FINAL REPORT

AMCO 02-R-280

MASTER

SODIUM-COOLED LMFBR CASK

RECOMMENDATIONS

January 1977

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1.0 INTRODUCTION

1.1 I-(182)-1 LMFBR Spent Fuel Shipping Cask

1.1.1 General

In April of 1970 a design study to establish the parameters of a shipping cask for LMFBR Spent Fuel Assemblies from FFTF and the first demonstration plant was initiated. The basic criteria presented were that the cask should be limited to 75 tons, and that the cask should be compatible with the FFTF Fuel Assembly design and the first demonstration LMFBR Fuel Assembly design. The study effort on LMFBR fuel reprocessing conducted by ORNL served, initially, as a primary guide, with respect to the features of seals, coolants and decay time. Other study efforts, requested by AEC from Savannah River and ARHCO were also applied to the design criteria. Inherent in the total effort were the requirements of FFTF with respect to cask handling and interface loading because of the status of the FFTF design at that time. Several features of the I-(182)-1 cask were defined on the basis of the above guidance plus additional features directed by the AEC. These features and their basis are briefly discussed below.

1.1.2 Coolant Selection

Although sodium, as the primary coolant, is outstanding and a most appropriate coolant for shipping LMFBR fuel, there remained the problem of obtaining design approval because;

1. Sodium leakage could result in additional fire energy input.
2. Cesium 137, one of the predominant fission product isotopes, is highly soluble in sodium.

Sodium was rejected as the primary coolant by AEC guidance.

Several alternatives were considered, including eutectics of lead and other metals, casting in zinc and aluminum, polyphenyl eutectics, some salt type eutectics, and gases, especially helium. With due consideration to the economics of these various coolants, especially operational requirements including cleaning (of sodium) and preparation for shipment, the organic biphenyl was selected with the biphenyl ether-biphenyl (Dowtherm A), reserved as an alternate. ORNL experiments

with biphenyl indicated that it was incompatible with sodium wetting the fuel assembly. Dowtherm A was found to be satisfactory and was chosen in spite of the problem of compatibility with respect to the reprocessing chemistry.

1.1.3 Canister Concept

The concept of a second containment vessel, the canister of I-(182)-1, was chosen on the basis of its secondary containment feature. The canister serves as a secondary barrier to release of radioactive material. In addition the canister concept was accepted by AEC on the basis of its potential capability. It might be separately loaded with fuel and installed in the shielded cask later, with a minimum holdup time between receiving and shipment of the cask at the reactor site.

A later observation was made that the gap between the canister and the cask shield actually served as a resistance limiting the heat input, from the 30 minute fire incident, to the coolant. Thus the pressure buildup in the canister occurs at a much slower rate than would be with the direct thermal contact from the exterior surface to the coolant.

1.1.4 Rotary Magazine

The rotary magazine concept was a direct result of designing the cask to be most compatible with the fuel handling operations at FFTF. The concept was established before the fuel handling procedure and facility at the demonstration plant was conceived. It was assumed that the fuel receiving would be into a hot cell but, because FFTF was established first, the hot cell interface with the loading port and rotary magazine was established as an adjustment to the cask design.

1.1.5 Coolant Expansion Void

A "zero" leakage goal for the cask design and the requirement that fuel pins be covered with coolant at all times led to the concept of the syphon connected coolant expansion void in the center of the rotary magazine. Early study work involved the concept of multiple diaphragm covered elements which would be squeezed to provide an expansion volume. With the selection of Dowtherm A as the coolant,

fabrication considerations and the requirement to verify, before each shipment, that the expansion volume was available required that this concept be dropped. The syphon connected volume can be checked and, most important, the evolution of radiolytic gas or the release of fission product gas can be accommodated without exposing fuel pins.

Thermal analyses of the I-(182)-1 cask in the fire accident indicated that the expansion volume was large enough to take the heat input of a two hour fire without excessive internal pressures. Thus no relief valves were provided in the initial design submittal to NRC. It was proposed, to NRC, that in an extended duration fire a relief valve effect would be obtained by stretching of the head cover bolts. This was rejected by NRC and relief valves were required and incorporated into the latest cask design.

1.1.6 Depleted Uranium Shield

The depleted uranium shield concept was initially based on just the extra shielding required over the fuel region of the cask. The fuel region was 4 feet in the longest core of any LMFBR Fuel Assembly concept. A reduction in shielding on either end, with due consideration to potential fuel redistribution in an accident situation, was considered. Later the AEC specified that the uranium should be extended as far (axially) as practical so that the cask might be used, as much as possible, for other fuel or radioactive material shipments. The alternative of an all steel cask (ref. ORNL-TM-3689) limited the payload with the AEC specified limit of 75 tons for the total cask weight.

1.1.7 Passive Cooling System

From the beginning of the cask study task it was directed by AEC that heat removal be by passive cooling means, namely fins and natural air convection flow. Early concepts of the cask cooling system assumed a staggered array of fins around the cask for optimum heat transfer in the horizontal shipping attitude. After review by the AEC it was directed that the fins be continuous so as to limit

the potential buildup of radioactive contamination and provide a smoother surface for decontamination effectiveness. The fin system, as finally designed, was continuous, as required, and as closely spaced as deemed practical for weld assembly. Carbon steel, with a conductivity of 28 BTU/ft-hr-°F, was specified over that of stainless steel, 9 BTU/ft-hr-°F.

Painting was specified to protect the surface.

1.1.8 Energy Absorber

The energy absorber design study for the I-(182)-1 cask covered several concepts starting with the tube-in-tube work of ORNL arranged as toroids on the ends of the cask. Fabrication difficulties and weight disadvantage ruled against the tube-in-toroid concept.

A state of the art heavy end "fin" concept was considered and found to result in very high G impact loadings.

A foamed aluminum (Ethyl Corp. developmental material) material was selected for testing by ORNL. Material of 20 ± 2 lb/ft³ was ordered in the form of two 4'x4'x1' blocks. The material delivered was three 3'x3'x1' blocks with a density range of 10 lb/ft³ at the center to 50 lb/ft³ at the edge. The concept had to be rejected due to the lack of uniformity in the foamed aluminum.

An alternate foamed material, polyurethane, was considered for awhile but the AEC then directed that the cask be designed with a non-combustible energy absorber. The ring configuration selected was based on drop experiments done on similar configurations by ORNL.

