle on which principle reliance is placed to retain the radioactive material during shipment. Does not allow buckling; accident resultant primary membrane stresses must be maintained below the lesser of 2.4 Sm or .7 Su; membrane plus bending must be less than 3.6 Sm and Su.

NOTES:

(1) Fully Applicable
(2) Partially Applicable
NA Not Applicable

TABLE 5-2 (Cont)

CODE, STANDARD OR REGULATION (CSR)	APPLICABLE PARAGRAPH OF DOCUMENT	DESCRIPTION OF REVISIONS NEEDED IN CSR	COMMENTS ON DOCUMENTS REVIEWED
Reg. Guide 7.8 Load Combinations for the Structural Analysis of Shipping Casks	All	NA	Load combinations for the structural analysis of shipping casks. A copy of the suggested load combinations for normal and accident conditions of transport is provided.
ASME BPV-III-1-ND, Nuclear Power Plant Components, Class 3	(See below)		If the canister is designed as an ASME Class 3 component, then this Subsection ND applies. (See remarks under Reg. Guide 1.26 in this compilation.)
Material	ND-2000$^{(1)}$	NA	Includes various tests required for materials of components, but does not apply to the design of a specific item.
Design	ND-3000$^{(2)}$	Revised CSR required. Depending upon the final process evolved for filling the canister, it may become necessary to revise this article to include treatment of design and testing entailing the process temperatures involved. More likely, a specific Code Case could be written for this if an existing one does not cover the process that will be evolved. (See discussion of ASME Code Cases N-47 through N-50 below.)	Applies to components that are intended for use at temperatures not greater than given in Tables I-7.0 and I-8.0 Appendix I. This is not inconsistent with the 350°C (662°F) assumed as equilibrium canister temperature during early storage period, but some of proposed canister filling processes will entail temperatures in excess of those given in the Tables.

NOTES:

(1) Fully Applicable
(2) Partially Applicable
NA Not Applicable

TABLE 5-2 (Cont)

CODE, STANDARD OR REGULATION (CSR)	APPLICABLE PARAGRAPH OF DOCUMENT	DESCRIPTION OF REVISIONS NEEDED IN CSR	COMMENTS ON DOCUMENTS REVIEWED
ASME BPV-III-1-ND, Nuclear Power Plant Components, Class 3 (Cont)			
Design (Cont)	ND-3131.2(1)	NA	Calls for proof tests to establish maximum allowable pressure if design cannot be adequately determined by rules given for analysis. (See ND-6900.)
Fabrication	ND-4000(1)	Depends on welding process used.	Article ND 4000 titled "Fabrication and Installation" contains subsection ND-4100 General Requirements covering acquisition of acceptable material for fabrication; Subsection ND-4200 "Forming, Fitting, and Aligning", covering material cutting, forming, and binding; forming tolerances, fitting and aligning of parts, requirements for welded joints in components, and welding end transitions; subsection ND-4300 Welding Qualifications covering types of processes permitted, welding qualifications, records, and identifying stamps, and general requirements for welding procedure qualification tests; Subsection ND-4400 rules governing making, examining, and repairing welds; Subsection ND-4500 brazing; Subsection ND-4600 heat treatment including welding preheat and postweld heat treatment; Subsection ND-4700 mechanical joints and Subsection ND-4800 expansion joints.

NOTES:

(1) Fully Applicable
(2) Partially Applicable
NA Not Applicable

TABLE 5-2 (Cont)

CODE, STANDARD OR REGULATION (CSR)	APPLICABLE PARAGRAPH OF DOCUMENT	DESCRIPTION OF REVISIONS NEEDED IN CSR	COMMENTS ON DOCUMENTS REVIEWED
ASME BPV-III-1-ND Nuclear Power Plant Components, Class 3 (Cont)			
Examination	ND-5000 In ND-5200 thru ND5242 it depends upon category chosen. All the rest except paragraphs: ND-5260, ND-5272, ND-5274 ND-5276, ND-5280 thru ND-5283.7 ND-5700 and ND-5720	NA	NDE requirements in addition to those listed in Section V.
Testing	ND-6000[1]	NA	Gives requirements and procedures for pressure testing of components for usual and special situations.
Protection Against Overpressure	ND-7000	NA	Gives requirements for protection against overpressure and for testing pressure relieving devices. However, it is considered undesirable to have any such device on the HLW canister.
Nameplates, Stamping and Reports	ND-8000[1]	NA	-

NOTES:

(1) Fully Applicable
(2) Partially Applicable
NA Not Applicable

TABLE 5-2 (Cont)