The Nuclear Regulatory Commission, with its first review of the I-(182)-1 SAR, requested that the hypothetical accident impact target surface be of such a yield strength as to result in maximum damage to the package. The side rings and end cylinders were relocated to give a longer stroke and also provided with "pads" on the ends to give a larger contact area on a low yield impact target (e.g. concrete).

1.1.9 Neutron Shield

The low neutron source strength of FFTF fuel (1×10^7 n/s per assembly) did not require shielding. The I-(182)-1 cask has no neutron shield. The plan, as directed by AEC, was to provide for the later addition of Water Extended Polyester (WEP) in the shroud of the shipping skid-enclosure. An amended SAR would be submitted to NRC to obtain design approval of the I-(182)-2 with WEP neutron shielding in the shroud at a later date.

Early work on the LMFBR cask study involved consideration of neutron shield materials suggested by ORNL. Preliminary consideration was given to the requirement for retaining the shielding in the fire accident or selecting a material which would be sacrificed in the fire, thus limiting the heat input to the cask. Typically these criteria resulted in a weight disadvantage due to the lesser neutron attenuation capability relative to water.

An exterior water neutron shield was rejected by AEC due to concern about loss of water in normal transport. Also, adding a neutron shield at a later date was more readily accomplished if it were in the enclosure shroud rather than a part of the cask structure.

1.2 I-(182)-1 Design Review Committee Recommendations

In August 1976 a design review committee met to review the I-(182)-1 cask design and make recommendations for future work on shipment of LMFBR spent fuel. Among the design features of concern were several which would dictate significant modifications to the I-(182)-1 design. These were:

1. Dowtherm A Coolant Selection.

Potential contamination of fuel handling and storage equipment at the reactor site. Special provisions for cleaning at the reprocessing site.

2. Neutron Shielding

Neutron shielding in the enclosure-shroud does not provide protection for workers at the loading and unloading site.

3. Cooling System

The fin and side energy absorber configuration does not allow adequate air circulation during loading-unloading or with an unfavorable cask attitude following an accident.

4. Manufacturing

Very expensive.

5. Operations/Maintenance

Time consuming.

Recommendations were made to overcome these objections. In view of the later decision to terminate redesign and licensing effort on the I-(182)-1 cask (October 1976) the applicable recommendations made by the committee were:

1. Pursue a sodium cooled modification of the I-(182)-1 design in order to accept higher power/shorter decayed CRBRP fuel assemblies.
2. Provide neutron shielding as an integral part of the cask.
3. Modify the fin and energy absorber design to assure adequate cooling following an accident.
4. To assure that ERDA and user organization needs are being met and to enhance data exchange and communication between responsible organizations, an LMFBR spent fuel shipping coordination meeting should be held at least semiannually. These meetings would be coordinated by ERDA Headquarters.
5. A detailed study of time, equipment, and manpower requirements for cask handling operations is needed. This study should look at reactor operations, reprocessing plant operations, and maintenance/preparation and QA operations.

1.3 Sodium Cooled LMFBR Spent Fuel Shipping Cask Design Criteria

1.3.1 Configuration

- a. Top Loading
- b. Double Closure (minimum)
- c. Removable Magazine (basket) Assembly
- d. Internal Coolant Expansion Chamber
- e. Integral Neutron Shield

- f. Integral Protective Structure (based on site handling accident considerations)

1.3.2 Transport

Design for unrestricted railway transport. Provide a skid-enclosure design with energy absorbing and load spreading structure, in conjunction with the cask structure, to assure containment under any credible accident incident.

1.3.3 Size

- a. Crane handling limit - 100 tons
- b. Railroad car capacity - 120 tons (including cask, skid-enclosure and auxiliary equipment)
- c. Skid-enclosure width - 10 feet maximum
- d. Length - for maximum anticipated LMFBR fuel assembly
- e. Payload - maximum number of assemblies fitting within above constraints.

1.3.4 Shielding (Ref. 49 CFR 173.393)

- a. Gamma shielding based on LMFBR fuel assemblies of 7.4 Mwt, 3 year operation at 80% power factor, 60 day decay
- b. Neutron shielding based on source strength of 2×10^6 n/s kg mixed oxide fuel

1.3.5 Decay Heat Removal

Provide passive cooling means, to the maximum extent practical, to assure containment following any credible accident situation. The maximum decay heat payload shall be based on the above assured containment.

Provide auxiliary cooling and heating means to satisfy operating requirements at the loading-unloading sites.

1.3.6 Regulations

Satisfy the requirements of the Nuclear Regulatory Commission as specified in 10 CFR 71, in applicable Nuclear Regulatory guides and as guided by NRC in the course of their Safety Analysis Report review.

1.4 Safety Analysis Report

1.4.1 General Preparation

The NRC, in its review of the I-(182)-1 Safety Analysis Report, was most critical with respect to the Structural Evaluation apparently because of the somewhat novel energy absorber concept and the complication of the canister and rotary magazine concepts. Complete detail involving generally accepted material properties, combined loads, rigorous analysis in lieu of experimental data and justification of all assumptions was requested. The other chapters were reviewed less critically but with no less concern for the possible release of radioactive material.

The I-(182)-1 Safety Analysis Report was the first to be received by NRC with their suggested format. The only modification made by NRC was - "The complete set of fabrication drawings should be omitted from the SAR and replaced with a composite sufficient for safety analysis purposes only." Seven fold out sheets were used to satisfy this NRC request. Additional sketches, with the separate analysis sheets, completed the necessary detail for NRC review purposes.

With due consideration of the experience with NRC on the review of the I-(182)-1 cask it should be possible to obtain design approval with only a reasonable re-submittal response effort following the first submittal. The following subsections summarize NRC requests made for additional information and also suggest guidelines for the preparation of the Safety Analysis Report on the Sodium Cooled LMFBR Spent Fuel Shipping Cask.

1.4.2 Chapter 0.0 General Information

The only NRC request was that "the total amount of fission products that would be available for release" be shown. The initial listing of kilocurie amounts was inadequate. The revised listing, down to curie quantities, was accepted. The data used was from an ORIGEN, Oak Ridge Isotope Generator, computer program.

The several other revisions made to Chapter 0.0 were necessary due to

design modifications arising from NRC criticisms in other chapters.

1.4.3 Structural Evaluation

The most critical review by NRC is on Chapter 1.0 Structural Evaluation. This is probably due as much to the complexity of the system as the seriousness of errors or offhand approximations and assumptions made in modeling and analyzing the cask system. With respect to the complexity, NRC requires an aid in following the design and evaluation. Their format includes the opening subsection, 1.1 Structural Design, divided into two parts: 1.1.1 Discussion, and 1.1.2 Design Criteria. It is suggested that the subsection 1.1 be arranged in an orderly fashion (related to the following analytical subsections in the same order) with reference to the composite drawings(s). Each point, part, subassembly and assembly to be analyzed should be identified and imposed loads under each different set of conditions (referred to in the regulations) tabulated with criteria for design acceptance, reference to subsection and page number of analyses, and results of the analysis.

The composite drawing(s) of the cask (also skid, enclosure and railroad car if they are vital to the structural evaluation) should be complete with respect to all parameters subject to the safety analysis. The composite drawing(s) must be controlled as a part of the Safety Analysis Report. Therefore, the fine details related to fabrication and assembly should be omitted from the composite so as to not encumber the NRC design approval process with amendments to the Safety Analysis Report during the manufacturing phase.

Having aided NRC in following through the complex structural evaluation with the above suggested format there remains the problem of quality or accuracy of the analysis. NRC may require a most rigorous analysis using finite difference computer techniques. Modeling of the cask, as it involves approximations and assumptions, must be justified, and, where the results may be critical, evaluated with respect to accuracy. Each subsection involving different models for analysis purposes should be prefaced with a model description and discussion to justify

the model. The chapter 1.0 appendix, subsection 1.10, may be used to avoid cumbersome or repetitive verbage, but the preface should be made with a reference to the appendix subsection and a summary of key points involved in the model. In essence, having identified and justified the analysis models, the applicant is more confident of his initial SAR submittal and NRC can complete its review process quickly.

1.4.4 Chapter 2.0 Thermal Evaluation

1.4.4.1 Cask Internal Pressure

All source of gas for estimating the cask pressure must be accounted for - fuel pin pressurant, fission product gases, cover gas, radiolytic production (Dowtherm A coolant) and coolant vapor pressure. Obtain an analytical expression for the cask pressure using all gas sources, volumes involved (including fuel pin gas plena), coolant expansion and container expansion. Differentiate to determine the sensitivity of various measurements and estimates.

If there are other closed volumes within the cask, determine the pressures there also. The I-(182)-1 cask has four regions - canister, between canister and shield, between closure heads and the depleted uranium insert cavity.

NRC required that the I-(182)-1 cask be provided with pressure relief devices regardless of the capability to meet the 30-minute hypothetical accident fire without exceeding pressure rating. "The cask has two pressure containing boundaries, both of which must be violated before the coolant or contents could be released. However, neither of these boundaries is equipped with a device to relieve excessive internal pressure. Since we are dealing with a liquid-filled, pressurized cavity, we believe that these boundaries should be provided with relief devices set above the maximum internal pressures expected as a result of the hypothetical accident conditions, but below the failure point of the boundaries. We find the solution proposed in your previous submittal, which relies on elastic stretching of the lid bolts as a means of releasing high pressures that could result for accidents more severe than the design basis accident, to be unacceptable."

1.4.4.2 Coolant Internal Expansion Void

An illustrated description of the coolant expansion system and its functioning under the different operations is important. These include tests to assure that the expansion chamber is available for use, coolant fill and topping procedures, in transit normal variations of temperature and attitude and an accident evaluation. With respect to the accident the thermal analysis of the

I-(182)-1 cask was done with the assumption that the expansion chamber was ruptured and the two upper fuel assemblies exposed to the cover gas, as with the cask horizontal.

1.4.4.3 Thermal Analysis

The SINDA, System Improved Numerical Differencing Analyzer, thermal analysis computer program used for the I-(182)-1 cask was accepted by NRC without comment. The detailed modeling of the cask and, particularly, the cases modeled were subject to critical review. "Solar heating of the cask should be included in the thermal analysis if the enclosure is to be omitted from the package definition. The cask designer's guide recommendation of a 24-hour average of 144 BTU/ft² hr to the subtended area of the cask may be used for steady state analysis. Assume an emissivity that might be expected with a soiled surface." "Provide a steady state analysis of the cask after the accident. Use a black surface emissivity as would be expected after a fire. Assume the syphon system is inoperative so as to result in the upper magazine positions being exposed to the cover gas."

"Provide an estimate of the coolant air temperature reaching the cask as might be expected with the cask on the skid and the enclosure in place."

It should be noted that NRC acceptance of the thermal analysis in the SAR is only relative to design approval. Final acceptance and licensing requires experimental verification of the cask heat removal capability as presented in section 7.1.6. Licensing would be for a heat load to be determined by test and the "Thermal Acceptance Criteria" of section 7.1.6.3 as evidenced by the NRC comment, "The thermal acceptance criteria of Section 7.1.6.3 should be clarified and the basis for the criteria established".

1.4.4.4 Thermal Stress

Additional comments from NRC regarding the thermal analysis were made with reference to the chapter 1.0 Structural Evaluation. It is required that the thermal and pressure data presented in chapter 2.0 be in appropriate detail

and clearly identified for the stress analysis. Cross-referencing should be made in both chapters.

1.4.5 Chapter 3.0 Containment

Refer to subsection 1.4.9 for applicable discussion.

1.4.6 Chapter 4.0 Shielding Evaluation

Since the application for design approval of the I-(182)-1 involved a low neutron source and no neutron shielding, the analysis of the Water Extended Polyester shielding in the skid-enclosure should have been omitted. This was done in Revision 1.

The only other comment by NRC was that the contribution of the coolant to the gamma attenuation should be omitted from the accident case due to exposure of some of the fuel with rupture of the coolant expansion chamber.

1.4.7 Chapter 5.0 Criticality Evaluation

Battelle-Columbus performed the criticality analysis for the I-(182)-1 SAR. The Battelle report was summarized for the SAR. NRC requested that the Battelle report be submitted as a supplement to the SAR.

The cask design feature involving eighteen poison columns in the rotary magazine was questioned with respect to disadvantage factors not being used. Since these poison columns were only proposed to reduce the subcritical multiplication of neutrons, they were eliminated for the Revision 3 submittal.