CODE, STANDARD OR REGULATION (CSR)	APPLICABLE PARAGRAPH OF DOCUMENT	DESCRIPTION OF REVISIONS NEEDED IN CSR	COMMENTS ON DOCUMENTS REVIEWED
ASME BPV-III-1-A, Nuclear Power Plant Components, Appendices	Article II-1000(1)	NA	Provides rules for substantiating, by experimental stress analysis, the critical or governing stress in parts for which theoretical stress analysis is inadequate or for which design rules are unavailable.
ASME Code Case 47, Class 1 Components in Elevated Temperature Service	(2)	Revised CSR required. Depending upon the process and service temperatures involved in the filling and sealing process for the HLW canister, this Code Case may provide adequate guidance for the canister design and testing. If not, it should be modified as necessary to cover the canister case.	Provides guidance for design and testing of Class 1 components for elevated temperature service. Can be applied to Class 3 components.
	3226(1)	NA	Gives pressure testing limitations.
ASME Code Case 48, Fabrication and Installation of Elevated Temperature Components	(2)	NA	Provides rules for fabrication and installation of ASME Section III, Class 1 (and therefore Class 3) elevated temperature components, including various materials qualification testing.

NOTES:

(1) Fully Applicable
(2) Partially Applicable
NA Not Applicable

TABLE 5-2 (Cont)

CODE, STANDARD OR REGULATION (CSR)	APPLICABLE PARAGRAPH OF DOCUMENT	DESCRIPTION OF REVISIONS NEEDED IN CSR	COMMENTS ON DOCUMENTS REVIEWED
ASME Code Case 49, Examination of Elevated Temperature Nuclear Components		NA	Refers to nondestructive examination of Section III, Class 1 (and therefore Class 3) elevated temperature components. Does not apply to design verification.
ASME Code Case 50, Testing of Elevated Temperature Components	6117[2] 6118 6200	Revised CSR required to permit testing at temperatures which exceed those for which S_m values are listed in Table I-1.0 However, such testing at service temperature would only be needed if satisfactory design verification could not be achieved analytically or by extrapolation of ambient temperature testing.	Gives rules for testing of closure welds, specially designed welded seals, and hydrostatic tests at elevated temperatures during the construction of Section III, Class 1 (and therefore Class 3) elevated temperature components.
ANSI N14.5 - 1974, Leakage Tests on Packages for Shipment of Radioactive Materials	6.2[1]	NA	Gives leakage test requirements for design verification, permissible leakage, and suggested test methods.
ANSI N101.3 - 1973, Guide to Principal Design Criteria for Nuclear Reprocessing Facilities		New CSR required. Similar document specially for HLW canister.	Provides guidance in the preparation of General Design Criteria (GDC) for nuclear fuel reprocessing facilities.

NOTES:

(1) Fully Applicable
(2) Partially Applicable
NA Not Applicable

TABLE 5-2 (Cont)

CODE, STANDARD OR REGULATION (CSR)	APPLICABLE PARAGRAPH OF DOCUMENT	DESCRIPTION OF REVISIONS NEEDED IN CSR	COMMENTS ON DOCUMENTS REVIEWED
ANSI N145-1973 Effects of High-Energy Radiation on the Mechanical Properties of Metallic Materials	All	NA	None
ANSI N305-1975, Design Objectives for Highly Radio-Active Solid Material Handling and Storage Facilities in a Reprocessing Plant	-	New CSR required. Include design objectives, criteria and verification requirements for HLW canister.	Defines design objectives related to facilities for the handling and storage of highly radioactive materials in solid form. However, this reference does not treat design verification.
ASTM E 8-1972 Tension Testing of Metallic Materials	All	NA	None
ASTM E 184-62(1968) Effects of High-Energy Radiation on the Mechanical Properties of Metallic Materials, Standard Rec. Practice for	All	NA	None
ASTM E208-69(1975) Conducting Drop-Weight Test to Determine Nil-ductility Transition Temperature of Ferritic Steels	All	NA	None

NOTES:

(1) Fully Applicable
(2) Partially Applicable
NA Not Applicable

TABLE 5-2 (Cont)

CODE, STANDARD OR REGULATION (CSR)	APPLICABLE PARAGRAPH OF DOCUMENT	DESCRIPTION OF REVISIONS NEEDED IN CSR	COMMENTS ON DOCUMENTS REVIEWED
ASTM A 262-75 Susceptibility to Intergranular Attack in Stainless Steels, Standard Rec. Practices for Detecting	All	NA	None