NRC commented that "Hanson-Roach cross sections are not acceptable for heterogeneous systems. The LMFBR cask criticality is satisfactory because of the large margin of $k_{eff} = 0.66$ ". Without the poison columns $k_{eff} = 0.74$.

NRC found an error in the fuel loading and fuel diameter used by Battelle in analyzing the benchmark experiment. Though this error was in the conservative direction, they did request a correction.

1.4.8 Chapter 6.0 Operating Procedures

The only NRC commentary on the operating procedures was related to 6.1.5.2 Topping Canister with Coolant and Preliminary Leak Check.

"Provide details, and specific instructions for the coolant fill procedure."

"Provide the equations used to calculate the coolant fill volume."

"Estimate the variations in the fill volume due to measurement and calculational errors. Determine the significance of these variations on the cavity pressures."

These requests were satisfied by expanding the subsections on cask preparation, 6.1.3.5 Partially Fill Canister with Coolant, the referenced subsection and Chapter 2.0, subsection 2.4.4.6 Internal Pressure Sensitivity was added.

1.4.9 Chapter 7.0 Acceptance Tests and Maintenance Program

Leak Testing (7.1.3)

"Determine the leak test sensitivity necessary to satisfy the leakage requirements of Regulatory Guide 7.4 for normal and accident conditions. Show that the proposed acceptance leak test sensitivity satisfies these requirements."

Component Tests (7.1.4)

"Show that the component acceptance leak test sensitivities satisfy the leakage requirements of Regulatory Guide 7.4."

Based on the above request and Regulatory Guide 7.4, which refers to ANSI N14.5 "Draft American Standard for Leakage Tests on Packages for Shipment of Radioactive Materials", Chapter 3.0 Containment was revised completely in Revision 4.

It should be noted that the acceptable helium leakage rates are to be used to determine leakage of radioactive material. Thus, it is impossible to verify a zero leakage of radioactive material without specifying an infinite leak test of absolute zero sensitivity. The leakages specified in Chapter 3.0 (Table 3.2) are based on the Plutonium containment requirement. The test leakage acceptance rates were determined and found to be reasonable with respect to available instrumentation, techniques and test times. Then, using the specified leakage allowance, the leak rates calculated in Chapter 3.0 were determined.

Acceptance Criteria (7.1.6.3)

"The thermal acceptance criteria is not stated with sufficient precision to enable our evaluation. The criteria should be stated so that significant parameters are specified. These should include: (1) the test heat load (at least equal to that requested for approval), (2) the temperature locations measured, (3) the method of comparison to the analytically predicted results, and (4) the basis for acceptance or rejection of a cask."

Valves and Gaskets on Containment Vessel (7.2.4)

"Provide a replacement schedule for all valve seals and gaskets. Although tests and inspection of these components are provided for in the maintenance plan, a replacement schedule, based on a reasonable expected service life, is necessary. Service to failure is not an acceptable maintenance plan for these components."

Thermal (7.2.6)

"Discuss in detail the method proposed to monitor the thermal performance of cask under continued use. Provide specifications for those procedures. Include in the specifications: (1) test or measurement procedures, (2) data monitored, (3) method of performance evaluation, and (4) the basis used for acceptance or rejection of a cask for continued use."

Revision 4 apparently satisfied NRC with respect to these items.

1.4.10 Chapter 8.0 Quality Assurance

The Quality Assurance chapter was submitted with Revision 4. Apparently NRC was satisfied since no comments nor requests for further information was made.

2.0 Sodium Cooled LMFBR Cask

2.1 General Description

The Sodium Cooled LMFBR Cask, illustrated in Figure 1, is an 80 ton (see 3.1) cylindrical shield assembly sized to hold eighteen LMFBR fuel assemblies within a fixed magazine in the cavity. Shielding is a combination of steel and depleted uranium for gammas and an outer water jacket for neutrons. The shielding is sized for 60-day decay spent fuel. Cooling fins on the exterior of the water jacket consist of rectangular pieces of stainless steel clad copper in a staggered spiral array which allows natural convection air flow with the cask in any orientation. Containment of the sodium coolant and radioactive contents is assured by the two closure heads and energy absorbing members on the cask exterior to prevent damage of the seals resulting from any credible accident situation.

The cask is designed to interface with a hot cell, after removal of the outer closure head, for loading and unloading operations. The neutron shield water jacket is provided with fittings to allow water circulation for reducing exterior working surface temperatures to tolerable levels after loading or prior to unloading operations. Steam can be circulated through the water jacket to allow liquid sodium filling (prior to fuel loading) or removal (after fuel unloading) for cask servicing.

The shipping skid and enclosure (not illustrated) would be comparable to the design for the I-(182)-1 cask with additional structural members such that credible railway transport accidents could not breach the cask seals nor rupture the containment vessel.

2.2 Magazine

The magazine is a hexagonal array of eighteen fuel positions, plus a central position, and the region around the nose pieces isolated to provide for sodium expansion in the event of a fire accident (see Figure 2). Vertical tubes at the corners of each hexagonal position are connected with perforated webs to

form a supporting structure holding hexagonal LMFBR fuel assemblies of 5.6 inches across the flats and 172 inches in length, with bowing of as much as 1 inch. The vertical tubes may contain a neutron poison to suppress subcritical neutron multiplication and also assure a subcritical condition in case of accidental compaction of a full loading of fuel.

Sodium volume changes under normal temperature variations are accommodated by an ullage which does not expose fuel pins in the horizontal attitude at 208°F. Analysis of the sodium temperature versus cover gas pressure indicates that normal transport extremes can be accommodated. The expansion volume of the magazine assembly is isolated by a burst disc and relief valve set at 500 psi. In the event of a fire accident, maximum pressure is 500 psi with the average sodium temperature at 1200°F (see section 3.2).

2.3 Gamma Shield

Gamma shielding is provided in the source region of the fuel by depleted uranium contained within a steel structure, which also serves as the containment structure. The uranium extends axially sufficient to assure required shielding in the event of a maximum credible fuel shift in an accident situation (11-inches forward and 4 inches rearward shifts possible). Beyond the full uranium thickness, it is tapered so that in combination with the steel the activated fuel assembly structure is adequately shielded (see 3.3).

The shielding indicated in Figure 1 satisfies the requirement for 60-day decay CRBRP fuel. Thirty-day decay fuel can be accommodated by increasing the uranium thickness by 0.5 inch, with a corresponding reduction in the outer steel shell thickness.

2.4 Neutron Shield

The neutron shield has been selected as boronated water-glycol which is not only the best, most economical, neutron shield available but is also a good natural convection heat transfer medium. The neutron shield annulus is established by a set of clad copper tubes which also serve as heat conduction paths in the

event of a loss of water accident. The outer shell is sufficiently thick to protect against loss of water under normal handling and transport incidents. An estimate of the neutron attenuation is given in section 3.4. The heat transfer conductance without the water in place is given in section 3.5.

It is noteworthy that the water shield concept as illustrated serves as a limiting heat transfer medium in a fire accident. Normal temperature variations in the water volume are accommodated by the expansion volume shown in the end view (Figure 1). In the event of a fire, the water is lost through a relief valve thus limiting the thermal conductance to that provided by the spacer tubes. Thus, heating of the sodium coolant is considerably less than alternate non-sacrificial neutron shield materials.

2.5 Cooling System

A passive cooling system comprised of many short fin pieces in a spiral, staggered array on the surface of the neutron shield water jacket is proposed (see detail C-C on Figure 1). The fin pieces are assumed to be nickel clad copper elements torch brazed to the water jacket. The spacing indicated is as close as practical for this assembly operation. The spiral staggered array is proposed as a configuration which is satisfactory for any cask orientation which may result from an accident situation. The condition analyzed (see section 3.5) is with the cask horizontal, as for transport. A peak surface temperature (base of fins on water jacket shell) of 270°F is estimated for an air temperature of 130°F and a decay heat load of 230,000 BTU/hr (twelve 60-day heat fuel assemblies and six blanket assemblies from the CRBR). The average sodium coolant temperature appears to be well under 400°F even under loss of neutron shield water conditions.

It is not practical to design a passive cooling system to limit the cask surface temperature to 180°F . It is suggested that the neutron shield water be circulated through a heat exchanger to reduce the cask temperature to tolerable working levels. This may be done only at the loading and unloading sites with auxiliary equipment, or it can be done continuously during transport. This latter

alternative allows the cask to be unloaded in a shorter time after receiving but at the expense of additional, complex equipment on the R.R. car transporter. If the cask is allowed to get hot in transit, a shroud is required to prevent uncontrolled access to the cask.

2.6 Energy Absorber

The energy absorber structures are illustrated in Figure 1 schematically to indicate a concept. The dimensions are not necessarily valid for the actual requirements. As shown, the ribs of the side energy absorbers can take up the energy of a sixty-foot drop (42 mph) without bottoming out on the neutron shield jacket. This implies that the impact load is to the energy absorbers as would be the case with a side drop onto a flat, unyielding surface.

3.0 Sodium Cooled Cask Analyses

3.1 Weight

3.1.1 Magazine (Reference Figure 2)

- o 48 S.S. tubes 1.75" O.D., .25" wall, 153" long
- o Boron carbide/graphite poison filled
- o 66 S.S. webs 2.87" x .25" x 150" reduced by 1500 2.5" holes
- o 6 S.S. walls 3" x .25" x 152"
- o 48 end caps and 1 central end closure 4900 lb
- o S.S. base assembly including grid plate, 4" nose piece tubes, side cylinder and base 1800 lb

Expansion Void (internal)

Central column 9120 in³

Base assembly 12440 in³

~ 12.5 ft³

Magazine Displacement

(Including void) 52780 in³

~ 30.5 ft³

3.1.2 Payload and Coolant

- o 18 CRBRP Fuel Assemblies
     1795 in³ displacement (each)
     18.7 ft³ (total)   8300 lb
- o Sodium   4200 lb
     72.7 ft³ (at 210°F, P = 57.8 lb/ft²)

Ullage, above fuel pins with the cask horizontal, allows normal temperature-density changes in sodium with a nominal internal pressure.

(See Section 3.2)   5.5 ft³

3.1.3 Shield Assembly

- o Uranium 41.25" I.D. Tapered (See Figure 1)
     71,655 in³   47,860 lb
- o S.S. Structure
     Payload cavity and interhead space
         218,852 in² (126.7 ft³) + 1963 in³
     Closure end edge wedge   2,482 in³
     Closure cyl (to taper)   62,864 in³
     Tapered portion   14,403 in³
     Cylinder   344,139 in³
     Base edge wedge   2,050 in³
     S.S. total 188,468 in³   53,713 lb
- o 41 copper tubes 5" O.D., .20" wall, 152" long
     + .01" Ni clad   20,588 in³   6,640 lb
- o S.S. Jacket   .5" wall, 154" long + end closures
     18,040 in³   5,140 lb
- o Neutron shield - water/glycol
     135,500 in³ (at p = 56.5 lb/ft²)   4,430 lb

o 4320 fin pieces 3.25" x .25" x 6"	
.01" Ni clad on copper	16,110 in ³
	6,730 lb
o Side energy absorber	
54 S.S. Ribs 1" x 12.31" x 11.5"	
54 S.S. Ribs 1" x 8.07" x 11.5"	
12,389 in ³	3,534 lb
2 S.S. Rings 80" O.D. 78" I.D. x 6" (?)	
3,723 in ³	1,061 lb
Shield Assembly Total	129,110 lb

3.1.4 End Energy Absorbers

Steel contained polyurethane foam.

Closure end 66" d x 20"

3,200 lb

Base end 55" d x 25"

2,600 lb

3.1.5 Weight Summary

Magazine	6,700
Payload	8,300
Sodium	4,200
Shield Assembly	129,110*
End Energy Absorbers	<u>5,800</u>
Total	~154,110 lb

*Includes

Depleted Uranium 47,860 lb

Water/Glycol 4,430 lb

The weight may be increased to obtain more heat transfer area from the fin pieces (see section 3.5) and to provide better protection of the fins by more side energy absorber structure. These could bring the weight up to 160,000 lb.