NOTES:

(1) Fully Applicable
(2) Partially Applicable
NA Not Applicable

TABLE 5-3
CODES AND STANDARDS
CANISTER MANUFACTURE AND USE

CODE, STANDARD OR REGULATION (CSR)	APPLICABLE PARAGRAPH OF DOCUMENT	DESCRIPTION OF REVISIONS NEEDED IN CSR	COMMENTS ON DOCUMENTS REVIEWED
Reg. Guide 1.71 Welder Qualification for Areas of Limited Accessibility	All	NA	This regulatory guide describes qualification and requalification required when the welders physical and visual accessibility to a weld are restricted.
ASME BPV-II Material Specifications	A,C	NA	Applicable to ordering of canister raw material. Part A Ferrous Materials Part C Welding Rods, Electrodes and Filler Metals Applicable parts dependent on choice of canister material.
ASME BPV-V Nondestructive Examination	Depends on method used	NA	
ASME BPV-IX Welding and Brazing Qualifications	Depends on welding process used.		Section IX of the Code would be used as the qualification standard for welding and brazing procedures, welders, brazers, and welding and brazing operators.
ANSI N45.2.11-1974	6(1)	NA	Establishes detailed requirements for design verification. Possibly useful to FRP in writing design specifications for canister.

NOTES:

(1) Fully Applicable
(2) Partially Applicable
NA Not Applicable

TABLE 5-3 (Cont)

CODE, STANDARD OR REGULATION (CSR)	APPLICABLE PARAGRAPH OF DOCUMENT	DESCRIPTION OF REVISIONS NEEDED IN CSR	COMMENTS ON DOCUMENTS REVIEWED
ANSI N45.2.13-1974, QA Requirements for Control of Procurement of Items and Services for Nuclear Power Plants	10.3.1(2) 10.3.3 A.4.d	Revision in CSR required. Include mention of fuel reprocessing and waste management facilities.	Treats procurement of items and includes design verification by supplier and qualification testing and acceptance testing at source. Possibly useful to FRP for processing of canister.
ASNDT			
SNT-TC-1A-1975 Recommended Practice for Nondestructive Testing Personnel Qualification and Certification	All	NA	None

NOTES:

(1) Fully Applicable
(2) Partially Applicable
NA Not Applicable

6.0 CONCLUSIONS AND RECOMMENDATIONS

6.1 Licensing and Regulatory Guides

Existing Federal regulations provide a basis for licensing and control of high-level waste (HLW) management. Appendix F of 10 CFR 50 clearly defines the overall HLW management policy. However, there is a need for further development of specific regulatory requirements and acceptance criteria. For example, it is recommended that requirements be established for a special license, or requirements be made obligatory through Regulatory Guides to:

a. Develop Design Criteria that are based on realistic and cost effective tradeoffs between canister design and materials, and the environments for which the canister is exposed during its three phases of use. These Design Criteria should delineate overall objectives, functional requirements, and acceptance criteria for the various HLW management options.

b. Develop a detailed Design Specification for design and fabrication of the HLW canister, which implements the requirements of the Design Criteria for a specific HLW Management System.

c. Require that HLW canisters be manufactured in accordance with the ASME Code Section III, Division 1, rules for either Code Class 3 vessels or in accordance with a new Code Section III subsection to be developed.

d. Control the solidification procedure for the HL waste and stabilize the waste which is placed in the HLW canister, to comply with requirements of 10 CFR 50, Appendix F. This will require the application of verified processes and processing to established acceptance criteria.

e. Control the seal welding for the HLW canisters in accordance with qualified procedures, including verification that acceptance criteria have been met.

It is further recommended that the Department of Energy establish acceptance criteria for HLW canisters at the Federal repository which will be reflected back into the canister design criteria.

6.2 Codes

It has been concluded that the canister should be designed as a pressure vessel with no provisions for overpressure protection devices such as relief valves or rupture discs. Since the canister is to be not only a pressure vessel but also a container of radioactive material, it is concluded that ASME Code, Section III, Division 1 rules are the most appropriate for detailed design and manufacture of the canister.

Code Class 3 rules (Subsection ND) provide for acceptable canister quality. However, it may prove advantageous to establish a new Subsection NH specifically for HLW canisters that include the general Code Class 3 rules, and also requirements reflecting the unique conditions, processes and acceptance criteria from the three phases of canister application. It is recommended that the selection of Subsection ND or NH be deferred until later in the Rockwell Hanford Operations Project.