3.2 Sodium Expansion Evaluation

Assuming an internal helium gas pressure of 10 psi at 210°F in the 5.5 ft³ ullage volume the gas quantity is 2.57 SCF. The internal pressure with this gas and an increasing sodium temperature would conform to:

$$P(\bar{T}) = P_0 V_0 (460 + \bar{T}) / 492 (V_c - W_{Na} / \rho(\bar{T}))$$

$$\rho(\bar{T}) = 59.5 - .00827\bar{T} \quad \text{Sodium Density}$$

$$V_c = 78.2(1 + \alpha T)^3 \quad \text{Sodium + Ullage Volume}$$

$$P_0 = 14.7 \text{ psi}, V_0 = 2.57 \text{ ft}^3$$

$$W_{Na} = 4200 \text{ lb} \quad \text{Sodium Loading}$$

$$\alpha = 9.7 \times 10^{-6} / ^\circ\text{F} \quad \text{S.S. 304 expansion coef.}$$

$$\bar{T} \text{ } ^\circ\text{F} = \text{Sodium \& Helium Temperature}$$

$$P(\bar{T}) = \text{Internal Pressure psi}$$

$$P(500^\circ\text{F}) = 21 \text{ psi}$$

$$P(840^\circ\text{F}) = 500 \text{ psi}$$

If the fuel pins should all be ruptured, the fission gas plena volume of 3.2 SCF is added to the cask ullage void. Also, the initial helium of 3.2 SCF and the fission product gases of 77.6 SCF is added to the cover gas quantity.

$$V_c = 81.4 (1 + \alpha T)^3$$

$$V_0 = 83.4 \text{ ft}^3$$

$$P(500^\circ\text{F}) = 355 \text{ psi}$$

$$\therefore P(637^\circ\text{F}) = 500 \text{ psi}$$

Thus, in case of an extreme shock loading, the expansion volume burst disc could be ruptured by the pressure of the additional internal gas. Having opened the expansion volume of 12.5 ft³, there is an additional gas volume of 12.5 SCF.

$$V_c = 93.9 (1 + \alpha T)^3$$

$$V_0 = 95.9 \text{ ft}^3$$

$$\therefore P(1000^\circ\text{F}) = 285 \text{ psi}$$

$$P(1371^\circ\text{F}) = 500 \text{ psi}$$

3.3 Gamma Shielding

The uranium shielding is extended with a reducing thickness beyond the blanket region of the fuel in the casks so as to limit the dose rate from the activated ends of the fuel assemblies to less than 10 mR/hr. at the edge of the railroad car. The handling socket of the FFTF fuel assembly is exposed to a larger activation than that of the CRBR assembly. Consider eighteen of these sources and the shield geometry near the cask closure end.

$$S_i = 3.00 \times 10^{13} \text{ Mev/sec} \quad \bar{E} = 1.27 \text{ Mev}$$

$$l = 25 \text{ cm} \quad \mu \rho Z = .71 \text{ self-absorption}$$

Source points are assumed at $Z = 15 \text{ cm}$ from the inside closure head and at the magazine coordinate positions $r = 20.3 \text{ cm}, 35.2 \text{ cm}, 40.6 \text{ cm} \quad \theta = 0, 30, 60, \dots$

Shielding consists of an average of 2 cm depleted uranium at $\bar{r} = 53 \text{ cm}$ and 30.7 cm steel at $51 \text{ cm} < r < 84 \text{ cm}$. At 1.27 Mev the attenuation parameters are, respectively, .065 and .051 cm^2/gm .

$$b = \mu \rho Z + \mu \rho X_u + \mu \rho X_{Fe} = 15.44$$

The dose rate from one source point

$$D_i = \frac{S B(b_i) e^{-b_i l}}{4 \pi R_i^2 C} \quad C = 600 \frac{\text{Mev/sec cm}^2}{\text{mR/hr}}$$

$$B(b_i) = 1 + .8457 b_i C^{.0411 b_i l} \quad (\text{Fe at 1.27 Mev})$$

$$R_i = 96.5 \text{ cm}$$

$$D_i = 2.16 \text{ mR/hr}$$

The other sources would add to a total of 10 mR/hr (calculation not included) reduced somewhat by the magazine structure, and reduced considerably if the cavity is filled with sodium.

Activation of the shield-orifice region of the CRBR fuel assembly is expected to be comparable to the above so that the uranium shielding shown (Fig. 1) is satisfactory. It may be desirable to shape the uranium insert differently with due consideration to source distribution, fabrication and assembly. Such design details would require more effort than could be accomplished for this report. As illustrated, the shielding is conservative.

The fueled region of the cask is shielded by the following materials:

Self-shielding	3.1 cm Fe equiv.
Uranium	11.1 cm
Steel	8.9 cm Fe
Copper & Water	1.2 cm Fe equiv.
Magazine-sodium	1.37 gm/cm^3 *

*Homogenized over total cavity including payload displacement volume to be used with point source locations and distance through cask cavity to shield wall.

The work sheets (next two pages) show a dose rate, for 60-day decay CRBR fuel, is less than 10 MR/hr at 3-ft. from the cask surface.

* P 3ft from cask surface
 $X = 0, Y = 193.04$

Report Title				Program	
				Work Order	
Prepared By	Date	Checked By	Date	Revision	Date

i	X_s	X_c	X_v	X_{F_2}	R	θ
1	3.1	15.24	11.1	10.1	157.84	0
2	3.1	86.36	11.1	10.1	228.24	0
3	3.1	32.51	11.4	10.4	175.73	3.31
4	3.1	67.82	11.4	10.4	210.88	2.76
5	3.1	12.70	14.7	13.4	159.14	7.34
6	3.1	48.26	14.7	13.4	194.11	6.01
7	3.1	83.82	14.7	13.4	229.14	5.09
8	3.1	25.91	16.3	14.8	178.07	9.86
9	3.1	63.50	15.7	14.3	212.83	8.23
10	3.1	39.12	18.3	16.7	197.27	11.89

Report Title				Program	
				Work Order	
Prepared By	Date	Checked By	Date	Revision	Date

Consider the three energy groups

j	E	μ_{Fe}	μ_U	a_1	a_2	C	S(60d)
1	1.5E	.045	.056	.2854	-.0219	620	8.50(14)
2	2.00	.040	.047	.3240	-.0084	660	1.90(14)
3	2.40	.038	.044	.3056	+.0039	700	1.34(14)

$$b = \sum \mu p x |_A \quad B(b) = 1 + a_1 b e^{a_2 b}$$

$$D = \frac{S/C}{4\pi R^2} B e^{-b}$$

i	b'	B	D _{1.5}	b	B	D _{2.0}	b	B	D _{2.4}
1	17.26, 4.37	.61	14.77, 5.23	1.86	13.89, 5.48	3.10			
2	21.53, 4.84	0	18.56, 6.15	.02	17.50, 6.73	.05			
3	18.72, 4.55	.12	16.05, 5.54	.44	15.11, 5.90	.80			
4	20.84, 4.77	.01	17.93, 6.00	.05	16.90, 6.52	.10			
5	22.07, 4.88	.01	18.85, 6.21	.04	17.73, 6.80	.08			
6	24.20, 5.07	0	20.75, 6.65	0	19.53, 7.44	.01			
7	26.34, 5.22	0	22.64, 7.07	0	21.33, 8.08	0			
8	25.04, 5.13	0	21.41, 6.79	0	20.14, 7.66	.01			
9	26.49, 5.23	0	22.73, 7.08	0	21.40, 8.11	0			
10	28.61, 5.36	0	24.48, 7.46	0	23.03, 8.70	0			
		0.9			3.0				5.2

Total Gamma Dose Rate

3ft from Cask Surface $< 10 \frac{mR}{hr}$

3.4 Neutron Shielding

An estimate of the neutron dose rate using the ORNL-TM-3503 (p.38) empirical formula and the Figure 1 parameters is as follows:

$$4\pi r^2 D(r) = e^{-\Sigma} c r_c \left[A_m (1 - e^{-\Sigma_m t_g}) e^{-\Sigma_e t_n} e^{-\Sigma_m t_o} + B_m e^{-\Sigma_m t_g} e^{-\Sigma_f t_n} \right]$$

Uranium Parameters

= neutron dose kernel, cm^2 $[(\text{mrem/hr})/(\text{neutrons/sec})]$

51.12 = r_c = inside radius of the cask, cm,

11.1 + 8.0 = t_g = thickness of the gamma shield, cm, (uranium + steel + C_u)

10.6 = t_n = thickness of the neutron shield, cm, (water equivalent)

2.1 = t_o = thickness of the outer steel liner, cm, (C_u tube + shell)

.015 = Σ_c = neutron absorption coefficient for the cask internals
(normally 0.015 cm^{-1} but 0.0225 cm^{-1} if the coolant is lithium or Dowtherm),

.169 & .102 = Σ_m = removal coefficient for the gamma shield (0.102, 0.070, and 0.169 cm^{-1} for iron, lead, and uranium).

.56 = Σ_e = removal coefficient of degraded neutrons in the neutron shield (0.52 , 0.35 , and 0.56 cm^{-1} for gamma shields of iron, lead, and uranium with a borated water neutron shield)

.14 = Σ_f = removal coefficient of uncollided neutrons in the neutron shield (0.14 cm^{-1} for a borated water shield),

.067 = A_m = dose conversion factor for degraded neutrons $[0.067, 0.10, \text{ and } 0.0139 (\text{mrem/hr})/(\text{neutrons cm}^{-2} \text{ sec}^{-1}) \text{ for iron, lead, and uranium}]$,

.12 = B_m = dose conversion factor for uncollided neutrons $[0.12 (\text{mrem/hr})/(\text{neutrons cm}^{-2} \text{ sec}^{-1})]$.

$$4\pi r^2 D(r) = .00092$$

Using a source strength of 2×10^9 n/s per metric ton and a subcritical neutron multiplication of 4 for a full load of 18 CRBR assemblies we have

$$S_n = 5.33 \times 10^9 \text{ n/s}$$

and, at

$$r = 193 \text{ cm}$$

$$D_n = 10.4 \text{ mrem/hr}$$

Total dose rate at 3 ft from the cask surface is

$$D_{\gamma} = 9.1 \text{ mR/hr}$$

$$D_n = \underline{10.4} \text{ mrem/hr}$$

$$D = 19.5 \text{ mrem/hr}$$

At 6 ft the total is estimated as

$$D \approx 9 \text{ mrem/hr}$$

which satisfies the regulatory requirement (49CFR179.393 j(3)).

3.5 Heat Transfer

Heat transfer is estimated on the basis of a payload of twelve 60-day decay CRBR fuel assemblies (5.24 kW each) and six blanket assemblies (0.75 kW each) for a total of 230,000 BTU/hr. Assuming a shroud around the cask with an 80-in. inside diameter and low-pressure-loss air access and exit at the bottom and top of the horizontal cask, the following natural convection flow parameters apply: Assume that the air flow is at 45° to the axis as directed by the fin orientation.

$$A_{\max} = L(R_o - R_i)2 \sin 45^\circ = 169 \times 7 \times 2 \sin 45^\circ / 144 = 11.62 \text{ ft}^2$$

The area is reduced by the fins

$$\Delta A_1 / A_1 = .25 \times 6 / (3.017 + 2.715) \times .5 \times 7 = .0748$$

$$\Delta A = 0.869 \text{ ft}^2 \quad A_{\min} = 10.75 \text{ ft}^2$$

The perimeter is the sum of the inner, outer, and fin piece lines. For each element

$$\text{Per}_1 = (3.017 + 2.715 + 2 \times 6) / 12 = 1.4777$$

$$A_1 = (3.017 + 2.715) \times .5 \times 7 / 144 = 0.1393$$

$$D_n = 4A_1 / \text{Per}_1 = 0.377 \text{ ft}$$

The flow path is assumed to be half of the average circumference at the 45° angle

$$\begin{aligned} L &= 2\pi(R_o + R_i) / 2 \times 2 \times 12 \sin 45^\circ \\ &= 13.51 \text{ ft} \end{aligned}$$

The flow resistance is assumed to be due to friction over the path length L plus entrance and exit losses over the series of $N/2$ (27) staggered fin pieces in the same path.

$$\Delta P = \frac{f W^2 L}{2g \rho A^2 D_n} + \frac{W^2 N}{2g \rho^2} \left[\frac{K}{A^2} + (A^{-1} - (A + \Delta A)^{-1})^2 \right]$$

The driving force for the air flow is assumed to be due to buoyancy as

$$\Delta P = H \Delta \rho = H \bar{\rho} \Delta T = H \bar{\rho} Q / W C_p \bar{T}$$

where the head rise H is,

by design of the shroud

$$H = 66 \text{ in, } 5.5 \text{ ft}$$

In the range of typical flow rate

$$f = 0.477 (W D_n / A \mu)^{-3}$$

$$\bar{\rho} = 39.5 / \bar{T} \text{ lb/ft}^3$$

$$C_p = 0.24 \text{ BTU/lb}^\circ\text{F}$$

$$\mu = 0.05 \text{ lb/ft hr}$$

$$K = 0.05 \text{ for rounded entrance losses}$$

$\bar{T}^\circ\text{R}$ is the absolute temperature

The flow rate W lb/hr is estimated by equating the pressure drop to the buoyancy for $Q = 230,000 \text{ BTU/hr}$

$$W = 3.470 \times 10^6 / \bar{T} (0.0790 + W^{-3})^{1/3}$$

$$\bar{T} = 460 + T_a + \frac{1}{2} \frac{Q}{W C_p}$$

$$\text{For } T_a = 130^\circ\text{F}$$

$$W = 9,983 \text{ lb/hr } T_{\text{max}}^{\text{air}} = 210^\circ\text{F}$$

The cask surface temperature depends on the fin area, S , fin efficiency, η , and heat transfer coefficient, h .