6.3 Standards

From this study a determination was made that a limited number of existing standards of the type, ANSI N, are pertinent to canister design, manufacture, and testing. However, it is recommended that ANSI N-Standards, specifically addressed to aspects of canister design, application and quality control, and acceptance for disposal, be generated and enforced by appropriate Regulatory Guides.

6.4 Design Criteria

In consideration of the potential output options at the Rockwell Hanford Operations Project for completion of HLW canister design development and design verification, establishment of Design Criteria is recommended to be the first step. These Design Criteria must be the keystone for canister design input and control.

It is recommended that the canister Design Criteria represent the constraints applicable to the canister resulting from realistic tradeoffs between canister design and requirements for (1) the waste processing facility, canister loading, and interim storage; (2) the shipping cask with loaded canister; and, (3) Federal repository canister handling and disposal. The Design Criteria should provide an adequate basis for conversion into Design Specifications. Acceptance criteria should be established for receipt of loaded canisters for disposal at the Federal repository.

6.5 Canister Design Features

Although this study deals with a seal-weld type canister closure, mechanical closure designs should also be included in follow-on investigations. A thorough study should be made of welding techniques to ensure that high integrity seal welds can be made by remote means, using remotely operated equipment. Remote nondestructive examination will also be required due to the high-level radiation of the loaded canister. The canister should be leak tested after closure (by use of helium leak detector or other means) to ensure that radioactive materials are adequately contained.

For this study an assumption has been made that suitable materials for canister fabrication include carbon steel, stainless steel, and Inconel and Incoloy series steels. A thorough follow-on study should be made of requirements and tradeoffs for each of the three phases of canister application to determine the most suitable material for canister fabrication.

It has been concluded that the use of an overpack as an integral part of the canister design results in an undesirable design and economic constraint on the waste management system. A thorough study should be made relative to overpacking of the canister in the form of removable shipping cask liners or containers holding more than one canister in disposal before a determination is made as to whether or not to employ any overpacking. The study should include such considerations as: (1) the necessity to construct an overpack to acceptance criteria equivalent to the canister, (2) usefulness in interim storage, (3) heat transfer problems, (4) advantages in canister handling, testing, transport, or geologic storage; and, (5) cost effectiveness.

7.0 REFERENCES

1. High-Level Waste Canister Envelope Study, Structural Analysis, Contract No. CPFF-626, Under Prime Contract No. EY-76-C-06-2130, Kaiser Engineers Report RHO-C-8, November, 1977.

2. Topical Report on Analysis of the Factors that Impact the Reliability of High Level Waste Canister Materials, W. K. Boyd and A. M. Hall, Battelle - Columbus Report RHO-C-7, September 19, 1977.

3. Proceedings of the International Symposium on the Management of Wastes from the LWR Fuel Cycle, CONF-76-0701, Denver, Colorado, July 11-16, 1976, sponsored by ERDA, arranged by ORNC. (Article on Waste-Solidification Technology, U.S.A. - J. L. McElroy, W. F. Bonner, and J. E. Mendel, Battelle Pacific Northwest Laboratories).

4. WASH-1539, Environmental Statement, Management of Commercial High Level and Transuranium-Contaminated Radioactive Waste, USAEC, September 1974.

Other documents used as reference material in preparing this report include the following:

a. WASH-1297, High-Level Radioactive Waste Management Alternatives, USAEC, May 1974.
b. Preliminary Engineering Studies of Shipping Casks for Transporting Canisters of Solidified High-Level Radioactive Waste, Final Report RHO-C-6, G. R. Bray and J. L. Ridihalgh, Science Applications, October 24, 1977.
c. Solidification and Disposal of High-Level Radioactive Wastes in the United States, Reactor Technology, Volume 13, No. 4, Winter 1970-71, K. J. Schneider.
d. Heat Transfer Analysis of the High Level Waste Canister - TRW, J. M. Bell, J. W. Nienberg, M. H. Seidman, L. C. Fink, Prime Contract EY-77-C-06-1030, Report RHO-C-9, November 2, 1977.

e. Determination of Performance Criteria for High-Level Solidified Nuclear Waste, NUREG-0279, J. J. Cohen et al, Lawrence Livermore Laboratory, July 1977.

f. The Disposal of Radioactive Wastes from Fission Reactors, Scientific American, Volume 236, Number 6, B. L. Cohen, June 1977.

g. Preliminary Review of ICPP Solids Storage Hazards, ICP-1038, G. E. Lohse, October 1968.

h. WASH-1279, Directory of Packagings for Transportation of Radioactive Materials, Division of Waste Management and Transportation, USAEC, October 1973.