Note that the radiant heat transfer is also ignored.

Continuing with the internal heat transfer, we have the conductances through the steel liner, U_1 , the uranium, U_2 , the stainless steel, U_3 , the water jacket (with clad copper tubes) U_4 , and the water jacket shell, U_5 . Assume an effective length $L = 154$ in., the same as the finned water jacket surface.

$$U_1 = 2\pi k L / \ln(R_o/R_i)$$

$$k_1 = 8.8 \text{ (stainless steel liner)}$$

$$R_o = 20.625", R_i = 20.125" \quad L = 12.83'$$

$$U_1 = 28,914 \text{ BTU/hr}^\circ\text{F} \quad \Delta T = 8^\circ\text{F}$$

$$k_2 = 20 \text{ (uranium shield)}$$

$$R_o = \text{variable (use average)}$$

$$R_o = 23.420 \quad R_i = 20.625" \quad L = 12.83'$$

$$U_2 = 12,691 \text{ BTU/hr}^\circ\text{F} \quad \Delta T = 18^\circ\text{F}$$

$$k_3 = 8.8 \text{ (stainless steel vessel)}$$

$$R_o = 27.5" \quad R_i = 23.42" \quad L = 12.83'$$

$$U_3 = 4,418 \text{ BTU/hr}^\circ\text{F} \quad \Delta T = 52^\circ\text{F}$$

$$k_5 = 8.8 \text{ (stainless steel jacket)}$$

$$R_o = 33.0 \quad R_i = 32.5 \quad L = 12.83'$$

$$U_5 = 46,477 \text{ BTU/hr}^\circ\text{F} \quad \Delta T = 5^\circ\text{F}$$

A natural convection of the water in the neutron shield can be expected to result in a very small temperature differential. Let us consider the loss of water case with the clad copper tubes serving as the conductance. Assume 36 tubes with two half-circle heat flow paths each:

$$h = 0.81 W C_p / A Pr^{2/3} (WZ/A\mu)^{.477}$$

(ref McAdams p. 270)

$$Z = \text{fin piece perimeter} = (.25\pi + 2 \times 3)/12$$

$$Z = .5654 \text{ ft. } \mu = .055 \text{ lb /ft hr at } 250^\circ\text{F}$$

$$Pr = .69, C_p = .2415 \text{ BTU/lb } ^\circ\text{F}$$

$$W/A = 928.3 \text{ lb/ft}^2 \text{ hr. } \therefore \underline{\underline{h = 2.94 \text{ BTU/ft}^2\text{hr}}}$$

$$\eta = (m\ell)^{-1} \text{ tank } m\ell \quad \text{for rectangular fins}$$

$$m\ell = \sqrt{h S_1 / k A \ell^{-1}}$$

$$h S_1 = \text{fin surface to air conductance}$$

$$k A \ell^{-1} = \text{cask body to fin conductance}$$

$$k \approx 200 \text{ BTU/ft hr}^\circ\text{F} \quad \ell^{-1} = 2 \text{ ft}^{-1}$$

$$A_1 = (\pi \times .125^2 + .25 \times 3)/144 = .00555 \text{ ft}^2$$

$$S_1 = Z\ell + A = 0.288 \text{ ft}^2 \text{ each fin piece}$$

$$m\ell = .6172 \quad \eta = .890 \quad \eta S = 0.256$$

The total number of the small fin pieces, over the 154 in. length of the neutron shield is 4320 for an effective surface area of 1107 ft².

Adding the 198 ft² of the neutron shield jacket surface, between the fin pieces, gives a total of 1305 ft². If the energy absorber ribs are ignored, with respect to their heat transfer capability, the surface temperature of the cask can be estimated from

$$T_s = T_{\text{air}} + Q/h(nS_f + S_e)$$

$$T_s = 270^\circ\text{F maximum}$$

$$\text{for } Q = 230,000 \text{ BTU/hr}$$

$$A = 0.20" \times 154" \times 2 \times 36/144 = 15.4 \text{ ft}^2$$

$$\ell = .2 / \ln (2.49/2.29) \times 12 = .625 \text{ ft}$$

$$U_4 = kA/\ell \quad k = 200 \text{ copper}$$

$$= 4925 \text{ BTU/hr}^\circ\text{F} \quad \Delta T = 47^\circ\text{F}$$

The sodium coolant temperature is expected to be in the range of 320°F on the bottom side of the cask to 400°F on the top side on the hottest day, $T_a = 130^\circ\text{F}$, with loss of water from the neutron shield.

Repeating the above calculation exercise with 9 in. vs. 6 in. fins indicates an increased air flow of 35% and a decreased surface temperature of 20° to 250°F at the top of the cask. This is probably not worth the additional weight and fabrication complication. The heat transfer gain is 25% at a 50% cost in weight. As illustrated, the temperatures and temperature differentials do not present any design or material selection problems. If it is desired to limit the normal operating temperature to a 200°F maximum, the neutron shield water may be circulated to a heat exchanger designed for 150,000 BTU/hr with air entry at 130°F and coolant entry at 200°F . In case of an accident (loss of coolant) passive cooling, comparable to the system analyzed above, can satisfy the regulatory requirements.

VIEW B

E - EXPANSION VOID
ISOLATED WITH BURST DISC

SECTION A-A

FIGURE-2
18 POSITION MAGAZINE
FOR
SODIUM COOLED LMFBK CASK

FIGURE-1
SODIUM COOLED LMFB CASK